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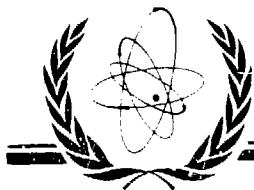
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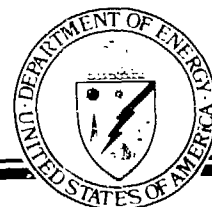
INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS

JUNE 3-21, 1985

Santa Fe, New Mexico and San Clemente, California



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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

FOREWARD

The fifth International Training Course on Implementation of State Systems of Accounting for and Control of Nuclear Materials (SSAC) was held June 3 through June 21, 1985, in Santa Fe and Los Alamos, New Mexico, and in San Clemente, California. The SSAC courses are sponsored by the US Department of Energy (DOE) in cooperation with the International Atomic Energy Agency (IAEA) to provide practical training in the design, implementation, and operation of systems for nuclear materials accounting and control.

Over 75 people took part in the 1985 course, including attendees, lecturers, workshop leaders, equipment demonstrators, and plant engineering and operations personnel. Nations represented in addition to the US were Bangladesh, Bolivia, Brazil, Canada, Czechoslovakia, Egypt, Hungary, India, Indonesia, Iraq, Japan, Korea, Morocco, Pakistan, the People's Republic of China, Philippines, Poland, Spain, Sweden, and Taiwan. Course attendees are involved in nuclear safeguards at either the national or facility level in their home countries, and many hold positions of major responsibility.

The 1985 SSAC course was conducted by the Los Alamos National Laboratory and the San Onofre Nuclear Generating Station of Southern California Edison. Course lecturers were selected not only from Los Alamos and the San Onofre Nuclear Generating Station, but also from the IAEA, DOE, the State Department, Nuclear Regulatory Commission (NRC), Sandia National Laboratories, Exxon Nuclear Company, and from nuclear organizations in Canada, Finland, Hungary, and Japan. Emphasis for the 1985 course was on safeguards procedures for item-control facilities, such as nuclear power reactors and experimental reactors. The course also included a section introducing application of safeguards at bulk processing facilities such as low-enriched uranium conversion and fuel fabrication plants.

The first two-thirds of the course was conducted in Santa Fe and Los Alamos, New Mexico. During this period, introductory course material was presented by lecturers from the State Department, NRC, the IAEA, Sandia National Laboratories, Canada, Finland, Hungary, and Japan. Attendees made brief presentations regarding nuclear activities in their home countries, and there were panel and small group discussions of the IAEA/state system interface. Several technical lectures were presented by people from Los Alamos and two were spent in Los Alamos safeguards laboratories where the attendees used nondestructive assay (NDA) instruments to measure typical nuclear materials.

The final week of the course was conducted in San Clemente, California. This period involved lectures and demonstrations by senior employees of the San Onofre Nuclear Generating Station of Southern California Edison. One morning was spent at Unit 2 of the San Onofre Nuclear Generating Station and

separate tours were made of the Southern California Edison Training Center at San Clemente. Lectures were also presented in San Clemente by representatives of the NRC, the IAEA, and the Exxon Nuclear Company. The course concluded with a two-day workshop on safeguards system design for a nuclear power generating station. Attendees were divided into four subgroups, each with a staff advisor, and each subgroup was asked to define (in some detail) the elements of their proposed safeguards system. Rapporteurs from each subgroup presented the proposed designs and the four approaches were summarized and compared by members of the course staff.

The course was structured to encourage discussion among attendees and between attendees and course staff. The formal interactions during the lectures and workshops and the informal interactions outside the classroom were of great value to both students and staff. The SSAC courses offer an almost unique opportunity for the sharing of safeguards knowledge and experience among people with differing professional and cultural backgrounds.

These published proceedings of the 1985 SSAC course can be obtained by writing to:

Safeguards Training Program
Q-1, MS E540
Los Alamos National Laboratory
Los Alamos, NM 87545

E. Arnold Hakkila
T. Douglas Reilly
January 21, 1986



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

ABSTRACT

This report incorporates all lectures and presentations at the International Training Course on Implementation of State Systems of Accounting for and Control of Nuclear Materials held June 3 through June 21, 1985, at Santa Fe and Los Alamos, New Mexico, and San Clemente, California. Authorized by the US Nuclear Non-Proliferation Act and sponsored by the US Department of Energy in cooperation with the International Atomic Energy Agency, the course was developed to provide practical training in the design, implementation, and operation of a state system of nuclear materials accountability and control that satisfies both national and international safeguards requirements. Major emphasis for the 1985 course was placed on safeguards methods used at item-control facilities, particularly nuclear power generating stations and test reactors. An introduction to safeguards methods used at bulk handling facilities, particularly low-enriched uranium conversion and fuel fabrication plants, was also included. The course was conducted by the University of California's Los Alamos National Laboratory and the Southern California Edison Company. Tours and demonstrations were arranged at the Los Alamos National Laboratory, Los Alamos, New Mexico, and the San Onofre Nuclear Generating Station, San Clemente, California.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

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S. Pillay, Los Alamos National Laboratory

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
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AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
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June 3 - 21, 1985

COURSE SYLLABUS

MONDAY, JUNE 3, 1985

- o REGISTRATION

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TUESDAY, JUNE 4, 1985

- o WELCOME (IAEA, DOE, LOS ALAMOS)
- SESSION 1 INTRODUCTION TO SSAC TRAINING COURSE
- o Course Subjects, Structure, Schedule, Materials
 - o Introduction of Course Staff
 - o Administrative Arrangements and Announcements
- SESSION 2 HISTORICAL DEVELOPMENT AND CURRENT TRENDS IN NUCLEAR SAFEGUARDS (A. Sessoms, US Department of State)
- o History of IAEA Safeguards
 - o Non-Proliferation Treaty
 - o US Policy on Nonproliferation and Safeguards
 - o Current Trends
- SESSION 3 INTRODUCTION TO IAEA SAFEGUARDS (C. Büchler, IAEA)
- o Objectives of IAEA Safeguards
 - o Diversion of Nuclear Material
 - o The IAEA Safeguards System
 - o Guidelines for State Systems of Accounting and Control
- SESSION 4 THE US NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM (C. N. Smith, US Nuclear Regulatory Commission)
- o The Need for National Nuclear Materials Safeguards
 - o Evolution of Safeguards in the US Private Sector
 - o US National MC&A Regulations
 - o Records and Reporting Requirements
 - o Auditing Procedures

IAEA RECEPTION, 6:00 PM TO 8:00 PM

WEDNESDAY, JUNE 5, 1985

SESSION 5 THE HUNGARIAN NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM
(O. Lendvai, National Atomic Energy Commission of Hungary)

- o Requirements and Regulations
- o SSAC Organization and Responsibilities
- o Facility Design Impact
- o Inspections
- o Reporting Requirements
- o SSAC Interface with Operator, IAEA, Other Countries

SESSION 6 WORKSHOP ON STATE SYSTEMS OF ACCOUNTING AND CONTROL

- o Brief Presentations by Course Attendees

SESSION 7 WORKSHOP ON IAEA/STATE SYSTEM INTERFACE

- o Subgroup Meetings to Identify Questions for Panel
- o Panel Discussion (IAEA, NRC, DOS, Hungary)

SECTION II: FUEL MEASUREMENT PROBLEMS AND METHODS

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SESSION 8 ELEMENTS OF NONDESTRUCTIVE ASSAY TECHNOLOGY (H. Smith,
Los Alamos)

- o Role of NDA
- o Passive and Active NDA methods
- o Gamma-Ray Assay Techniques and Instruments
- o Neutron Assay Techniques and Instruments
- o Relative Advantages of Different NDA Techniques

SESSION 9 NDA INSTRUMENTS FOR MEASUREMENT OF UNIRRADIATED FUEL
(H. Menlove, Los Alamos)

- o Enrichment Meter Verification Measurements
- o Autoradiography
- o Fuel Assembly Measurement with Neutron Coincidence Collar
- o Experience at Operating Facilities

FRIDAY, JUNE 7, 1985

SESSION 10 TOUR LOS ALAMOS SAFEGUARDS RESEARCH AND DEVELOPMENT LABORATORY;
DEMONSTRATION AND USE OF NDA INSTRUMENTS FOR ASSAY OF
UNIRRADIATED FUELS

- o Uranium Enrichment Measurement
- o LWR Fuel Assembly Measurement with Neutron Coincidence Collar
- o Gamma-Ray Spectroscopy Measurement of Plutonium Isotopic Composition
- o Measurement of Highly Enriched Uranium with Active Well Coincidence Counter
- o Materials Accounting Simulation

SATURDAY AND SUNDAY, JUNE 8-9, 1985

- o FREE: OPTIONAL TOURS OF ALBUQUERQUE AND NORTHERN NEW MEXICO

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(R. Gutmacher, Los Alamos)

- o Key Accountability Points in LWR Fuel Fabrication
- o Chemical Assay Techniques for Total Uranium
- o Mass Spectrometric Determination of Uranium Isotopic Composition
- o Precision and Accuracy of Chemical Assay Techniques

SESSION 12 SAFEGUARDING NUCLEAR POWER STATION (G. Whan, University of New Mexico)

- o Characteristics of Nuclear Power Stations
- o Typical Fuel Inventory at Nuclear Power Station
- o Characteristic Gamma-Ray and Neutron Emissions from Irradiated Fuel
- o Principles to Relate NDA Measurements to Fuel Quantities

SESSION 13 GAMMA-RAY ASSAY OF IRRADIATED FUEL (P. Rinard, Los Alamos)

- o Gamma-Ray Activity and Profile Measurement
- o Fission Product Measurement
- o Isotopic Correlations
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SESSION 14 NEUTRON ASSAY OF IRRADIATED FUEL (G. Bosler, Los Alamos)

- o Coincidence and Single Neutron Measurements
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- o Isotope Correlation Technique

TUESDAY, JUNE 11, 1985

SESSION 16 TOUR OF THE OMEGA WEST REACTOR (OWR)

- o Irradiated Fuel Measurement Demonstrations at Materials Testing Reactor: Cerenkov Viewing Device, Total Gamma and Neutron Activity Measurements, Fission Product Measurements
- o Afternoon Tour of Science Museum and Bandelier National Monument

DINNER AT CASA DEL MIRADOR

SECTION III: REACTOR SAFEGUARDS PRACTICE

WEDNESDAY, JUNE 12, 1985

SESSION 17 PHYSICAL PROTECTION IN RELATION TO IAEA SAFEGUARDS
(C. Sonnier, Sandia National Laboratories)

- o Role of Physical Security in the State System
- o Interface between Physical Security, Containment and Surveillance, and Materials Accountancy
- o Physical Security Techniques

SESSION 18 MATERIAL CONTROL AND ACCOUNTING AT A RESEARCH REACTOR
(H. Kuroi, Japan Atomic Energy Research Institute)

- o Research Reactor and Critical Facility Description
- o Material Control and Accounting Requirements
- o Personnel and Resource Requirements
- o Records, Reports, Data Forms, Written Procedures
- o Operating Practice and Experience
- o Problems Encountered at State and Facility Level

SESSION 19 DESCRIPTION OF AN SSAC AT THE FACILITY LEVEL FOR RESEARCH REACTORS AND CRITICAL FACILITIES (E. Weinstock, Brookhaven National Laboratory)

- o Facility Description
- o MC&A Requirements
- o Recommended System Design Criteria

SESSION 20 IAEA SAFEGUARDS EXPERIENCE AT RESEARCH REACTORS AND CRITICAL FACILITIES (W. Theis, IAEA)

- o Facilities under IAEA Safeguards
- o Report and Records Requirements
- o Inspection Activities and Techniques
- o Measurement Instruments and Experience

THURSDAY, JUNE 13, 1985

SESSION 21 MATERIAL CONTROL AND ACCOUNTING AT A CANDU REACTOR (E. Payne, Atomic Energy of Canada, Ltd.)

- o CANDU Power Station Description
- o Safeguards Procedures
- o Control Areas, Seals
- o Physical Inventory Verification
- o Measurement Instruments in Use

SESSION 22 IAEA SAFEGUARDS EXPERIENCE AT CANDU REACTORS (L. Wredberg, IAEA)

- o Facilities under IAEA Safeguards
- o Report and Records Requirements
- o Inspection Activities and Techniques
- o Measurement Instruments and Experience

SESSION 23 MATERIALS ACCOUNTING AND CONTROL FOR LIGHT-WATER REACTORS AND
SPENT FUEL STORAGE FACILITIES IN FINLAND (M. Riihonen,
STUK-Finnish Centre for Radiation and Nuclear Safety)

- o Power Station Description
- o Safeguards Organizational Structure
- o Records, Reports, Data Forms, Written Procedures
- o Control Areas, Seals
- o Physical Inventory Verification
- o Measurement Instruments in Use

SESSION 24 IAEA SAFEGUARDS EXPERIENCE AT LWR POWER REACTORS (W. Theis,
IAEA)

- o Facilities under IAEA Safeguards
- o Report and Records Requirements
- o Inspection Activities and Techniques
- o Measurement Instruments and Experience

FRIDAY, JUNE 14, 1985

SESSION 25 EXPERIENCE AND PROBLEMS SAFEGUARDING REACTOR FACILITIES

- o Subgroup Meetings to Identify Questions for Panel
- o Panel Discussion (IAEA, Japan, Canada, BNL)

TRAVEL TO SAN DIEGO, CALIFORNIA

SATURDAY AND SUNDAY, JUNE 15-16, 1985

- o FREE: OPTIONAL TOURS OF SAN DIEGO

TRAVEL TO SAN CLEMENTE, CALIFORNIA

SECTION IV: FACILITY VISIT AND WORKSHOP

AT SAN ONOFRE NUCLEAR GENERATING STATION (SONGS)
San Clemente, California

MONDAY, JUNE 17, 1985

SESSION 26 INTRODUCTION TO SAN ONOFRE NUCLEAR GENERATING STATION
(S. Wylie, SONGS)

- o Facility Description and Photographic Slide Tour

SESSION 27 NUCLEAR FUELS MANAGEMENT, PROGRAM ADMINISTRATION, AND IAEA
INTERFACE AT THE SAN ONOFRE NUCLEAR GENERATING STATION
(S. Wylie and SONGS staff)

- o Material Control and Accountability at SONGS
- o Safeguards Organizational Structure
- o Fuel Handling
- o Reports, Records, Procedures
- o IAEA Interface

SESSION 28 TOUR OF THE SONGS TRAINING CENTER/REACTOR TOUR PREPARATION
(S. Wylie and SONGS Staff)

TUESDAY, JUNE 18, 1985

SESSION 29 TOUR OF THE SONGS REACTOR FACILITY (S. Stilwagen and SONGS Staff)

SESSION 30 U.S. NUCLEAR REGULATORY COMMISSION INSPECTION PROCEDURES AT THE SONGS (L. Norderhaug, NRC)

- o Records and Reporting Requirements
- o Inspection Activities and Experience
- o Measurement Instruments and Experience

SESSION 31 IAEA INSPECTION PROCEDURES AT SONGS (L. Wredberg, IAEA)

- o Records and Reporting Requirements
- o Inspection Activities and Experience
- o Measurement Instruments and Experience

WEDNESDAY, JUNE 19, 1985

SESSION 32 INTRODUCTION TO EXXON NUCLEAR FUEL FABRICATION FACILITY
(R. Schneider, Exxon Nuclear Company, Inc.)

- o Facility Description
- o LWR Fuel Fabrication Process Description
- o Photographic Slide Tour of Facility
- o Model Plant, KMP/MBA/MCA Structure

SESSION 33 NUCLEAR MATERIAL CONTROL AND ACCOUNTING AT EXXON NUCLEAR, I
(R. Schneider, Exxon Nuclear Company, Inc.)

- o Basis for MC&A at a Bulk-Handling Facility
- o Accounting Systems - NMRS and NICS
- o Physical Inventory Taking
- o IAEA Inspection
- o Safeguards Organization

SESSION 34 NUCLEAR MATERIAL CONTROL AND ACCOUNTING AT EXXON NUCLEAR, II
(R. Schneider, Exxon Nuclear Company, Inc.)

- o Measurements - Sampling, Analytical, NDA and Volume
- o Measurement Control
- o MUF Evaluation
- o Acct. Forms, Composition Codes and Tamper-Indicating Seals

SESSION 35 SAFEGUARDS INSPECTION EXPERIENCE AT US NUCLEAR FACILITIES
(IAEA, NRC, SONGS, EXXON NUCLEAR)

- o Panel Discussion

THURSDAY, JUNE 20, 1985

SESSION 36 INTRODUCTION TO THE MC&A SYSTEM DESIGN WORKSHOP
(K. K. S. Pillay, Los Alamos)

- o Objectives of Workshop
- o Model LWR Facility Assumptions
- o Preparation of Design Information Questionnaire
- o Instructions to Subgroups

SESSION 37 MC&A SYSTEM DESIGN WORKSHOP
(K. K. S. Pillay, Workshop Coordinator)

- o Organizational Design
- o MBA and KMP Design
- o Reports and Records
- o Physical Inventory Procedures
- o Measurements Required
- o Prepare Subgroup Design Report
- o Select Rapporteur

AWARDS BANQUET

FRIDAY, JUNE 21, 1985

SESSION 38 REPORTS OF MC&A SYSTEM DESIGN WORKSHOP SUBGROUPS

- o Subgroup Reports Presented by Rapporteurs
- o Discussion of Subgroup Reports

SESSION 39 COURSE EVALUATION, DISCUSSION, AND WRAP-UP

- o Attendees Complete Detailed Evaluation Questionnaire
- o Discuss Course Benefits and Suggestions for Course Improvements

END OF COURSE



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

REGISTRATION
MONDAY 3 JUNE
1500-1800

TUESDAY 4 JUNE	WEDNESDAY 5 JUNE	THURSDAY 6 JUNE	FRIDAY 7 JUNE	SATURDAY 8 JUNE	SUNDAY 9 JUNE	MONDAY 10 JUNE	TUESDAY 11 JUNE	WEDNESDAY 12 JUNE
			BUS TO LOS ALAMOS			BUS TO LOS ALAMOS		
0815 WELCOME AND ORIENTATION	5 THE HUNGARIAN NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM (LENDVAY)	8 ELEMENTS OF NONDESTRUCTIVE TECHNOLOGY (SMITH)	10 LOS ALAMOS SAFEGUARDS ORIENTATION AND BRIEFING (SMITH, REILLY)			11 THE CHEMICAL ASSAY OF UNIRRADIATED FUEL MATERIALS (GUTMACHER)	16 OMEGA WEST REACTOR TOUR AND SPENT FUEL MEASUREMENTS DEMONSTRATION (BOSLER, STAFF)	17 PHYSICAL PROTECTION IN RELATION TO IAEA SAFEGUARDS (SONNIER)
0900 LOS ALAMOS DOE IAEA								
0915 1 INTRODUCTION TO SSAC TRAINING COURSE REILLY								
1000 BREAK	BREAK	BREAK	10 (CONTINUED) LOS ALAMOS NDA DEMONSTRATIONS AND MC & A SIMULATION (LOS ALAMOS STAFF)			BREAK		BREAK
1100 2 HISTORICAL DEVELOPMENT AND CURRENT TRENDS IN NUCLEAR SAFEGUARDS SESSIONS	6 WORKSHOP ON SSAC BRIEF PRESENTATIONS BY ATTENDEES (MEYERS)	9 NOA INSTRUMENTS FOR MEASUREMENT OF UNIRRADIATED FUEL (MENLOVE)				12 REACTOR OPERATION AND UNIRRADIATED FUEL CHARACTERISTICS (FOLEY)		18 MATERIAL CONTROL AND ACCOUNTING AT A RESEARCH REACTOR (KUROH)
1200 LUNCH	LUNCH	LUNCH	LUNCH	FREE TIME OPTIONAL TOURS	FREE TIME OPTIONAL TOURS	LUNCH	LUNCH	LUNCH
1330 3 INTRODUCTION TO IAEA SAFEGUARDS (BUECHLER)	7 WORKSHOP ON IAEA STATE SYSTEM INTERFACE COURSE (SUBGROUPS)	FINISH PRESENTATIONS BY ATTENDEES (SESSION 6)	10 (CONTINUED)			13 GAMMA-RAY ASSAY OF IRRADIATED FUEL (RINARD)	14 NEUTRON ASSAY OF IRRADIATED FUEL (BOSLER)	19 DESCRIPTION OF A SSAC AT THE FACILITY LEVEL FOR RESEARCH REACTORS AND CRITICAL FACILITIES (WEINSTOCK)
1430 BREAK								
1500 BREAK	7. PANEL DISCUSSION IAEA STATE SYSTEM INTERFACE (SESSIONS)	FREE TIME				BREAK		BREAK
1530 4 THE U.S. NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM (SMITH)						15 ISOTOPE CORRELATION AND MASS SPECTROMETRY TECHNIQUES FOR IRRADIATED FUEL ASSAY (DERON)	20 IAEA SAFEGUARDS EXPERIENCE AT RESEARCH REACTORS (THEIS)	
1700			BUS BACK TO SANTA FE				DINNER AT CASA DEL MIRADOR	
1800 IAEA RECEPTION								
2000								



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS

June 3 - 21, 1985



	THURSDAY 13 JUNE	FRIDAY 14 JUNE	SATURDAY 15 JUNE	SUNDAY 16 JUNE	MONDAY 17 JUNE	TUESDAY 18 JUNE	WEDNESDAY 19 JUNE	THURSDAY 20 JUNE	FRIDAY 21 JUNE
0830	21. MATERIAL CONTROL AND ACCOUNTING AT A CANDU REACTOR (STIRLING)	25. WORKSHOP ON EXPERIENCE AND PROBLEMS SAFEGUARDING REACTOR FACILITIES (SUBGROUPS)			28. INTRODUCTION TO SAN ONOFRE NUCLEAR GENERATING STATION (SONGS) (WYLIE)	29. TOUR OF THE SONGS REACTOR FACILITY (STILWAGEN, STAFF)	32. INTRODUCTION TO EXXON NUCLEAR FUEL FABRICATION FACILITY (SCHNEIDER)	36. INTRODUCTION TO THE MC & A SYSTEM DESIGN WORKSHOP (PILLAY)	38. REPORT OF MC & A SYSTEM DESIGN WORKSHOP SUBGROUPS
0930		BREAK							
1000	BREAK	25. PANEL DISCUSSION EXPERIENCE AND PROBLEMS SAFEGUARDING REACTOR FACILITIES			BREAK		BREAK	BREAK	BREAK
1030	22. IAEA SAFEGUARDS EXPERIENCE AT CANDU REACTORS (WREDBERG)		FREE TIME	FREE TIME	27. NUCLEAR FUELS MANAGEMENT, PROGRAM ADMINISTRATION, AND IAEA INTERFACE AT THE SONGS (WYLIE, STAFF)		33. NUCLEAR MATERIAL CONTROL AND ACCOUNTING AT EXXON NUCLEAR, L (SCHNEIDER)	37. MC & A SYSTEM DESIGN WORKSHOP (PILLAY)	38. (CONTINUED)
1200	LUNCH	LUNCH			LUNCH	LUNCH	LUNCH	LUNCH	LUNCH
1330	23. MATERIALS ACCOUNTING AND CONTROL AT AN OPERATING LWR NUCLEAR POWER STATION (VUORINEN)	TRAVEL TO SAN DIEGO			28. TOUR OF THE SONGS TRAINING CENTER (WYLIE, STAFF)	30. U.S. NRC INSPECTION PROCEDURES AT THE SONGS (NORDERHAUG)	34. MC & A AT EXXON NUCLEAR, IL (SCHNEIDER)	37. (CONTINUED)	39. COURSE EVALUATION, DISCUSSION, AND WRAP-UP
1500	BREAK					BREAK	BREAK		
1530	24. IAEA SAFEGUARDS EXPERIENCE AT POWER REACTORS (THEIS)					31. IAEA INSPECTION PROCEDURES AT SONGS (WREDBERG)	35. PANEL DISCUSSION SAFEGUARDS INSPECTION EXPERIENCE AT U.S. NUCLEAR FACILITIES		
1800									
1700									
1830								AWARDS BANQUET	



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SSAC SCHOOL LIBRARY

We assembled a number of books, reports, conference proceedings, and journals pertaining to Nuclear Material Accounting and Control for the students' information. These items were placed in the lecture hall in Santa Fe, and all course attendees are invited to browse through the materials at any time. Extra copies of lecturers' handouts were also placed in the library display.

Listed below were the items in the library display:

PREVIOUS SSAC COURSE PROCEEDINGS:

- "International Training Course on Nuclear Materials Accountability for Safeguards Purposes," LA-8620-C Volumes I and II (1980).
- "Advanced International Training Course on State Systems of Accounting for and Control of Nuclear Materials," LA-8901-C (1981).
- "International Training Course on Implementation of State Systems of Accounting for and Control of Nuclear Materials," LA-9609-C (1982).
- "International Training Course on Implementation of State Systems of Accounting for and Control of Nuclear Materials," LA-10135-C (1983).

PROCEEDINGS OF INMM MEETINGS (INSTITUTE OF NUCLEAR MATERIALS MANAGEMENT):

Palm Beach, FL, June 30-July 2, 1980
San Francisco, CA, July 13-15, 1981
Washington, DC, July 18-21, 1982
Vail, CO, July 10-13, 1983
Columbus, OH, July 15-18, 1984

"Measurement Technology for Safeguards and Materials Control," American Nuclear Society Topical Conference, Kiawah Island, SC, November 26-30, 1979, NBS Special Publication 582.

"Safeguards Technology: The Process-Safeguards Interface," ANS INMM-DOE Conference, Hilton Head Island, SC, CONF 831106, Nov. 28-Dec. 2, 1983.

Nuclear Materials Management (INMM Journal)

Summer 1982 Issue
Winter 1982 Issue
Spring 1983 Issue
Summer 1983 Issue
Winter 1983 Issue
Spring 1984 Issue
Summer 1984 Issue

SAFEGUARDS SYSTEMS:

- "The Structure of Safeguards Systems," James P. Shipley, Los Alamos report LA-7337-MS (1978).
- "Some Safeguards Approaches and Evaluation Methods," James P. Shipley, Nucl. Mat. Mgmt. VIII, No. 4 (1979).
- "Performance Analysis of Nuclear Materials Accounting Systems," Donald D. Cobb and James P. Shipley, Nucl. Mat. Mgmt. VIII (1979).
- "Coordinated Safeguards for Materials Management in a Mixed-Oxide Fuel Facility," James P. Shipley, et al., Los Alamos report LA-6536 (1977).
- "Coordinated Safeguards for Materials Management in a Fuel Reprocessing Plant," E. Arnold Hakkila, et al., Los Alamos report LA-6881 (Volumes I and II) (1977).
- "Materials Management in an Internationally Safeguarded Fuels Reprocessing Plant," E. Arnold Hakkila, et al., Los Alamos report LA-8042 (Volumes I, II, and III) (1980).
- "Coordinated Safeguards for Materials Management in a Nitrate-to-Oxide Conversion Facility," Hassan Aly Dayem, et al., Los Alamos report LA-7011 (1978).
- "Coordinated Safeguards for Materials Management in a Uranium-Plutonium Nitrate-to-Oxide Conversion Facility," Hassan Aly Dayem, et al., Los Alamos report LA-7521 (1979).
- "Concepts for Inventory Verification in Critical Facilities," Donald D. Cobb, et al., Los Alamos report LA-7315 (1978).
- "Integrated International Safeguards Concepts for Fuel Reprocessing," E. Arnold Hakkila, Los Alamos report LA-8955; Sandia Laboratories report SAND81-1311 (1981).
- "Design and Evaluation of An Integrated Safeguards System," Jack T. Markin, et al., Vol I: Principles; Vol II: Applications; Los Alamos report LA-9564 (1984).

NUCLEAR SAFEGUARDS RESEARCH AND DEVELOPMENT:

- "Safeguards in the Seventies - A Bibliography of LASL Safeguards R&D Publications," Sharon L. Klein, Los Alamos report LA-8663-MS (1981).
- "Safeguards and Security Status report August 1981-January 1982," James P. Shipley, Los Alamos report LA-9373-PR (1982).
- "Safeguards and Security Status report February-July 1982," James P. Shipley and Darryl B. Smith, Los Alamos report LA-9595-PR (1983).
- "Safeguards and Security Status report, August 1982-January 1983," Darryl B. Smith, Los Alamos report LA-9821-PR (1983).

"Safeguards and Security Status report, January-December, 1983,"
Darryl B. Smith, Los Alamos report LA-10170-PR (1984).

Los Alamos Science, Vol I, No. 1, Reprint Issue on Nuclear Safeguards
(1980).

NDA INSTRUMENTATION:

"Description and Operation Manual for the Active Well Coincidence
Counter," Howard O. Menlove, Los Alamos report LA-7823-MS (1979).

"Description and Performance Characteristics for the Neutron Coincidence
Collar for the Verification of Reactor Fuel Assemblies,"
Howard O. Menlove, Los Alamos report LA-8939-MS (1981).

"Gamma-Ray Measurements with the Segmented Gamma Scan," E. Ray Martin,
David F. Jones, and Jack L. Parker, Los Alamos report LA-7059-M
(1977).

"Measurement of Uranium and Plutonium in Solid Waste by Passive Photon or
Neutron Counting and Isotopic Neutron Source Interrogation,"
Tom W. Crane, Los Alamos report LA-8294-MS (1980).

Neutron Measurement Techniques for the NDA of Irradiated Fuel
Assemblies," John R. Phillips, et al., Los Alamos report LA-9002-MS
(1981).

"The Feasibility of NDA Measurements in Uranium Enrichment Plants,"
Roddy B. Walton, Los Alamos report LA-7212-MS (1978).

"NDA Verification and Assay Systems for Spent Fuels," Donald D. Cobb,
et al., Los Alamos report LA-9041 (1982).

"Active Neutron Coincidence Counting for the Assay of MTR Fuel Elements,"
R. Sher, Los Alamos report LA-9665-MS (1983).

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

"High-Level Neutron Coincidence Counter (HLNCC) Implementation: Assay of
the Plutonium Content of Mixed-Oxide Fuel Assemblies," John E. Foley
and Gerald E. Bosler, Los Alamos report LA-9318-MS (1982).

"Application of the Active-Well Coincidence Counter (AWCC) to
Highly-Enriched Uranium Metal," Howard O. Menlove and
Gerald E. Bosler, Los Alamos report LA-8621-MS (1981).

"Automated Instruments for In-Line Accounting of Highly-Enriched Uranium
at the Oak Ridge Y-12 Plant," Phyllis A. Russo, et. al., Los Alamos
report LA-10243-MS (1985).

"Liquid Sample Shuffler," Tom W. Crane, Los Alamos report LA-10291-MS
(1985).

"Irradiated Fuel Examination Using the Cerenkov Technique,"
Nicholas Nicholson and Ed Dowdy, Los Alamos report LA-8767-MS (1981).

"Vehicle Monitors for Domestic Perimeter Safeguards," Paul Fehlau,
et al., Los Alamos report LA-9633-MS (1983).

NDA TRAINING: (Brochures Available)

"Fundamentals of Passive NDA of Fissionable Material,"
Ronald H. Augustson and T. Douglas Reilly, Los Alamos report
LA-5651-M (1974). (2 copies)

"A Guide to Gamma-Ray Assay for Nuclear Material Accountability,"
T. Douglas Reilly and Jack L. Parker, Los Alamos report LA-5794-M
(1975).

Laboratory Manual: "Fundamentals of Nondestructive Assay of Fissionable
Materials Using Portable Instrumentation," Last Given October, 1984.

Laboratory Manual, "Gamma-Ray Spectroscopy for Nuclear Material
Accountability", Last given in December 1984.

Laboratory Manual: "Neutron Assay of Nuclear Material," Last Given,
June, 1983.

Laboratory Manual: "NDA For IAEA Inspectors," Last Given in April 1985.

Laboratory Manual: "Attributes and Semi-Quantitative Measurements of
Nuclear Materials," Last given, March, 1985.

Course Manual: "Materials Accounting for Nuclear Safeguards," Last
given, April 1985.

BOOKS:

A Guidebook to Nuclear Reactors, A. V. Nero (1979).

Active NDA of Nuclear Materials, T. Gozani NUREG/CR-0602, (1981).

Handbook of Nuclear Safeguards Measurement Methods, D. Rogers, ed. NUREG/
CR-2078 (1983).

The Detection of Fissionable Materials by Nondestructive Means, R. Sher
and S. Untermeyer II, ANS Monograph (1980).

Selected Measurement Data for Plutonium and Uranium, M. Anderson and
J. Lemming, Mound Laboratory report MLM-3009 (1982).

An Introduction to NDA Instrumentation, Tom R. Canada, IAEA (1984).

ESARDA:

Bulletin, Number 2, June 1982
Bulletin, Number 3, November 1982
Bulletin, Number 4, April 1983
Bulletin, Number 5, October 1983

Bulletin, Number 6, April 1984
Bulletin, Number 7, October 1984

ESARDA 4th Symposium Proceedings 1982 (ESARDA 15)
ESARDA 5th Symposium Proceedings 1983 (ESARDA 16)
ESARDA 6th Symposium Proceedings 1984 (ESARDA 17)
ESARDA Joint Specialists' Meeting 1984 (ESARDA 18)

SPECIAL REPORTS:

"Detail Description of an SSAC at the Facility Level for a Low-Enriched Uranium Conversion and Fuel Fabrication Facility," R. Jones and Eugene Weinstock, ISPO-200 (1984).

"Documentation and Analysis of IAEA Safeguards Implementation at the Exxon Nuclear Fuel Fabrication Plant," Richard A. Schneider, Exxon report XN-NF-84-86 (1984).

IAEA LITERATURE:

The Statute

"The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Nonproliferation of Nuclear Weapons," IAEA INFCIRC/153 (Corrected) (1972). (3 copies)

"The Agency's Safeguards System," IAEA INFCIRC/66/Rev. 2 (1968).

"Nonproliferation and International Safeguards," IAEA Public Information Booklet for the 1978 UN General Assembly Session on Disarmament (1978).

"IAEA Safeguards Glossary," IAEA/SG/INF/1, (1980).

"Guidelines for States' Systems of Accounting for and Control of Nuclear Materials," IAEA/SG/INF/2, (1980).

"IAEA Safeguards, An Introduction," IAEA/SG/INF/3, (1981).

Nuclear Safeguards Technology, IAEA Symposium Proceedings, Vols I and II, (1982).

SSAC group photo outside the Santa Fe Hilton Hotel.





SSAC Course Sponsors and Staff: Left to Right-Bob Keepin, LANL; Gene Bates, IAEA; Carlos Büchler, IAEA; Allen Sessoms, DOS; David Myers, DOE; and Arnie Hakkila, LANL.



SSAC Course Administrative Support: Left to Right-Luz Woodwell, Linda Robinson, and Cathy McCabe.



"Oh My Aching Head." SSAC Course Administrative Support, Cathy McCabe.

Course Elements: Lectures



Classroom scene at Santa Fe Hilton.



A. Hakila, SSAC Course Director, lecturing.

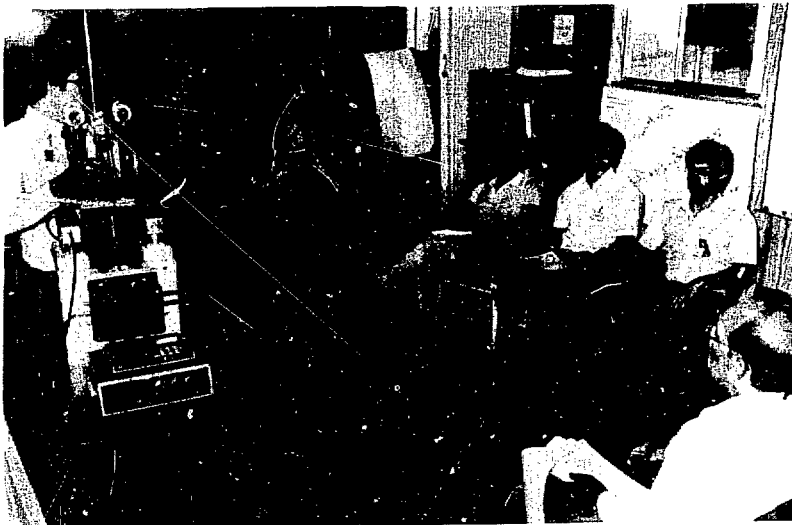
Course Elements



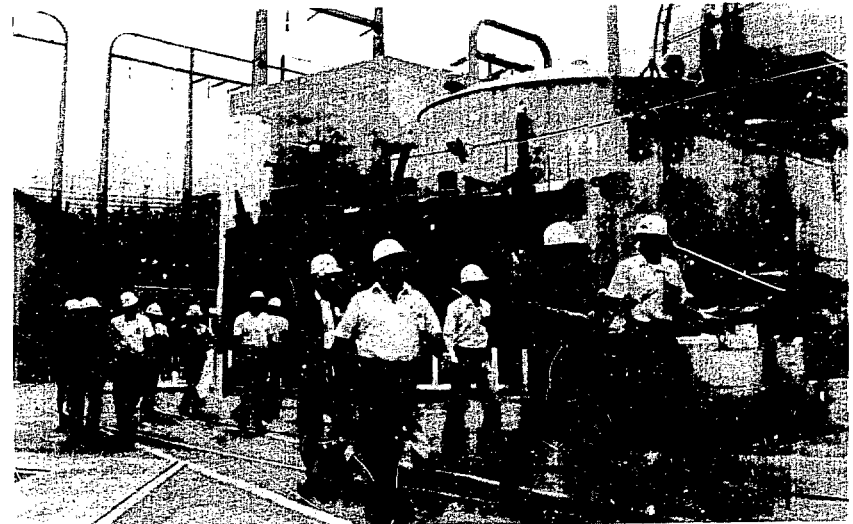
Subgroup discussions.



Panel discussions.



Experimental demonstrations.



Tours.

Course Elements: Parties and Informal Discussions



Krzysztof Zarnowiecki and Ryszard Zarucki entertain group with Polish folk songs.



Course Director, Doug Reilly, leads group in singing American folk songs.



"Which one are we singing next? I didn't know that one of the course requirements was to sing in English."

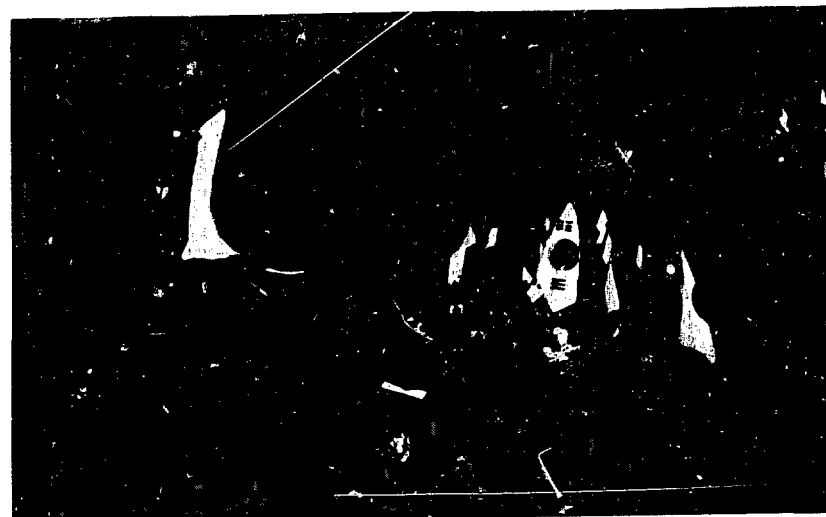


International Relations: Iraq, Korea, Pakistan, Taiwan, and Egypt.

Final Awards Banquet



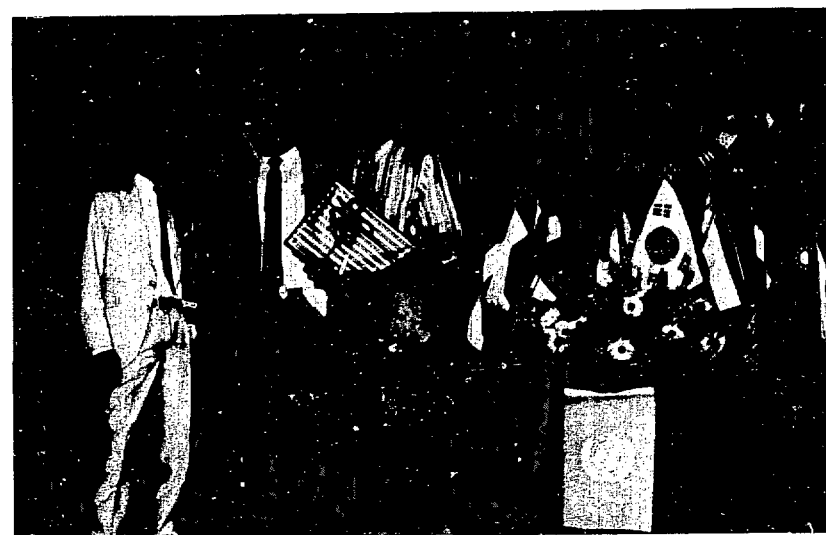
Final banquet was a seafood dinner at Delaneys overlooking yachts in Dana Point Marina.



Bob Keepin delivering banquet address.



Chuang (Jack) Hung-Chan receiving diploma and telling history of his flag.



Even the course staff got awards.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 1: INTRODUCTION TO SSAC TRAINING COURSE

The IAEA scientific advisor will present a brief overview of the SSAC training courses sponsored by the IAEA.

The course schedule and syllabus will be described briefly. The course staff will be introduced. Time will be available for administrative announcements and participant questions.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 1: INTERNATIONAL SSAC TRAINING COURSES

Olan Gene Bates

International Atomic Energy Agency

On behalf of the Safeguards Training Section of the Department of Safeguards, I would like to add my welcome to those already given by the previous speakers. It has been a real pleasure to work closely with the organizers of this training course. After having heard the excellent overview of the training that is in store for you, I think you would agree with me that this is going to be a very interesting and informative course in which there will be ample opportunity for learning about SSACs through the various lectures, discussions, workshops, and tours. I would also like to reinforce the encouragement that you ask questions and give all of us the benefit of your experience. As was pointed out, several of you bring substantial knowledge in certain areas to this course and so we are looking forward to your contribution to the discussions at the appropriate times.

As has already been said, this is not the first SSAC training course to be offered. So I would like to just very briefly comment on what training has been conducted, who has participated, and the relationship of the different training courses in the hopes that, should you consider attending another SSAC course, you will be informed on the similarities and differences of the various SSAC training courses.

State systems of accounting for and control of nuclear material (SSAC) are widely recognized as a vital element in achieving effectiveness and credibility for international safeguards. A nation's SSAC must also satisfy national objectives, such as accounting for all safeguarded material and detecting losses or unauthorized removal of material.

In recent years the Agency has placed an increasing emphasis on assisting member states in developing their state systems of accountancy and control (SSAC) of nuclear material. The programme actually began with a panel meeting in Tokyo in November 1973 at which the essential requirements of state systems in compliance with international safeguards agreements were outlined and discussed. In July 1975 the responsibilities and activities of state systems were discussed in greater detail at an advisory group meeting in Brno, Czechoslovakia, with emphasis on some optional state system activities.

In 1975, in the course of the Brno advisory group meeting, it was noted by several participants that there was a need for an Agency-sponsored training programme related to the organization and operation of state systems, and a separate advisory group meeting on that subject was recommended.

Preparation for the advisory group on training programmes began almost immediately, with the calling of an informal meeting of consultants in early December 1975. The advisory group meeting took place in Vienna in March 1976, and it was opened by the Director General of the IAEA, who noted that the establishment of nuclear materials accountancy and control systems in states embarking on nuclear power programmes was a very important element in international safeguards. He expressed the belief that the Agency could play an important role in the development of such systems, and that one way of providing this assistance was through the organization of a series of training courses.

The advisory group on SSAC training gave primary attention to the organization of a basic training course for member states whose nuclear power programmes are still in the early stages. The advisory group also made several suggestions relating to more advanced training courses or seminars. It was expected that the first advanced course would be held in early 1978, which in fact it was.

The first "Basic Training Course for Personnel of State Systems of Accounting for and Control of Nuclear Material" was held in Vienna in November 1976 and was followed by visits to selected nuclear facilities in the Federal Republic of Germany. The course was attended by 27 individuals from 18 member states. Primary emphasis was given to two topics, namely 1) Fundamentals of Nuclear Materials Accountancy, and 2) Organization, Activities and Responsibilities of the State System. Guest lecturers described material control systems in their countries, and discussed materials accountancy practices and problems in their state systems, and numerous discussion periods were scheduled in order to encourage audience participation. In addition to the primary topics, lectures were also presented on the history of international safeguards, on physical protection of nuclear materials and nuclear facilities, and on a description of the Agency's international safeguards system.

The second basic training course was held in IAEA Headquarters in May/June 1977. No facility visits were arranged, but the basic course content was the same as that given in the first training course. This second basic training course was attended by 35 persons from 25 countries.

Formal training courses are not the Agency's only activity related to state systems of nuclear materials accountancy and control. In the autumn of 1977 some 17 persons participated in a month-long study tour involving three days of lectures in Vienna, two weeks of lectures and facility visits in the Soviet Union, three days in the German Democratic Republic and three days in Czechoslovakia. This four country study tour included extensive visits to operating nuclear facilities, and provided participants with the opportunity to see actual operating nuclear material control systems in action.

Several states have requested, formally or informally, that they be granted fellowships or similar awards permitting extended visits to some country (or several countries) with an operating state system, or that the Agency arrange for suitable experts in the field to visit their country for an extended consulting period, possibly as long as one year. There are, however, difficulties in these approaches. Nearly all of the major industrial countries consider that any extended fellowship-type visit would permit the recipient to have an unacceptable degree of access to commercially sensitive infor-

mation. Thus while such visits are by no means precluded, the duration is more likely to be three weeks rather than three months to one year. At the same time, moreover, most experts in the field who might consult on the development of a state system are urgently needed where they are, and are more likely to be available for consulting visits of three weeks rather than three months. Nevertheless, in spite of these difficulties, technical cooperation projects have been arranged; others are currently under development. This is a very important complementary measure to the SSAC training courses.

Actually, the first training course organized and conducted in a member state was in 1978 in Yalta, USSR, under Soviet Union support. The training course evolved naturally out of the two previous courses which were held in Vienna in 1976 and 1977. Subsequently, three additional basic training courses have been held in Yalta as indicated in Table I.

In 1979, a more advanced SSAC course emphasizing safeguards at bulk-processing facilities was presented in Richland, Washington. Sponsored by the US Department of Energy (DOE) in cooperation with the IAEA, the 1979 course featured tours of the Exxon Nuclear Company fuel fabrication plant in Richland, as well as lecturers from the IAEA, Exxon, and Battelle Pacific Northwest Laboratories.

Starting in 1980, SSAC training courses in the US came under the auspices of the Nuclear Nonproliferation Act of 1978. Los Alamos National Laboratory was asked by the US Government to take the lead in planning and presenting these courses, which have been sponsored by DOE in cooperation with the IAEA. The SSAC course working committees include representatives of DOE, IAEA, Los Alamos, US/NRC, US/DOS, ACDA, and Exxon Nuclear. Course lecturers have been provided by the IAEA, Los Alamos, US Government, foreign governments and industrial organizations, as well as for other US national laboratories and US industry. The emphasis of SSAC courses offered in 1980 and 1982 was on safeguarding item-dominant facilities, such as power and research reactors and spent-fuel storage facilities. The 1981 and 1983 course material emphasizes safeguards techniques at bulk facilities, and in both 1981 and 1983 (as in 1979), Exxon Nuclear provided many of the technical lecturers and allowed access to their plant for tours and demonstrations.

By the end of this year, a total of 13 SSACs will have been conducted. Looking at Table II you can see the degree of participation of member states in these courses. Including this course, 242 participants from 52 different member states have attended these courses. This says to me, that a substantial, maybe even surprising, number of member states are interested in and support SSAC training. It goes without saying then that the subject of state systems of nuclear materials accountancy and control continues to be of vital concern to all, and planning for future training courses will be continued.

In conclusion, I would like to thank you very much for attending this course and I am looking forward to working with you over the next three weeks to make this course worthwhile for all of us.

TABLE I

List of SSAC Training Courses

	1976	1977	1978	1979	1980	1981	1982	1983	1984	1985
1. <u>Vienna</u>										
November	X									
May		X								
2. <u>USA</u>										
July/Richland				X						
May/Santa Fe					X					
April/Santa Fe, Richland						X				
March/Santa Fe							X			
October/Santa Fe, Richland								X		
June/Santa Fe, San Clemente										X
3. <u>Yalta, USSR</u>										
November			X							
October						X				
October							X			
October									X	
4. <u>Japan</u>										
March/Tokai, Tokyo										X

TABLE II

PARTICIPATION BY COUNTRIES IN SSAC COURSES 1978 - 1985

COUNTRIES	1978	1979	1980	1981		1982		1983	1984	1985	
	USSR	USA	USA	USA	USSR	USA	USSR	USA	USSR	Japan	USA
Australia								1		1	
Bangladesh								1			1
Belgium				1							
Bolivia					1						1
Brazil		2	2	1		2		2			1
Bulgaria					2				1		
Canada		1	1	1		2		1			1
Chile		1	1								
China										2	2
Czechoslovakia	3	2			1	2	1	1	2		1
EURATOM			2								
Egypt	1		2	2				1	3		3
Federal Rep. of Germany			1								
Finland						1	1				
France		1				1		1			
German Dem. Rep.	1				2	1			2		
Greece	1										
Hungary			1		2		1		1		1
India	2	1	2	1	1		2	2	3		1
Indonesia	2	1	1							3	2
Iran	1										
Iraq	1			2	1	2	1	1			1
Israel		1	1	1							
Italy			1	1	1	1					

(continued)

PARTICIPATION BY COUNTRIES IN SSAC COURSES 1978 - 1985

COUNTRIES	1978	1979	1980	1981		1982		1983	1984	1985	
	USSR	USA	USA	USA	USSR	USA	USSR	USA	USSR	Japan	USA
Japan			1	1	1	4				6	1
Kenya			1								
Korea											1
Libya					2		1	1			
Luxembourg			1	2							
Malaysia				1		1		1		1	
Mexico	1	1		1							
Morocco											1
Netherlands		1									
Pakistan		1	2	2		2		2			1
Peru	1										
Philippines			1		1	1	1	3		3	2
Poland	2	2		2			1	2	2		2
Portugal		1	1								
Rep. of Korea		1	2	1		1		1		4	
Romania					1		1		2		
South Africa						1		1			
Spain											1
Sweden		1		1	1	1		2			1
Switzerland			1	1							
Taiwan			2	2		2		3			
Thailand											
Turkey	1		1	1	1	1			1	2	
U.S.S.R									2		
United Kingdom		1									
Venezuela											1
Yugoslavia	1			1	1	2			1		
Zaire											1
TOTAL	18	19	28	26	19	28	10	27	18	22	27

= 242



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 2: HISTORICAL DEVELOPMENT AND CURRENT TRENDS IN NUCLEAR SAFEGUARDS

The historical development of national and international nuclear materials safeguards systems will be presented. Emphasis will be placed on the interaction of political, economic, and technical factors and the interrelation of national (state) and international systems. The impact of current political, institutional, and technical trends will be discussed.

After the session, participants will be able to:

- Outline the need and rationale for both national and international nuclear safeguards systems, and the need for coordination, consensus, and cooperation.
- Trace the development of international safeguards and the role of the IAEA in implementing international safeguards inspection and verification.
- Trace the development of US policy regarding nonproliferation and nuclear safeguards.
- Describe the interdependence of technology, institutional arrangements (including treaties), and national policy in the achievement of safeguards and nonproliferation goals.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 2: THE DEVELOPMENT AND IMPLEMENTATION OF
INTERNATIONAL SAFEGUARDS

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The unique institution of International Atomic Energy Agency (IAEA) safeguards has been commented on at length by both technicians and politicians. IAEA safeguards have variously been extolled as a great international breakthrough and model for other arms control verification regimes and alternatively as a not very credible arrangement with many weaknesses and contradictions and which cannot, essentially by definition, do the job it's supposed to do. The 1981 bombing by Israel of the safeguarded Iraqi research reactor at Tuwaitha is used by both extremes to support their positions.

As a physicist who has been working in this area, among others, for several years it seems to me that neither side is giving the reality of IAEA safeguards a fair deal. The bottom line on safeguards is to ensure that nuclear material, especially plutonium or highly enriched uranium (>20% U-235 but usually 93% U-235) under safeguards, is not used to make nuclear explosives or to further other military purposes. In the first instance they are pretty good and do represent a major concession by sovereign states; but the structure of the safeguards regime was designed in another era. The system has its flaws technically, but it is also a political system which, because of its dual nature, can and does do a very useful and important job for the international community.

• Arms Control Agreements

In order to put the IAEA safeguards regime into context we should look back to the events that led to the founding of the Agency and those which have had significant effects on its evolution. I will emphasize a few key events.

A Nuclear Chronology

1938	Hans Bethe Discovery of the Thermonuclear process of the sun.
1939	Hahn, Meitner and Frisch discovery of fission.
1940	Einstein/Szilard letter to Roosevelt
1941	Establishment of the Manhattan District
1942	First Chain Reaction
Jul 1945	Trinity Test

Aug	1945	Hiroshima
Aug	1945	Nagasaki
Aug	1949	The first Soviet test
Mar	1952	The first British test
Nov	1952	The first US H-bomb test
Aug	1953	The first Soviet H-bomb test
Dec	1953	Eisenhower Atoms for Peace program
Jul	1957	Establishment of the IAEA
Feb	1960	The first French test
	1961	Irish proposal at the UN for control of nuclear weapons -- precursor of NPT
Jun	1961	Antarctic treaty
Oct	1963	Limited test ban treaty
Oct	1964	The first Chinese test
May	1966	The first Chinese H-bomb test
Feb	1967	Treaty of Tlatelolco (signed)
Oct	1967	Outer Space Treaty
Mar	1970	Non-Proliferation Treaty
	1970	First French H-bomb test
May	1972	Seabed Treaty
Oct	1972	SALT I
Oct	1972	ABM Treaty
May	1974	Indian Test
Jul	1974	Threshold Test Ban Treaty (not ratified)
May	1975	First NPT Review Conference
May	1976	Protocol to ABM Treaty
May	1976	Underground Threshold Treaty (not ratified)
Apr	1977	Carter nonproliferation statement, INFCE initiated

Aug	1977	Soviet Union satellite photographs of possible test site in the South African Kalahari Desert
	1978	US Nonproliferation Act
Jun	1979	SALT II (not ratified)
Sep	1979	South Atlantic flash
Nov	1979	Publication of the Progressive Article on H-bomb design
Mar	1980	Conclusion of INFCE
Aug	1980	Second NPT Review Conference
Jun	1981	Israeli bombing of Iraq reactor
Jul	1981	Reagan statement on nonproliferation
Sep	1982	US reassessment of participation in IAEA Feb 1983
Aug/Sep	1985	3rd NPT Review Conference

Atoms for Peace: President Eisenhower proposed the establishment of an international agency to which nations would contribute from their stock-piles of fissile material. The agency would be responsible for allocating these nuclear materials "to serve the peaceful pursuits of mankind," so that they would be unavailable for military uses. The USSR initially refused to consider this proposal apart from the general issues of nuclear energy control and the prohibition of nuclear weapons. It subsequently reversed itself, but only after the U.S. made clear that it would proceed with or without Soviet cooperation.

The IAEA, founded in 1957, does not distribute significant quantities of nuclear materials. In fact it does not physically possess nuclear material. However, arrangements are made through it for the supply of materials from one country to another under certain controls. The agency also applies controls to any nuclear facility in a country when that country requests that it do so.

Tlatelolco: Momentum for this treaty originated in the Cuban missile crisis. In November 1962, Bolivia, Brazil, Chile, and Ecuador submitted a draft resolution to UNGA First Committee. The final treaty was largely a product of efforts by Mexican Ambassador Garcia Robles. The treaty provides for IAEA safeguards on all peaceful nuclear activities in the zone. However, the treaty allows for peaceful nuclear explosions (PNEs). It has been waived into force by all signatories except Argentina, Brazil and Chile. Cuba has not even signed the treaty.

The Treaty on the Nonproliferation of Nuclear Weapons (NPT): This treaty, which is the cornerstone of the nonproliferation regime, requires all peaceful facilities in NNWS to be put under IAEA safeguards. It prohibits NNWS from acquiring nuclear weapons or other nuclear explosive devices. The

implementation of the NPT through the IAEA is codified in a document called INFCIRC/153, the so called Blue Book. Since over 130 countries are parties to the NPT, the safeguards provisions in INFCIRC/153 have established the norm for the application of safeguards, even in those countries that are not parties to the treaty where safeguards are implemented pursuant to the older INFCIRC/66. These countries, which do not accept safeguards on all their peaceful facilities but only on some, complicate the safeguard regime.

The major component in the present nonproliferation regime is the restraint shown by nations in not developing nuclear weapons and the confidence other nations have that their neighbors will continue to act with restraint in this area. Making legally binding international commitments such as those embodied in the NPT and Tlatelolco treaties lend significant weight to a nation's statements about not developing nuclear weapons. However, as is clear in all such arrangements, some form of independent verification is essential if the regime is to be stable.

In its clearly defined and fairly narrow context IAEA safeguards provide important independent verification for essential aspects of both the NPT and Tlatelolco Treaty.

- Role of the IAEA

In order to simplify the discussion I will focus on type safeguards. The role of the IAEA is limited to the verification of commitments with respect to the agreements voluntarily entered into by sovereign nations. It is to assure, as far as it is able, that materials under its control are used for the stated peaceful purposes.

The commitment to allow inspection is as important as the technical efficacy of the verification techniques. If the techniques are not credible, however, the value of the commitment is debased.

Figure 1 gives a schematic of the IAEA's safeguards system and how it interacts with national state systems of accounting for and control of nuclear material (SSACs).

- A Simple Fuel Cycle

Figure 2 shows schematically a typical light water reactor once through fuel cycle. Imagine if, in addition to the enrichment facility shown in the figure, another smaller enrichment plant existed. If that smaller plant were under safeguards as well as the facilities shown, the problem raised for safeguards would be fairly easily manageable. If however all the facilities except this smaller plant were under IAEA safeguards, a fundamentally different situation exists.

- Enrichment Plant Safeguards

A short digression will serve to illustrate the problem. In NPT countries the method chosen for safeguarding enrichment plants (only gas centrifuge enrichment plants are currently safeguarded by the IAEA) is to get assurance that highly enriched uranium (HEU) is not present. This can be done by detecting either neutrons or gammas. A very efficient way is detecting the gammas given off by U-235 and by U-238 daughters (See figure 3).

The U-238 part of the relationship is determined by gammas from the TH-234 or PA-234M daughters. If the daughter gammas are in equilibrium with the U-238 parent activity (takes about 160 days - this is never a problem since times from conversion to enrichment tend to be long), then the daughter gammas give a measure of U-238 content. As can be seen from figure 3, there are no gamma peaks from U-235 close to those of PA-234M or TH-234. After normalization, the enrichment is given by I in the formula in figure 3, where the half-lives are $T(U-238) = 4.468$ billion years and $T(U-235) = 0.704$ billion years.

An IAEA inspector can make a measurement of enrichment using this technique in a matter of minutes with instruments that have been developed for the IAEA through the US safeguards support program.

In the full scope safeguards scenario one does not worry very much about excess LEU production. However, the addition of an unsafeguarded enrichment plant adds profound complications. If this unsafeguarded plant were fed with LEU (3%) instead of natural uranium, its capacity to produce HEU (93%) would be increased by over a factor of 4. A small pilot plant would thus become a fair sized production plant for unsafeguarded HEU, which can be fairly easily converted to nuclear weapons.

Ensuring that significant amounts of LEU are not being produced in the safeguarded plant and then sent to the unsafeguarded one poses an entirely new class of safeguards problems which the IAEA and some member states are just beginning to grapple with. The situation actually faces the IAEA at the present time.

In another example, suppose a country has only a small, unsafeguarded enrichment plant. A serious concern then arises with respect to the generally benign low-enriched uranium. If this country "acquires stocks of LEU (3%)", this small plant raises fundamental proliferation concerns. Questions about whether the LEU is being shipped to another unsafeguarded plant for conversion to HEU or for the production of unsafeguarded plutonium must be addressed by the international community.

How, then, does the system control commerce in LEU, especially to countries with small unsafeguarded enrichment facilities? There are various methods, including generally accepted export control regimes, but nothing is perfect. Such unsafeguarded facilities pose fundamental challenges for the nonproliferation regime.

In spite of difficulties such as this one posed by countries without full scope safeguards, IAEA safeguards provide significant and important assurance that NNWS are living up to their obligations. It should be emphasized that the IAEA safeguards system is but one part of the independent verification regime. National technical means of monitoring events play a vital role in keeping the international community aware of activities that might cause concern.

However, notwithstanding these assurances, circumstances may exist where one nation will not feel comfortable about its neighbor's intentions. This may result from a long history of hostility in certain regions or from concerns about political stability. A nation may then take actions outside of the regime, or in spite of it, to get the assurance it seeks. Generally, this

should not be seen as a condemnation of the nonproliferation regime, but rather more likely as a statement about relations between nations. It is not possible to construct a system that is all things to all people. We should endeavor to ensure that what we do in the safeguards area, and more generally in the nonproliferation regime, maximizes the assurance that can be given and minimizes the chance of misinterpretation of a nation's actions by its neighbors.

It is important to note that the independent verification efforts of the nonproliferation regime do not stop with, nor are limited to, the detection of diversion of nuclear materials from civil programs. These efforts also do not stop with respect to a proliferating state should that state actually test a nuclear explosive device. However, the nature of the efforts may change.

Efforts to Improve Safeguards: The Case for Nuclear Weapons States (NWS) Voluntary Offers

The US was the first NWS to accept IAEA safeguards on its civil nuclear facilities. The USSR is the most recent. In the US, safeguards are applied as they are in NNWS at those facilities which the Agency inspects. The purpose in these cases is not nonproliferation as such. It is to demonstrate that NWS don't view safeguards as punitive. They are a country's public affirmation of the peaceful intent of its nuclear programs and facilities.

NWS voluntary offers also provide a very useful opportunity for the IAEA to develop advanced techniques and to apply safeguards at advanced nuclear fuel cycle facilities.

• Conclusion of Safeguards

Each year the IAEA publishes its safeguards implementation report (SIR). It is a long, technical document which opens with its conclusion. The conclusion for 1983:

In 1983, as in previous years, the Secretariat, in carrying out the safeguards obligations of the Agency, did not detect an anomaly which would indicate the diversion of a significant amount of safeguarded nuclear material - or the misuse of facilities or equipment subject to safeguards under certain agreements - for the manufacture of any nuclear weapons, or for any other military purpose, or for the manufacture of any other nuclear explosive device, or for purposes unknown. With the exception of two cases where the Agency was unable to draw conclusion for part of the year, it is considered reasonable to conclude that nuclear material under Agency safeguards in 1983 remained in peaceful nuclear activities or was otherwise adequately accounted for.

* Is the first sentence of the conclusion technically credible?
Yes.

* Can it be generalized to mean that there has been no diversion (second sentence)? If the IAEA inspectorate as a whole follows well defined procedures, and if at least a significant number of facilities were inspected, and if no

obvious flaws in the system exist that would allow straight-forward concealment of clandestine activities - then the answer is yes. I personally believe that this is presently the case.

* Is the conclusion scientifically verifiable in the sense that an independent observer can take the results of a year's work, do checks, including some measurements, and come to the same conclusion? Probably not.

* Most importantly, is the conclusion politically credible? My answer is that it is adequate for the role safeguards play in the international nonproliferation regime and in the peaceful uses of nuclear energy. However, if a nation wanted an essential certainty, based on independent verification, that things were a certain way with respect to activities in another nation, this may not be the way to do it.

An important statement about the IAEA safeguards regime that one can make is that it is the only regime in which an international organization performs on site inspection within the boundaries of sovereign nations for the purpose of verifying compliance with treaty or other international commitments. This demonstrates that at least it can be done.

A Schematic of the IAEA Safeguards System

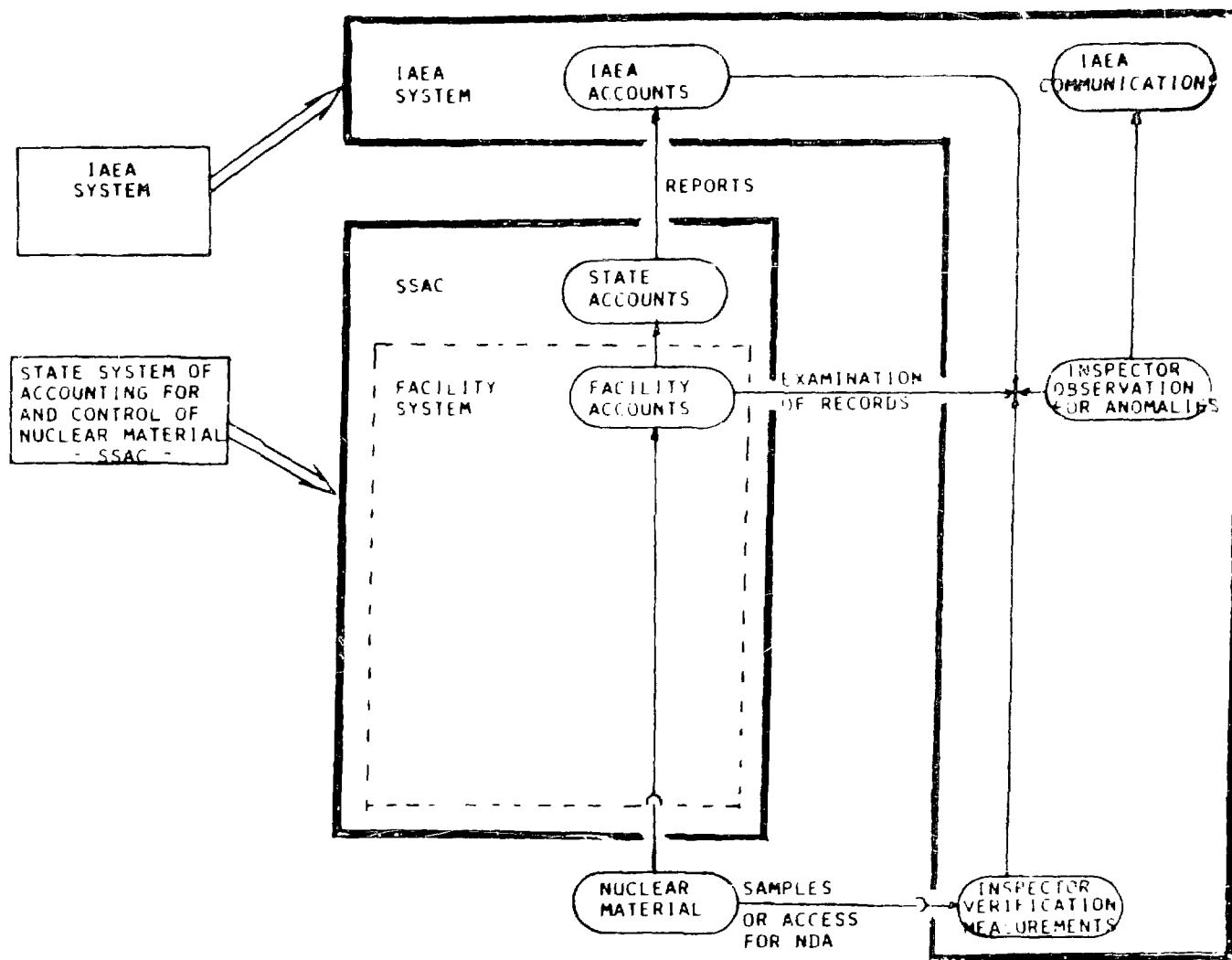
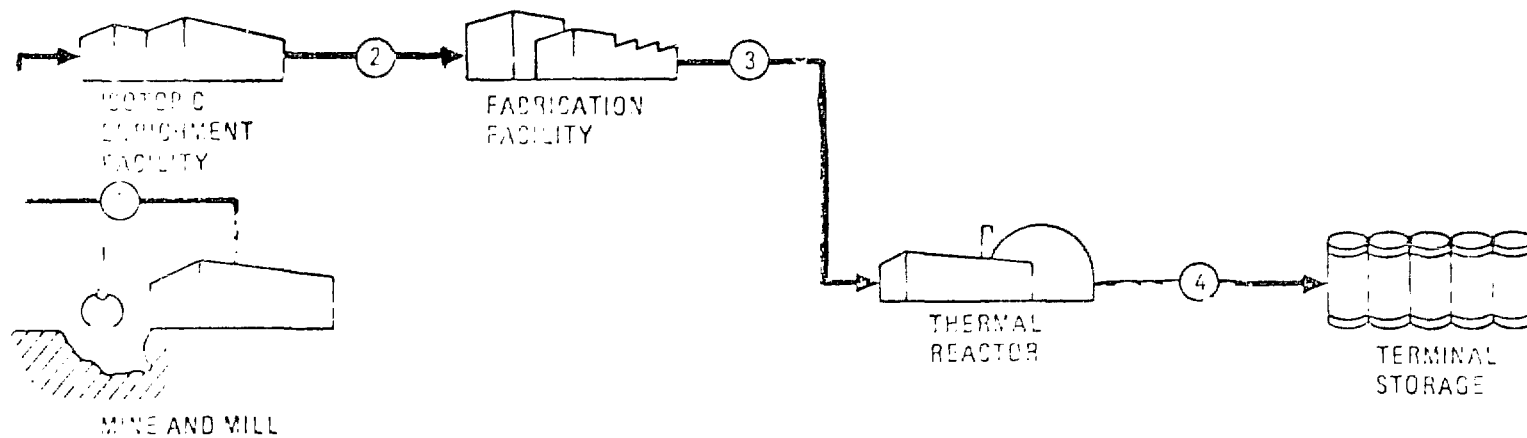


FIGURE 1.



	STREAM				
	1	2	3	4	5
FISSILE MATERIAL					
TYPE	U-235	U-235	U-235	Pu	FISSION PRODUCTS
ENRICHMENT (%)	NATURAL	3	3	0.8, 100	—
MASS (KG)*	10,740	8,220	8,100	2,150 2,470	9,550
COMPOSITION	UF ₆	UNH	UO ₂	(U,Pu)O ₂	—

* PER 10 GW(e) — YR

System description and material balance for LWR once-through.

FIGURE 2.

FIGURE 3.

Table 5.34 Values of the gamma-ray abundances of ^{235}U , $^{234\text{m}}\text{Pa}$, and ^{234}Th ^a

Isotope	Energy (keV)	Br (γ/dis) (% error in Br) ^b
^{235}U	143.78	0.1067 (0.4)
	163.36	0.0506 (0.3)
	185.72	0.576 (0.4)
	194.94	0.00624 (1.0)
	202.13	0.0103 (2.1)
	205.31	0.0494 (0.5)
$^{234\text{m}}\text{Pa}$	258.30	0.000770 (0.8)
	742.30	0.000870 (2.0)
	766.50	0.00343 (1.2)
	1001.40	0.00889 (0.7)
	1738.5	0.000242 (3.0)
	1831.9	0.000179 (4.8)
^{234}Th	63.33	0.0425 (4.9)
	92.367 ^c	—
	92.792 ^c	—

^aTaken from Ref. 116.^bThe error is the external relative standard deviation (in %).^cDragnev and Damjanov (Ref. 112).

Source: Ref. 116.

$$I = \left[1 + k \frac{I_{\gamma}(238)}{I_{\gamma}(235)} \right]^{-1}$$

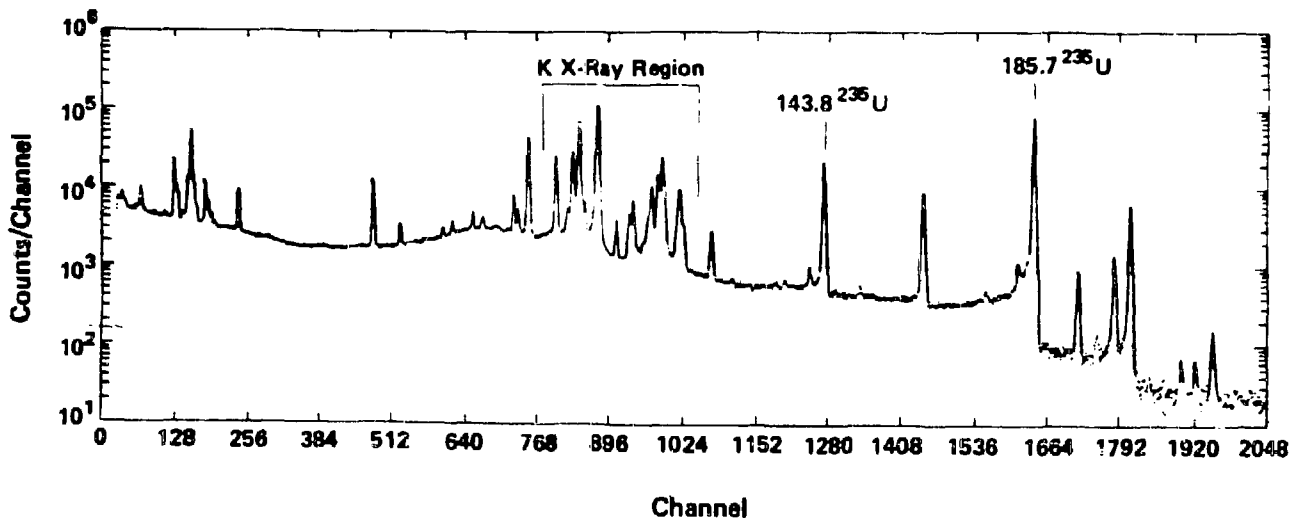


Fig. 2. Pulse-height distribution of gamma rays from high-enriched (93% ^{235}U) uranium, measured with a high-resolution germanium detector. Gamma-ray energies (in keV) are noted on some of the peaks. The prominent peak at 186 keV is used for passive assay of ^{235}U .



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 3: INTRODUCTION TO IAEA SAFEGUARDS

The objectives, legal documents, and methods of the IAEA safeguards system will be discussed. This session will emphasize guidelines for SSAC organization to meet the obligations arising from safeguards agreements concluded by a state with the IAEA.

After the session, participants will be able to:

- Understand the basic objectives of IAEA safeguards and the legal basis for their application.
- Understand the IAEA safeguards system.
- Describe the major requirements of nuclear material accounting and control.
- Describe the requirements placed on the SSAC by the IAEA.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 3: RELATIONS BETWEEN SSAC AND THE IAEA

Carlos Büchler

International Atomic Energy Agency

I. OBJECTIVES OF IAEA SAFEGUARDING

A. Introduction

Nuclear and non-nuclear material, services, facilities, equipment, and information which are to be used for legally defined purposes may be deliberately diverted from these purposes. The actions aimed at the detection and deterrence of this diversion are known as safeguards.

Potential diverters are facility operators, individuals or groups of individuals, and states. IAEA safeguards are aimed at the timely detection of diversion in or by states having undertaken to accept safeguards in accordance with an agreement between the IAEA and the state and at the deterrence of such diversion by the risk of early detection by the IAEA.

B. Safeguards in the Statute of the IAEA

The statute authorizes the IAEA "to establish and administer safeguards designed to ensure that special fissionable and other materials, services, equipment, facilities, and information made available by the Agency or at its request or under its supervision or control are not used in such a way as to further any military purpose; and to apply safeguards, at the request of the parties, to any bilateral or multilateral arrangement, or at the request of a state, to and of that state's activities in the field of atomic energy."¹ The statute, therefore, limits the application of safeguards to IAEA sponsored projects and to activities for which a specific request is made by a state.

The IAEA shall, according to the statute,² enter into an agreement with the state or group of states submitting a project, which agreement shall include undertakings that "the assistance provided shall not be used in such a way as to further any military purpose" and "the project shall be subject to the safeguards provided for in Article XII, the relevant safeguards being specified in the Agreement."³

Further, the state specifies the IAEA safeguards rights and responsibilities concerning projects and arrangements.³ These rights and responsibilities include, inter alia, the use of inspectors "who shall have access at all times to all places and data ... as necessary to account for source and special fissionable materials supplied and fissionable products and to determine whether there is compliance with the undertaking against use in furtherance of any military purpose."⁴

C. Project Agreements, Safeguards Transfer Agreements and Unilateral Submissions to IAEA Safeguards

Since 1961 the IAEA has entered into "projects agreements" for the supply of materials, equipment and facilities made available by or through the IAEA, "safeguards transfer agreements" in which two states transfer to the IAEA their safeguards responsibilities set forth in their cooperation agreements; and agreements for "unilateral submission" by a state to IAEA safeguards of certain facilities, nuclear material or of all the state's nuclear activities.

All such agreements are based on the safeguards system which the IAEA set up in 1961,⁵ extended in 1964,⁶ revised in 1965⁷ and extended in 1966⁸ and in 1968.⁹ This system⁵⁻⁹ does not specify further than the Statute does⁴ either the objective of safeguards or the conclusion of the IAEA verification activities in stipulating that nuclear material, facilities, and equipment shall not be used to further any military purpose and whether there is compliance with the terms of the agreements (the undertaking by a state has been explicitly stated in "safeguards transfer agreements" concluded since 1975^{10,11} as not to use nuclear material, facilities, and equipment for the manufacture of nuclear weapons, or to further any other military purpose, or for the manufacture of any other nuclear explosive device).

D. Safeguards Agreements Pursuant to the Treaty on the Non-Proliferation of Nuclear Weapons

The Treaty on the Non-Proliferation of Nuclear Weapons (NPT) entered into force in March 1970.¹² Each non-nuclear weapon state party to the Treaty undertakes to accept safeguards, as set forth in an agreement to be negotiated and concluded with the IAEA for the exclusive purpose of verification of the fulfillment of its obligations assumed under the Treaty with a view to preventing diversion of nuclear energy from peaceful uses to nuclear weapons or other nuclear explosive devices.¹³ Procedures for the safeguards required shall be followed with respect to all source or special fissionable material whether it is being produced, processed or used in any nuclear facility or is outside any such facility. The safeguards required shall be applied on all source or special fissionable material in all peaceful nuclear activities within the territory of such a state, under its jurisdiction, or carried out under its control anywhere.

Each state party to the Treaty also undertakes not to provide source or special fissionable material, or equipment or material especially designed or prepared for the processing, use, or production of special fissionable material, to any non-nuclear weapon state unless the source or special fissionable material is subject to the required safeguards.¹⁴

At the time of the entry into force of the NPT, most of the governments concerned expressed the view that the IAEA safeguards system was insufficiently defined. All members of the IAEA were therefore invited to take part in a specially convened Safeguards Committee. The Committee agreed on "the structure and content of the agreements between the Agency and states required in connection with the Treaty on the Non-Proliferation of Nuclear Weapons,"¹⁵ which has served as a basis for every agreement concluded in connection with the NPT.

The basic undertaking by the State in NPT safeguards agreements is to "accept safeguards, in accordance with the terms of the agreement, on all source or special fissionable material in all peaceful nuclear activities within the territory of the state, under its jurisdiction, or carried out under its control anywhere, for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other nuclear explosive devices."16

The objectives of safeguards are further defined in these agreements to be the "timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection."17 The inclusion of the expression "for purposes unknown" is very important for the practical application of safeguards for it means that the IAEA does not have to attempt to determine the use to which diverted material is put and, in particular, does not have to determine whether diverted nuclear material is for "the manufacture of nuclear weapons or of other nuclear explosive devices." In addition, it is not an objective of IAEA safeguards to determine who is responsible for any diversion.

The agreements provide for "the use of material accountancy as a safeguards measure of fundamental importance, with containment and surveillance as important complementary measures"18 and also provide that "the technical conclusion of the Agency's verification activities shall be a statement, in respect of each material balance area, of the amount of material unaccounted for over a specific period, giving the limits of accuracy of the amounts stated."19

E. Implementation of Safeguards by the IAEA

The IAEA safeguards system is laid down in two IAEA documents, INFCIRC/66/Rev. 29 and INFCIRC/153.15 The first document forms the basis for bilateral agreements, transfer agreements, and unilateral submission agreements under which equipment, facilities, nuclear and other materials, are subject to safeguards. The second document forms the basis of all agreements required by Article III.I of the NPT, under which all nuclear material in all peaceful nuclear activities of a state is subject to safeguards. INFCIRC/153 obliges the IAEA to draw from its verification activities a technical conclusion in respect to nuclear material for each material balance area. INFCIRC/66/Rev. 2 does not include the required specifics of a conclusion, but the IAEA is obliged by the Statute to make a determination of compliance and, where noncompliance has been concluded, to report to the Board of Governors. INFCIRC/66/Rev. 2 provides the IAEA with means to draw in respect to nuclear material the same type of technical conclusion as required by INFCIRC/153. The IAEA has to judge in each particular situation whether the application of its nuclear material verification procedures permits it to fulfill the responsibility of safeguarding equipment, facilities, and non-nuclear material.

Implementation of nuclear material safeguards requires quantification of the objectives for each situation. To provide guidelines for the implementation requires identification of the possible strategies that a state may adopt for diverting nuclear material and specification of the measures that the IAEA must employ in its safeguards system in order to be able to counter successfully these diversion strategies. These subjects are treated in the following sections.

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II. DIVERSION OF NUCLEAR MATERIAL

A. Introduction

In the context of IAEA safeguards, the state with its corresponding capabilities and resources is considered as the primary potential diverter, and the probability of attempted diversion is considered small but finite. The purpose of diversion is assumed to be the acquisition of nuclear material for uses proscribed by the relevant safeguards agreement.

B. Diversion Strategies

The plans for diverting nuclear material and for either delaying the detection of the diversion or avoiding it are known as diversion strategies.

Diversion strategies could involve a single facility or a number of facilities cooperating in the diversion and its concealment. Diversion could involve material already in a form suitable for the intended use or in a form requiring further processing before such use. This further processing could be undertaken immediately, or the diverted material could be stockpiled for processing and used at a later time. The diverter may attempt to use safeguarded facilities to process material which has been diverted at another safeguarded facility, or material which either is at the starting point of safeguards or has already undergone some processing and which must be under safeguards but has not been declared by the state. Such an attempt would provide the IAEA with a chance to detect at a facility material which had not previously been in a safeguarded facility or material which had been previously diverted.

The material might be diverted in a single removal or repeated removals. Immediate detection by the IAEA can only be possible if it applies strict containment and surveillance measures. Verification of the physical inventory and of the material balance provides for a delayed opportunity for detection of diversion.

To conceal the removal of nuclear material the diverter may present evidence that the material;

- (a) was never received at the facility in question;
- (b) was shipped to some other facility or facilities;
- (c) was discarded or accidentally lost; or
- (d) is still present at the facility:
 - (i) with complete items missing;
 - (ii) with part of the items missing;
 - (iii) with portions of material from all items missing;
 - (iv) with a combination of (i), (ii) and (iii);
 - (v) by substituting, for the diverted material, non-nuclear material of lesser value to the diverter;
 - (vi) by borrowing the needed quantity of material from another facility and returning it after inventory verification has passed.

The strategy of concealment that gives the inspector only one opportunity to detect the concealment maybe called final concealment, as opposed to temporary concealment. The recording of fictitious discards is an example of final concealment. If the fictitious discard is not detected at the time of

the discard itself, it will never be detected, because no second opportunity for verification will exist. The falsification of inventory data, in contrast, is an example of temporary concealment and transfers the diversion into the next material balance period, where it has a second chance of being detected. In temporary concealment the facility operator must continue to attempt to conceal the removal until he can achieve final concealment.

1. Falsification of Records and Reports. The concealment of the removal of nuclear material which had previously been included in the records and reports available to the IAEA would presumably involve some falsification of these documents as part of the diverter's attempts to conceal the shortage from the IAEA and, in particular, to avoid detection by audit. Such falsification can be classified as understatements or overstatements of inventory or flows and introduction of "mistakes" in the transcription of data or in calculations.

In cases where the facility receives material from unsafeguarded facilities, the operator may understate receipts by not recording all receipts or by recording smaller than actual quantities for some receipts. Another possibility would be to arrange for a receipt to arrive just prior to a physical inventory to replace material already removed and to record the transfer as a receipt which occurred after the inventory.

In cases when the facility ships material to facilities which are not subject to IAEA safeguards, the operator may record non-existing transfers. Other possibilities would be to record measured discards in excess of those which occurred or to record shipments as having occurred just prior to the physical inventory taking, but hold the material and ship it after the inventory.

There are many possibilities for the falsification of records by the introduction of "mistakes:" recording a number and reporting a different one, recording correctly a series of numbers and recording an incorrect total, recording a correct net weight and analysis and recording an incorrect total, etc.

2. Deceiving IAEA Measurements Concealment strategies could also involve attempts to deceive IAEA measurements with respect to either the completeness or the correctness of the measurements. Examples are partial or periodic bypassing of flow key measurement points, alteration of containers, biasing of instruments and biasing of sampling devices.

3. Declaring Diverted Material as MUF A diverter could choose to divert material without alteration of the inventory and inventory change data and allow the removal to be shown as MUF. This strategy may, or may not, be supported by inflation of the measurement uncertainties and might be supported by explanations designed to portray the MUF as being due to legitimate causes.

C. Importance of Diversion

The importance of the diversion depends on the type and amount of the diverted material. Materials, e.g., plutonium and highly enriched uranium, which are of immediate use for nuclear explosive devices represent a greater hazard than does material which requires a lengthy and complex process to be used for these devices.

Rough estimates of the times required to convert different materials to material suitable for nuclear explosive devices are given in Table 1. The times listed in Table 1 are dependent, among other factors, upon the amount of materials involved and capabilities of the facilities carrying out the processing. If the necessary processing is carried out in a large unsafeguarded facility, the shorter times in each range would apply. If done in a large safeguarded facility by unreported introduction and removal of the material at less than full capacity rate, the intermediate times in each range might apply. If the processing is carried out in small unsafeguarded facilities or activities, the longer time would apply. These times provide the basis for the requirements for the timeliness of detection by the IAEA of diversion and, hence, for the frequencies of verification by the IAEA of its containment and surveillance measures and of physical inventories.

TABLE 1

IMPORTANCE OF DIVERSION*

Required Conversion of Nuclear Material to the Forms Suitable for the Manufacture of Nuclear Explosive Devices	Material Form	Approximate Range of Times Required to Convert Nuclear Material to the Form Suitable for Manufacture of Nuclear Explosive Devices
Physical change; or chemical and physical change, but no purification	Plutonium and enriched metal oxide or solution	Days to weeks
Chemical and physical change with purification	Irradiated fuel, radioactive solution, cold scrap	Weeks to months
Isotopic, chemical, and physical change	Natural and low- enriched	Less than one year

* Based on the approximate time required to convert the material suitable to manufacture of nuclear explosive devices.

III. IMPLEMENTATION OF SAFEGUARDS

A. Introduction

The IAEA safeguards system must enable the IAEA to verify that a state has complied with its undertaking as specified in the relevant safeguards agreement. The safeguards responsibilities and rights of the IAEA cannot, therefore, be delegated to the state or to any organization to which the state has delegated the state's responsibilities. The IAEA system has been conceived to ensure the timely detection of diversion that might be attempted by the wide range of strategies described in Section II. For these reasons the IAEA must verify the completeness, formal correctness and validity of the information (including all records and reports) made available by the state, regardless of the nature or level of the verification activities carried out by the state.

By means of its safeguards system, the IAEA shall be able to verify, in particular, that:

- (a) the quantities of nuclear materials imported into a state, produced within state or otherwise becoming subject to safeguards in any peaceful nuclear activity are not understated by the state;
- (b) quantities of nuclear materials on which safeguards are to be terminated, e.g., exports or consumption, are not overstated by the state; and
- (c) physical inventories are not overstated by the state, at intervals appropriate for satisfying the requirements for the timely detection of diversion.

Essential elements of the IAEA safeguards system are:

- (a) a Safeguards Agreement between the IAEA and the state, including Subsidiary Arrangements and Facility Attachments;
- (b) provision by the state to the IAEA of all information relevant to the operator's accountancy, containment, and surveillance of the material according to state's regulations, which must be in compliance with the terms of the Agreement; and
- (c) verification by the IAEA that the state is complying with the basic understanding as laid down in the Agreement.

The different types of safeguards agreements have been described in Section I. Section III. B describes the operator's measures of accountancy, containment, and surveillance. Section III. C and D describe, respectively, the information to be provided by the state and the verification to be carried out by the IAEA.

B. Accountancy, Containment, and Surveillance of Nuclear Material

Accounting for nuclear material is defined as the knowledge of the material's identity, composition, quantity and location. Agreements of the

INFCIRC/153 type require that "the state shall establish and maintain a system of accounting for and control of all nuclear material subject to safeguards...."¹ They prescribe that the system shall be based on a structure of material balance areas, a measurement system, a records and reports system, and a system of control by the state that the accounting procedures are being operated correctly. INFCIRC/66/Rev. 2² does not refer explicitly to a state's system of accounting for and control of nuclear material or to some of the above elements of such a system, but it does prescribe the accounting and operating records to be kept by the state and the accounting and operating reports to be submitted by the state to the IAEA.

The undertaking by the state in an INFCIRC/153 type agreement requires the state "to accept safeguards..., on all source or special fissionable material...."³ Such agreements also specify the starting point of safeguards⁴ and the conditions for the termination of safeguards⁵ and for exemptions from safeguards.⁶ Similar provisions exist in the agreements of the INFCIRC/66/Rev. 2 type.

The basic principle of the accountancy system required by INFCIRC/153 is the operator's recording at the facility and the state's reporting to the IAEA, for each material balance area, initial inventories of nuclear material and subsequent inventory changes. Additions to and subtractions from the initial inventory yield the "book inventory,"⁷ the amount of nuclear material which, according to the operator, is expected to be in a given facility or a given material balance area. Periodically, the facility operator takes a physical inventory⁸ in the material balance area by measuring the nuclear material which is present. For facilities having nuclear material in unsealed bulk form, because of the measurement uncertainties, there is usually some difference between the book inventory and the physical inventory. There may also be discrepancies for other reasons, e.g., failure to measure parts of the inventory or an unmeasured loss of material. The difference between book inventory and physical inventory is the "material unaccounted for,"⁹ abbreviated to "MUF". As a variable derived from measurements themselves, subject to uncertainties.

INFCIRC/153 provides definitions for the fundamental concepts of material accountancy, namely: book inventory, physical inventory,⁸ material unaccounted for,⁹ adjustment,¹⁰ batch,¹¹ batch data,¹² corrections¹³ enrichment,¹⁴ inventory change,¹⁵ key measurement point,¹⁶ material balance area,¹⁷ nuclear material,¹⁸ shipper/receiver difference,¹⁹ and source data.²⁰

Containment, as employed by the state or the operator, is understood as the restriction of the movement of or access to nuclear material. Containment measures are used by facility operators for physical protection of the material, safety of personnel, and convenience of operational procedures. In general, containment measures are not provided specifically for safeguards purposes, but their existence in a facility often simplifies surveillance for safeguards.

Surveillance means instrumental or human observation to indicate the movement of nuclear material. Surveillance may indicate the effectiveness of containment and, therefore, has for the operator the same use as containment.

Both containment and surveillance are, for the IAEA, important measures complementary to material accountancy.²¹ They should not impose any physical restriction on the movement of or access to material, but they have to provide to the IAEA information as to whether such movement or access occurred while inspectors were not present, in order to preserve the integrity of prior measurements of nuclear material by the IAEA and to provide the IAEA with knowledge of material flows at important points in a fuel cycle.

C. Information

Both documents, INFCIRC/66/Rev. 2 and INFCIRC/153, require that the state:

1. Provide the IAEA with information in respect to facility design features and other information relevant to safeguards;
2. Arrange that records are kept in respect of each material balance area; and
3. Provide the IAEA with reports in respect of nuclear material based on the records kept.

INFCIRC/153 prescribes the required design information²² and the required systems of records²³ and of reports.²⁴ Member states have further advised the IAEA on the detailed design information to be provided by the States.²⁵ The IAEA Secretariat has prepared design information questionnaires for different types of facilities.²⁶ The IAEA Secretariat has established model Subsidiary Arrangement and Facility Attachments,²⁶ which contain, inter alia, reporting forms and explanations for their use.²⁷

D. Verification

Although INFCIRC/153 does not contain a formal definition of verification, it does specify the activities, including independent measurements, to be used by the IAEA for achieving verification, and it does specify that verification applied to the location, identity, quantity, and composition of all nuclear material subject to safeguards.^{23/29/30}

Accordingly, the IAEA's verification process consists of:

1. Examination of the information provided by the state in:
 - i) Design Information;³¹
 - ii) Accounting Reports;³²
 - iii) special reports;³³
 - iv) amplification and clarification of reports;³⁴ and
 - v) advance notification of international transfers.^{35,36}
2. Collection of information by the IAEA in:
 - i) inspections for verification of design information;³⁷
 - ii) ad hoc and routine inspections;^{38,39}
 - iii) special inspections.⁴⁰
3. Evaluation of the information provided by the state and collected in inspections for the purpose of determining the completeness, correctness, accuracy, and validity of the information provided by the state.

The purpose of inspections of facilities "to verify design information"³⁷ is to enable the IAEA to evaluate the validity of the design information made available to the IAEA. This verification is carried out with respect to design information submitted for existing and new facilities and for subsequent modifications of these facilities. The purpose of the examination of design informations is:

1. to identify the features of facilities and nuclear material relevant to the application of safeguards to nuclear material in sufficient detail to facilitate verification;
2. to determine material balance areas to be used for IAEA accounting purposes and to select those strategic points which are key measurement points and which will be used to determine the nuclear material flows and inventories;
3. to establish the nominal timing and procedures for taking of physical inventory for IAEA accounting purposes;
4. to establish the records and reports requirements and records evaluation procedures;
5. to establish requirements and procedures for verification of the quantity and location of nuclear material; and
6. to select appropriate combinations of containment and surveillance measures and the strategic points at which they are to be applied.

Accounting reports provide information on the initial inventory,⁴¹ inventory changes,⁴² and material balances.⁴³

Ad hoc inspections by the IAEA are carried out in order to verify the information contained in the initial report and to identify and verify changes that have occurred since the date of the initial report. Ad hoc inspections are also carried out for the purpose of identifying and, if possible, verifying the quantity and composition of nuclear material involved in international transfers.³⁸ In the case of transfers out of a state, these inspections, including the affixing of seals by the IAEA, are to be carried out at the time the material is being prepared for shipping. In the case of transfers into a state these inspections are to be carried out at the time that material is unpacked.^{44,36}

The purpose of routine inspections by the IAEA is:

1. to verify that the information contained in the reports submitted by the state to the IAEA is consistent with the accounting and operating records maintained by the state;
2. to verify the location, identity, quantity, and composition of all nuclear material subject to safeguards; and
3. to verify information on the possible causes of material unaccounted for, shipper/receiver differences, and uncertainties in the book inventory.³⁹

Special inspections are to be carried out by the IAEA:

1. to verify information contained in special reports; and
2. to collect additional information when the IAEA considers that the information provided by the state and the information obtained through routine inspections are not adequate for the IAEA to fulfill its responsibilities.⁴⁰

The activities of the IAEA in the course of ad hoc, routine, and special inspections are in general for the purpose of collecting information whereby the IAEA can independently establish that the information provided by the state is:

1. complete in that it covers all nuclear material that has been present in the material balance area;
2. formally correct in terms of being free of mistakes;
3. valid with respect to the actual location, identify, quantity and composition of all nuclear material subject to safeguards; and
4. accurate in terms of the conformity of the measurement data of the state (random and systematic errors) with internationally accepted measurement accuracy.

These activities include: examining records, making independent measurements on all nuclear material subject to safeguards using IAEA equipment and also state's or operator's equipment by verifying its proper functioning, calibration and procedures; obtaining samples and ensuring their proper collection, treatment, handling and shipping; using and servicing IAEA surveillance equipment; affixing and removing IAEA seals; and using other objective methods which become available,^{29,30} in particular containment and surveillance measures to be used to help ensure the completeness of flow measurements.⁴⁵

The IAEA shall "make every effort to ensure optimum cost-effectiveness"⁵¹ and, in order to ensure it, should use, among other means, "the concentration of verification procedures in those stages in the nuclear fuel cycle involving the production, processing, use or storage of nuclear material from which nuclear weapons or other nuclear explosive devices could readily be made, and minimization of verification procedures in respect of other nuclear material on condition that this does not hamper the Agency in applying safeguards under the Agreement."⁵² Therefore, the statements on material unaccounted for and its limits of accuracy must not necessarily be based on equally intensive verification activities in all types of facilities or for all types of nuclear material. These activities must, however, in all cases enable the IAEA to satisfy the objective of safeguards, i.e., the timely detection of diversion of significant quantities of nuclear material."⁵³ In structuring its verification system, the IAEA takes into account not only whether material can be readily made into nuclear weapons or explosives but also the relationship between various parts of the nuclear fuel cycle. For example, although low-enriched uranium cannot be directly fabricated into nuclear weapons, its value as a starting point for the production of plutonium or for further enrichment cannot be overlooked.

To achieve optimum cost-effectiveness while ensuring the capability to detect the range of diversion strategies identified in Section II, the IAEA's verification system involves two different types of approaches, depending upon the type of nuclear facility. For facilities in which nuclear material is produced, such as enrichment facilities and power reactors and the larger research reactors, and for chemical reprocessing facilities where the material produced in reactors is separated from the other component of the irradiated fuel, the verification of all flows is of critical importance. In other types of facilities, the primary inspection activity is inventory verification.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 4: THE US NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM

This session will describe the state system of accounting and control (SSAC) for reactors and fuel cycle facilities in the licensed, commercial sector of the US nuclear community. Details of the material control and accounting (MC&A) measures dealing with the national safeguards program will be discussed. Also, the relationship will be described between the national safeguards program and the international safeguards program of the US SSAC.

After the session, participants will be able to:

- Understand the need for a state system.
- Understand the basic MC&A elements in an SSAC.
- Understand which MC&A elements serve the country's national interests and which serve IAEA safeguards.
- Understand the US SSAC.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 4: THE U.S. NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM

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I. INTRODUCTION

The purpose of Session 4 is to provide course participants with details of the U.S. national material control and accounting (MC&A) system. Each of the basic elements of the national MC&A system will be discussed in some detail in this paper. The purpose served by the Fundamental Nuclear Material Control (FNMC) Plan will be developed and the participants will be provided with information concerning the level of detail and information currently required in the FNMC Plan. The relationship will be described between the national safeguards program and the international safeguards program of the U.S. SSAC.

Two handouts will be utilized to aid course participants in understanding the U.S. national MC&A system. The first handout is a consolidation of all of the U.S. national MC&A regulations. The purpose of this document is to provide the participants with a single document of the various sections of the U.S. national MC&A regulations which, by the way, were implemented in three different time periods during the 1970's. The arrangement of the regulations in the consolidated regulations document has been made the same as that for the second handout, Regulatory Guide 5.45, "Standard Format and Content for the Special Nuclear Material Control and Accounting Section of a Special Nuclear Material License Application." The Regulatory Guide 5.45, which was developed and published by the U.S. Government in 1974, is intended to provide the licensed sector of the nuclear community with information regarding the amount and level of detail that must be provided by the facility in its FNMC Plan. This FNMC Plan is required by the national regulations.

II. U.S. NATIONAL MC&A REGULATIONS

In order to deter, detect, prevent, and respond to subnational attempts at theft of nuclear material or sabotage of facilities, an organized, national program of materials accounting and control and physical protection is needed. While these are terms that are generally used to describe the basic components of the U.S. national safeguards program, many aspects in the area of material control and accounting are similar and complimentary to the international safeguards program that is generally considered to be a part of a State's System of Accounting and Control.

In the U.S. national MC&A system, we generally define material control as that part of the safeguards program encompassing management and process controls to (1) assign and exercise responsibility for nuclear material, (2) maintain vigilance over the material, (3) govern its internal movement, location, and utilization, and (4) monitor the inventory and process status of all nuclear material. Material accounting is defined as that part of the safeguards program encompassing the procedures and systems to (1) perform nuclear material measurements, (2) maintain records, (3) provide reports, and (4) perform data analysis to account for nuclear material.

The material control system should contribute to deterrence by providing a means of readily detecting unauthorized removals of SNM, and tracing and identifying suspects, thus deterring those who fear exposure. By maintaining continuous vigilance over material, monitoring process operations, and establishing cross-checks over material movements, material transactions, and administrative controls, the material control system can provide early warning of attempt at theft or diversion. Full use of process monitoring information can provide additional safeguards alarms and can improve data analysis capabilities. Thus, the material control system should contribute to the prevention function by providing timely information to improve material loss alarm responsiveness, leading to the interruptions of attempts to steal or divert material. The material control system, by continuous monitoring and vigilance, should play a major response role in the rapid discovery of a loss of material. Material control should also play an important short-term assurance role by providing continuous indication of effective system operations and by confirming material status between physical inventories.

The material accounting system should contribute to deterrence by providing an after-the-fact detection capability for significant material loss and by discouraging those who desire anonymity after committing a theft. In the case of a hoax, the material accounting system plays an important prevention role in combating the alleged theft by providing records of material quantities and locations to assist in the verification of plant holdings. With respect to the response function, the material accounting system, especially the records, can contribute in a major way to after-the-fact loss detection, to the precise assessment of losses or alleged losses, and to the identification of suspects. However, it is in the area of assurance that material accounting makes its greatest contribution to safeguards. The primary role of material accounting is to provide long-term assurance, through records of holdings verified by physical inventories, that material is present in assigned locations and in correct amounts. In addition, shipper-receiver comparisons provide assurance that material has not been lost or stolen in-transit and that overstatements of a plant's shipments or understatements of receipts are not being used to disguise a material loss or theft.

The national material control and accounting regulations of the U.S. Nuclear Regulatory Commission (NRC) are contained in Part 70, Title 10 of the U.S. Government Code of Federal Regulations, Chapter 1 (10 CFR Part 70). Since 10 CFR Part 70 deals with all aspects of special nuclear material (SNM), it also contains regulations for applications, licensing, inspection, violations, health and safety matters, etc., in addition to national MC&A requirements. These MC&A regulations apply to special nuclear material in all licensed phases of the nuclear fuel cycle. For example, a few of the MC&A requirements apply to the power reactor phase of the fuel cycle, and to facilities with sealed SNM sources. However, most of the MC&A regulations in 10 CFR Part 70 apply to the bulk-handling facilities involved with conversion, fabrication, scrap recovery, and/or reprocessing operations. The emphasis in this paper will be on the MC&A system for U.S. licensed, bulk-handling facilities which are authorized to possess and use special nuclear material in a quantity exceeding one effective kilogram.

The national regulations in 10 CFR Part 70 include requirements for the following MC&A elements:

1. Records
2. Reports
3. Written Procedures
4. Facility Organizational Structure
5. Material Control Areas
6. Measurement System
7. Measurement Quality Assurance Program
8. Limit of Error Calculation
9. Physical Inventory
10. Accounting System
11. Internal Controls
12. Management Activities
13. Submission of FNMC Plan

Now I will discuss some details of each of the above MC&A elements and give references to the applicable requirements in 10 CFR Part 70. The requirements in some of the MC&A elements are different for low enriched uranium than for high enriched uranium or plutonium.

The records regulations, which are under 10 CFR 70.51(b) and (e), and 70.57(b), address the need for records of SNM receipts and shipments, beginning and ending SNM inventories, material added to or removed from process, material balance components, data and information from the measurement system and its quality assurance program, etc. Also, records must be maintained for training and measurement qualifications. The regulations also address the period of time which the various records must be retained by the facility.

The regulations dealing with national level reports are found in 10 CFR 70.52, 70.53, and 70.54. These reports concern the loss or theft of SNM, material unaccounted for (MUF) exceeding its measurement uncertainty (LEMUF), LEMUF exceeding specified limits, the semi-annual Material Status Reports (Form NRC-742), and the Nuclear Material Transaction Reports (Form NRC-741) for transfers of SNM.

Written procedures, 10 CFR 70.51(c), are required to be established, maintained, and followed for all material control and accounting activities at bulk-handling facilities.

10 CFR 70.58(b) and 70.57(b)(1) contain the facility organizational requirements which address the roles of the safeguards manager and the measurement quality assurance manager, the need for separation of functions within the organization, and the need for written delegation of MC&A responsibilities and authority.

The material control area regulations, which are contained in 10 CFR 70.58(d), address the need for physical and administrative controls for SNM by establishing material balance areas (MBA) and item control areas (ICA) within each plant. The custody of the SNM within each MBA or ICA is the responsibility of a single individual.

10 CFR 70.58(e) contains the MC&A regulation concerning the measurement system which each facility must establish, maintain, and follow so that the SNM present at the facility can be measured. The measurement system must have the capability to determine the element and fissile isotope content of special nuclear material received, produced, transferred between MBAs, shipped, on inventory, etc. One objective of the U.S. national MC&A regulations is to assure that all inputs to the material balance calculations are based on measurements and that the material unaccounted for (MUF) value resulting from the material balance is a meaningful value.

Each bulk-handling facility is required to have a quality assurance program for the MC&A measurement system (also called a measurement control program). The requirements in 10 CFR 70.57(b) (3-11) address the need to obtain representative samples, the calibration of each measurement technique, the monitoring and control of each technique during its use throughout the material balance period, the generation of bias correction data, the generation of information for random and systematic errors for use in determining the measurement system uncertainty (limit or error), etc.

The limit of error associated with the material unaccounted for (LEMUF) represents the uncertainty in the facility's measurement system as applied to the special nuclear material involved in the facility's operations during a material balance period. The requirements concerning LEMUF are found in 10 CFR 70.51(e). It should be noted that the U.S. only requires LEMUF calculations associated with the entire plant. LEMUF is expressed in the same units as MUF; e.g., in grams or kilograms uranium, uranium 235, or plutonium, as appropriate.

The LEMUF calculation is complex compared to the calculation of the MUF value. Since many of the inputs to the material balance expression (from which the MUF value is determined) are related, there are covariance effects which have to be dealt with properly in the LEMUF calculation in order to obtain a meaningful LEMUF value. Session 30 of this course will provide more information on the subject of calculation of the LEMUF.

The regulations in 10 CFR 70.51(e)(3) and 70.51(f) address the subject of physical inventory. In the U.S., the licensed, bulk-handling facilities usually take a physical inventory every two (2) months for plutonium and high enriched uranium ($\geq 20\%$ enriched) and every six (6) months for low enriched uranium ($< 20\%$ enriched). The inventory requirements include procedures to assure that the quantity of SNM in each item on inventory is a measured value, each item is only listed once on the inventory, all material containing SNM is inventoried, etc. Also, the requirements specify that the book inventory record must be reconciled and adjusted to the results of the physical inventory; note that if this requirement is not carried out, there is no way of obtaining the MUF value.

The basic accounting system required in the U.S. provides records and reports necessary to locate SNM and to close a measured material balance around each material control area and the total plant. The accounting system includes centralized records using double-entry bookkeeping practices, subsidiary accounts for each MBA and each ICA, and procedures for reconciling subsidiary accounts to control accounts as well as reconciling the accounts to the results of the physical inventory. Accounting records are required for quantities of SNM on inventory and for quantities added to and removed from the facility's various processing operations. The accounting system regulations also require that the MUF value, resulting from reconciling the book inventory to the results of the physical inventory, be calculated within 30 calendar days after the start of the physical inventory. The U.S. regulations concerning the accounting system are found in 10 CFR 70.51(e) and 70.58(k).

The requirements concerning internal controls for SNM are contained in 10 CFR 70.51(e)(1), and 70.58(g), (h) and (i). These requirements address procedures for SNM received by the facility operator and for SNM shipped from the facility, including the evaluation of shipper/receiver differences. Each facility must maintain a system to provide knowledge of the identity, quantity and location of all items containing SNM, including the generation and disposition of each item. There are requirements for limiting the accumulation of scrap materials, controlling and using tamper-safing seals, and the documentation of all transfers between material control areas including authorized signatures.

The management activities deal with development, revision, implementation and enforcement of control and accounting procedures at each facility. The safeguards manager and other facility management must approve in writing the MC&A procedures and revisions to the procedures. The regulations in 10 CFR 70.57(b)(2) and 70.58(c) also require a review and audit of the entire MC&A program at the facility every twelve (12) months by knowledgeable, independent persons. The results of the review and audit must be documented and reported to the corporate and facility management. The NRC also reviews the reported findings as part of its inspection program.

Finally, the MC&A regulation 10 CFR 70.58(1) requires each bulk-handling facility to submit a description of the program for control of and accounting for SNM to the U.S. NRC for review and approval. This submittal has become known as the Fundamental Nuclear Material Control (FNMC) Plan.

Each of the above 13 elements comprising the U.S. national MC&A regulations plays a role in the overall U.S. safeguards program which is designed to deter, detect, prevent, and respond to the unauthorized possession or use of significant quantities of nuclear materials through theft or diversion, and the sabotage of nuclear materials and facilities.

III. THE FNMC PLAN

Now that you have some idea of the types of MC&A information required by the national program under 10 CFR Part 70, I want to discuss the relationship of the regulations and the FNMC Plan.

The facility's FNMC Plan plays an important role in the U.S. SSAC program. Essentially the FNMC Plan describes "what is done and how it is done" in the facility in order to meet or accomplish the intent of the national MC&A regulations in 10 CFR Part 70. Therefore, while each FNMC Plan addresses the same MC&A elements, the measures described in each FNMC Plan for each MC&A element can be different from facility to facility.

One of the functions that the NRC Licensing staff must perform is to evaluate the measures described in the FNMC Plan by the facility operator in order to assure that the facility's MC&A measures meet the MC&A requirements in the national regulations. When all of the MC&A measures in the FNMC Plan have been determined by the NRC Licensing staff to be acceptable, the FNMC Plan is incorporated as a part of the facility's license. This means that the facility must follow all of the MC&A measures described in the approved FNMC Plan for all of its material control and accounting activities. The NRC Inspection staff inspects the licensed facility against its approved FNMC Plan, the MC&A regulations in 10 CFR Part 70, and any applicable license conditions. The NRC inspection program will be discussed in detail in Session 35 of this course.

As one might expect, there can be a lot of activity associated with keeping an approved FNMC Plan up to date. Changes made to the FNMC Plan by the facility operator have to be submitted in writing to the NRC for evaluation and approval. When a facility makes changes to existing processing operations, installs new operations, or decides to stop handling SNM, the FNMC Plan has to be revised to describe the activities involving control and accounting of the SNM for the modified or new operations or for the decommissioning effort.

Now I would like to discuss briefly how the licensed, bulk-handling facility operator prepares the FNMC Plan. As mentioned before, each U.S. facility authorized to possess and use SNM in a quantity exceeding one effective kilogram must have an approved FNMC Plan. The basic guidance document, which each facility uses in preparing its FNMC Plan, is the Regulatory Guide 5.45. Chapters 3-11 of Regulatory Guide 5.45 provide the level of detail required for the MC&A elements that must be included in the FNMC Plan. The facility's FNMC Plan will usually contain 9 chapters, with each chapter addressing one of the 9 MC&A elements discussed above. The FNMC Plan currently does not contain chapters on Records, Reports, and Written Procedures since these three MC&A elements are straightforward and do not need elaboration in the FNMC Plan. In addition to Regulatory Guide 5.45, there are many other NRC regulatory guides which provide information on specific MC&A elements. In Session 37 of this course, the preparation of the FNMC Plan will be discussed in more detail.

IV. U.S. INTERNATIONAL MC&A REGULATIONS

For completeness I want to briefly mention international safeguards and its relationship to the national safeguards program. 10 CFR Part 75, Safeguards on Nuclear Material-Implementation of U.S./IAEA Agreement, contains the international safeguards requirements for the private sector nuclear facilities in the U.S. Essentially, the regulations in 10 CFR Part 75 correspond to the articles in the IAEA document, INFCIRC/153. The eight basic elements for implementing international safeguards are:

- A measurement system for the determination of the quantities of nuclear material received, produced, shipped, lost or otherwise removed from inventory, and the quantities on inventory;
- The evaluation of precision and accuracy of measurements and the estimation of measurement uncertainty;

- Procedures for identifying, reviewing, and evaluating differences in shipper/receiver measurements;
- Procedure for taking a physical inventory;
- Procedures for the evaluation of accumulations of unmeasured inventory and unmeasured losses;
- A system of records and reports showing, for each material balance area, the inventory of nuclear material and the changes in that inventory including receipts into and transfers out of the material balance area;
- Provisions to ensure that the accounting procedures and arrangements are being operated correctly; and
- Procedures for the provision of reports to the Agency.

A comparison of these international MC&A elements with the U.S. national MC&A elements shows a strong relationship between the two sets. In the U.S., many of the MC&A procedures that have been implemented for national MC&A purposes (to satisfy the requirements in 10 CFR Part 70) are also used for purposes of international safeguards (to satisfy the requirements in 10 CFR Part 75).

It should be pointed out that the IAEA approved Facility Attachment for the licensed facility is incorporated as a condition of license, just like the FNMC Plan. Therefore, licensees in the U.S. that have been selected under the U.S./IAEA Agreement must follow the Facility Attachment for international safeguards as well as the FNMC Plan for national safeguards.

V. SUMMARY

The U.S. national MC&A system has been described, including details of each basic MC&A element and the role of the licensed facility's FNMC Plan.

The relationship between the U.S. national MC&A requirements in 10 CFR Part 70 and the U.S. international MC&A requirements in 10 CFR Part 75 has been described.

The U.S. SSAC for licensed facilities consists of the national MC&A system and, when applicable, the international safeguards program.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



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APPENDIX TO SESSION 4: U. S. NATIONAL MATERIAL CONTROL AND ACCOUNTING
REGULATIONS 10 CFR PART 70 CONSOLIDATION

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CONSOLIDATION OF U. S. MATERIAL CONTROL AND ACCOUNTING
REGULATIONS IN 10 CFR PART 70

I. Basic Material Control and Accounting Requirements

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- b. Records [70.51(b)]
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- g. Physical Inventory [70.51(f), 70.51(e)(3)]
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70.51(e)(4)(i), 70.51(e)(4)(ii)]
- i. Records [70.57(b)(12), 70.51(e)(4)(iii-v)]
- j. Internal Controls [70.58(g), (h), (i), 70.51(e)(1)(i, iv, v, vi)]
- k. Management [70.58(c), 70.57(b)(2)]
- l. Submittals [70.58(l)]

*Denotes specific section of 10 CFR Part 70

SPECIAL NUCLEAR MATERIAL CONTROL AND ACCOUNTING

I. Basic Material Control and Accounting Requirements

a. Definitions

1. "Additions to material in process" means receipts that are opened except for receipts opened only for sampling and subsequently maintained under tamper-safing, and opened sealed sources.
2. "Enrichment category" for uranium-235 means high-enriched uranium—that uranium whose isotope content is 20 percent or more uranium-235 by weight, and low-enriched uranium—that uranium whose isotope content is less than 20 percent uranium-235 by weight.
3. "Element" means uranium or plutonium.
4. "Fissile isotope" means (i) uranium-233 or (ii) uranium-235 by enrichment category.
5. "Limit of error" means the uncertainty component used in constructing a 95 percent confidence interval associated with a quantity after any recognized bias has been eliminated or its effect accounted for.
6. "Material balance" means a determination of material unaccounted for (MUF) by subtracting ending inventory (EI) plus removals (R) from beginning inventory (BI) plus additions to inventory (A). Mathematically,

$$MUF=BI+A-EI-R$$

7. "Material in process" means any special nuclear material possessed by the licensee except in unopened receipts, sealed sources, and ultimate product maintained under tamper-safing.
8. "Physical inventory" means determination on a measured basis of the quantity of special nuclear material on hand at a given time. The methods of physical inventory and associated measurements will vary depending on the material to be inventoried and the process involved.
9. "Removals from material in process" includes measured quantities of special nuclear material disposed of as discards, encapsulated as a sealed source, or in other ultimate product placed under tamper-safing or shipped offsite.
10. "Tamper-safing" means the use of devices on containers or vaults in a manner and at a time that ensures a clear indication of any violation of the integrity of previously made measurements of special nuclear material within the container or vault.

11. "Ultimate product" means any special nuclear material in the form of a product that would not be further processed at that licensed location.
 12. "Unopened receipts" means receipts not opened by the licensee, including receipts of sealed sources, and receipts opened only for sampling and subsequently maintained under tamper-safing.
 13. "Measurement" includes sampling and means the determination of mass, volume, quantity, composition or other property of a material where such determinations are used for special nuclear material control and accounting purposes.
 14. "Measurement system" means all of the apparatus, equipment, instruments and procedures used in performing a measurement.
 15. "Reference standard" means a material, device or instrument whose assigned value is known relative to national standards or nationally accepted measurement systems.
 16. "Traceability" means the ability to relate individual measurement results to national standards or nationally accepted measurement systems through an unbroken chain of comparisons.
 17. "Random error" refers to the variation encountered in all measurement work, characterized by the random occurrence of both positive and negative deviations from a mean value.
 18. A "systematic error" is a constant unidirectional component of error that affects all members of a data set; its value can, in some instances, be estimated by the deviation of the mean of a measurement process from a reference value. A systematic error whose value has been determined in this manner is called a bias, whose effect can be corrected for.
 19. "Uncertainty" is the extent to which a measurement result is in doubt because of the effects of random error variances and the limits of systematic errors associated with a measurement process, after the measurements result has been corrected for bias.
 20. "Calibration" means the process of determining the numerical relationship between the observed output of a measurement system and the value, based upon reference standards, of the characteristics being measured.
- b. Records
1. Each licensee shall keep records showing the receipt, inventory (including location), disposal, acquisition, import, export, and transfer of all special nuclear material in his possession regardless of its origin or method of acquisition.

2. Records which are required by the regulations in this part or by license condition shall be maintained for the period specified by the appropriate regulation or license condition. If a retention period is not otherwise specified by regulation or license condition, such records shall be maintained until the Commission authorizes their disposition.
3. Records of receipt, acquisition, physical inventory, or import of special nuclear material which must be maintained pursuant to paragraph (b)(1) of this section shall be maintained as long as the licensee retains possession of the material and for five years following transfer or export of such material. Records of inventory maintained to demonstrate compliance with paragraph II (j)(2) shall be maintained for six months.
4. Records of export of special nuclear material shall be maintained for five years after such event.
5. Records of transfer of special nuclear material to other persons shall be maintained by the licensee who transferred the material until the Commission authorizes their disposition. Records required by paragraph II (j)(4) shall be maintained for five years.
6. Records of disposal of special nuclear material shall be maintained in accordance with § 20.401(c) of this chapter.

c. Annual Inventory

Except as required by paragraph II, each licensee who is authorized to possess at any one time and location special nuclear material in a quantity totaling more than 350 grams of contained uranium-235, uranium-233, or plutonium, or any combination thereof, shall conduct a physical inventory of all special nuclear material in his possession under license at intervals not to exceed twelve months.

d. Written Procedures

Each licensee who is authorized to possess at any one time special nuclear material in a quantity exceeding one effective kilogram of special nuclear material shall establish, maintain, and follow written material control and accounting procedures which are sufficient to enable the licensee to account for the special nuclear material in his possession under license.

e. Reports

1. Each licensee shall report immediately to the appropriate NRC Regional Office listed in Appendix A of Part 73 of this chapter by telephone and telegraph, mailgram, or facsimile any case of accidental criticality and any loss, other than normal operating loss, of special nuclear material.

2. Each licensee who possesses 1 gram or more of contained uranium-235, uranium-233, or plutonium shall report immediately to the appropriate NRC Regional Office listed in Appendix A of Part 73 of this chapter by telephone and telegraph, mailgram, or facsimile any theft or other unlawful diversion of special nuclear material which he is licensed to possess, or any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion of such material in accordance with the procedures in § 73.71 of this chapter.
3. Each licensee who is authorized to possess at any one time and location special nuclear material in a quantity totaling more than 350 grams of contained uranium-235, uranium-233, or plutonium or any combination thereof, shall complete and submit to the Commission Material Status Reports on Form NRC-742, in accordance with printed instructions for completing the form, concerning special nuclear material received, produced, possessed, transferred, consumed, disposed of or lost by the licensee. All such reports shall be made as of June 30 and December 31 of each year and shall be filed the U. S. Department of Energy, P. O. Box E, Oak Ridge, Tennessee, 37830 within thirty (30) days after the end of the period covered by the report. The Commission may permit a licensee to submit Material Status Reports at other times when good cause is shown.
4. Each licensee who transfers and each licensee who receives special nuclear material shall complete and distribute Nuclear Material Transaction Report on Form NRC-741, in accordance with printed instructions for completing the form, whenever he transfers or receives a quantity of special nuclear material of 1 gram or more of contained uranium-235, uranium-233, or plutonium. Each licensee who transfers such material shall submit a copy of Form NRC-741 to the U. S. Department of Energy, P. O. Box E, Oak Ridge, Tennessee, 37830, and three (3) copies to the receiver of the material promptly after the transfer takes place. Each licensee who receives special nuclear material shall submit a copy of Form NRC-741 to the U. S. Department of Energy and to the shipper of the material within ten (10) days after the special nuclear material is received.
5. Each licensee subject to the requirements of § II shall submit a report, in accordance with paragraph (5)(i) or (5)(ii) of this section, to the appropriate NRC Regional Office listed in Appendix A of Part 73 of this chapter, with a copy to the Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, within 30 calendar days after the start of each ending physical inventory required by § II (g).
 - (i) If the material unaccounted for exceeded both its associated limit of error and 200 grams of plutonium or U-233, 300 grams of high enriched uranium or uranium U-23, contained in high enriched uranium, or 9,000 grams of U-235 contained in low enriched uranium, a statement of the probable reasons for the material unaccounted for and actions taken or planned with respect to the material unaccounted for; and

- (II) If for any material the limit of error of the material unaccounted for balance exceeds any applicable limits specified in § II (f) or approved pursuant to § II (f), a statement of the probable reasons for the limit of error and actions taken or planned with respect to the limit of error.

II. Fundamental Nuclear Material Control Program

a. Preamble

Each licensee who is authorized to possess at any one time and location special nuclear material in a quantity exceeding one effective kilogram, and to use such special nuclear material except those uses involved in the operation of a nuclear reactor licensed pursuant to Part 50 of this chapter and those involved in a waste disposal operation, and as sealed sources, shall establish, maintain, and follow written material control and accounting procedures in compliance with the fundamental nuclear material controls requirements specified in paragraphs (b) through (k) of this section and such other controls as the Commission determines to be essential for the control of and accounting for special nuclear material.

b. Organization

1. The overall planning, coordination, and administration of the material control and accounting functions for special nuclear materials shall be vested in a single individual at an organizational level sufficient to assure independence of action and objectiveness of decisions. In manufacturing organizations, such individual shall be independent of individuals or units that are solely responsible for production functions.
2. Material control and accounting functions shall be identified and assigned in the licensee organization to provide a separation of functions so that the activities of one individual or organizational unit serve as controls over and checks of the activities of other individuals or organizational units.
3. Material control and accounting functional and organizational relationships shall be set forth in writing in job descriptions, organizational directives, instructions procedure manuals, etc. Such documentation shall include position qualification requirements and definitions of authorities, responsibilities, and duties. Delegations of material control and accounting responsibilities and authority shall be in writing.
4. The licensee shall assign responsibility for planning, developing, coordinating, and administering the measurement quality assurance to an individual in his organization who has no direct responsibilities for the operation of the analytical laboratory or for the processing of material, holds a position at an organizational level which will permit independence of action and objectivity of decision and has authority to obtain all the information required to monitor and evaluate measurement quality as required by this section.

c. Material Control Areas

Material Balance Areas (MBA) or Item Control Areas (ICA) shall be established for physical and administrative control of nuclear material.

1. Each MBA shall be an identifiable physical area such that the quantity of nuclear material being moved into or out of the MBA is represented by a measured value determined pursuant to paragraph (d) of this section.
2. The number of MBAs shall be sufficient to localize nuclear material losses, or thefts and identify the mechanisms.
3. The custody of all nuclear material within any MBA or ICA shall be the responsibility of a single designated individual.
4. ICAs shall be established according to the same criteria as MBAs except that control into and out of such areas shall be by item identity and count for previously determined special nuclear material quantities, the validity of which shall be assured by tamper-safing unless the items are sealed sources.

d. Measurements

A system shall be established, maintained and followed for the measurement of all special nuclear material received, produced, transferred between MBAs, transfers from MBAs to ICAs, on inventory, or shipped, discarded, or otherwise removed from inventory and for the determination of the limit of error associated with each such measured quantity except for plutonium-beryllium sources: samples which have been determined by other means to contain less than 10 grams U-235, U-233, or plutonium each; and reactor-irradiated fuels involved in research, development, and evaluation programs in facilities other than irradiated-fuel reprocessing plants. The system shall provide for sufficient measurements to substantiate the quantities of element and isotope measured and the associated limits of error.

e. Measurement Quality Assurance

1. The licensee shall assure that any person who contracts to perform materials control and accounting measurement services conforms with applicable requirements of paragraphs (2) through (5) and (8) through (9) of this section and § II (1)(1). Conformance shall include reporting by the contractor of sufficient error data to allow the licensee to calculate bias corrections and measurement limits of error. All statistical studies shall be reported or references in the measurement report submitted to the licensee, who shall have access to the contractor's supporting control data. The licensee shall perform reviews to determine the adequacy of the contractor's program and audits to verify conformance with all aspects of the program. Reviews and audits shall be performed at intervals not to exceed 12 months. Results shall be documented and reported to licensee management.

2. In order to assure that potential sources of sampling error are identified and that samples are representative, process and engineering tests shall be performed using well characterized materials to establish or to verify the applicability of existing procedures for mixing and for sampling special nuclear materials and for maintaining sample integrity during transport and storage. The program shall assure that such procedures are maintained and followed, and that sampling is included in the procedures for estimating biases, limits for systematic errors, and random error variances.
3. The program shall include provisions for the review and approval, before use, of written procedures for:
 - (i) Preparing or acquiring, maintaining, storing and using reference standards,
 - (ii) Calibrating measurement systems, performing bulk measurements, obtaining samples, and performing compositional analyses,
 - (iii) Recording, analyzing and reporting the program data and information, and
 - (iv) Controlling measurement performance.
4. To assure the adequacy of each measurement system with respect to process flows, sampling and measurement points, and nominal material compositions, engineering analyses and evaluations shall be made of the design, installation, preoperational tests, calibration, and the operation of each system. Such analyses and evaluations shall be repeated whenever a significant change is made in any component of a system. The results of these analyses and evaluations shall be recorded.
5. Procedures and performance criteria shall be established for the training, qualifying, and periodic requalifying of all personnel who perform sampling and measurements for materials control and accounting purposes. Qualification results shall be recorded.
6. The program shall generate current data on the performance of measuring processes, including, as appropriate, values for bias corrections and their uncertainties, random error variances, limits for systematic errors, and other parameters needed to establish the uncertainty of measurements pertaining to materials control and accounting. The program data shall reflect the current process and measurement conditions existing at the time the control measurements are made. Measurements which are not controlled by the program shall not be used for materials control or for accounting purposes. The program shall include:
 - (i) The ongoing use of standards for calibration and control of all applicable measurement systems. Calibrations shall be repeated whenever any significant change occurs in a measurement system or when program data, generated by tests performed

at a predetermined frequency, indicate a need for recalibration. Calibrations and tests shall be based upon reference standards.

- (ii) A system of control measurements to provide current data for the determination of random error behavior. On a predetermined schedule, the system shall include the replicate analysis of process samples, the replicate weight or volume measurement of bulk quantities of material, and the analysis of replicate process samples.
7. The program data generated during the current material balance period shall be used for the determination of the limit of error of the plant material balance. Measurement error data collected and used during immediately preceding material balance periods may be combined with current data provided that the measurements are in statistical control, i.e., when repeated samples from the portion of the measurement system under test behave as random samples from a stable probability distribution. Under such conditions, data sets may be combined provided that the parameter estimates based on the current set of data and the previous set of data are not significantly different on the basis of appropriate statistical tests performed at a level of significance of 0.05.
 8. The licensee shall evaluate with appropriate statistical methods all program data and information, and relevant process data used to establish bias corrections and their associated uncertainties, random error variances, limits of systematic error, and other parameters pertaining to special nuclear materials control and accounting measurements, and to control measurement performance. Bias corrections shall be made by an appropriate statistical procedure.
 9. The licensee shall establish and maintain a statistical control system, including control charts and formal statistical procedures, designed to monitor the quality of each type of program measurement. Control chart limits shall be established to be equivalent to levels of significance of 0.05 and 0.001. Whenever control data exceed the 0.05 control limits, the licensee shall investigate the condition and take corrective action in a timely manner. The results of these investigations and actions shall be recorded. Whenever the control data exceed the 0.001 control limits, the measurement system which generated the data shall not be used for material control and accounting purposes until the deficiency has been corrected and the system has been brought into control at the 0.05 control level.
- f. Limit of Error
1. Establish and maintain a system of control and accountability such that the limits of error for any material unaccounted for (MUF) ascertained as a result of the material balances made pursuant to paragraph (g) of this section do not exceed (i) 200 grams of plutonium or uranium-233, 300 grams of high enriched uranium or uranium-235 contained in high enriched uranium, or 9,000 grams of uranium-235

contained in low enriched uranium, (ii) those limits specified in the following table, or (iii) other limits authorized by the Commission pursuant to paragraph (f)(2) of this section:

Material Type	Limit of Error of MUF on Any Total Plant Inprocess Material Balance* Percent
Plutonium element or uranium-233 in a chemical reprocessing plant.....	1.0
Uranium element and fissile isotope in a reprocessing plant.....	0.7
Plutonium element, uranium-233, or high enriched uranium element and fissile isotope - all other.....	0.5
Low-enriched uranium element and fissile isotope - all other.....	0.5

2. An applicant or a licensee subject to the requirements of paragraph (f) of this section may request limits higher than those specified in paragraph (f)(1) of this section. The requested higher limits shall be based on considerations such as the type and complexity of process, the number of unit operations, process throughput quantities, process recycle quantities, and the technology available and applicable to the control and accounting of the material in the process. The Commission will approve higher limits if the applicant demonstrates:

- (i) That he has made reasonable efforts and cannot meet the limits of error of MUF specified in paragraph (f)(1) of this section; and
- (ii) That he has initiated or will initiate a program to achieve improvements in his material control system so as to meet the limits specified in paragraph (f)(1) of this section.

3. Within 30 calendar days after the start of each ending physical inventory required by paragraph (g) of this section:

*As a percentage of additions to or removals from material in process, whichever is greater.

- (i) Calculate, for the material balance interval terminated by that inventory, the limit of error of material unaccounted for for each element and the fissile isotope for uranium contained in material in process.

g. Physical Inventory

Each licensee shall:

1. Establish physical inventory procedures to assure that:

- (i) The quantity of special nuclear material associated with each item on inventory is a measured value;
- (ii) Each item on inventory is listed and identified to assure that all items are listed and that no item is listed more than once;
- (iii) Cutoff procedures for transfers and processing are established so that all quantities are inventoried and none are inventoried more than once;
- (iv) Cutoff procedures for records and reports are established so that all transfers for the inventory and material balance interval and no others are included in the records; and
- (v) Upon completion of the inventory, all book and inventory records, both total plant and material balance area, are reconciled with and adjusted to the physical inventory.

2. Establish inventory procedures for sealed sources and containers or vaults containing special nuclear material that provide for:

- (i) Identification and location of all such items;
- (ii) Verification of the integrity of the tamper-safing devices for such items;
- (iii) Reverification of identity and quantity of contained special nuclear material for each item not tamper-safed, or whose tamper-safing is found to have been compromised;
- (iv) Verification of the correctness of the inventory records of identity and location for all such items; and
- (v) Documentation in compliance with the requirements of paragraphs (g)(2)(i), (ii), (iii), and (iv) of this section.

3. Establish inventory procedures for special nuclear material in process that provide for:

- (i) Measurement of all quantities not previously measured by the licensee for element and fissile isotope; and

- (ii) For the material whose content of element and fissile isotope has been previously measured by the licensee but for which the validity of such previously made measurements has not been assured by tamper-safing, verification of the quantity of contained element and fissile isotope by remeasurement.
4. Conduct physical inventories according to written inventory instructions for each inventory which shall:
- (i) Assign inventory duties and responsibilities;
 - (ii) Specify the extent to which each material balance area and process is to be shut down, cleaned out, and/or remain static;*
 - (iii) Identify the basis for accepting previously made measurements and their limits of error;
 - (iv) Designate measurements to be made for inventory purposes and the procedures for making such measurements; and
 - (v) Identify the means by which material on inventory will be listed to assure that each item is inventoried and that there is no duplication.
5. Conduct physical inventories made in accordance with the criteria for physical inventories set forth in paragraph (g) of this section at intervals determined from the start of the beginning inventory to the start of the ending inventory not to exceed:
- (i) 2 calendar months for plutonium except for plutonium containing 80 percent or more by weight of the isotope Pu-238, uranium-233 and for uranium enriched 20 percent or more in the isotope uranium-235 (except as provided in paragraph (g)(5)(ii) of this section); and
 - (ii) 6 calendar months for uranium enriched less than 20 percent in the isotope uranium-235; for plutonium, U-233 and high-enriched uranium in that portion of an irradiated-fuel reprocessing plant from the dissolver to the first vessel outside of the radiation shielded portion of the process; and for plutonium containing 80 percent or more by weight of the isotope Pu-238.
- h. Accounting System
1. A system of records and reports shall be established, maintained, and followed which will provide information sufficient to locate special nuclear material and to close a measured material balance around each

*No process shutdown and/or cleanout for inventory is required if requirements with respect to MUF and the limit of error of MUF as specified in paragraph (f)(1)(ii) of this section are met using other inventory methods.

material balance area and the total plant. Such a system shall include:

- (i) A centralized accounting system employing double-entry book-keeping;
 - (ii) Subsidiary accounts for each material balance area and item control area;
 - (iii) Procedures for the reconciliation of subsidiary accounts to control accounts at the end of each accounting period; and
 - (iv) Procedures for reconciliation of control and subsidiary accounts to the results of physical inventories.
2. Maintain procedures which shall include:
- (i) Records of the quantities of special nuclear material added to or removed from the process;
 - (ii) Inventory records for the quantity of special nuclear material in process;
 - (iii) Means for control of and accounting for internal transfer documents.
3. Within 30 calendar days after the start of each ending physical inventory required by paragraph (g) of this section:
- (i) Calculate, for the material balance interval terminated by that inventory, the material unaccounted for (MUF) for each element and the fissile isotope for uranium contained in material in process;
 - (ii) Reconcile and adjust the book record of quantity of element and fissile isotope, as appropriate, to the results of the physical inventory.
1. Records
1. Within 30 calendar days after the start of each ending physical inventory required by paragraph (g):
- (i) Complete and maintain for a period of five years material balance records for each material balance showing the quantity of element and fissile isotope, as appropriate, in each component of the material balance, with the associated limit of error for the material unaccounted for both in terms of absolute quantity of element and fissile isotope and relative to additions to or removals from material in process for the interval, where results of limit of error calculations are recorded in sufficient detail to permit an evaluation of sources of error.

- (ii) Complete and maintain for a period of five years a record summarizing the quantities of element and fissile isotope, as appropriate, for ending inventory of material in process, additions to material in process during the material balance interval and removals from the material in process during the material balance interval; and
 - (iii) Complete and maintain for a period of five years a record summarizing the quantities of element and fissile isotope, as appropriate, in unopened receipts (including receipts opened only for sampling and subsequently maintained under tamper-safing), and ultimate products maintained under tamper-safing, or in the form of sealed sources;
2. A records system shall be provided in which all data, information, reports, and documents generated by the measurement quality assurance program shall be retained for five years, except for those records pertaining to the training and qualification of personnel who perform measurement activities pursuant to § II (e)(5). Those records shall be retained for two years. Records shall include a summary of the error data utilized in the limit of error calculations performed for each material balance period. The records system shall be organized for efficient retrieval of program information. Each reported result shall be readily relatable to the original measurement data and to all relevant measurement control information, including pertinent calibration data. Records shall be available for NRC inspection.
- j. Internal Controls
1. Procedures shall be established, maintained, and followed to:
- (i) Assure accurate identification and measurement of the quantities of special nuclear material received and shipped by a licensee;
 - (ii) Review and evaluate shipper-receiver differences on an individual container or lot basis, as appropriate, on a shipment basis, and on a cumulative basis for shipments of like type material;
 - (iii) Take appropriate investigative and corrective action to reconcile shipper-receiver differences that are statistically significant at the 95 percent confidence level except those shipments which involve differences of 50 grams or less of U-235, U-233, or plutonium; and
 - (iv) Maintain records of shipper-receiver difference evaluation, investigations, and corrective actions on file at the plant for a period of five years.
2. A system of storage and internal handling controls shall be established, maintained, and followed to provide current knowledge of the

identity, quantity, and location of all special nuclear material contained within a plant in discrete items and containers.

3. Procedures for special nuclear material scrap control shall be established, maintained, and followed to limit the accumulation and the uncertainty of measurement of these materials on inventory. Such procedures shall include:

- (i) Identification and classification of special nuclear material scrap;
- (ii) Regular processing and recovery of scrap so that no item of such scrap generated in the licensee's plant measured with an uncertainty of greater than ± 10 percent remains on inventory longer than six months when such scrap contains plutonium, U-233, or uranium enriched 20 percent or more in the isotope U-235 or twelve months when such scrap contains uranium enriched less than 20 percent in the isotope U-235 or plutonium containing 80 percent or more by weight of the isotope Pu-238.

4. Maintain procedures which shall include:

- (i) Procedures for tamper-safing containers or vaults containing special nuclear material not in process, which include control of access to the devices and records of the date and time of application of each device to a container or vault; unique identification of each such item; inventory records showing the identity, location, and quantity of special nuclear material for all such items; and records of the source and disposition of all such items;
- (ii) Unique identification of items or containers containing special nuclear material in process; inventory records showing the identity, location, and quantity of special nuclear material for all such items; and records of the source and disposition of all such items;
- (iii) Documentation of all transfers of special nuclear material between material balance areas to show identity and quantity of special nuclear material transferred;
- (iv) Requirements for authorized signatures on each document for transfer.

k. Management

1. A management system shall be established, maintained, and followed to provide for the development, revision, implementation and enforcement of nuclear material control and accounting procedures. The system shall include:

- (i) Provisions for written approval of such procedures and any revisions thereto by the individual with overall responsibility for the material control and accounting function and by licensee plant management.
 - (ii) Provision for a review at least every 12 months of the nuclear material control system by individuals independent of both nuclear material control management and personnel who have direct responsibility for the receipt, custody, utilization, measurement, measurement quality, and shipment of nuclear material. Such a review shall include a review and audit of material control and accounting procedures and practices and an audit of the nuclear material records. The results of the review and audit along with recommendations for improvements shall be documented, reported to the licensee's corporate and plant management, and kept available at the plant for inspection for a period of five years.
2. Provisions shall be made for management reviews to determine the adequacy of the measurement quality assurance program and to assess the applicability of current procedures and for planned audits to verify conformance with all aspects of the program. These reviews and audits shall be performed at intervals not to exceed 12 months. Audits and reviews shall be performed by trained individuals independent of direct responsibility for the receipt, custody, utilization, measurement, measurement quality and shipment of special nuclear material. The results of reviews and audits shall be recorded and reported to licensee management.
1. Submittals

Each applicant or licensee subject to the requirements of § II (a) shall submit a full description of his program for control of and accounting for special nuclear material in his possession under license to show how compliance with the requirements of this section will be accomplished. This program shall be followed by the licensee after the program is approved by the NRC.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 5: THE HUNGARIAN NATIONAL MATERIAL CONTROL AND ACCOUNTING SYSTEM

The history, requirements, organization, and operation of the state system of accounting and control in Hungary will be described. Emphasis will be placed on the safeguards system at power reactor facilities; however, the scope of the Hungarian nuclear program will be described. The interactions between the state, the IAEA, the facility, and supplier safeguards authorities will be discussed.

After the session, participants will be able to:

- Understand the organization and operations of the Hungarian SSAC.
- Understand the reporting and inspection requirements for nuclear materials control and accounting.
- Understand the interface between the SSAC and the IAEA, the facility operator, and supplier state authorities.



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OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



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SESSION 5: THE HUNGARIAN NATIONAL NUCLEAR MATERIAL
CONTROL AND ACCOUNTING SYSTEM

Otto Lendvai

National Atomic Energy Commission of Hungary

I. INTRODUCTION

We are so happy to advise others that occasionally
we even do it in their interest.

Jules RENARD: Journal

This lecture is intended to provide a brief description of the Hungarian SSAC, its history, structure and functions. Some facets of our operational experience will be highlighted specifically. I wish to state from the outset that I cannot but provide somewhat biased information, influenced by the specific situation in our state. Even where information provided may sound like presentation of a good and ready recipe, it should be regarded in the light of conditions prevailing in your own system. Thus, for example, in a state where only some of the nuclear activities are subject to IAEA safeguards, the SSAC's tasks will be formulated in a different way. Most probably they will be rather complicated. My domestic experience extends only to a system under full INFCIRC/153 safeguards.

II. BEGINNING AND HISTORY

Historians relate not so much what is done as what
they would have believed.

Benjamin FRANKLIN: Poor Richard's Almanack

The first thing to ask oneself when embarking upon setting up an SSAC or improving an existing one is: "What is the purpose?" Specifically in the case of starting afresh, should the new body serve only national and IAEA safeguards, or would it be intended for other aims as well--like providing data to Health and Safety authorities or even to enforce H+S regulations?

Another question: "Is it thought that whatever existed in the field before should be preserved intact as much as possible, or would it be more important to streamline the system so as to adapt it to its new functions?"

Naturally, the answers may vary. In some cases, even the posing of these questions may appear superfluous--but that also is some kind of answer. Only after clarification of this issue should the organizational side be tackled. Even before J. N. Parkinson, it was known that it is usually easier to create new organs and organizations than to introduce any functional changes into an existing framework.

Of course, many of these questions are decided at some higher governmental or even legislation levels, largely determining what the necessary structures will look like. This makes it even more desirable that there should exist at all levels an awareness of the requirements and limitations posed by the magnitude and type of the national nuclear economy, including all associated R+D.

The outcome may also be influenced by whether IAEA safeguards--or even creation of an SSAC by itself--are considered just as a kind of unavoidable evil or as a means to achieve a positive and desirable purpose.

This latter was the attitude in Hungary at the time an SSAC as required by the Safeguards Agreement was to be created. The Hungarian government considers NPT as a treaty that corresponds to our national interests.

But the setting up of an SSAC had to be done shortly after a major reorganization of the NAEC as a whole--equivalent, as it was, to severe trimming. Authorities never like to go back on recent decisions. Still, it was clear that a fresh start had to be made, at least as far as accounting was concerned. Before that, nuclear materials were accounted for, together with other radioactive materials, not by mass but by their radioactivity. It was also regarded as desirable to keep the SSAC as transparent as possible to not cause unnecessary problems; that required autonomy and separation from the handling of other (for example, commercial) data.

It was clear that acceptance of IAEA safeguards would have to evoke concerted action of various national offices and departments. This involved foreign affairs, trade, industry, security, legal matters, and finance (not only to obtain funds to run the system, but also because that is the ministry responsible for customs offices, etc).

The first step was establishment by the government of a temporary inter-departmental committee, composed of deputy ministers (under the chairmanship of the Chairman of the NAEC), to set up coordinated regulations that would cover the field. Secretary to that committee was a staff member of the NAEC, who also chaired a group of experts nominated by the same deputy ministers. In fact, the experts had first to get acquainted with the matter. After that, they produced good work. The committee itself met in plenary session only at two occasions.

The outcome was that the NAEC was made responsible for all matters related to IAEA safeguards, including the related international contacts. Actual accounting was to be done, however, at the Institute of Isotopes of the Hungarian Academy of Sciences, hereinafter just "the Institute", which was already entrusted with the task to keep records on all radioactive material in Hungary. The Institute is the sole distributor of radioactive materials in Hungary--not only with regard to international transfers, but also domestically. Only nuclear power plant fuel is exempt from this restriction. Within the Institute, a special unit was set up consisting of two people working full time on safeguards only, and of three to four more who have regular, long-term, but part-time assignments associated with safeguards (this stands for the head of the unit as well). Together with several staff members of the NAEC, the members of this unit were also commissioned to conduct on-site inspections at all locations holding nuclear materials. Every facility and other location holding nuclear materials was requested to nominate a person responsible for safeguards matters and for contacts with the national authorities. In the absence of such a nomination, the actual head of the facility or location was considered to bear this responsibility in person.

The Institute--in its capacity as an R+D institution--was supposed to be capable of conducting the necessary measurements. As to accounting, we de-

cided to adopt for domestic reporting the same philosophy, the same system, and even the same codes as the IAEA.

This latter decision was not made by default. We had reason to believe that the system developed by the Agency within the framework of INFCIRC/153 was sufficient while being limited to really necessary matters. Moreover, this obviated the need for recoding and extensive cross-referencing. It made life easier for the national office, for the IAEA inspectorate, and, this is important, for facility operators as well.

It was repeatedly suggested to the NAEC to incorporate into the set of data collected and regularly processed by the system some supplementary data, such as consignment numbers and other shipment data. After all, the Institute was entrusted with its safeguards task just because it regularly receives this information, using it not only to confirm reported data, but also to detect nonreported events. Still, it was decided to keep these data on the fringe of the system because of their lack of universality and homogeneity. (The idea keeps cropping up periodically.)

Since the time when the system was set up, major changes in the legal structure did not occur. The status of the NAEC was somewhat modified in the mid-seventies; at present, some developments indicate that it may be elevated again. In essence, the NAEC is a governmental agency, serving as advisory organ to the government. In the field of NM safeguards it exercises the functions of a national authority. In other fields, such as H+S problems of nuclear energy, the NAEC acts rather as a coordinating organ, bringing together the viewpoints of all ministries and other national authorities concerned.

The legal status of the NAEC is laid down in its Statute. This was somewhat modified during the last fifteen years, but never substantially. In 1980, an Act on Atomic Energy was promulgated by the National Assembly (the Hungarian parliament); a regulation issued subsequently by the government provided executive details. The Act did not change the NAEC Statute, which is in force in its present form since 1978.

Larger changes have been--and are going to be--caused by the expansion of activities in the nuclear field. The commissioning of our first nuclear power plant was a major quantitative and qualitative step, requiring corresponding changes in our national system. This we saw in advance and prepared for it to be able to cope with the new situation adequately. However, we now recognize that we have been slow to respond to more gradual changes. They were partly due to domestic developments, but there is also a trend towards more explicit specification of the Agency's requirements in order to strengthen the credibility of safeguards. The Hungarian NAEC is at present reviewing our work in this light. Within this year some changes will occur. Thus, our system is not to be perceived as closed and immutable--but rather as an organism capable of adequate responses to changes in our environment.

III. NUCLEAR ACTIVITIES: THE SYSTEM TO BE CONTROLLED

One has to look out for engineers: they begin with sewing machines and end up with an atomic bomb.

Marcel PAGNOL: Critique des Critiques

At the time when implementation of IAEA safeguards in Hungary began, we had two focal points of atomic research activities (at Budapest and Debrecen). Of these two, the institutions at Debrecen handle only minor quantities of

fissionable and source materials. In Budapest, there was a Central Institute of Physical Research (CIPR) and the Institute of Isotopes, co-located at the same site, but otherwise independent. The latter was created by the NAEC, the former by the Hungarian Academy of Sciences and subsequently passed over to the NAEC. Not long before safeguards started, both institutions were passed over to the Academy, leaving the NAEC without a laboratory base of its own.

The CIPR had a research reactor rated at 2 MW and a whole range of laboratories. The Institute also had its laboratories, but in addition, it acted as a central distributor for all radioactive substances in Hungary.

Hungary has uranium ores, which are mined and processed to the stage of ore concentrate, but these activities do not fall under safeguards.

Some dozens of plutonium-containing neutron sources were used at various locations. Thorium is used in batches of several hundred kilograms by the industry. And finally, small quantities of both source and special fissionable materials were handled and used at many places all over the country.

In fact, at the beginning one of the major problems was to find out what nuclear materials were present in Hungary. True, all such materials are radioactive and should have been under control, but small batches of less active materials were not subject to that.

We decided to start by setting up a very tight system and to include into the national inventory even the smallest items and batches to get a reliable picture. Another starting point was that it is usually much easier to relax existing regulatory provisions and procedures than to tighten regulations after getting people used to a loose way of handling matters.

This seems to have been the right approach. We managed to get a reasonably reliable picture of the national inventory of nuclear materials, although for several years some additional quantities were unearthed, sometimes at most unlikely places. Of course, we did our best to have small quantities exempted, but only when being sure that they would not subsequently emerge at places where they could complicate the accounting.

At present, the situation is as follows: the research reactor has been upgraded to 5 MW. It is used both for R+D and for the routine production of radioactive isotopes, especially of short-living ones for medical diagnostic and therapeutic purposes. In addition, we now have a training reactor at the Budapest Technical University, a nuclear power plant with two reactors, and a second such plant under construction. Further NPPs will follow. At the R+D side, CIPR has set up two critical facilities, one of them with some thousands of fuel pins. The national inventory has also significantly increased simply due to accumulation of spent fuel of two different basic types from the research reactor. This latter causes safeguards problems, because it is intended to have it hermetically encapsulated. Our arrangements with the Agency should then incorporate practicable provisions for that case, but the procedures have not yet been agreed upon. There was also a growth in the use of neutron sources. Now we give preference to sources not subject to safeguards, but the majority of the existing ones contain safeguarded plutonium.

We have in early years found out, together with the IAEA, that facilities and material balance areas (MBAs) are best defined on the principle of one-to-one correspondence. So at present the system includes five facilities with five MBAs to match. One of these includes all minor "other locations."

On the quantitative side, our system had to handle at the beginning some hundreds of batches containing more than a thousand items, although most of the batches contained one item only or a comparatively small quantity of bulk research material. We had less than two kilograms of plutonium and about two tons of enriched uranium.

Now we have hundreds of tons of enriched uranium and an ever-growing number of batches at the nuclear power reactors. The inventories at all other facilities have also increased, but at a lower rate. More important, however, is the fact that before the commissioning of the NPP the rate of changes was very slow. In other words, the number of PIL and ICR lines to be processed and transmitted to the IAEA was small. Now it is very much larger and keeps growing.

Some more data: our NPPs are WWER-440 PWRs. Summing the figures for the two reactors in operation (they constitute one facility), their cores contain about 700 fuel assemblies. Annually, about one third of this quantity is discharged (meaning that burn-up and Pu accumulation has to be reported). Receipts of an equal number of fresh assemblies and, in later years, outgoing shipments also add to the number of ICR lines. At the same time, the number of physical inventory listing (PIL) lines for each such facility keeps increasing until some maximum size, which will depend upon the length of cooling period before shipping back to the original supplier.

This supplier is, in our case, the USSR. Our arrangements--contractual and at the level of governmental agreements--provide for assured fuel supplies during the lifetime of the NPPs and also for reshipment of spent fuel. This suits us well. We have embarked upon manufacturing machinery for NPPs and may be interested in the production of specific kinds of fuel cycle equipment but do not plan to set up a closed fuel cycle of our own.

As to other nuclear materials, the USSR is our major supplier of nonfuel items, but we also maintain import/export contacts with other countries. In addition, we have several geological prospecting companies that possess and use Pu-containing neutron sources for work conducted abroad. Under the terms of the safeguards agreements, this is also to be reported to the IAEA.

IV. INSTITUTIONAL FRAMEWORK AND STRUCTURE

An administration, like a machine, does not create. It carries on.

Antoine de SAINT-EXUPERY: Flight to Arras

Our laws and regulation--including the State of the NAEC--state that responsibility for the organization and functioning of a national system of accounting for and control of nuclear materials in Hungary rests with the NAEC. The NAEC is responsible directly to the government. To the extent necessitated by the lack of back-up institutions of its own, the NAEC may delegate functions to specialized institutions, in agreement with relevant authorities. International contacts in safeguards matters are handled entirely by the NAEC. This means direct relations with the IAEA and with national authorities of the same standing in other states. It hardly needs mentioning that whenever responsibilities and interests of other Hungarian national bodies and authorities are involved, the NAEC has to coordinate its actions with them. More specifically, the NAEC negotiates and signs on behalf of Hungary Subsidiary Arrangements to the Safeguards Agreement. It forwards to the Agency all relevant communications, including routine and special reports, notifications, etc. It receives from the IAEA all statutory notifications and statements.

In our understanding, a special role is assigned to NAEC representatives during on-site inspections: they must represent the national interests vis-a-vis both the inspectors and facility operators.

The NAEC has a Secretariat, within which a separate division handles all matters connected to safeguards. Right now, this division consists of two

persons. In some cases, another staff member of the NAEC deputizes for the SG director, but this is becoming increasingly difficult. At present, efforts are under way to enlarge the division to some extent (a tedious problem), if only because this is a job requiring training and experience in widely diverging fields.

The unit responsible for the actual accounting, etc., works in the Institute, which is several miles of city traffic away from us. It has a mainstay full-time staff of two persons, plus four more, for which SG work is only part of their routine tasks. Of course, for special jobs--such as R+D or temporary participation in safeguards-related activities--other people may be recruited into temporary teams, but this usually goes along with the need for additional efforts to ensure that the peculiarities of the SG approach are properly reflected in the work.

It is worth mentioning that we regard the Institute unit not merely as a subordinated group. Whenever, for one reason or another, the NAEC safeguards division is not able to act, the Institute team can and does take over almost any of our functions. Especially in the first years of the existence of our SSAC, this took place quite often, but even right now it is important, permitting to keep our system smaller and improving its internal coherence.

As to the technical basis for the work to be conducted: the NAEC Secretariat has the usual set of office equipment, from phone and word-processor to telex, photocopiers and data terminal (actually dubbing for a PC).

The Institute team has access to the Institute's mainframe computer and also a PC/terminal unit of the same type as the AEC, plus another PC. Whenever necessary, access to quite sophisticated measuring equipment is also provided, but they complain about shortage of run-of-the-mill equipment for routine verification measurements.

Communications between the major SG units are provided by phone and telex. We could use the phone link to exchange computer data, but for the time being find it more convenient and reliable to transmit data on floppy disks.

The staff of the Institute is not directly subordinated to the NAEC. However, the Secretary General of the Academy is a member of the NAEC, thus providing a link to the Institute from a higher authority level.

The third institutional level is at the facilities and "other locations." One or more persons are nominated by the facility management to be responsible for the control of nuclear material at the facility and for contacts with us. In many cases, the same persons are also responsible for local enforcement of H+S rules. The NAEC could refuse to accept a given nomination, but so far there were no real problems in that respect. The same persons are responsible for such matters as correctness of inventory taking, timely preparation of NM reports to the national authority, provision of assistance required by national and IAEA inspectors in order to discharge their obligations, etc. In the absence of national authority staff, these persons are entitled to act during IAEA inspections as representatives of the national authorities at facility level. To that end, the national authority keeps these people fully informed of all relevant legal and other developments. Copies of the SG Agreement, Subsidiary Arrangements and of the relevant Facility Attachment are deposited at all facilities in Hungarian and English, together with national regulations and instructions. Formal training and/or examinations have not been introduced, but routine on-the-job discussions are usually conducted in such a way as to deepen understanding of all related issues; familiarity with the essential documents is checked. Free expression of opinion is encouraged (including objections that may be raised on the local level against specific demands of the national authorities).

This seems to have provided these people with a good motivation. We have had no serious difficulties at this level of our system. Also, in our experience, these persons receive all necessary support from their management. Still, there are problems, and we try to keep the amount of paperwork at their level down. They are expected to be able to measure the quantities of NM under their supervision, and at large facilities they are usually equipped to perform that. At smaller places, they usually can do no more than believe whatever has been stated to them by suppliers. This should increase in such cases the burden on the Institute, but, as said above, this latter also has its limitations.

V. ACCOUNTING

He hath a good judgment that relieth not wholly
on his own.

Thomas FULLER: Gnomologia

As stated above, our internal accountability system is modeled after the IAEA's. This requires that all and any NM should be accounted for in one or another MBA defined for that purpose. We have five of these. One is pretty straightforward, representing the Training Reactor. Two pertain to the CIPR. Of these, one includes the fuel inventory of the research reactor; the other covers everything else, including samples temporarily placed into the reactor for irradiation.

The fourth MBA represents the NPP with its two reactors. The reason for putting them together is that they both are accessible for inspection from the same reactor hall and that the fresh fuel for both reactors is stored at the same place. Since reloading of both reactors is performed at different times, this calls for special procedures with regard to physical inventory taking. This is done in any calendar year by "interim" inventory taking and verification of the core contents of whichever reactor is opened and reloading first, sealing the reactor and completing the physical inventory taking when the second reactor is reloaded and ready to be closed for the next campaign.

The fifth MBA (centered at the Institute of Isotopes) covers all other safeguarded material in Hungary. This includes (1) the Institute's own NM, and (2) material stored there temporarily (for redistribution or before ultimate disposal), or just "running through" as in the case of import of fuel for the Research Reactor. Finally, it encompasses the nuclear materials pertaining to "other locations," that is, to various small users of safeguarded nuclear material all over the country. They have the accounting status of inventory key measurement points (KMPs) of the MBA HU-C, as it is coded. There are about twenty of them.

It might have been tidier to set up separate MBAs for the Institute and for all "other locations," but this is one of the matters you see so clearly in retrospect only. Moreover, up to now, this arrangement did not cause us (or the IAEA) any serious troubles.

At the "other locations," the flow of NM is usually negligible. They are bound to keep records on the nuclear material on separate pages of a "Radio-isotope Accounting Book" required by the H+S regulations. Inventory taking must be performed once a year at all such places simultaneously and usually causes no problems because quantities are small, both in terms of weight and of the number of items. The results of inventory taking and inventory changes are to be reported to the Institute. No nuclear material balance reports are

to be provided. The forms used are the same as those used by the IAEA, partly even with English headings.

The other MBAs also send in the usual sets of forms. The accounting for NM pertaining to the MBA HU-C is performed by the same personnel that keep the records for the other MBAs and the "national" accounts. This would require a closer auditing of their data by NAEC staff, and that is, in fact, scheduled.

We make great efforts to keep the system transparent. To mention only some of the factors:

At the NPP we require full batch follow-up. At other places it has not been introduced formally and there is no such intention, either. But we place great importance upon being able to trace each batch from any point in time back to its origin and forward to the present.

To assist us in that, batch names used in Hungary are nonrepetitive in the sense that there never are in Hungary two different batches with the same name. (At least, not simultaneously, except for momentary instances, for example, after receipt of an imported batch bearing a name already "occupied." In such cases, one of the batches in question--usually the new one--is immediately renamed). This is achieved for nonfuel materials by notifying the facilities what batch names they may give to batches.

Power plant fuel assemblies are normally accounted for under the manufacturer's serial numbers. The much smaller fuel assemblies at the Research and Training Reactors are listed individually by their serial numbers, but they are accounted for in "running account" batches: for example, one such batch for all fresh fuel assemblies of the same type stored together. When more fuel arrives, the batch is increased; when fuel is removed (for example, to be loaded into the core) the batch is correspondingly decreased.

Another case: we try to run our work so that reports to the IAEA would match those sent to the Institute from the other MBAs by entry number and number of report (besides all other data, of course). This very much reduces the auditing and reconciliation effort.

You may have noted that in the description above the term "national" accounts was used in inverted commas. In fact, for safeguarded NM we do not produce overall "national" records--almost all we need is provided for in the MBA-wise records. Actually, nationwide records are also maintained, but for nonsafeguarded material. They relate to material exempted from safeguards; there is also a ledger and a batchwise register of terminations. They are similar in structure to the MBA-wise record (we use the same algorithms) and are maintained for various statistical purposes, but also for practical reasons. We do not want "exempted" material to pop up as if out of nowhere in the midst of safeguarded nuclear material; this might complicate up our accounts.

Up to now, all reporting was done on hard copy. We have, however, started to report to the IAEA on magnetic disks (at first, for test purposes only). On the domestic level, we are also changing over to disks as data carriers from the NPPs to the Institute and between the latter and the NAEC. This entails some small changes in the rules governing hard copy reporting, in order to keep intact the entry and report number correspondence mentioned above.

VI. INTRODUCTION OF E D P

It is critical vision alone which can mitigate the unimpeded operation of the automatic.

Marshall McLuhan: Magic that Changes Mood

in the first years of its existence in fact, up to the commissioning of the NPP, accounting was done in the Institute, essentially manually. They maintained card files, totalling sheets, ledgers, etc. It was done quite well; the IAEA did not complain (neither did the staff). But even if they could have coped with the increasing workload for some more time, we wanted to put more emphasis on the control part of our overall task, that is, upon local audit and verification. We wanted to entrust our accountant with conducting more of such audits (but also with simpler verification tasks). To that end, we had to relieve him from manual drudgery.

We went the usual (up to several years ago) way for institutions not having a computer of their own, hiring the services of a competent computer center to set up the necessary software and subsequently to run the system for us. The attempt did not even fail; they prepared a beautiful software set, which did almost all we had specified. They were all set to run it. But we found two major faults: firstly, the preparation and dispatch of reports to the IAEA would have involved too many steps, including repeated transmission of data between different locations. Secondly, it would have involved very high-running costs.

To some degree, slowness could have been overcome with the help of still more money (for priority rating) and of reliable data links between several institutions. The running costs we could hardly do anything about. In addition, we would have had to get several terminals.

By then, we found out that if we acquired intelligent terminals (at the price of just one year's running costs at the computer center) we could use them as PCs, capable to do the things we needed. So we decided to cut our losses and postponed the switchover for a year. We obtained the hardware and charged one of our physicists to prepare the software (in BASIC, so as to keep it transparent and intelligible to others). As a side effect, we found out that we could use the PCs for other tasks, too.

We negotiated and tested with the IAEA reporting on magnetic disks, which--again in our special situation--seem to be the best carrier medium. Magnetic tape went along with red tape (customs, etc.). Disks enabled us to dispose of all that.

We know that we are by far not through yet. The system is in its initial stage and will have to be expanded to cover needs that we already perceive and others that we do not yet know about. But it is something we now have well in hand and head, and which--we already know--is amenable to future expansion (something we regarded as more than questionable with the system worked out by the computer center).

Reports sent in by MBAs and "other locations" are processed by the accountancy group in different ways, depending upon their source.

The essential operations are checking of the data, updating of the MBA inventory records, and preparation of reports to the Agency.

Data checking consists of syntactic checks and of comparison of data with earlier ones and with some "design" data. In particular, batches are expected not to appear out of nowhere nor disappear without trace. Internal shipments should match receipts. KMP totals should remain within design and operational limits. In addition, data should just make sense. An oil or water prospecting company will hardly need enriched uranium fuel rods. A batch of uranium oxide powder may be split into two, while a Pu-Be neutron source may not.

Data are checked against corroborating information available at the Institute. Any shipment of nuclear material abroad requires written permission issued by the NAEC, usually on a case-by-case basis. (For certain batches,

such permissions may be granted for repeated crossing of the border within a specified period--for example, in cases of Pu-Be sources remaining in Hungarian possession, etc.). Data are also checked against information from other sources. Because the Institute is a clearing house for all exports and imports of radioactive substances, reports can be checked versus actual shipping documents. H+S information is also used for confirmation of data.

A special situation arose with the NPP. There, the data were processed by computer--and large amounts of data, at that. It would have been foolish to try to cross-check these data manually on a batch-by-batch basis. At that time, even the reports from us to the IAEA were already produced at the Paks NPP. (Actually, we let some errors slip through).

EDP will permit us to introduce a more thorough checking of the data reported. With all that has been said above, we intend to step up the number of tests and checks to which incoming data are submitted. Most of them will be performed at the Institute, but I intend to introduce more cross-checking of our own. EDP will be used also to assist audit of MBA and local data by producing special selective printouts for such occasions.

Still, we do not intend to put everything on the computer. Even after the initial transition period, it remains a question of manual versus programming and running time. Some tasks are best left as they are.

We estimated the advantages that could be derived from inclusion of corroborating data into our database. It appears that now, and in the foreseeable future, the growth in database complexity would by far outweigh the advantages. It was decided, therefore, not to go beyond the data set specified by the Code 10 of the model Subsidiary Arrangements. Instead, in the course of processing, special printouts will be produced to facilitate the use of additional evidence.

At the facility level, the records are, with one exception, produced and updated manually. They consist of the ledgers prescribed by the SG Agreement, supplemented at smaller facilities by batchwise listings and at larger ones by card files on individual batches. In addition, these records are supplemented by data needed to satisfy the facility management's own needs. Introduction of EDP at this level is also envisaged.

Because the Institute is also a facility and MBA, it goes without saying that the keeping of their own internal MBA records will also be computerized.

At the NPP, EDP was introduced from the very beginning. However, the computer designated for the job was not installed in time, so they had to resort to various temporary arrangements. This decreased efficiency and led to errors, some of them puzzling. Thus, just in the printout of a report transmitted to the IAEA one line was missing while the copy retained at the plant contained this line.

Even apart from errors, the system is still in a development stage. True, at the NPP all documents requested by the Agency inspectorate are produced by EDP. However, up to now their system was not fully integrated internally. Their database was split in a way requiring repeated input of the same data. This has been overcome recently.

There remained the problem of data transmission to the Institute. We found the Institute's phone links to be unreliable. Once more, magnetic carriers seemed to be the best answer. This, of course, requires compatible disk systems. Instead of obtaining a PC/terminal of the same type as ours, the NPP opted for different ones. At present, the problem is close to solution. The SG team at the Institute now uses a second PC of a type to which the plant SG unit also has access. So there is going to be one more data transfer on mag-

netic disks--of a format different from the one adopted between our NAEC and the IAEA, with transcoding to be done at the Institute. In addition to all that, the NPP accounting unit now wishes to confer to the PC all its specifically safeguards-oriented data, using to the extent possible the software developed at the Institute (this is facilitated by us starting in BASIC).

As you see, it is all a bit complicated. I have reported it to show that running an economy/budget system requires additional ingenuity and workload for all concerned. There is an advantage; by the time the system settles down it is really adequate to the job at hand. Moreover, there is greater chance the people involved understand it--with the benefit that when further adaptations are made, they will be done intelligently and without fear of introducing hidden errors.

VII. BASIC ACCOUNTING

Sufficiency's enough for men of sense.

EURIPIDES: The Phoenician Women

We had the temerity to commit to a PC handling of data that usually are considered as meriting a mainframe computer. We had to consider very carefully what to put on machine and where to retain manual operations.

The hardware consists of two identical units, each comprising a PC with rather limited capabilities (8-bit processor, 20 KByte ROM, 40 KByte RAM, twin floppy disk unit with approximately 250 KByte 8" disks) plus a printer. One such unit would have been enough, but we wanted 100 per cent backup and the possibility to do some work of our own at the NAEC.

The data required by Code 10 (plus some more, if needed) for any single MBA in Hungary can be easily accommodated and processed with such a PC--except for the NPP, due to the growing inventory of spent fuel. However, at the NPP the nature of operations to be accounted for is limited, the inventory is very much uniform in character--and this permits us to introduce for this MBA special algorithms, should they become necessary. (Right now we can manage to handle this MBA as any other; later we may obtain PCs with larger computing power and fast external storage.) The essential idea would still remain the same: namely to keep the system self-contained to the extent possible. This is not regarded as a dogma; thus, some operations like, for example, sorting of data by batch numbers may be done for convenience at larger computer facilities.

The main functions of our programs--as they stand now--are:

- tabulated data input into temporary files and check of syntax;
- recoding the content of the temporary files and loading it into the data base;
- producing the ICRs, PILs and material balance reports (MBR) in accordance with Code 10 in labeled format on floppy disks;
- producing current book inventory listings;
- listing reports or parts of them from the data base;
- producing the general ledgers;
- handling of terminations and exemptions; keeping of nationwide accounts for these (as pseudo-MBAs);
- handling of corrections.

The structure of the system of programs may be seen on the figure, where:

REPIN : tabulated data input from domestic hard-copy format into
(transitory) feed storage,
SINT : checks the syntax of the input into the feed storage,
SEM : checks the semantics of the reports in the feed storage,
EDI : is a program for modifying reports in the feed storage,
LPR : produces reports from feed storage in IAEA hard-copy format,
TAGCNV : writes onto disks reports to the IAEA in labeled format,
DBCNV : transfers feed storage data to the database, recodes into DB format,
DBHCNV : transfers data back to feed storage (again, with recoding);
DBLIST : lists reports or parts of them from the database,
GL : produces the General Ledger,
BIL : produces book inventory listings,
MBR : produces material balance reports,
TFS : feed storage (on floppy disk),
DB : data base (on floppy disk).

The database is organized in sequential files. The data elements within records are of variable length, with separators between them. Such a format might cause serious breakdowns if used in a fully integrated EDP system, where many operations follow each other automatically. We hope to avoid them partly by introducing intermediate checks, but mostly by the step-by-step processing as depicted and under human supervision. In terms of machine time, this is going to work very much slower than full-fledged mainframe systems. We hope, however, to be faster in total clock and calendar time due to desk-top operation with immediate reaction.

Actually, the picture may be regarded as showing merely a skeleton system. We intend to add further features, beginning from the very moment the operations start.

As mentioned already, the software is written in BASIC. We are aware that this causes longer running times but we wanted, above all, an internal transparency of the system to all people concerned. We wanted independence, flexibility, the possibility of further development, and the "portability." We are willing to pay the price. Also, checks have shown that input quality very much improves if the rhythm dictated by the machine is not all too fast, that is, if the person at the keyboard feels that the computer needs about the same time to "swallow" the input as the one he or she is going to need to key in the next entry.

After a time, we may have some or more programs rewritten in machine code. But we intend to always have a complete and up-to-date software set in BASIC (or perhaps in PASCAL, or whatever of that kind) so as to augment understanding of the structure and working of our system.

The Institute dispatches processed data to the NAEC office. In addition to hard copy, this now includes disks with the reports to be sent to the IAEA. The reports are transcribed onto disks specially marked for the purpose and mailed. We have tested whether it is possible to send such disks by ordinary mail, if packed between aluminum sheets. Up to now, all test disks arrived in good condition. (However, this may be a very special case--just look at a map of Europe).

In addition to that, the information on the disks is used to update the NAEC's database. This should be identical to the one maintained at the Institute of Isotopes; periodic comparisons are provided to ensure that this remains so. The database is used by NAEC to extract and to process information for internal purposes.

All this also means that the database is at any time stored at two different locations. Together with the usual back-up copies, this should ensure preservation of the data under any conceivable conditions except those where the data would be of no use.

All software, including programs developed at the NAEC for its own use, will also be stored at both places, permitting it to perform all operations at either of the two sites.

VIII. INSPECTION

Detection is, or ought to be, an exact science.

Sir Arthur CONAN DOYLE: The sign of Four

The word "control" in the SSAC logo refers to a whole range of activities. Some of them are similar to those of the IAEA inspectorate, that is, consist of a verification of data recorded at the various NM locations and reported to national authorities. Others go beyond that--and there are also some differences of principle.

The IAEA is required to exercise a kind of impartial mistrust vis-a-vis all governments taken separately, so as to uphold the credibility of its international safeguards. Only in the case of international transfers does the Agency receive direct, independent confirmation of transactions reported by one of the parties.

A national authority is in a somewhat better position, because it can rely upon more information channels to indicate hidden activities violating the obligations taken upon itself by the state.

In addition, it may initiate direct institutional and ad personam sanctions against violators.

But, of course, these presuppose detection of such violations. Thus, the SSAC must participate in inspections conducted by the Agency and perform its own inspections, as necessary. At the beginning of this year, the frequency and intensity of domestic inspections was reappraised by the NAEC. It was decided to increase both. Not as if there had been any significant cases of violations, and not even as if there was a founded suspicion that the absence of "cases" merely signifies ignorance. It was simply determined that we should present a better image (to ourselves and to the outside world), and that this may even lead to additional benefits.

It was, for instance, established that the plutonium inventory that we are responsible for has been overestimated. Neutron source technology has been repeatedly improved, permitting to produce sources that contain less plutonium for the same neutron flux. For some sources, our book data indicate a plutonium element weight greater than the gross total weight of the source.

No false hopes, please. We cannot offer yet a method permitting good (and acceptable to the IAEA) nondestructive quantitative determination of the Pu content of such sources. This also implies that we cannot yet use such techniques, for example, for checking Pu-Be sources brought back from abroad.

We are in a better position with regard to "open," that is, bulk materials. However, their total quantities on hand are small. A lot of these are just stored, without being used in a way that would alter their quantity or batch composition.

The vast majority of nuclear material in Hungary is in the form of fresh or irradiated power reactor fuel assemblies. We receive them from abroad and for all practical purposes intend to ship them abroad again as soon as possi-

ble after use in reactors. If we can be reasonably sure that none of these assemblies has been dismantled and that they all are at any given point in time located where they are supposed to be--then we have done a major step towards discharging our obligations. We call this the problem of positive identification.

The Hungarian NAEC, research institutions, and industry have sponsored development of an underwater telescope designed to be used precisely for identification of nuclear fuel assemblies (and other items) stored under water. This permitted us to create order in spent fuel storages. (Before that, we were not aware of how many spent reactor fuel assemblies were actually misplaced; eventually we found them all, but many were not at the precise locations where they were supposedly stored.) Agency inspectors became addicted to the use of these instruments.

As to going further in measuring techniques we do not intend to introduce NDA for our own SG purposes, except under special circumstances, for example, such as accidental destruction of one or more fuel assemblies. However, we have provided the possibility for the installation of gamma-spectrometers and the like at the NPP. Recently, the operator of the NPP has become interested in performing such measurements in order to optimize fuel utilization. We shall participate, since this might provide us with data and experience in this field, to be put to use, should the need arise. At the Training Reactor, work is going on with the aim to develop both transmission (for fresh fuel) and emission (for spent fuel) gamma-tomography to check the internals of fuel assemblies. Although this might be of large interest for states having reactors with fuel assemblies that can be dismantled, we intend to support this work further as a contribution of our country to the overall development of safeguards.

IX. INTERNATIONAL RELATIONS

We shall be judged more by what we do at home
than by what we preach abroad.

John F. KENNEDY: State of the Union; 14.I.1963

First, our relations with the IAEA. We regard them as good and hope that the Agency is of the same opinion. Of course, there is always some ambivalence in our relations; a Hungarian member of the Board of Governors participates in the making of decisions, which the Director General and his Secretariat are supposed to execute. On the other hand, Agency inspectors have considerable rights and privileges when visiting Hungarian facilities. We tried not to cause problems at either of the levels.

I have already mentioned some facets of IAEA inspections. There are some more.

After a time, inspection routines settle down. Still, there are gradual changes in approach--and special occasions also occur. Inspectors may want some special operation to be performed; they may pose some other unusual request. For such cases, our instructions to NAEC and facility staff are: whatever is foreseen in the relevant Facility Attachment should be performed or provided; if not in--use sound judgment. If it is not dangerous and does not cause substantial difficulties and costs, have it done. But make it clear that this is a favor to the IAEA inspectorate and should not be construed as establishing law by precedent--even when some such additional operations are performed repeatedly.

As to the inspectorate: we did not impose restrictions with regard to the designation of inspectors to Hungary (for example, by nationality), considering them to be deleterious to the cause of SG and nonproliferation. There were no attempts to designate to Hungary inspectors from states with which my country does not have diplomatic relations.

Over the years, the number of inspectors designated to Hungary has grown. Some of them have never yet been there (at least, not in their official capacity); some have not visited us for many years. We maintain a current list of inspectors designated to Hungary; all relevant institutions and facilities have copies of it. In order to be able to act fast if an Agency inspection were to be scheduled at short notice, the list contains the data usually needed for entry passes (nationality, birth data, numbers of national passport and Laissez-Passer). This list is updated whenever necessary and a copy sent to the Agency as well, to invite correction of any errors or obsolete data.

The example just mentioned stands for other matters where we have set up and maintained with the Agency running, working contacts, without keeping this very formal. It extends to cases of mutual notifications, provided for in the SG Agreement and its subsidiary documents. Sometimes we would be supposed to send to Vienna an advance notification concerning a specific event--but not all data are yet available. At other times, the Agency must at short notice reschedule an inspection. In both cases, we apply the principle: get information through to the partner by the shortest route possible (and compatible with security, if that is involved), even if it is incomplete. Pass on additional data as soon as available and dispatch an official notification when all data elements are assembled together.

This attitude extends also to other communications. Authorities originating information handled by us or otherwise coresponsible for it know the extent of our duties and to not impose formal restrictions or other hindrances that would add to the workload without improving data protection.

The channels and modes of data transmission used by us must, of course, suit the type of data. This ranges from diplomatic pouch to ordinary mail. Some communications may be subject to specific rules. Thus, for example, brief notifications to the Agency, which may be due in cases of nuclear accidents, would be transmitted by telex but using appropriate code.

The NAEC enjoys autonomy in matters relating to negotiations with the Agency. Whenever the prerogatives of other authorities or commercial interests of facility operators are involved, the matters are cleared with those concerned; after that, the negotiating teams and persons receive their authorization from the NAEC.

In my experience, IAEA Secretariat staff are tough partners in negotiations, but I also think that this is good; the result is usually an arrangement that neither side has cause to regret later. In cases when action is due but formal negotiations take time, we have more than once agreed to start on an "ad hoc" basis, postponing legal confirmation and in the meantime acquiring experience on how the matters work out in reality.

We have found that other states with whom we get into contact on safeguards matters usually (not always, though) also show a predisposition towards a businesslike way of maintaining bilateral and multilateral relations. We see this particularly in matters concerning international transfers of NM. With a number of states we have working arrangements (to a large extent formalized) to notify each other whenever material is transferred from one state involved to the other. Receipt of such notifications is confirmed. This simplifies and shortens the preparation of reports on such events to the IAEA and considerably reduces the number of unconfirmed transactions.

Sometimes we encountered in this field purely practical difficulties. We would welcome the periodic publication of an updated list containing some essential data on the national authorities of various states responsible for safeguards. To be really useful, it should provide the names and ranks of officials to be contacted on matters of various levels of importance--with addresses, phone and telex numbers. In this respect, an example to follow has been set by the Convention on the Physical Protection of Nuclear Materials.

X. INSIDER EXPERIENCE

Experience is the name everyone gives to their mistakes.

Oscar WILDE: Lady Windermere's Fan

The tasks to be performed by an SSAC are by themselves often difficult. And nevertheless, there is always the danger of making them more difficult than strictly necessary. Overdoing not only adds to the workload, it can cause frictions in relations with others--first of all, with operators.

Surely, we in Hungary were not the first to find out that an SSAC has to create and preserve a kind of image that would help its work. This is especially important in cases when you wish or have to keep your staff small. A good image affects not only the relations with the outside world, but also your own staff. Ambitious people may be prone to some showmanship or activities that are not strictly necessary. Excellent R+D work has been done in the field of safeguards that has no real chance of being put to direct use. Cutting that down may save money, but could lose competent people.

As to the image we tried to create of our SSAC in general, this was as follows. We tried to explain the why's and how's of IAEA safeguards to all those concerned; in addition to improving the general level of understanding of all these matters, it had the somewhat paradoxical result that many started to believe there is much more behind what we are giving out.

Another aspect was that we did not shirk away from accepting responsibility--including the one for inevitable mistakes. Also, we never made great fuss about errors in NM handling detected by us. However, on one or two occasions, when we established that efforts were made to cover up errors, we took appropriate action. Even then, we avoided harsh threats because, after all, we had not in all these years had a single case when we had reason to suspect that someone tried to sabotage our work, let alone to divert nuclear material.

Turning now to a less philosophical topic, we had to make some special efforts to cut down superfluous work. I mentioned before that we started by creating a very tight and rigid system, hoping that time and experience would show us where we might and should relax regulations. The first part worked out well. As to the second, at the end of the seventies, we started to look for possibilities to get rid of work that was not really necessary. One of the first things we noticed was a disproportionately large number of very small batches of NM.

When we started, we made people report everything. And we were cautious in asking the IAEA for exemption of nuclear material from safeguards. Over the years, we mapped the pathways of NM, so we were able to request not only exemption of small quantities under the quantity limit, but also exemption and even termination by virtue of non-nuclear use. Many batches were disposed of entirely. In grams and kilograms, all this did not amount to much, but it considerably reduced the total paperwork associated with the R+D institution MBAs.

It took three years, not merely due to the "measure twice before cutting" approach, but also because some people thought they might become less important by not having to account for nuclear material.

The general rule was: try to get rid of all batches that are smaller than half the "accounting unit"--gram or kilogram--and also of batches comprised of a moderate number of items in this magnitude range. At the beginning, we thought of reducing the number of decimals to be reported. (Reactor fuel weight data are reported and recorded down to tenths of grams. Other NM may vary, but even MBRs are prepared down to milligrams and grams for special fissionable and source material, respectively). Later, we realized that cutting decimals would not offer substantial savings; instead, we would have to calculate and report rounding adjustments. The only exception: at the NPP MBA all materials are reported in tenths of grams and kilograms. There is practically no NM flow between this MBA and the rest, so it does not afflict the others.

Together with other states, we tried to set up some lower cut-off level for the reporting of international transfers when the quantities are very small and/or when the materials crossing the border are not subject to IAEA safeguards in the country of origin, for example, having been exempted from safeguards.

A lot of goodwill has been shown by everyone, including the IAEA, but no general solution has been devised yet.

Still another matter: my estimation of the tasks to be performed by the SSAC seems to confirm that for a small state it is quite difficult to find good people for the job. There might not be enough work for a full-time bookkeeper, neither for a specialist in particular inspection work. At another time the requirement is for a person able to participate in a discussion about safeguards to be applied at some facility that is still in the planning stage. Even then, the SSAC representative should be able to participate intelligently in the drafting of safeguards procedures for it.

So you need versatile people. But a large part of the job will be repetitive and boring--something that is usually disliked just by the kind of people you need. So we arrive back at what was said in the beginning of this chapter. It is difficult to put together a good team, even if you are not hamstrung by overzealous and narrow-minded bureaucrats (as you see them) who try to cut down on governmental expenditure (which is actually their job), and still try to preserve a sensible level of redundancy and try to get good equipment.

Again, in our experience, aside from being the most expensive, the best is all too often not good enough. It is usually devised to be pampered and protected, just like the proverbial "modern" safety pin that has to contain three precision ball bearings and two ICSSs, and must be submitted twice a year for recalibration. To illustrate what I mean, I will bring just one example. The PCs we bought were not specially selected. We just took what was immediately available at a reasonable price. But since then, we have had opportunity to find out that not only versatility, speed, and memory size count, but reliability and ruggedness. To mention just one aspect, this last winter was a pretty cold one with power shortages and breakdowns in many European countries. Hungary was among the luckier ones, but every now and then some emergency switching occurred with accompanying power glitches and surges. Now it appeared that some equipment in the lowest price range remained almost unaffected, while others were extremely sensitive, losing or garbling programs and data in memory. Unfortunately, information about sensitivity of particular brands and types of computers and PCs to brief power failures is not easy to obtain, except, perhaps, if you specifically ask the manufacturers about it.

XI. CONCLUSION

Unhappy the general who comes on the field of
battle with a system.

NAPOLEON: Maxims

I have tried to present a picture of the structure and functioning of the SSAC in a European state with a population somewhat below 10 million. Activities in the nuclear field do not extend to the nuclear fuel cycle outside reactors--except for small-scale work in terms of nuclear material quantities. Accordingly, our NAEC has up to now devoted to the domestic inspection aspect less attention than we now consider desirable. We were not doing badly, but we do strive to improve our system. We keep learning.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 6: BRIEF PRESENTATIONS BY COURSE ATTENDEES

The course participants will be asked to make brief presentations on any subject of interest, such as nuclear technology and SSAC activity in their home countries. This is an opportunity to become better acquainted with their fellow participants and learn something about nuclear activities in the nations participating in this course. We hope to have one or more presentations from every country represented.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 7: WORKSHOP ON IAEA/STATE SYSTEM INTERFACE

Participants and panel members will be divided into five subgroups. Each subgroup will develop a list of questions for the panel. Then, following a short break, the panel will be convened under the session chairman to respond to subgroup questions, as well as questions from the floor. Discussion will focus on safeguards interactions between the IAEA, state authorities, and facility operators. Full participation by course attendees is encouraged.



Questions for panel were first developed in small subgroup discussions.



Ben Agu, IAEA, moderated the panel, which included: Left to Right- Carlos Büchler, IAEA; Bob Keepin, LANL; Ben Agu; Otto Lendvai, NAEC; and Mike Smith, NRC.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 8: ELEMENTS OF NONDESTRUCTIVE ASSAY TECHNOLOGY

This session will provide an introduction to nondestructive assay methods and instruments as they are applied to nuclear safeguards.

After the session, participants will be able to:

- Discuss the general principles and major applications of NDA.
- Describe situations in which NDA is particularly useful for nuclear safeguards purposes.
- Distinguish between various passive and active gamma-ray and neutron NDA methods.
- Describe several NDA instruments that measure gamma rays, and identify assay situations particularly suited to gamma-ray techniques.
- Describe several NDA instruments that measure neutrons, and identify assay situations particularly suited to neutron techniques.
- Compare the advantages and disadvantages of various NDA methods for different types of nuclear materials.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 8: ELEMENTS OF NONDESTRUCTIVE ASSAY (NDA) TECHNOLOGY

C. R. Hatcher and H. Smith
Los Alamos National Laboratory

I. INTRODUCTION

A. History of NDA

When we speak of NDA in the context of nuclear safeguards, we are generally referring to the use of nuclear radiation to measure the quantity of fissionable material present in a given sample or container. NDA techniques have been applied to special nuclear materials in the U.S., at least to some extent, since the initiation of the Manhattan project. But it was not until 1967, when the Office of Safeguards and Material Management was established by the AEC, that effort was focused on the development of NDA techniques specifically designed for nuclear safeguards. The Los Alamos National Laboratory has been involved in this development work since its inception and is currently the DOE lead laboratory for nuclear materials measurement and accounting. Other laboratories in the U.S., including Brookhaven, Hanford, Argonne, Livermore, Idaho, Oak Ridge, Savannah River, New Brunswick, Mound, and NBS, have also contributed, as have instrument companies such as Eberline, Canberra, National Nuclear, and IRT. Significant advances in NDA technology also continue to be made by organizations in Europe, as well as in other countries such as the USSR, Canada, and Japan.

B. Uses of NDA

NDA methods are widely used throughout the nuclear fuel cycle primarily because they are able to measure the material in its existing form, and they provide rapid assay results. Applications include:

- ore location and assay
- process control
- quality control
- health and safety
- criticality
- material accounting
- containment
- waste disposal

C. Strengths and Limitations of NDA

During the past few years, NDA techniques have become established as a fundamental element of nuclear safeguards programs throughout the world. NDA is particularly well suited for safeguards applications when:

- It is difficult to obtain a representative sample for chemical analysis; for example, consider non-uniform solids, material in sealed containers, spatially distributed material, or valuable finished products.
- There is a need for many repetitive measurements, such as might be the case for receiving stations, process lines, or waste streams.
- There is a need for timely material accountability; i.e., it is desirable to close material balances in a matter of hours rather than days or weeks.

The arguments in favor of NDA are compelling, but certain limitations and consequences must also be noted:

- Adequate NDA standards are frequently not available in the particular geometry, material form, and isotopic composition that would be ideal for instrument calibration. Consequently, variations in these factors may complicate instrument calibration and data interpretation and, in many cases, may limit the accuracy of the assay result.
- Absolute calibration of NDA instruments usually depends in the final analysis on chemical methods, such as gravimetric analysis, fixed stoichiometry, titration, and mass spectrometry.
- The large quantity of data made practical through NDA can be fully utilized only if adequate computer methods are used for assisting with data analysis and interpretation.

D. Classifications of NDA Methods

Figure 1 shows, in block diagram form, a classification of NDA techniques into passive and active gamma-ray and neutron methods and calorimetry. According to Fig. 1, passive NDA methods include all techniques that derive their primary information from the natural radioactive decay of the sample, whereas, active NDA methods include techniques that derive their primary information from the interaction of an external radiation source with the sample. Similar classifications have been made by Dragnev^(1,2) and by Smith and Canada.⁽³⁾

In Fig. 1, applications are listed for each of the NDA techniques shown, along with one instrument (in parenthesis) that makes use of the principle. The information presented in Fig. 1 covers most of the widely used NDA methods, but for the sake of brevity, is not totally inclusive. In following sections, each of the techniques shown in Fig. 1 will be discussed, and the advantages and disadvantages of various NDA methods will be compared for different types of nuclear materials.

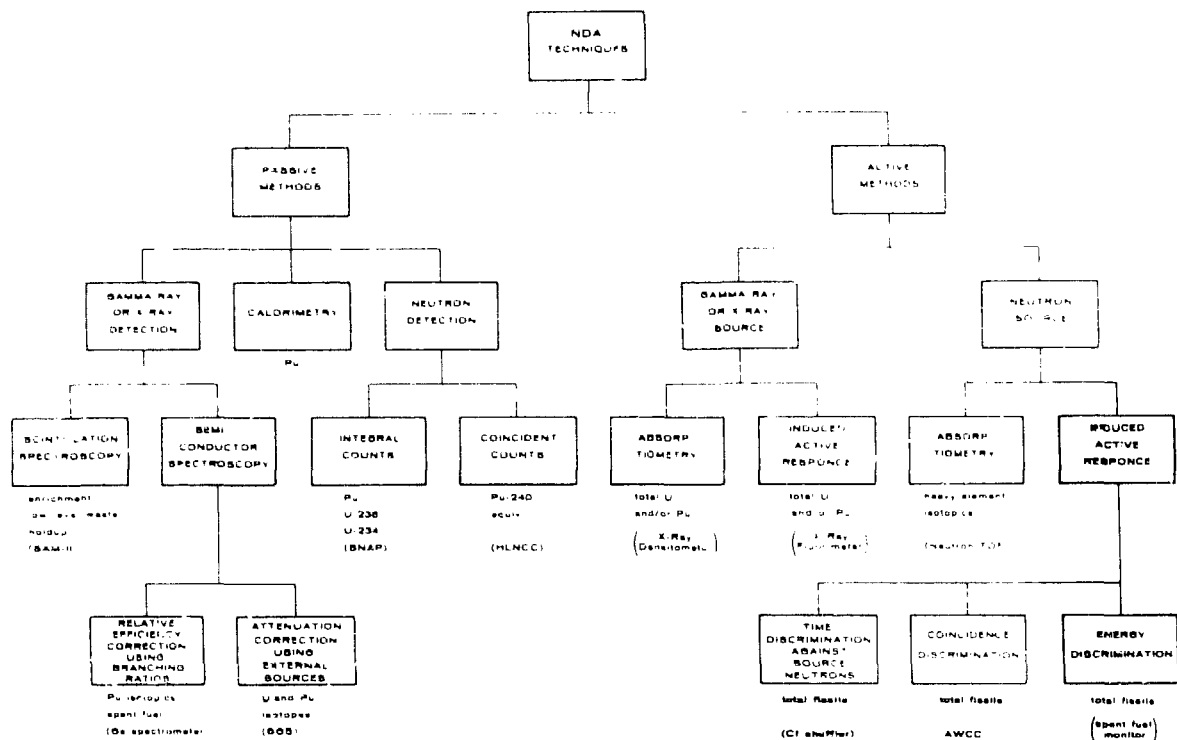


Fig. 1
Classification of nondestructive assay (NDA) techniques
used for the measurement of fissionable materials.

II. PASSIVE GAMMA-RAY METHODS

A. General Information

All isotopes of uranium and plutonium are radioactive and decay by alpha emission, beta emission, or spontaneous fission. Following either alpha or beta emission, the nucleus is sometimes left in an excited energy state, which then decays by the emission of gamma rays to the ground state. Each isotope has a unique decay scheme, and when gamma rays are detected, a determination of gamma-ray energies provides a way of identifying the specific isotopes present.

Table I lists some of the gamma-ray energies and emission rates for uranium, plutonium and americium isotopes that are commonly used for the NDA of these materials.⁽⁴⁾ A more complete list of gamma-ray energies and other properties of heavy element nuclides can be found in reference 1.

Fig. 2
Arrangement for
measuring ^{235}U
enrichment of
 UO_2 powder in
cans. The
Eberline SAM-II
electronics unit
is used with a
NaI-photomultiplier
detector mounted
vertically under
the sample can.

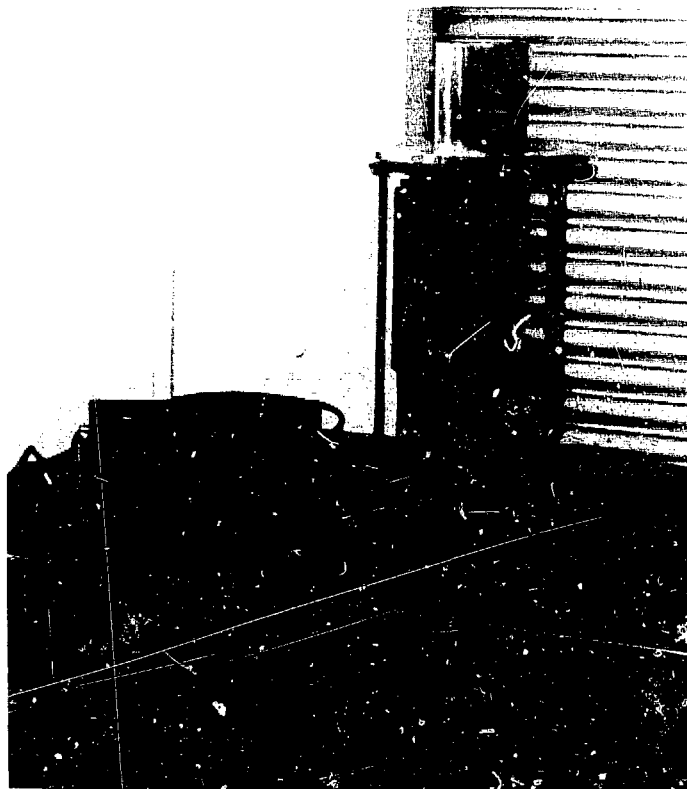


Fig. 3
Characteristic
pulse height
spectrum obtained
with a NaI
detector viewing a
 ^{235}U sample.
The curve was
obtained with a
SAM-II by varying
one of the single
channel analyzers.

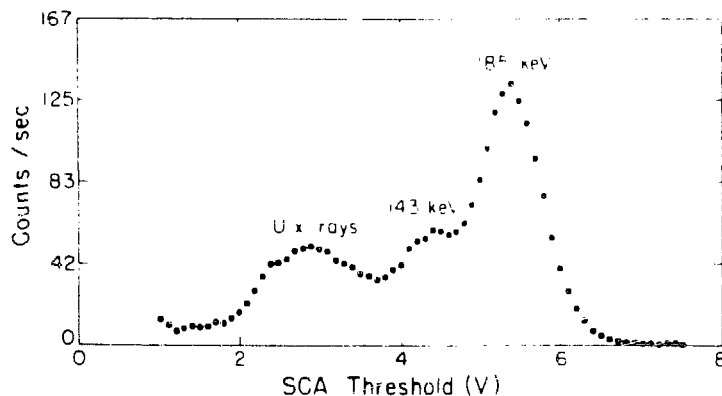


TABLE I
URANIUM, PLUTONIUM, AND AMERICIUM
GAMMA-RAY EMISSION RATES

Isotope	Half-life (years)	Activity Level (Ci/g)	Principal Gamma Rays (keV)	Emission Rate (gamma rays/s/g)
^{235}U	7.04×10^8		185.7	4.5×10^4
^{238}U	4.47×10^9		766.4	39
			1000.1	103
^{238}Pu	87.79	17.1	152.8	6.4×10^6
			766.4	1.5×10^5
			1000.1	8.2×10^3
^{239}Pu	24082	0.0621	129.3	1.4×10^5
			375.0	3.6×10^4
			413.7	3.5×10^4
^{240}Pu	6537	0.228	160.4	3.5×10^4
			642.3	1.2×10^3
^{241}Pu	14.35	103.4	148.6	7.3×10^6
^{242}Pu	3.79×10^5	0.0039	none	
^{241}Am	434.1	3.42	368.6	2.6×10^5
			662.4	4.4×10^5

B. Scintillation Spectroscopy

1. Enrichment Meter. Figure 2 shows the Eberline SAM-II, one of the simplest and most widely used instruments in nuclear safeguards. It consists of a NaI detector (typically 1.27 cm thick by 3.81 cm diameter), a two channel analyzer, and an up-down scaler. A typical pulse height spectrum obtained with a uranium sample using the SAM-II is shown in Fig. 3.

The SAM-II and similar types of instruments built by other manufacturers are capable of accurately measuring uranium enrichment. For samples that are thick relative to the penetrating depth of the 186 keV U-235 gamma ray and for fixed detector-sample geometry, the count rate due to 186 keV gamma rays is proportional to enrichment; this linear relationship between enrichment and count rate is referred to as the enrichment meter principle. Calibration of enrichment meters is accomplished using two or more enrichment standards having container walls similar or preferably identical to those of the unknown samples.⁽⁵⁾

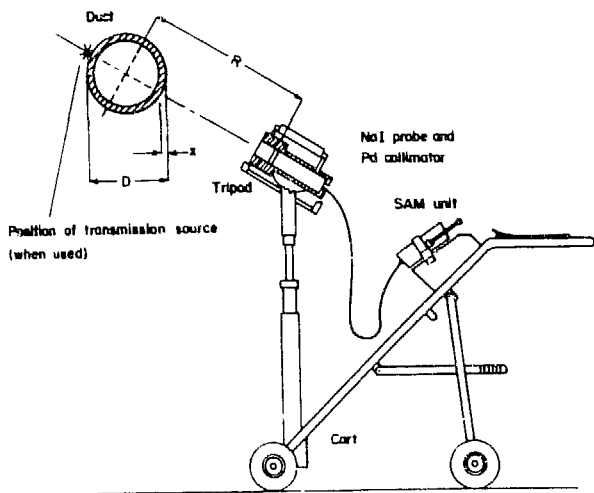


Fig. 4

Holdup of nuclear material in process lines can be measured using portable scintillation detection equipment.



Fig. 5

Scintillation detectors are used in portal monitors and in hand-held survey meters. The portable instrument shown was developed by Los Alamos Group Q-2 and manufactured by National Nuclear Corporation.

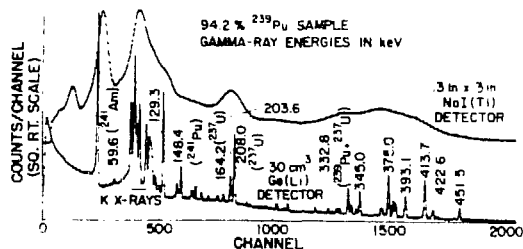


Fig. 6

Illustration of plutonium gamma-ray spectra as measured with a NaI detector (upper curve) and a Ge(Li) detector (lower curve), showing the capability of high resolution Ge spectroscopy to determine energies and relative intensities of individual gamma lines in complex spectra.

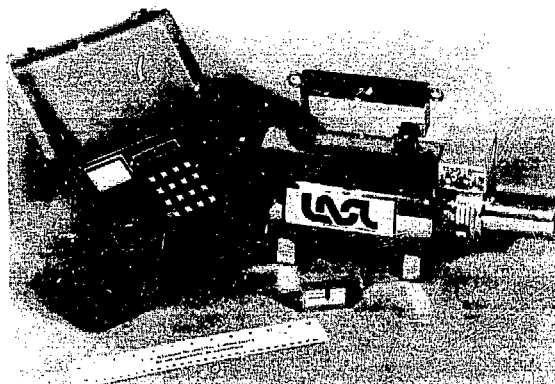


Fig. 7

The battery operated 1000 channel pulse height analyzer developed in H-Division at Los Alamos has been adapted for use by the IAEA. Small intrinsic Ge detectors, such as this 30 cm³ unit from Princeton Gamma Tech can be operated in any position.

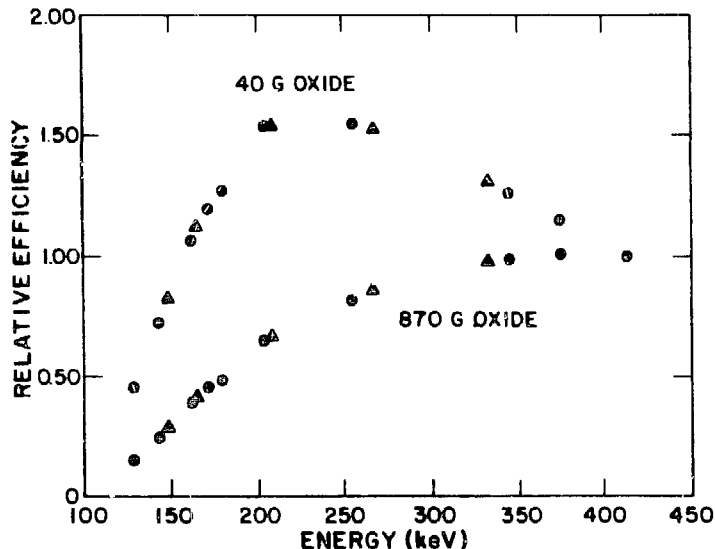
2. Other Applications. Although the enrichment meter is perhaps the most well known, gamma-ray instruments using sodium iodide and other scintillation detectors have been used for many other applications in safeguards, such as assaying low level waste, monitoring effluents, and estimating the holdup of nuclear material in processing plants (Fig. 4). Because they are portable, simple, and reliable, instruments employing scintillation detectors are also used as portal monitors and survey meters (Fig. 5).

C. Semiconductor Spectroscopy

1. Methods Using Relative Efficiency Correction. High resolution gamma-ray spectroscopy (HRGS) using semiconductor detectors, such as intrinsic Ge and Ge(Li), provides significantly better energy resolution than can be achieved using scintillator detectors, as is demonstrated by the two plutonium gamma-ray spectra shown in Fig. 6. Energy resolution (FWHM) for high quality coaxial Ge detectors is on the order of 0.8 keV for 122 keV gamma-rays and 1.7 keV for 1.33 MeV gamma rays. In the last few years, portable HRGS systems have become available through the advent of intrinsic Ge detectors (that can be transported at room temperature) and smaller multichannel analyzers (Fig. 7).

If one measures the photopeak areas of gamma rays from different isotopes, it may be possible to determine isotopic ratios using a technique known as the relative efficiency correction. The relative efficiency correction factor (Fig. 8) includes effects due to attenuation in the sample, attenuation in external absorbers, and the detector sensitivity, all of which vary as a function of gamma-energy and measurement geometry.

Fig. 8
Relative efficiency curves taken with a 200 mm² x 10 mm-deep planar Ge detector, showing dependence on gamma-ray absorption in the sample. Circles are points from ²³⁹Pu; triangles are from ²⁴¹Pu and ²⁴¹Pu-²³⁷U. The circular points and triangular points for each sample are normalized at 332 keV.



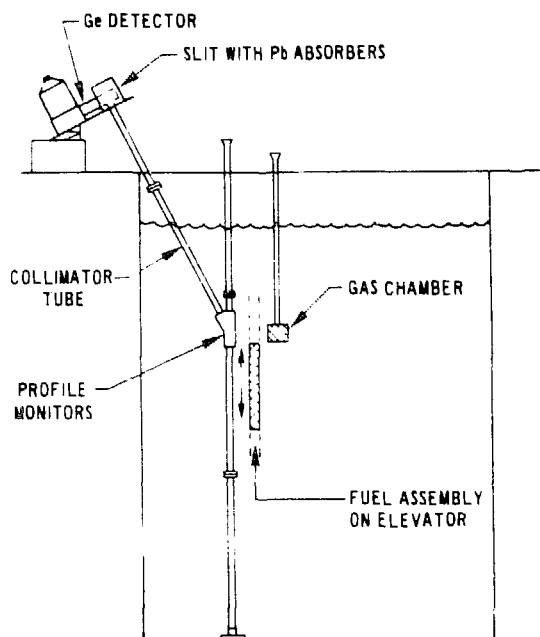


Fig. 9

Arrangement of Ge detector and slant collimator pipe for observing fission product gamma rays from spent-fuel assemblies stored underwater. Ion chambers and fission chambers are placed in the vertical pipes.

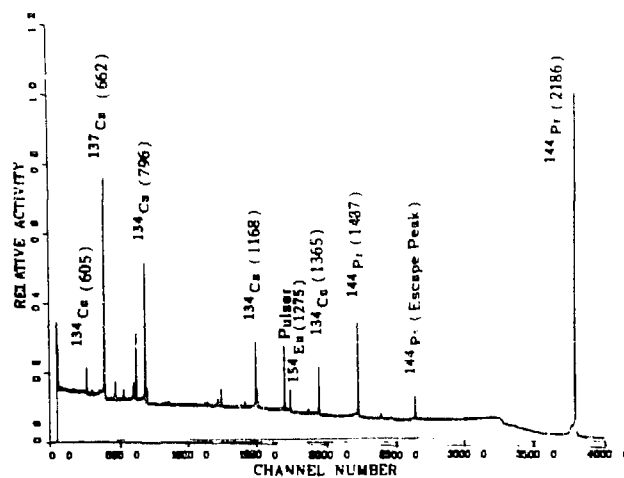


Fig. 10

Characteristic gamma-ray spectrum observed with spent fuel. From fission product isotopic ratios, it is possible to verify cooling time and burnup.

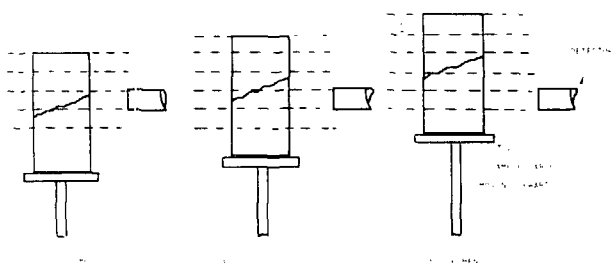


Fig. 11

In the segmented-gamma-scanner, the sample container is stepped vertically past the fixed Ge detector and transmission source. A separate gamma-ray absorption correction is made for each vertical segment.



Fig. 12

Segmented-gamma-scanner systems suitable for assaying a variety of container sizes were developed at Los Alamos and are now commercially available. The transmission source is to the right of the barrel, and the Ge detector is to the left of the barrel.

Isotopic ratios are calculated from the equation

$$\frac{I_1}{I_2} = \frac{A_1}{A_2} \frac{(BR)_2 \epsilon_2}{(BR)_1 \epsilon_1} \frac{\lambda_2}{\lambda_1},$$

where

I_1/I_2 = isotopic ratio

A_1/A_2 = ratio of photo peak areas

BR_n = branching ratio for particular gamma ray from isotope n

λ_n = decay constant of isotope n

ϵ_1/ϵ_2 = relative efficiency correction factor for two gamma rays used in calculation.

The relative efficiency correction factor can be experimentally determined from the above equation by using gamma rays of different energies from the same isotope, and setting $I_1/I_2 = 1$. This technique has been successfully applied for measuring Pu isotopic ratios and spent-fuel fission product isotopic ratios⁽⁶⁾ (see Figs. 6, 9, and 10). In these applications, a single gamma-ray spectrum contains the peak area data for both the relative efficiency correction and the determination of isotopic ratios.

The relative efficiency correction method works well when:

- a. isotopes are uniformly distributed throughout the sample,
- b. at least one isotope in the sample has two or more prominent gamma rays in the appropriate energy range, and
- c. the objective is to determine isotopic ratios in a sample, as opposed to the total amount of an isotope present in a sample.

2. Methods Using External Sources to Correct for Sample

Attenuation. To deal with situations in which the three conditions listed in the previous paragraph are not met, techniques have been developed to correct for gamma-ray absorption in the sample using external radiation sources. One such instrument, the segmented-gamma-scanner (SGS),⁽⁷⁾ was designed for assaying uranium and/or plutonium waste in a variety of container sizes and matrices (see Figs. 11 and 12). The idea is to divide the sample into a series of horizontal segments and to assay each segment, one at a time, with a self-absorption correction separately determined for each segment. For assaying ^{235}U using the 186 keV gamma ray, the SGS uses a ^{168}Yb external radiation source with gamma rays at 177 and 198 keV.

Other instruments that use external radiation sources to correct passive gamma assays for absorption in the sample include the solution assay systems, PUSAS and USAS.⁽⁸⁾ All of the instruments

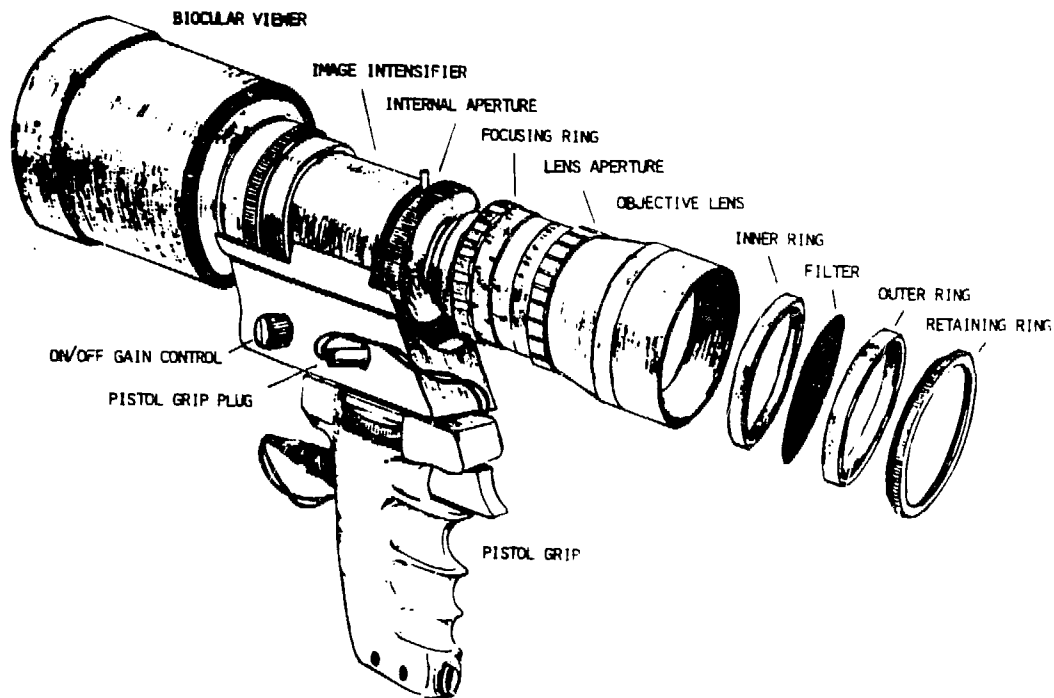


Fig. 13

The hand-held Cerenkov detector developed by Los Alamos Group Q-2 for the IAEA is used to observe light produced by irradiated fuel assemblies stored underwater. Later models permit the attachment of a film camera, so that the inspector has a permanent record.

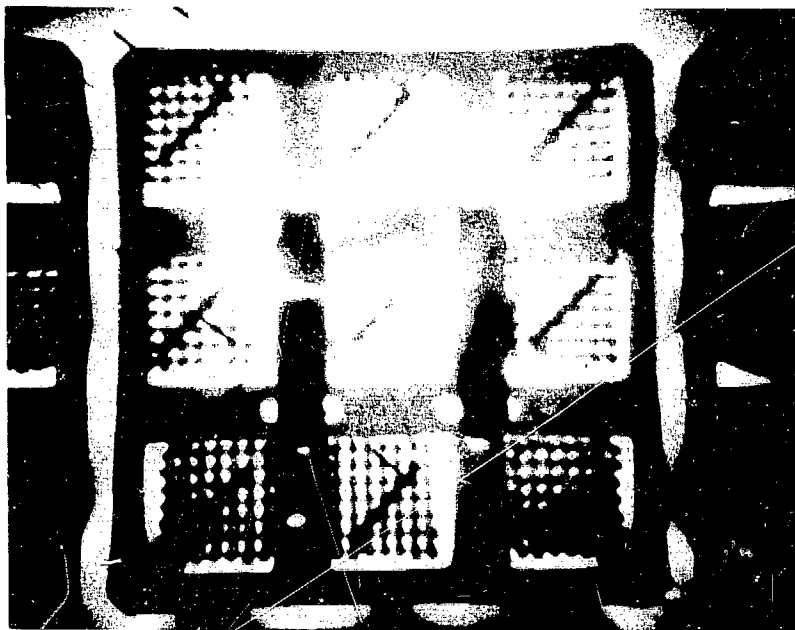


Fig. 14

Image recorded by the Cerenkov detector when observing BWR fuel assemblies, looking down into a storage pond from a bridge crane.

in this category give isotopic information. In size and complexity, they are comparable to some of the active interrogation instruments such as x-ray densitometers and active well coincidence counters.

D. Gross Gamma-Ray Techniques

A few instruments used in safeguards applications measure gross gamma-ray fields (or dose rates), rather than counting individual gamma rays. The Cerenkov detector, shown in Fig. 13, is used to obtain an image from light produced by spent-fuel assemblies stored underwater. The detector consists of a telephoto lens coupled to an image intensifier tube that amplifies the Cerenkov light intensity, so that it can be easily seen in a darkened fuel storage pond (Fig. 14). This instrument permits IAEA inspectors to verify that spent-fuel assemblies are intact and are highly radioactive, without placing any instrumentation underwater.⁽⁹⁾

Ion chambers have also been used to measure the gamma-ray fields produced by spent-fuel assemblies. In particular, ion chambers are used to determine axial gamma-ray activity profiles, which closely resemble the burnup profiles of spent-fuel assemblies.

III. PASSIVE NEUTRON METHODS

A. General Information

Neutrons originate in special nuclear materials primarily because of spontaneous fission and (α, n) reactions, see Tables II, III, and IV.⁽¹¹⁾ Passive neutron measurements can be influenced by neutron multiplication in the sample and by the presence of neutron moderators, reflectors, and absorbers in or near the sample. Compared with gamma rays, neutrons are much more penetrating in high-Z materials, and it is this characteristic that makes passive neutron techniques invaluable for assaying large heterogeneous samples of plutonium.

Whereas gamma-ray energies allow one to identify isotopic content, passive neutron energies contain no isotopic information. As a result, neutron assays involve counting rather than spectroscopy, and for this reason, passive neutron hardware is usually simpler than high resolution gamma-ray hardware. The most commonly used neutron detector for NDA instrumentation is the gas proportional counter, typically ^3He or $^{10}\text{BF}_3$. This type of detector is chosen because of its relatively high efficiency for detecting thermal neutrons, insensitivity to gamma rays, reliability, and long-term stability.

TABLE II

SPONTANEOUS FISSION OF FISSIONABLE ISOTOPES

Isotope	Half-Life (yr)	Spontaneous Fission Half-Life (yr)	$\bar{\nu}$ (Spontaneous Fission)	Spontaneous Fissions per g-s
^{232}Th	1.41×10^{10}	$\sim 10^{21}$		$\sim 10^{-8}$
^{234}U	2.47×10^5	2.0×10^{16}	~ 2	2.8×10^{-3}
^{235}U	7.04×10^8	1.9×10^{17}	~ 2	2.96×10^{-4}
^{236}U	2.40×10^7	2.0×10^{16}	~ 2	2.8×10^{-3}
^{238}U	4.47×10^9	9.86×10^{15}	1.95	5.64×10^{-3}
^{238}Pu	87.79	4.7×10^{10}	2.26	1.1×10^3
^{239}Pu	2.41×10^4	5.5×10^{15}	2.2	1.0×10^{-2}
^{240}Pu	6537	1.17×10^{11}	2.17	4.71×10^2
^{241}Pu	14.35	5.0×10^{15}	2.2	1.1×10^{-2}
^{242}Pu	3.79×10^5	6.8×10^{10}	2.16	8.0×10^2
^{241}Am	434.1	2.0×10^{14}	2.3	0.27
^{242}Cm	.4456	7.2×10^6	2.65	7.8×10^6
^{244}Cm	17.6	1.4×10^7	2.84	4×10^6
^{252}Cf	2.646	86	3.8	6.14×10^{11}

TABLE III

(v,n) YIELDS FROM OXIDES AND FLUORIDES

Material	Yield (neutrons/s/g)
$^{234}\text{UO}_2$	~ 14
$^{234}\text{UF}_6$	5.8×10^2
$^{235}\text{UF}_6$	12.2×10^{-2}
$^{238}\text{UF}_6$	12.9×10^{-3}
$^{238}\text{PuO}_2$	1.4×10^4
$^{238}\text{PuF}_4$	2.1×10^5
$^{239}\text{PuO}_2$	45
$^{239}\text{PuF}_4$	4300
$^{240}\text{PuO}_2$	170
$^{240}\text{PuF}_4$	1.6×10^4
$^{241}\text{PuO}_2$	10
$^{241}\text{AmO}_2$	3754
$^{242}\text{PuO}_2$	10

TABLE IV
NEUTRON EMISSION RATES FOR Pu METAL, PuO₂, AND PuF₄

Isotope	Wt%	Neutron Rate for 100 g of Pu (n/s)		
		Metal (Spontaneous Fission)	PuO ₂ (α, n)	PuF ₄ (α, n)
²³⁸ Pu	0.3	746	4 200	630 000
²³⁹ Pu	75.6	2	3 400	325 080
²⁴⁰ Pu	18.0	18 400	3 060	288 000
²⁴¹ Pu	5.0	0	50	5 000
²⁴² Pu	<u>1.1</u>	<u>1 900</u>	<u>10</u>	<u>1 000</u>
	Total	21 048	10 720	1 249 080

B. Passive Methods That Detect Single Neutron Counts (SNAP)

The Shielded Neutron Assay Probe (SNAP), shown in Figs. 15 and 16, can be used for assaying total plutonium when the chemical and isotopic composition are known, and suitable standards are available.⁽¹²⁾ In assaying plutonium, neutrons from both the spontaneous fission of even Pu isotopes and (α, n) reactions in the sample are measured. The SNAP detector consists of two ³He proportional tubes in a cylindrical polyethylene moderator encased in a Cd shield. To achieve directionality and reduce background, a 240° polyethylene shield can be placed around the inner cylinder.

Neutron counting has been applied to the assay of spent LWR fuel assemblies using ²³⁵U fission chamber detectors. Neutrons originate primarily from isotopes of curium and plutonium and can be correlated with burnup via a power law relationship.⁽¹⁰⁾

More specialized applications of singles neutron counting include the measurement of ²³⁴U enrichment (by observing α, n neutrons in UF₆), and the assay of total uranium in low enriched scrap (based on the spontaneous fission of ²³⁸U). These applications are considered special because, in general, passive neutron signals from uranium are too weak to give reliable assays.

C. Passive Methods That Detect Coincident Neutron Counts (HLNCC)

The purpose of using coincidence counting for passive neutrons is to discriminate against single (α, n) neutrons, while detecting coincident neutrons due to spontaneous fission. For assaying plutonium, coincidence counting is generally better than singles counting because coincidence methods are less sensitive to variations in low Z matrix materials and less sensitive to changes in neutron background.

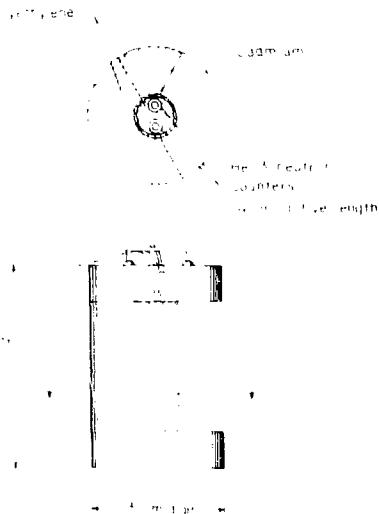


Fig. 15
Assembly drawing of the Shielded Neutron Assay Probe (SNAP), showing two ^3He proportional tubes inside a cadmium shield.

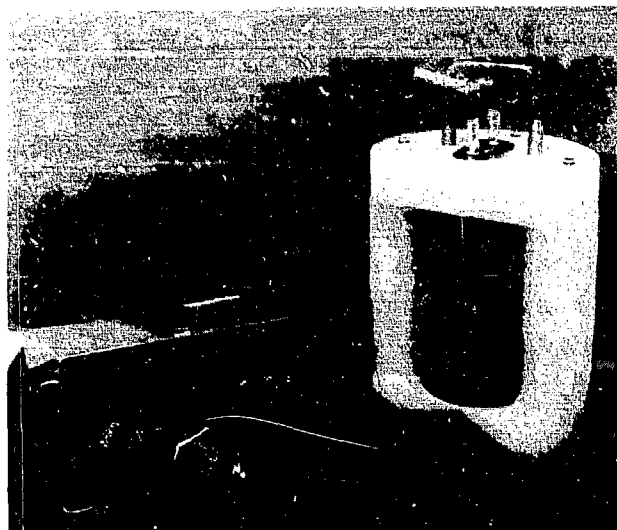


Fig. 16
Shielded Neutron Assay Probe (SNAP) connected to the Eberline SAM-II electronics unit. The SNAP can be used to assay plutonium samples by counting both (α, n) and fission neutrons.

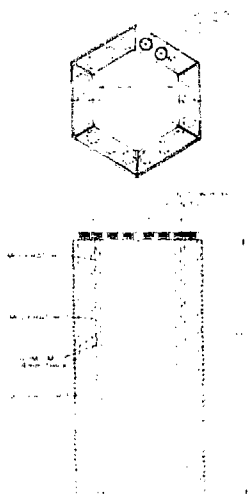


Fig. 17
The High-Level Neutron Coincidence Counter (HLNCC) contains 18 ^3He proportional tubes in a hexagonal polyethylene moderator. Coincidence counting allows one to detect fission neutrons from ^{240}Pu and other even isotopes, while discriminating against (α, n) neutrons in the sample.

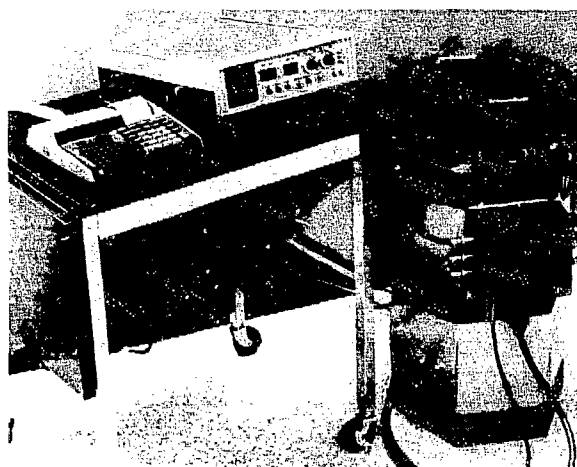


Fig. 18
Portable assay station consisting of a High-Level Neutron Coincidence Counter (HLNCC), shift register electronics, and HP-97 calculator. Designed for use by the IAEA, this arrangement allows inspectors to make rapid assays of plutonium samples under field conditions.

Since the efficiency for detecting coincidences is approximately equal to the square of the efficiency for detecting singles, coincidence counters are designed to achieve high singles counting efficiency by using well-counter geometry, i.e., by surrounding the sample with detectors.

The High-Level Neutron Coincidence Counter (HLNCC) shown in Figs. 17 and 18 is a compromise between the highest obtainable efficiency and portability.⁽¹³⁾ This detector, designed for use by IAEA inspectors, consists of 18 ^3He detectors in a hexagonal polyethylene shield. Larger, more efficient well counters have been designed for other applications.

Well counters are used to assay ^{240}Pu effective, a quantity defined by the equation

$$^{240}\text{Pu eff} = 2.5\ ^{238}\text{Pu} + ^{240}\text{Pu} + 1.7\ ^{242}\text{Pu} .$$

To obtain a complete plutonium assay, well counter measurements are frequently made in conjunction with isotopic ratio measurements using gamma spectroscopy (Fig. 19).

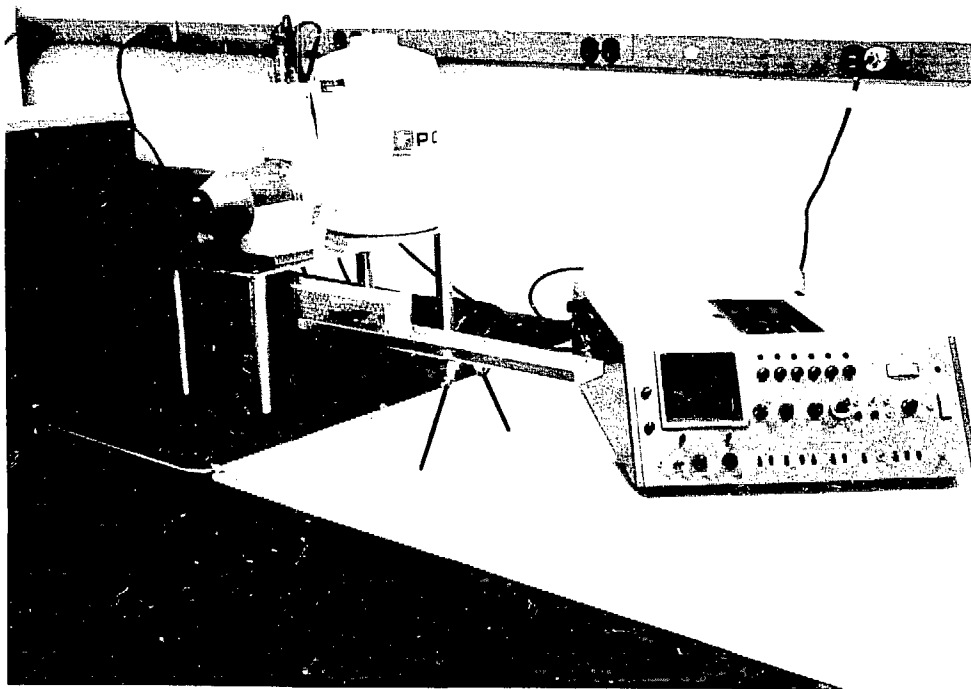


Fig. 19

Plutonium assays using the HLNCC depend on knowledge of plutonium isotopic ratios. Here, an HLNCC measurement is made on a zero power reactor fuel drawer, while isotopic ratios are being determined using an intrinsic Ge detector.

One of the more important improvements in neutron well counter technology has come about through the development of shift-register coincidence circuitry.⁽¹³⁾ This technique allows higher coincidence counting rates than is possible with conventional coincidence circuits, and as a result, can be used to assay very large Pu samples, with minimum deadtime correction.

Passive coincidence counters have also been built using plastic scintillators that detect both fast neutrons and gamma rays from spontaneous fission. Although this approach has advantages in certain applications, well counters have become more widely accepted for precise assays of plutonium, because of their insensitivity to gamma rays and their long-term stability.

IV. CALORIMETRY

The heat generated by natural radioactive decay is the basis of one of the most precise methods for assaying plutonium. Almost all of the power generated in typical Pu samples comes from alpha decay, with minor amounts coming from beta decay and spontaneous fission. Table V gives the specific power (watts/gram) for plutonium and americium isotopes of interest to calorimetry measurements.⁽¹⁴⁾ For "reactor grade" plutonium, the specific power is about 15 mW/g, which is sufficient to allow calorimetric measurements of even subgram samples.

TABLE V
SPECIFIC POWERS OF PLUTONIUM AND AMERICIUM

Isotope	Half-Life (yr)	Specific Power (W/g)	Uncertainty in Specific Power (%, 1σ)
²³⁸ Pu	87.79	5.6716×10^{-1}	0.10
²³⁹ Pu	24 082	1.9293×10^{-3}	0.27
²⁴⁰ Pu	6 537	7.098×10^{-3}	0.2
²⁴¹ Pu	14.35	3.390×10^{-3}	0.06
²⁴² Pu	379 000	1.146×10^{-4}	--
²⁴¹ Am	434.1	1.1423×10^{-1}	0.14

One of the unique advantages of calorimetry compared with other NDA methods is that absolute calibration can be performed using electrical standards; and the precision obtainable under laboratory conditions is 0.1 to 0.2%, a figure that is difficult to achieve by most other methods. On the other hand, for large samples of PuO₂, the measurement time is long, approximately two to four hours, due

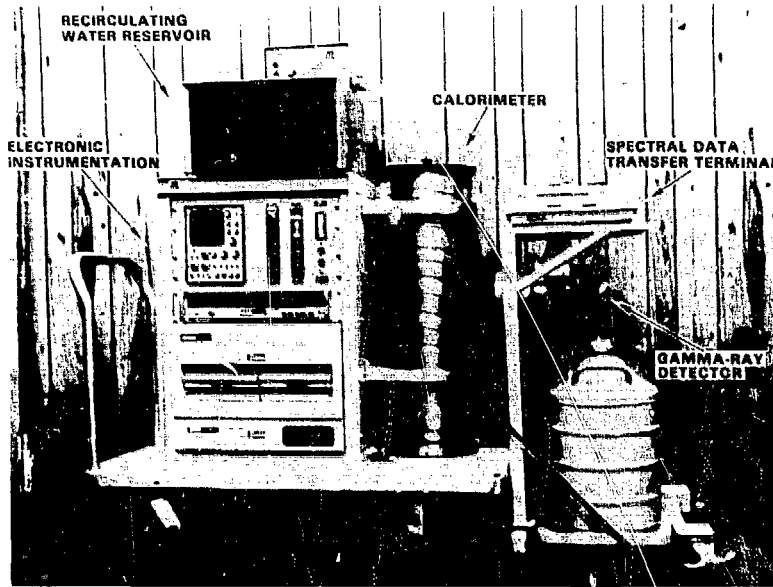


Fig. 20

Mound Laboratory instrument for assay of plutonium. The calorimeter measures power in watts, while the Ge spectrometer determines watts/gram based on Pu isotopic ratios.

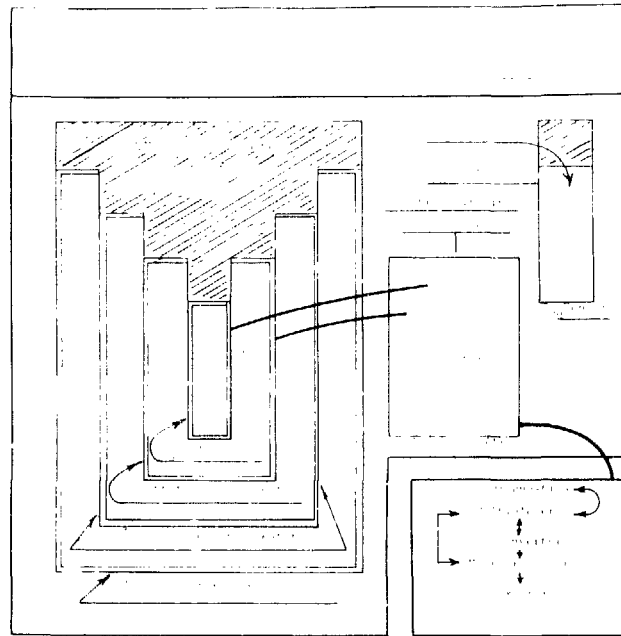


Fig. 21

Argonne design of a portable small sample calorimeter for the IAEA uses concentric aluminum cylinders, each with precise temperature control.

to the thermal properties of the sample material itself. Furthermore, calorimetry requires accurate knowledge of isotopic content, and absolute accuracy of total Pu is typically limited to about 1% because of uncertainties in determining the fraction of ^{238}Pu and ^{241}Am . Calorimeters are designed to give optimum performance (in terms of precision and measurement time) for a particular sample size, shape, and weight, and have lower precision when applied to other types of samples.

Figure 20 shows an instrument developed at Mound Labs for simultaneously measuring both plutonium isotopic ratios (using gamma-ray spectroscopy) and the power generated in the sample due to radioactive decay.⁽¹⁵⁾ The calorimeter consists of a constant temperature water bath that contains the sample well and temperature sensors. When a Pu sample is placed in the calorimeter, the sample comes to an equilibrium temperature that is measurably different from the reference temperature. Mound quotes an uncertainty in assaying grams of Pu of 1% for 11-cm-diameter containers with a measurement time of four hours.

Portable calorimeters have been developed at Argonne for use by the IAEA.⁽¹⁶⁾ The design consists of a series of concentric aluminum cylinders, each separately insulated and temperature controlled (Fig. 21). When placed in the inner-most cylinder, a plutonium sample reduces the electrical power required to maintain constant temperature. These instruments have a precision of about 1% and an equilibrium time of 20 minutes for samples of a few grams.

V. ACTIVE GAMMA-RAY AND X-RAY METHODS

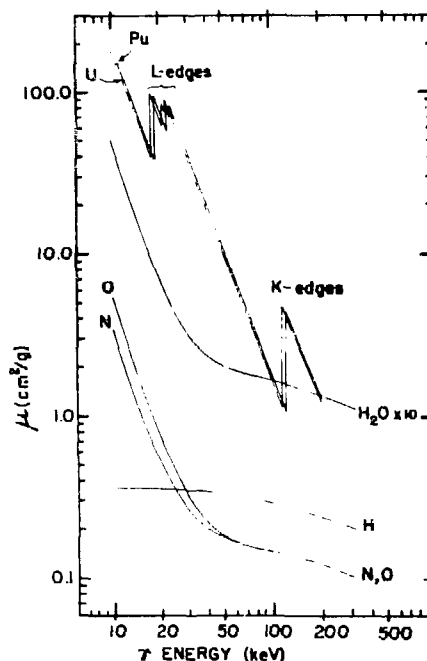
A. Absorption Spectrometry (Including X-Ray Densitometry)

Referring back to Fig. 1, active gamma-ray and x-ray methods can be further divided into absorption spectrometry (absorptiometry) and induced active response techniques.

Gamma-ray absorption measurements are not commonly used as a stand-alone assay technique in nuclear safeguards, since they are capable of little more than verifying the amount of high Z material in a sample. However, gamma-ray absorption techniques are widely used in many non-safeguards applications, such as in thickness gauges.

By contrast, x-ray absorption techniques are finding increasing application in safeguards, based largely on work at Los Alamos, where both K-edge and L-edge x-ray densitometers have been developed.⁽¹⁷⁾ X-ray densitometers are used to determine the amount of elemental uranium or plutonium in a sample by measuring the transmission of the sample at photon energies just above and below either the K or L absorption edge, see Table VI and Fig. 22. Measurements are made with a high resolution gamma-ray detector, external radiation sources, and suitable collimators for viewing the

Fig. 22
X-ray mass absorption coefficients for uranium, plutonium, and typical low Z matrix materials. X-ray densitometers have been designed that use both the K-edge and L-edge regions to determine elemental concentrations of U and Pu.



CONCEPTUAL DIAGRAM K-EDGE DENSITOMETER

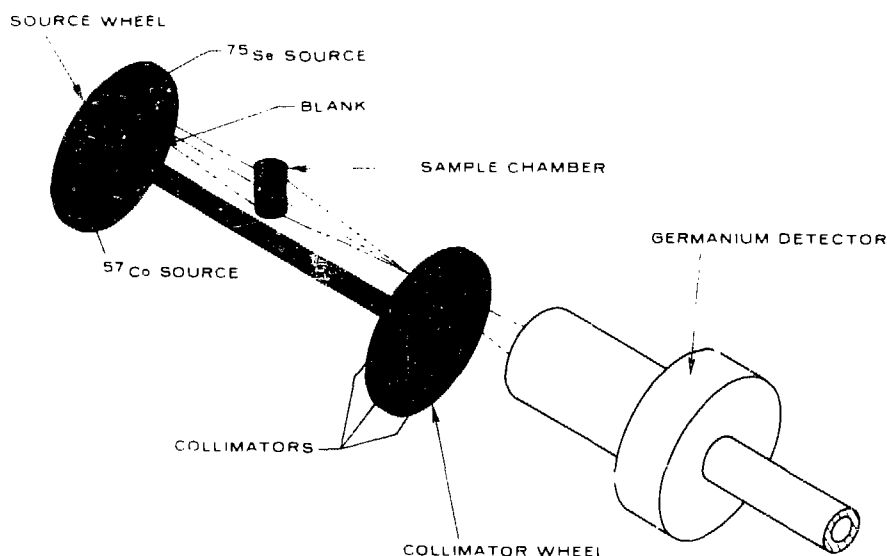


Fig. 23
In the K-edge densitometer installed at Tokai, isotopic sources and collimators are automatically rotated into position to measure x-ray attenuation of the sample just above and below the plutonium K-edge. Here a ⁷⁵Se source is positioned to measure the transmission at 121.1 keV.

TABLE VI
L_{III} AND K ABSORPTION EDGE ENERGIES FOR
URANIUM AND PLUTONIUM

<u>Absorption Edge</u>	<u>Element</u>	<u>Edge Energy (keV)</u>	<u>$\Delta\mu$ (cm²/gm)</u>
L _{III}	U	17.17	54.60
	Pu	18.05	51.90
K	U	115.60	3.65
	Pu	121.76	3.39

sample. For a K-edge assay of plutonium concentration, the sample transmission is measured at 121.1 keV (⁷⁵Se) and 122.1 keV (⁵⁷Ce), see Fig. 23. Plutonium concentration (g/liter) is calculated from the equation

$$\rho = \frac{-\ln(T_2/T_1)}{\Delta\mu x}$$

where

T_2/T_1 = ratio of sample transmission at 122.1 keV and 121.1 keV

$\Delta\mu$ = difference in mass attenuation coefficient at 122.1 and 121.1 keV

x = sample thickness.

Precisions on the order of 0.5% can be obtained in a 30 minute assay.

Most of the K-edge instruments have been designed to measure discrete liquid samples of a few ml; however, an in-line instrument has been installed at the Savannah River Plant for test and evaluation (Fig. 24).

SRP DENSITOMETER

SCHEMATIC OF THE IN-LINE X-RAY ABSORPTION-EDGE DENSITOMETER FOR THE SAVANNAH RIVER PLANT

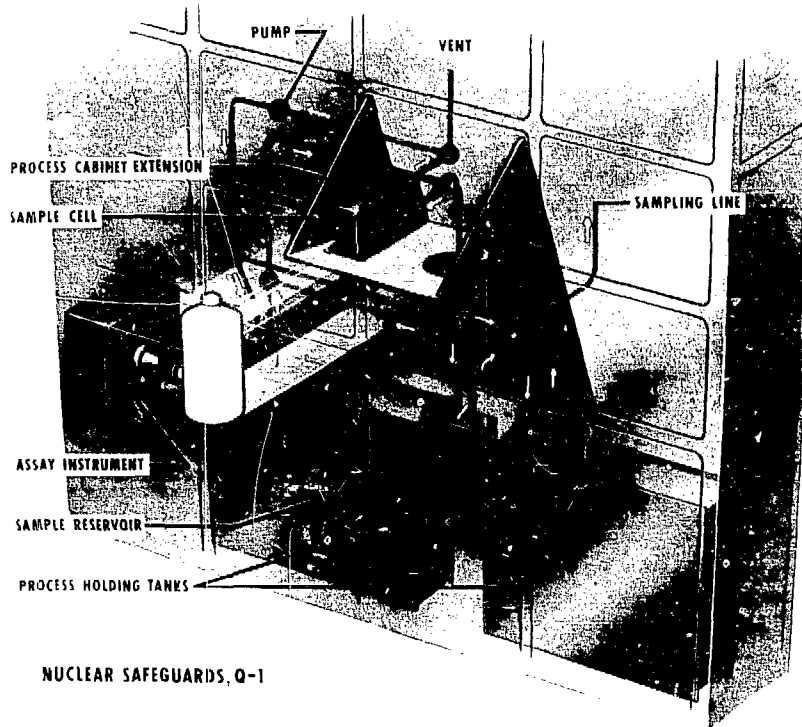


Fig. 24

The in-line K-edge densitometer installation at the Savannah River Reprocessing Plant. Plutonium product solution can be pumped from either of the process holding tanks through the sample cell, which is contained in a small extension of the process cabinet. The assay instrument sits on a shelf outside the process cabinet, allowing transmission measurements of the sample cell.

For solutions having plutonium densities below 20 g/l, the sample thickness required to perform accurate K-edge assays becomes impractically large. Table VI shows that the values of $\Delta\mu$ at the L_{III} edge are about 15 times greater than $\Delta\mu$ at the K-edge, allowing sample thicknesses to be reduced accordingly for L-edge assays.

L-edge densitometers use the bremsstrahlung from low energy x-ray generators to determine the sample transmission as a function

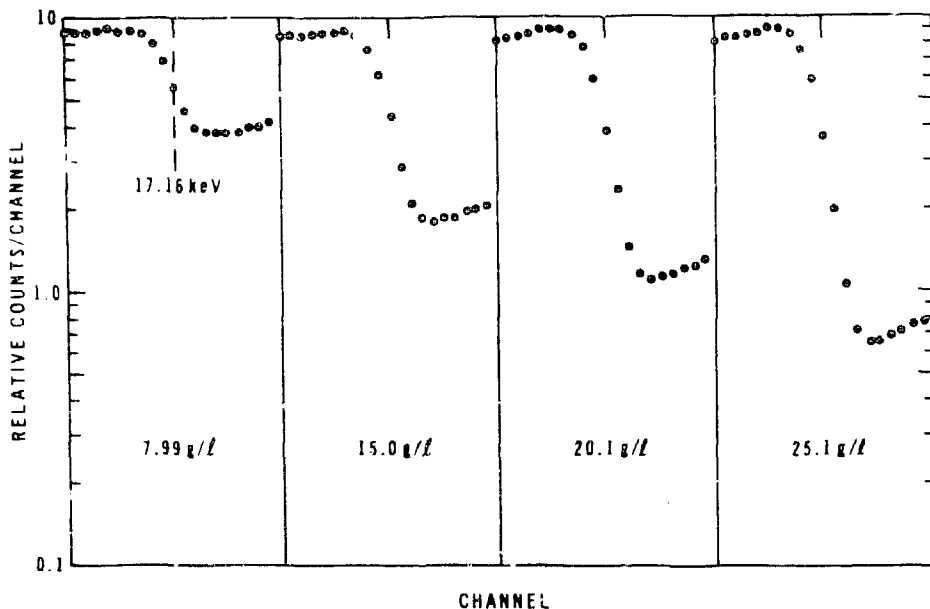


Fig. 25

X-ray transmission spectra for 2-cm-thick uranium solutions near the L_{III} absorption edge of uranium (17.16 keV), taken with the instrument shown in Figure 26.

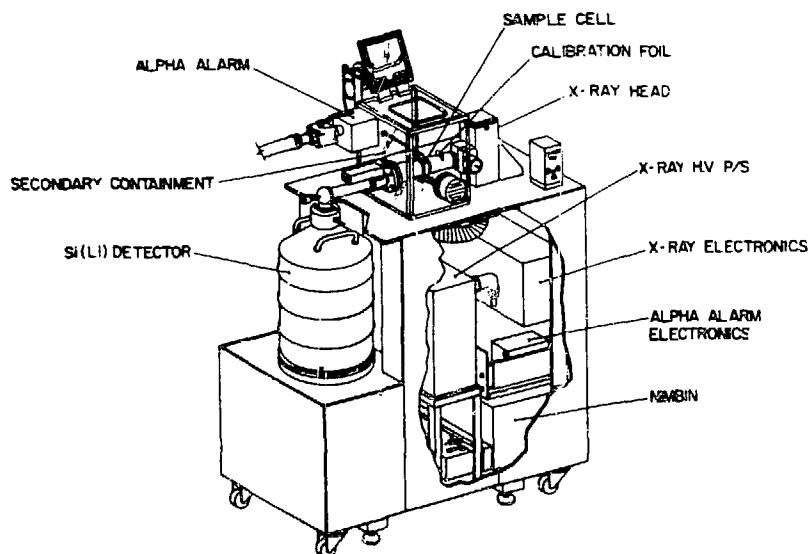


Fig. 26

L-edge densitometer designed for measuring uranium or plutonium concentrations in low density solutions. The secondary containment box which encloses the sample cell is designed to be mounted to the rear of a glovebox, and solution from the glovebox is piped into the sample cell.

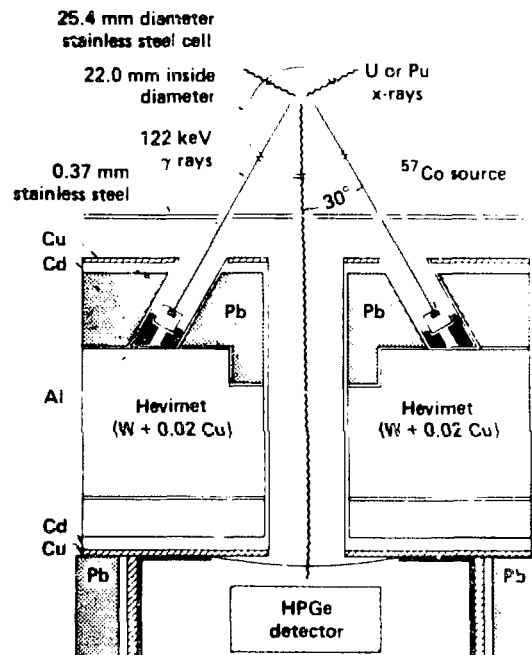
of energy across the absorption edge (Fig. 25). The use of a continuous x-ray spectrum provides a complete assay in one measurement, and the same x-ray machine can be used for assaying plutonium, uranium, and mixed U-Pu solutions (Fig. 26). Unlike K-edge assays, L-edge assays are sensitive to low Z matrix constituents because of the rapidly changing photo-electric and Compton cross-sections of low Z materials in this energy region.

B. Induced Active Response (Including X-Ray Fluorescence)

Active interrogation methods using gamma-ray or x-ray sources are not widely used in safeguards today, although one method was utilized in the past and another method looks promising for the future. In the early 1970s, IRT developed a barrel scanner using bremsstrahlung from a 10 MeV electron linac to cause photo-fission in ^{235}U , ^{238}U , and ^{239}Pu .⁽¹⁸⁾ Between each linac pulse, neutrons from the sample were moderated and counted with BF_3 counters. Although this technique gives fairly accurate assays of scrap materials, it has largely been replaced by active neutron methods.

A method that holds promise for future safeguards applications is x-ray fluorescence. For many years x-ray fluorescence has been used as a laboratory tool for chemical analysis based on the fact that characteristic x-ray energies depend on atomic number. Most previous applications used electron bombardment to produce the characteristic K or L x-rays; hence, careful sample preparation was required, much as is the case for alpha counting. More recently, low energy gamma-rays from isotopic sources have been used as the excitation source,⁽¹⁹⁾ allowing measurement of plutonium and uranium in their existing containers (Fig. 27).

Fig. 27
Arrangement of ^{57}Co excitation sources and Ge detector used by Camp and Ruhter⁽¹⁹⁾ for x-ray fluorescence measurement of uranium and plutonium solutions contained in a cylindrical (pipe) geometry.



X-ray fluorescence has been shown to have high precision over a wide dynamic range of uranium and plutonium concentrations. In mixed uranium-plutonium solutions, the method is capable of giving an accurate determination of the Pu/U ratio. However, to give accurate assays of individual concentrations of uranium and plutonium, techniques must be developed to correct for absorption in the sample of both the excitation gamma rays and the fluorescent x-rays.

VI. ACTIVE NEUTRON METHODS

A. General Information

Active neutron methods constitute one of the most powerful techniques for the assay of fissionable materials, and the number of active neutron instruments used in safeguards grew rapidly during the 1970s. Initially, active neutron assays were performed using reactors and positive ion accelerators as the source of neutrons, and these methods are still used when the ultimate in sensitivity is needed for small sample assay. However, for most safeguards applications, instruments that use isotopic neutron sources are preferred because of improved size, cost and reliability. Characteristics of the most commonly used isotopic neutron sources are shown in Table VII.⁽¹⁾

TABLE VII
COMMON ISOTOPIC NEUTRON SOURCES

<u>Source</u>	<u>Approximate Average Energy</u>	<u>Half-Life</u>	<u>Maximum Typical Strength n/s</u>
²⁵² Cf	Fission (2 MeV)	2.6 yr	5×10^9
²³⁸ Pu-Li(α ,n)	0.5 MeV	88 yr	2×10^6
²³⁸ Pu-Be(α ,n)	5 MeV	88 yr	10^6
²⁴¹ Am-Li(α ,n)	0.5 MeV	458 yr	5×10^5
¹²⁴ Sb-Be(γ ,n)	23 keV	60 days	5×10^8

The primary instrument design problem in active neutron interrogation is in discriminating against source neutrons, while detecting neutrons produced by induced fissions in the sample. As shown in Fig. 1, the three methods used for discriminating against source neutrons are:

- coincidence discrimination,
- time discrimination, and
- energy discrimination.

In the following sections, instruments will be discussed that employ each of the above techniques.

B. Coincidence Discrimination (Random Driver, AWCC)

The random driver (Fig. 28) was one of the first active neutron assay instruments developed specifically for safeguards. The Los Alamos design⁽²⁰⁾ uses an AmLi neutron interrogation source of about 5×10^5 neutrons/s and large plastic scintillator neutron detectors, shielded from the sample with lead to reduce gamma-ray sensitivity. By demanding that two neutrons be counted with a 50 ns

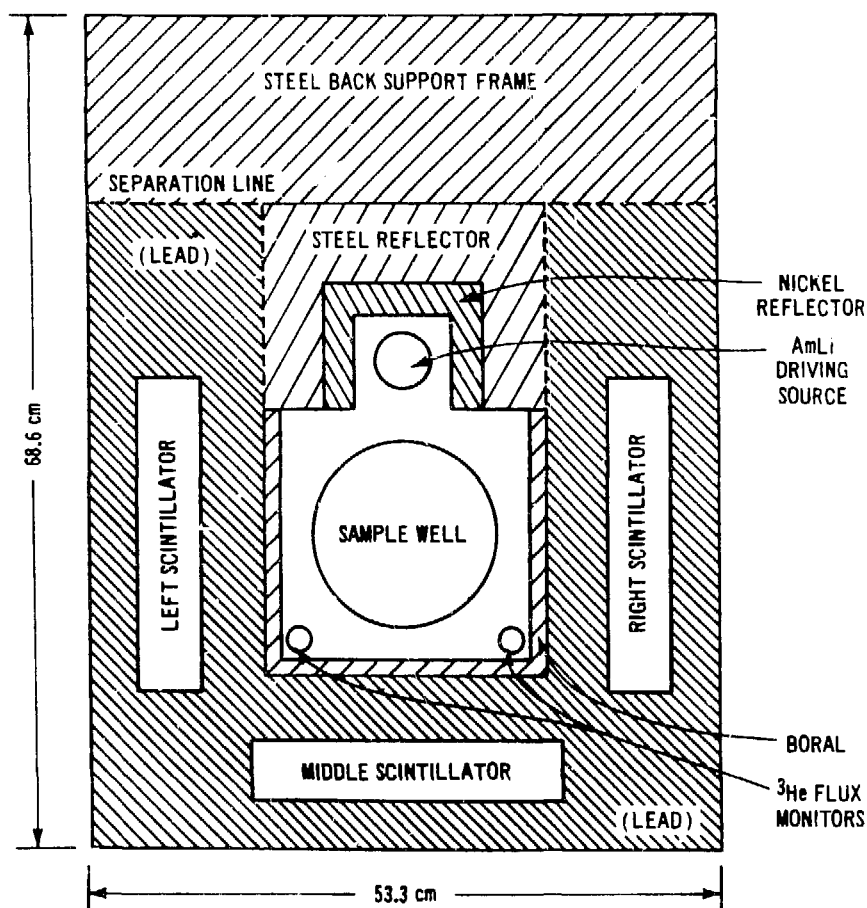


Fig. 28
Random driver of recent design used by Los Alamos for assay of plutonium samples. One of the first active neutron instruments designed for safeguards applications, random drivers have been widely used for the assay of uranium scrap materials in fuel fabrication facilities.

coincidence resolving time, single neutrons from the AmLi source are discriminated against, while multiple neutrons from induced fission in the sample are detected. Other random driver designs that omit the lead shielding are able to assay samples of approximately 0.1 gram of ^{235}U , but are also sensitive to gamma-ray attenuation in the sample. Random drivers have been widely used for assaying uranium scrap and waste materials, and have been designed to accommodate a variety of container sizes, ranging from a few liters to 200-liter waste barrels.

Most random drivers have incorporated into their design ^3He detectors used to correct the assay for homogeneous moderating material in the sample and a means of rotating the sample in order to provide more uniform neutron irradiation. Some random drivers have also used temperature sensors to correct for detector temperature dependence. Although the smaller random drivers are suitable for van installations, their basic design using large scintillator detectors and lead shielding does not lend itself to portability.

The Active Well Coincidence Counter (AWCC) was designed as a simpler, more easily transported replacement for the random driver (Figs. 29, 30). The design of the AWCC is very similar to passive neutron well coincidence counters (Section III.C), except that two small AmLi neutron sources (approximately 5×10^4 n/s) are placed in end plugs above and below the sample well.⁽²¹⁾ A shift register coincidence unit effectively discriminates against single neutrons from the AmLi sources while detecting coincident neutrons from fissions in the sample. Compared with the random driver, the AWCC is less accurate for small samples, and it is slightly more expensive; but it is far more portable, rugged, and reliable; and, for these reasons, more suitable for IAEA applications.

An instrument based on the same principles as the AWCC was designed to assay fresh LWR fuel assemblies (Fig. 31). All of the instruments in this class can be used in the passive mode (by removing the AmLi neutron sources) to assay plutonium.

C. Time Discrimination (Cf Shuffler)

One of the oldest techniques for assaying fissionable materials involves irradiating samples with neutrons from reactors or particle accelerators and then counting either delayed neutrons or gamma rays from the sample. Delayed neutrons are emitted by neutron unstable fission products with half-lives ranging from 0.25 to 56 s. For ^{235}U , the ratio of delayed neutrons to prompt neutrons is about 1/120, and for ^{239}Pu , about 1/335. Delayed gamma rays are emitted by many different fission products with half-lives ranging from seconds to years.

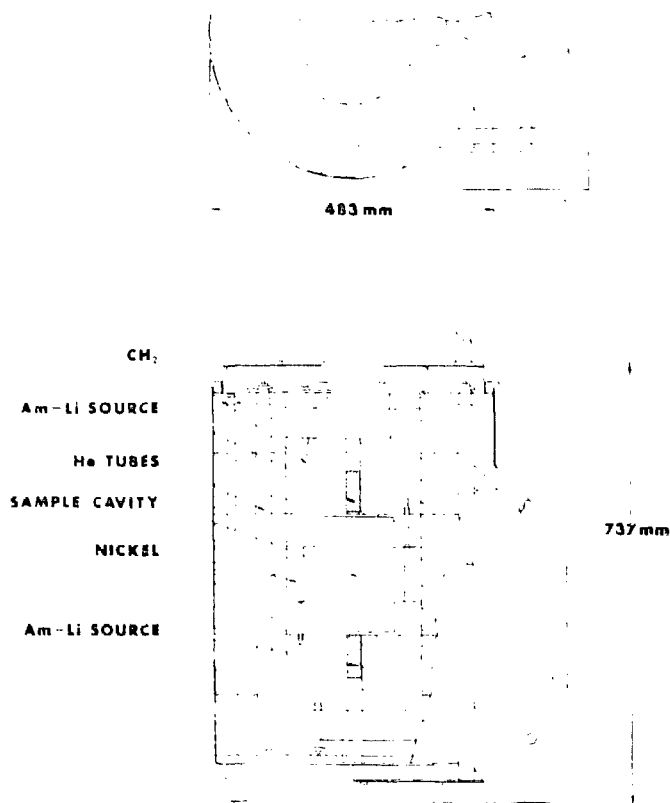
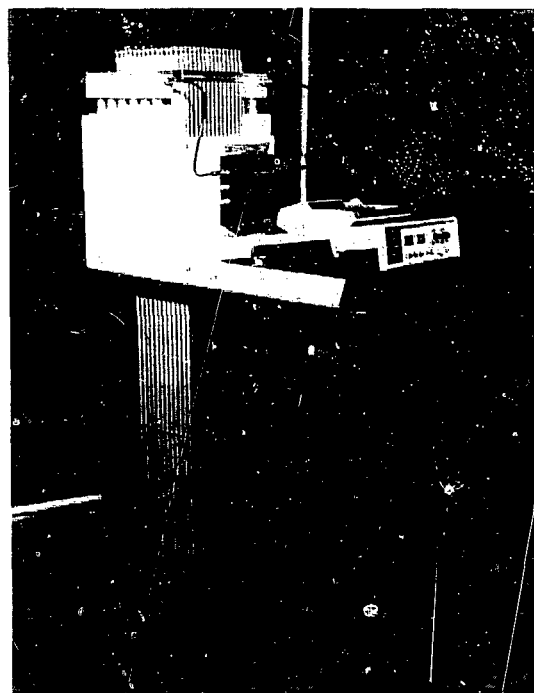


Fig. 29
Cross-section of Active Well Coincidence Counter, showing location of neutron sources in end plugs above and below the sample well.

Fig. 31
Active neutron instrument for the assay of fresh fuel assemblies. It uses a single AmLi neutron source on one side of the fuel assembly and six ^3He detectors in polyethylene slabs on the other three sides of the assembly. The electronics package is the same for this instrument, the Active Well Coincidence Counter, and the High-Level Neutron Coincidence Counter.



Fig. 30
The Active Well Coincidence Counter was designed as a transportable instrument for IAEA use in assaying uranium samples.



With the production of large ^{252}Cf sources in the late 1960s, it became possible to use isotopic sources in delayed neutron and gamma-ray assay instruments. The Cf shuffler (Figs. 32, 33) uses a 1 mg ^{252}Cf source (approximately 2.5×10^9 n/s), a U-shaped source transfer tube, and ^3He proportional counters around the sample well.⁽²²⁾ A motor-driven cable moves the source into position where it irradiates the sample for a few seconds, then quickly withdraws the source to a shielded position. With the source removed, the ^3He detectors are gated on to count delayed neutrons, and the whole cycle is then repeated many times for an assay. Typical irradiation and counting cycles are about 10 seconds each, although optimum times may vary depending on sample characteristics.

Cf shufflers are among the most sensitive assay instruments, capable of measuring 1 mg of ^{235}U in small containers, with an assay time of approximately 30 minutes. Using source-tailoring techniques such as Ni reflectors and CH_2 moderators, one can adjust the energy spectrum of the interrogating neutrons to obtain either a thermal or a fast neutron assay. Shufflers have been designed for a wide range of material types, including ^{235}U scrap from fuel fabrication facilities, hot waste barrels containing uranium and plutonium, and highly enriched spent-fuel elements and waste canisters (Fig. 34). For measuring spent fuel, the Cf source must be large enough to overcome neutron background in the sample due to plutonium and curium isotopes.⁽²³⁾

Another type of instrument using ^{252}Cf neutron irradiation was developed for assaying fast breeder reactor fuel rods (Fig. 35). Here the technique was to continuously move a fuel rod through a shielded neutron irradiator, past a NaI detector where delayed gamma rays are counted. Individual fuel pellets can be scanned and total fissile accurately measured for both materials accounting and quality control.⁽²⁴⁾

D. Energy Discrimination (Spent-Fuel Assay System)

Several active neutron instruments have been designed that discriminate against source neutrons on the basis of neutron energy. One fairly recent example⁽²⁵⁾ is an instrument designed at Oak Ridge for the assay of spent breeder reactor fuel subassemblies. It uses four large ^{124}Sb -Be photo-neutron sources, each of which produce about 10^9 neutrons/s. Detectors are methane-filled, proton-recoil proportional counters, capable of discriminating against the 23 keV neutrons from the sources, while detecting the more energetic neutrons from induced fissions in the spent fuel. Developed for performing assays at the head end of a reprocessing plant, the system is capable of 5% precision in a 20-minute measurement time.

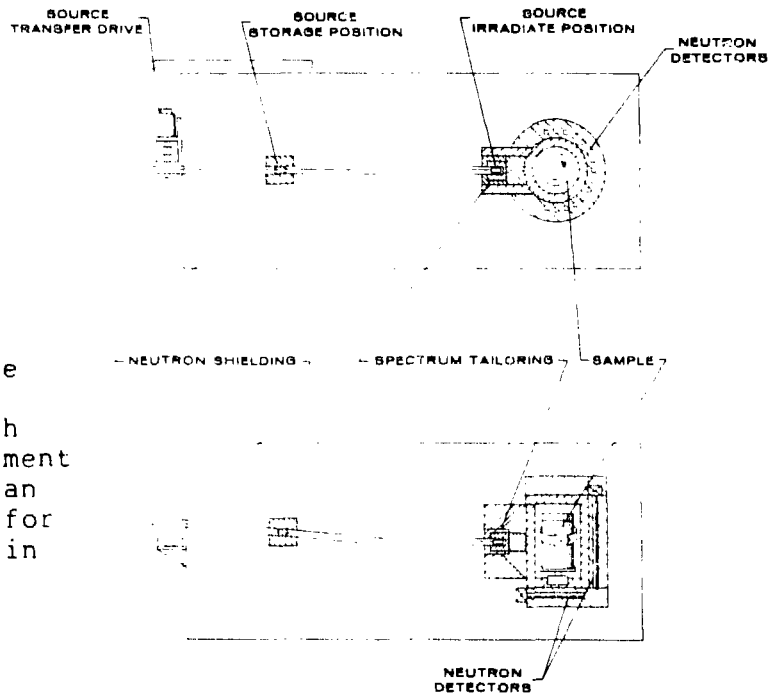


Fig. 32
Top and side views of the californium shuffler installed at the Savannah River Plant. The instrument was designed to provide an assay precision of 0.3% for uranium scrap and waste in 18 cm x 30 cm cans.



Fig. 33
Complete Savannah River Cf shuffler showing terminals, electronics rack, and hoist for lowering samples into the measurement well.

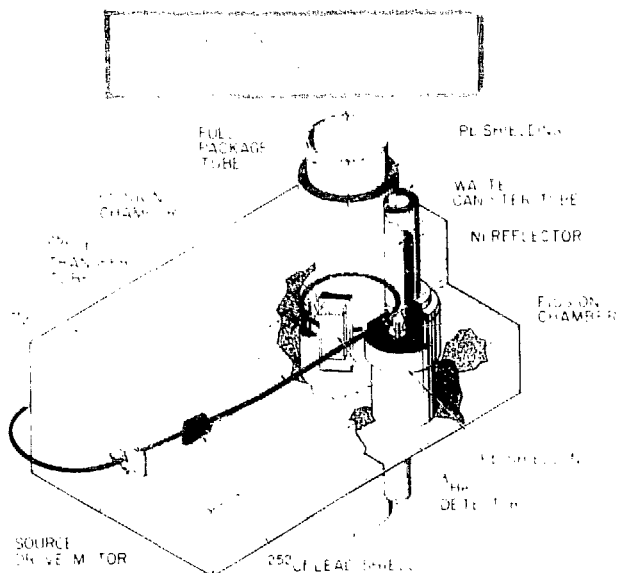


Fig. 34

Large Cf shuffler system for the assay of either spent-fuel assemblies or waste canisters. To achieve a more uniform assay of assemblies, the Cf source moves around the fuel package tube during the irradiation cycle. This instrument is to be installed in the Florinel and Storage (FAST) Facility at the Idaho Chemical Processing Plant.

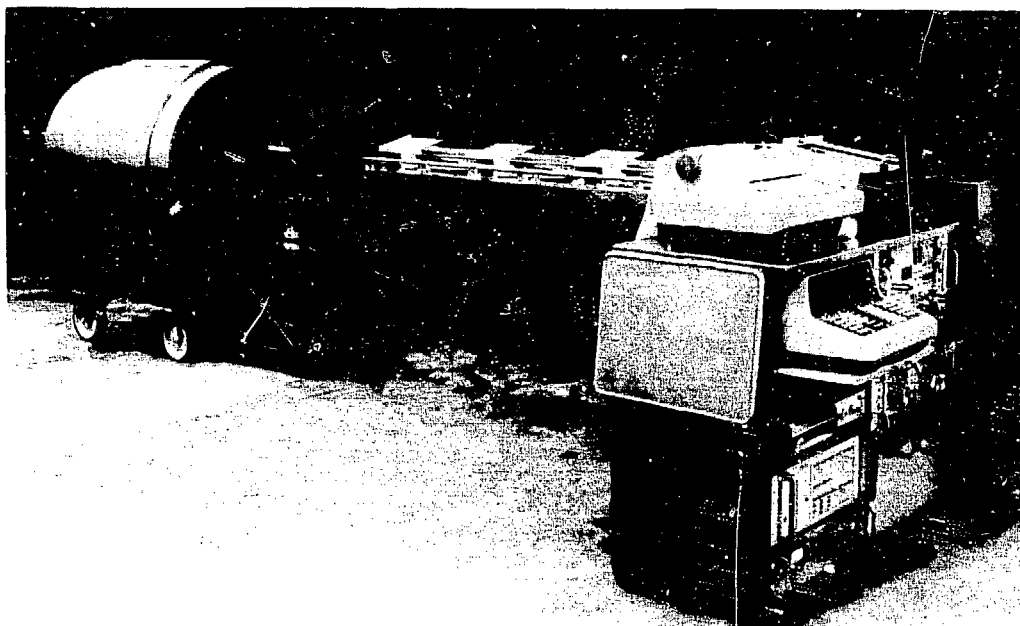


Fig. 35

Fuel rod scanner developed for the Fast Flux Test Facility (FFTF) at Hanford, Washington. The hybrid assay instrument uses a Cf source for fast neutron interrogation and measures delayed gamma rays with two large NaI detectors. The system determines plutonium fissile content in a fuel rod to better than 0.5% accuracy and has been used to assay many thousands of rods.

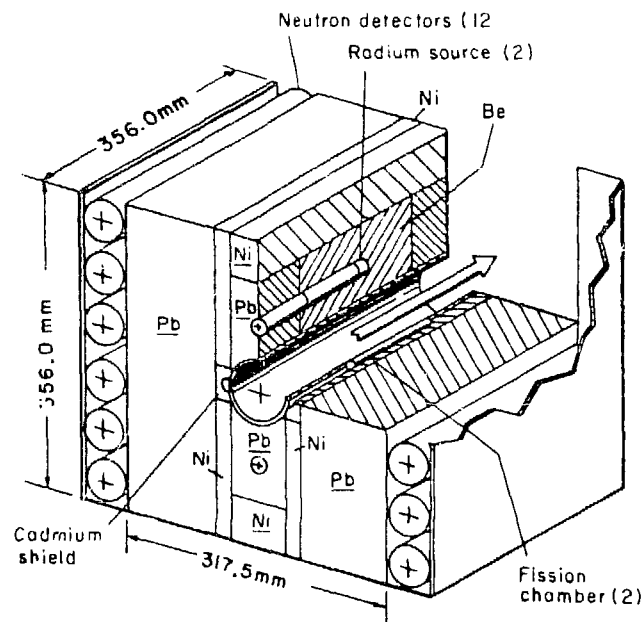


Fig. 36

^{226}Ra -Be neutron source and ^4He detector arrangement used by Los Alamos (26) for assay of ^{235}U content of irradiated Rover (nuclear rocket) fuel. The ^4He proportional counters detect fission neutrons while discriminating against the less energetic source neutrons.

An earlier system designed at Los Alamos for assaying irradiated nuclear rocket fuel worked on the same principle, but was physically much smaller (Fig. 36). It used two small ^{226}Ra -Be(γ ,n) sources and twelve ^4He gas proportional counters.(26) The technique of energy discrimination against source neutrons has also been applied by Los Alamos in instruments for assaying fresh LWR fuel assemblies and for well logging of uranium ore formations.

VII. CONCLUSIONS

The interpretation of essentially all assays depends on having information about the sample, other than the quantity being measured. Usually, the more well characterized a sample is from a physical and chemical standpoint, the more accurately it can be assayed, either destructively or nondestructively.

The selection of one NDA technique over another commonly depends on what one knows and does not know about the characteristics of the sample material. From the standpoint of NDA, the most important characteristics of a sample are its spatial properties (size, shape, uniformity, etc.), its isotopic properties (ratios of fissionable isotopes), and its matrix properties (atomic numbers, density, unusual non-fissioning isotopes, etc.).

If a sample to be assayed has well determined isotopic properties but poorly defined spatial properties, then a neutron method or calorimetry is likely to provide the best assay. Consider, as examples, the dependence of passive neutron and calorimetric assays on precise knowledge of plutonium isotopic content, but their relative independence of sample uniformity.

When neither spatial nor isotopic properties are well known, a combination of gamma-ray spectroscopy and active neutron methods is likely to provide the most accurate assay and greatest level of confidence.

Variations in high Z or high density matrix materials (that are not properly corrected for) increase the uncertainty in gamma-ray assays more than in neutron assays. Variations in low density matrix materials, such as water, oxygen, and fluorine, increase the uncertainty in neutron assays more than in gamma-ray assays. Thus, a high density matrix tends to favor neutron assay, while a low density matrix tends to favor gamma-ray assay.

The relationship between gamma-ray assay, neutron assay, and calorimetry is complementary in nature. When the characteristics of a sample (spatial, isotopic, or matrix) limit the accuracy of one NDA approach, another approach can generally be used with a high degree of confidence.

A large number of NDA instruments have been developed over the past decade and many are now commercially available. Future development work is expected to produce a new generation of NDA equipment that will be easier for inspector to independently calibrate, operate, and maintain, and also to lead to new approaches for potential problem areas, such as large bulk processing facilities.

Table VIII graphically summarizes points discussed above. For Case I, gamma-ray scattering and absorption parameters are measurable or known, but isotopic composition of fissionable and matrix materials is unknown. In this case several different gamma-ray techniques can be applied to give the typical assay results indicated. For Case II, isotopic composition of fissionable and matrix materials is known, but gamma-ray scattering parameters of the sample are not known. In this case passive or active neutron techniques and calorimetry are applicable. For Case III, neither gamma-ray scattering parameters nor isotopic composition of the sample are known. In this case, high resolution gamma-ray spectroscopy can be used to detect the presence of certain isotopes, and active neutron methods may be able to give an estimate of effective fissile content.

TABLE VIII

CONDITIONS UNDER WHICH VARIOUS NDA METHODS ARE USEFUL

	<u>Measurable or Known Sample Characteristics</u>	<u>Useful NDA Methods</u>	<u>Typical Assay Results</u>
Case I	Key gamma-ray scattering/ab- sorption charac- teristics	Passive gamma	Enrichment Isotopic ratios, Iso- topic content
		X-ray	Pu, U concen- tration
		Passive gamma and Passive neutron	Pu mass
		Passive gamma and Calorimetry	Pu mass
Case II	Isotopic composition	Passive neutron	Pu mass
		Calorimetry	Pu mass
		Active neutron	U-235 mass Pu mass
Case III	Nil	Passive gamma	Presence of certain iso- topes
		Active neutron	Effective fissile

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 9: NDA INSTRUMENTS FOR MEASUREMENT OF UNIRRADIATED FUEL

This session will provide an overview of passive gamma-ray and neutron measurement techniques for the assay/verification of fresh uranium and plutonium fuel materials.

After the session, the participants will be able to:

- Enumerate the safeguards motivations for the nondestructive assay/verification of unirradiated fuel materials, including complete fuel assemblies.
- Describe the reactor fuel categories that have been addressed by nondestructive assay/verification techniques.
- Describe several gamma-ray instruments used to measure the enrichment of uranium fuel materials.
- Describe the limitations of gamma-ray measurements of fuel materials and complete fuel assemblies.
- Describe neutron assay methods commonly used in fuel fabrication facilities to assay different types of reactor fuel.
- Point out differences and limitations in neutron and gamma-ray assay instrumentation.
- Discuss the precision and accuracy of NDA instruments for unirradiated fuel measurement.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 9, PART I: GAMMA-RAY TECHNIQUES FOR ASSAY/VERIFICATION OF UNIRRADIATED URANIUM AND PLUTONIUM FUELS

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I. INTRODUCTION

Nondestructive assays of unirradiated nuclear fuel and fuel components are utilized for quality control in the fabrication process and for materials accounting and control and safeguards verification prior to installation of the fuel into the reactor. Gamma-ray-based assays of fuel assemblies are, however, necessarily confined to investigation of the surfaces of the assemblies, due to the limited penetrability of the gamma radiation of interest. This disadvantage is especially acute for uranium-bearing fuels, because the gamma radiation from the fissile isotope of interest, ^{235}U , has the relatively low energy of 186 keV. Thus, verification of ^{235}U enrichment and/or total ^{235}U assay cannot be made on the full nuclear fuel volume without the impractical procedure of dismantling the assemblies. The situation with plutonium-bearing fuels is less severe because the radiation from the fissile isotope of interest, ^{239}Pu , includes gamma rays of higher energy, such as 414 keV.

In general, however, complete material accounting of unirradiated nuclear fuel by gamma-ray assay techniques is not feasible. Full interrogation of fuel assemblies must be carried out using neutron assay techniques, since the neutron radiation is sufficiently penetrating to provide useful information from the full volume of fuel material. Nevertheless, gamma-ray-based measurements on fresh fuel can be very useful for verification and attribute checks on fuel materials and as a complement to certain neutron-based fuel assays.

In this session we describe a variety of gamma-ray-based nondestructive measurements that can be carried out on unirradiated nuclear fuel. Measurements on both uranium and plutonium-bearing fuel will be addressed. One gamma-ray-related technique--autoradiography--has useful application for almost all types of nuclear fuel, so this technique is dealt with in some detail.

II. PASSIVE GAMMA-RAY COUNTING

A. Uranium Enrichment Measurements

A measurement of the ^{235}U enrichment in uranium-bearing fuel involves the determination of the intensity of the 186-keV gamma ray emitted by the fuel material. Specifically, if the uranium sample is thick compared with the mean-free path of 186-keV photons in the sample material, then the 186-keV intensity emitted from a fixed area of such a thick uranium sample is proportional to the ^{235}U enrichment of the sample.^{1,2} Numerous systems have successfully applied this concept with a variety of electronics and detectors to a wide range of problems, such as enrichment measurements of fuel pins, fuel plates, containers of UO_2 , cylinders of UF_6 , and in-line monitoring of liquid UF_6 . The key elements in reliable enrichment measurements are (1) achievement of an

appropriately "thick" sample (that is, several mean-free paths for the 186-keV gamma ray), (2) maintenance of a constant viewing geometry in both the calibration and measurement procedures (accomplished with an appropriate collimation system), (3) determination and correction for cladding or other variable-thickness material between the detector and the uranium being measured.

Qualitative verification of fuel enrichment can be performed for the fuel rods in peripheral rows of light-water reactor (LWR) assemblies by straightforward application of the enrichment measurement principles.^{1,2} Scanning of individual fuel plates in materials testing reactor (MTR) fuel is also feasible and will give a more complete picture of the total fissile content of that type of fuel assembly.

B. Plutonium Isotopic Determination

The assay of plutonium-bearing materials focuses on the radiative emission of specific isotopes. This is particularly true in the neutron assays of nuclear fuel where passive neutron assays measure the amount of the spontaneously fissioning isotopes (^{240}Pu , ^{242}Pu , ^{238}Pu), and active neutron interrogation is sensitive to the fissile isotopes (^{239}Pu , ^{241}Pu). The neutron assay techniques alluded to here will be discussed in detail in the next session. For determination of total plutonium from passive assays, one must also know the isotopic composition of the samples being assayed. This isotopic composition can be easily determined by passive gamma-ray counting of the plutonium-bearing sample. Various analytical techniques are available for plutonium isotopic composition determination³ using high-resolution gamma-ray spectroscopy and known nuclear data on the gamma-ray emissions of the various plutonium isotopes.⁴

At present, the isotopic data quoted by the facility operator (rather than that which could be measured by the inspector) are used in conjunction with neutron measurements to obtain total plutonium mass. High-resolution gamma-ray spectroscopy is used periodically to verify the quoted isotopic data. This procedure avoids the significant degradation of the precision of the neutron measurement by the less precise field measurements of isotopic composition. An example of plutonium isotopic verification in conjunction with neutron assay of plutonium-bearing fuel is depicted in Fig. 1, where fuel plates for a zero-power-plutonium reactor (ZPPR) are shown being assayed first by a passive neutron coincidence counter, followed by an isotopic determination through high-resolution gamma-ray spectroscopy.

C. Passive Gamma-Ray Assay of MTR Fuel

An MTR-type fuel element is a boxed array of parallel fuel plates with uniform gaps between the plates. There may be from 10 to 25 plates per element; the usual number is about 20. Each plate is of sandwich construction, consisting of an inner layer of enriched uranium alloy (approximately 20% uranium by weight and ^{235}U enrichment as high as 93%) and two outer layers of aluminum cladding. The plates are typically 60 cm long by 6 cm wide, and the thickness of each of the layers is 1-2 mm. The 2-3-mm gaps between the plates make the overall dimensions of the element 60 by 6 by 6 cm. Typical uranium loadings per plate are about 13-15 g total uranium. Some fuel elements may contain a few empty plate positions or dummy plates (pure aluminum) in place of the fuel-bearing plates.

In the fuel fabrication stage, the loading uniformity and the total fissile content of the fuel plates can be determined by careful scanning of the plates with a collimated gamma-ray detector. However, even though the fuel plates are thin, their attenuation of the 186-keV gamma radiation is

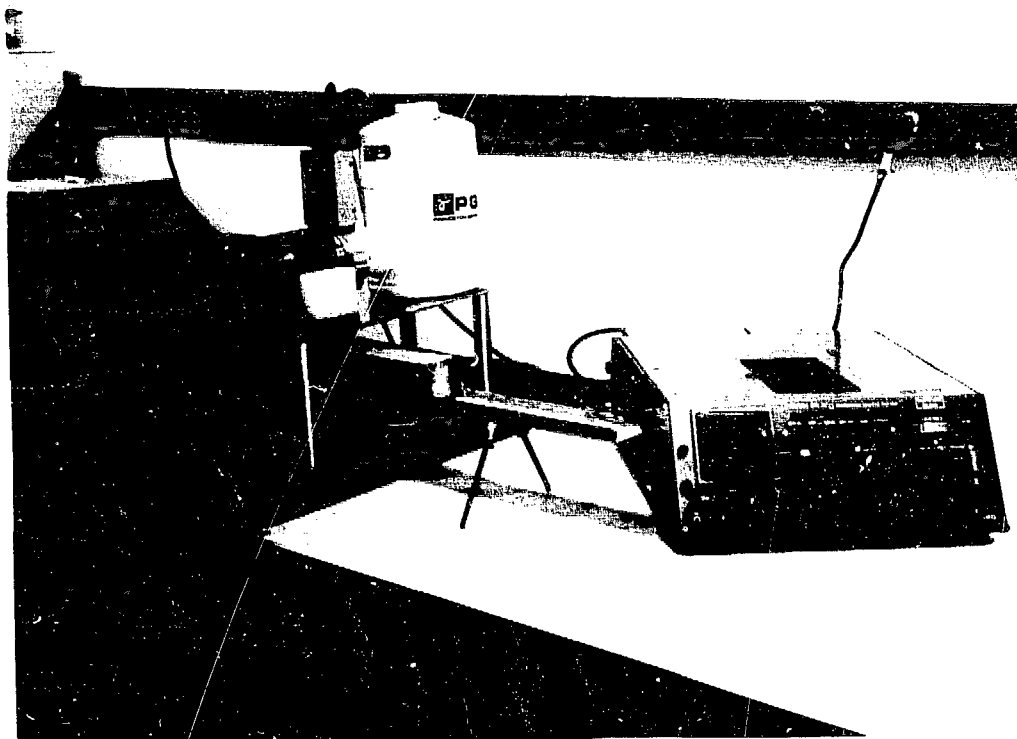


Fig. 1. Setup for the simultaneous assay of ZPPR fuel by passive neutron coincidence counting and high-resolution gamma-ray spectroscopy. The neutron count gives the effective ^{240}Pu content of the fuel, and the gamma-ray analysis gives the isotopic composition, thereby providing the total plutonium content of the fuel.

significant and must be corrected for. This can be accomplished by the use of a radioactive source (most conveniently a ^{235}U source) positioned behind the plate being assayed. In this way, the transmission property of each plate segment is determined, thereby providing sufficient information for correction of the assay results for self-absorption of the 186-keV gamma ray by the fuel-plate meat and cladding. General details of such transmission corrections may be found in Ref. 5, and specific application to MTR fuel plates is discussed in Ref. 6.

The measurement requirement more commonly encountered in a reactor facility is the quantitative assay of complete MTR fuel assemblies. Such a measurement can be performed by passive gamma-ray counting in a reproducible geometry wherein the full assembly is viewed by the collimated detector. The measurement configuration is depicted schematically in Fig. 2. A number of factors must be considered in such an assay. Especially important is the transmission correction, which is quite significant because many fuel plates are involved. The most convenient transmission source is another MTR assembly (as implied by the figure). The transmission count through the fuel assembly being assayed is then obtained through subtraction of the 186-keV gamma-ray intensity without the transmission source present from that with it in place.

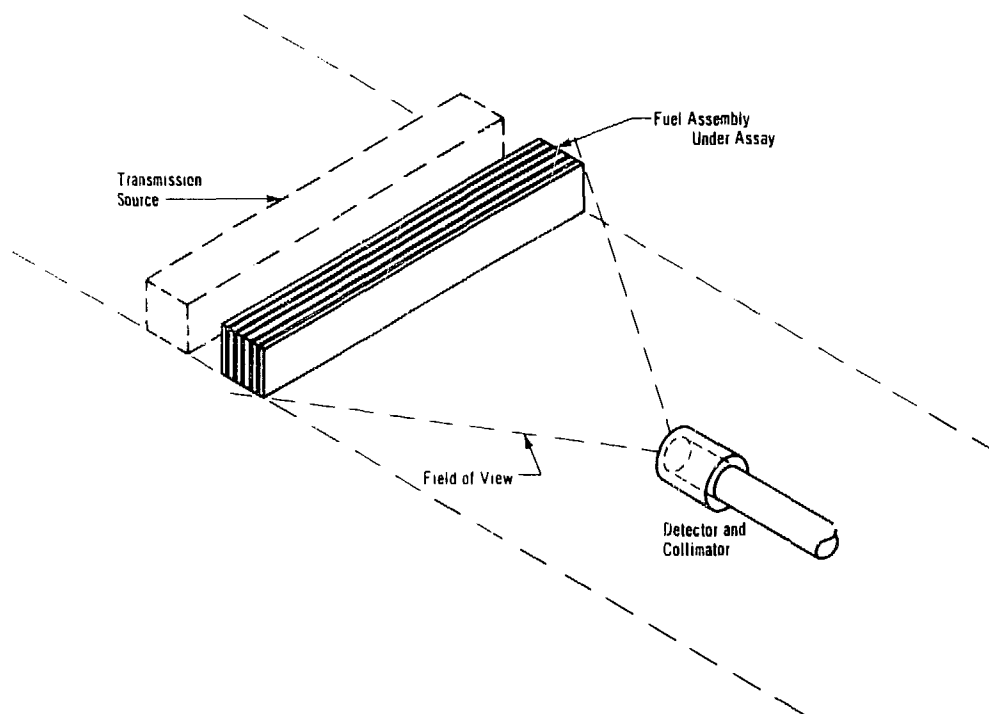


Fig. 2. Schematic for the passive gamma-ray counting of MTR fuel assemblies. The gamma-ray detector is collimated to view the entire assembly during the count. A second assembly is indicated as a possible source for the measurement of the transmission correction.

The data-acquisition procedure is the same as that outlined in Ref. 6 for the MTR fuel plates. An additional consideration is a correction for gamma-ray buildup in the fuel assembly. This phenomenon arises when gamma rays, which would normally travel out of the detector field of view, are scattered within the assembly so as to be detected, thereby augmenting (or "building up") the gamma-ray signal. This type of distortion of a measured gamma-ray intensity arises in any extended source. In the MTR fuel assay case, it can be corrected by calibrating the assay apparatus with standard MTR assemblies. A thorough discussion of the data analysis and theory underlying MTR fuel assembly assays by passive gamma counting may be found in Ref. 7.

Another interesting approach to MTR fuel assay for inspection purposes is discussed in Ref. 8. The approach involves the use of K-absorption-edge densitometry⁹ to determine the total (elemental) uranium content of an MTR fuel assembly. Densitometry measurements have demonstrated⁸ that the technique is sufficiently sensitive to detect the absence of one fuel plate from the assembly. However, densitometry is not routinely used to inspect MTR fuel because it is a more complex technique than passive gamma-ray counting.

D. Passive Gamma-Ray Assay of Fast Critical Assembly (FCA) Fuel

Passive gamma-ray counting techniques have been investigated for the quantitative verification of plutonium in FCA fuel drawers.¹⁰ These fuel

drawers (an example of which is shown in Fig. 3) contain several plutonium-bearing fuel plates. Further description of FCA facilities is given in the next section on autoradiographic techniques. The plutonium content of the fuel drawer can be assayed by using an enrichment type of measurement. The ^{239}Pu and ^{241}Pu content of the entire drawer and the isotopic ratio of the two can be measured simultaneously. The counting rate is proportional to the product of the fraction of plutonium isotope to the heavy metal (enrichment) and the surface area of the fuel plates viewed by the detector. Moreover, if the plutonium fraction, the type of alloy, and the widths of the fuel plates are the same, then the counting rate is proportional to the total length of the fuel plates in the drawer.

The plutonium can be assayed in the experimental setup shown in Fig. 4. The drawer is placed sideways so that the edges of the plates face the detector. The detector-to-drawer distance is sufficiently large so that the $1/r^2$ variations across the drawer are negligible. If the plutonium mass fractions and isotopic enrichments of the fuel plates are similar and the ^{241}Am buildup is known, then the total plutonium content of the drawers can be inferred from the gamma-ray intensity of the surfaces being viewed. Figure 5 shows the relative counting rate of the ^{239}Pu 414-keV gamma ray vs the total ^{239}Pu content of the drawer under such controlled conditions, and the proportionality between the two quantities is evident. For drawers containing two rows of plutonium plates, the ^{239}Pu assay (using the 414-keV gamma ray) and the

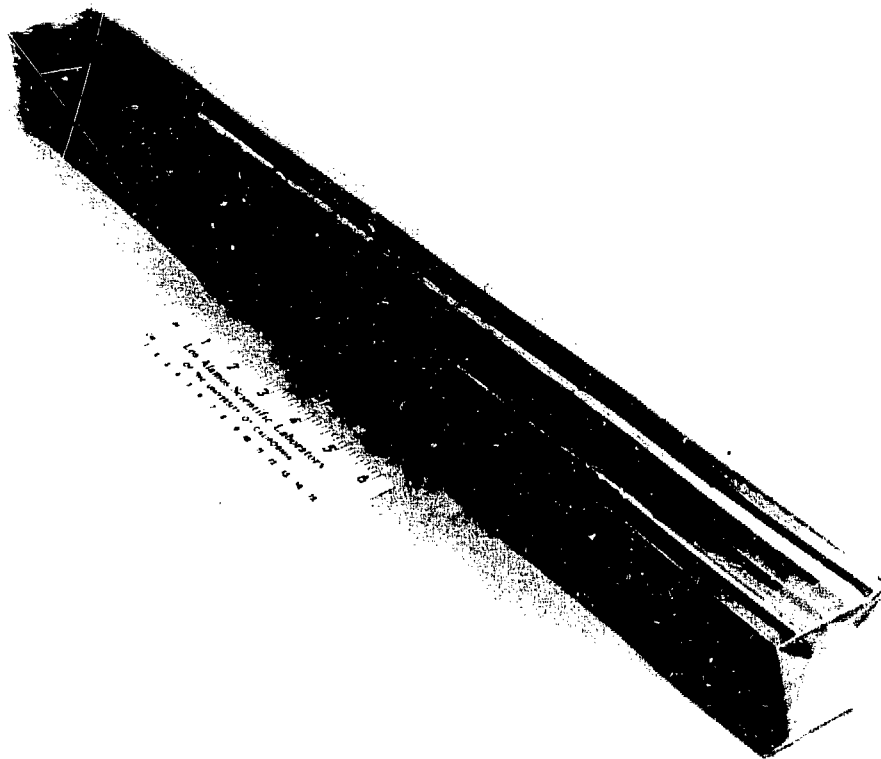


Fig. 3. A typical fuel drawer for a FCA. The drawer contains a number of plutonium metal plates and is part of a large matrix of drawers in the critical assembly.

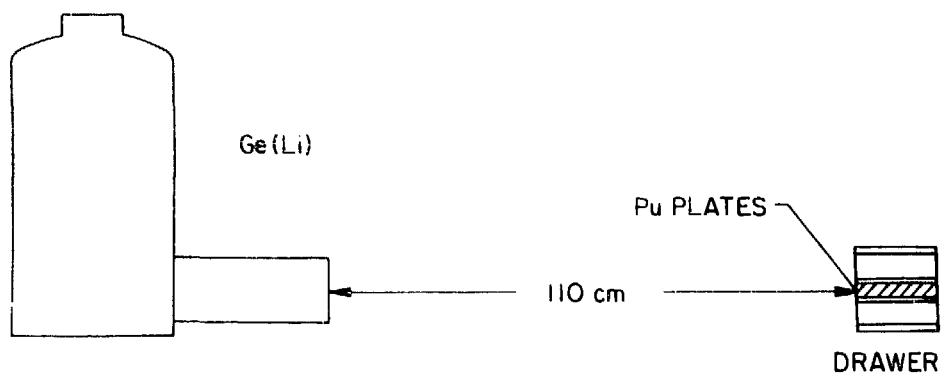


Fig. 4. Schematic of the experimental setup for passive gamma-ray counting of FCA fuel drawers. The fuel plates are viewed edge-on in a far-field counting geometry.

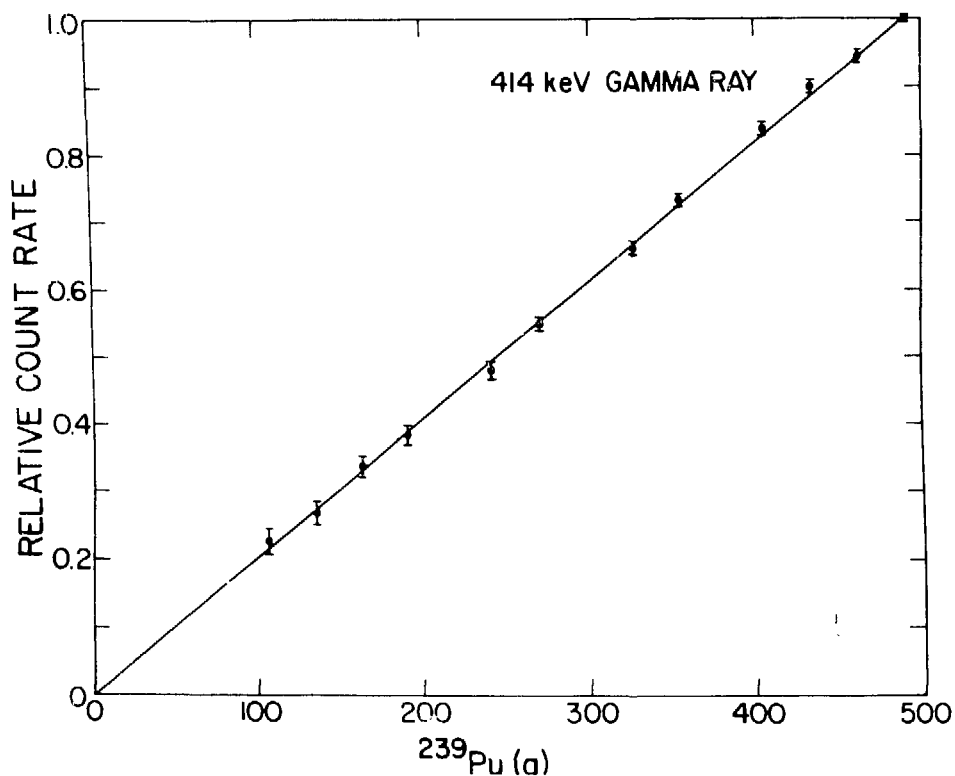


Fig. 5. A plot of the gamma-ray count rate in the experimental setup of Fig. 4 vs the total plutonium content of various FCA fuel drawers, with other experimental conditions kept constant, drawer-to-drawer.

^{241}Pu assay (using the 208-keV gamma ray) can be performed with 10 precisions in the 1% to 2% range in just a few hundred seconds of counting.

The plutonium fuel drawers and certain types of storage canisters can be assayed rapidly by gamma measurements. The main advantage of the gamma assay is that it uniquely identifies the fissile isotopes and can be used to determine the isotopic ratios and (with proper calibration) the isotopic content of the fuel containers. The unique gamma signatures of various plutonium isotopes make plate substitution difficult. However, fuel plates are dense, and the gamma rays give only a verification of the plutonium on the outer surface on the top edge of the plates (see Fig. 5).

III. AUTORADIOGRAPHY

A. Introduction

The exposure of photographic film to the spontaneously emitted x and gamma radiation from fuel elements containing uranium and/or plutonium produces an image of the fuel material that is in close contact with the detecting film. This technique of producing a radiograph of the fuel with its own radiation is called autoradiography. The optical density of the resulting image will depend on the degree of exposure of the film and hence on the amount of radioactive material present near the film and the exposure time.

Autoradiography has been used to study the distribution of radioactive elements in reactor fuel materials.¹¹ More recently, autoradiographic methods have been applied both to inventory verifications of plutonium-containing fast critical assembly fuels^{12,13} and even to verification of LWR fuel assemblies.^{13,14}

B. Autoradiography of Plutonium-Bearing Fuels

The best example of plutonium-bearing fuel in large quantity is in a critical facility in which the characteristics of plutonium-fueled fast-breeder power reactors are studied. The cores involved in these studies may contain as much as 3000 kg of plutonium. They thus present an extremely difficult problem in materials accounting, with many thousands of fuel plates to be verified in a safeguards inspection. These fuel plates are assembled in reactor drawers of the type shown in Fig. 3, and the drawers are loaded into the reactor matrix, which is divided into two arrays of stainless steel tubes (see Figs. 6 and 7). When not in use, the fuel plates are placed in storage canisters for vault storage. An example of such a storage canister is shown in Fig. 8.

Autoradiography of plutonium-bearing fuel uses spontaneously emitted 60-keV gamma rays from ^{241}Am to expose an x-ray film. This technique has been used to verify plutonium plates in ZPPR drawers and storage containers.^{12,14} Exposure time using industrial type x-ray film is 10-15 min for reactor drawers and 30-45 min for vault canisters.

In the ZPPR assembly there is sufficient clearance between the contents of a drawer and the supporting matrix above it to insert a strip of prepackaged film without removing or handling the drawer itself. Figure 9 is a composite photographic print of several autoradiographs performed on fuel drawers inside the reactor. The drawer shown in Fig. 9-A contains a single row of standard 0.13-in.-thick ZPPR fuel plates. Figure 9-B shows a drawer containing two rows of three plates each. In both cases the image of the fuel plate edges is clearly defined. Figure 9-C shows an autoradiograph of a drawer with no fuel

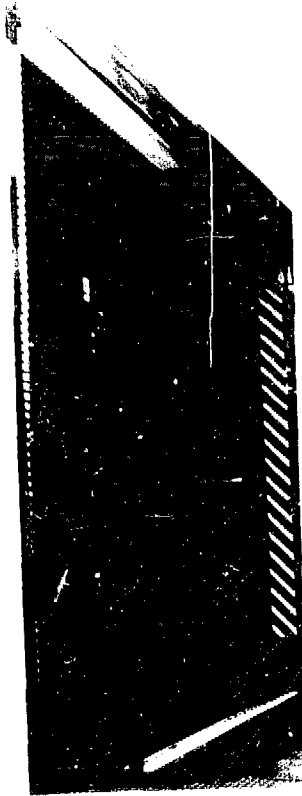


Fig. 6. One-half of a FCA matrix with a few drawers projecting from the matrix tubes.

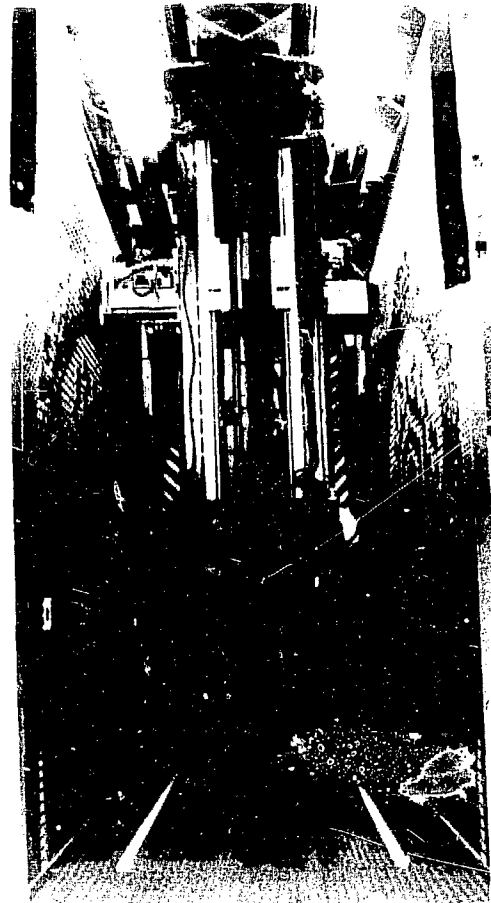


Fig. 7. A view of a FCA with the two halves of the matrix separated. The matrix loader, for automatic fuel drawer handling, is shown between the separated matrix halves.

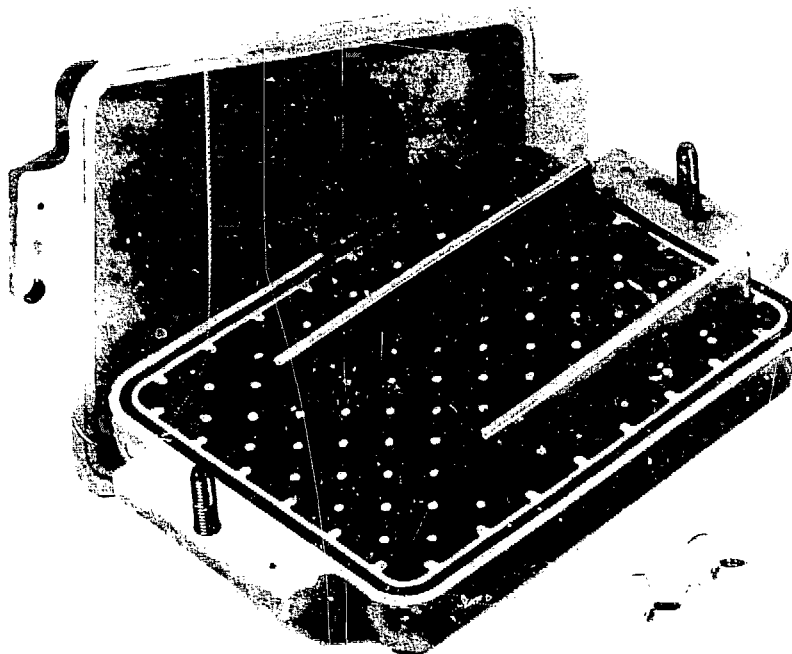


Fig. 8. A fuel-plate storage canister for the FCA at Argonne National Laboratory.

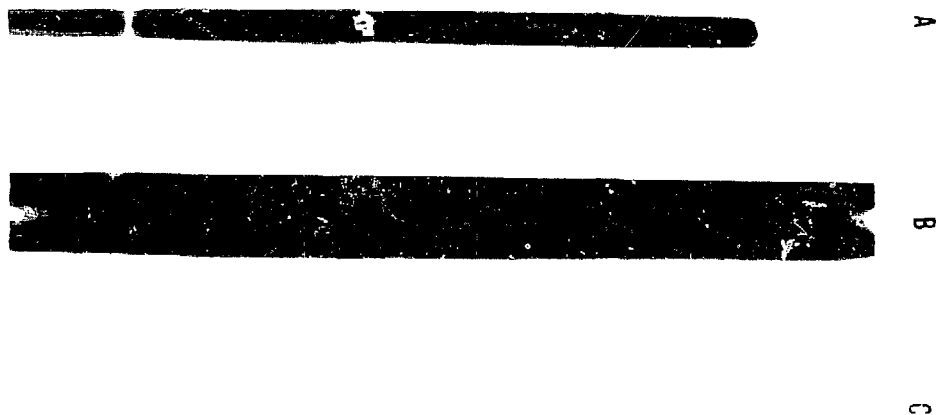


Fig. 9. A composite print of several autoradiographs taken of FCA fuel drawers inside the critical assembly. For details of each autoradiograph, see text.

in it, but with a drawer containing fuel in the matrix position above it. The absence of fuel is apparent. Although a very faint image of the fuel in the drawer above is visible, the steel matrix wall attenuates the gamma radiation so that a significant image is not produced.

For the verification of plutonium plates inside storage canisters (cf. Fig. 8), the film can be placed beneath the aluminum canister without removing the canister from its vault position. Figure 10 shows an autoradiograph of a canister filled with 24 4-in. plates. Each plate image has a contribution from both direct and scattered radiation. The image of the four outside plates is lighter than the 20 interior plates, because plates with only one neighbor have less scattered radiation contributing to their image density. Figure 11 shows a partially filled canister. The missing plate (third from bottom, right side) is easily detected. Figure 12 shows the results of replacing a fuel plate with a stainless steel "dummy" (center, right side). An image of the dummy plate is formed by scattered radiation from its neighbors. However, because the dummy plate emits no radiation, its image is less dense than that of a normal fuel plate. Note that the images of the plates adjacent to the dummy are also less dense than normal because there is no scattered radiation from the dummy. The effect of substituting lead or uranium plates for fuel plates is the same as that shown for stainless steel. Substitution of low-Z material such as aluminum will give a result more nearly resembling a vacancy. Although

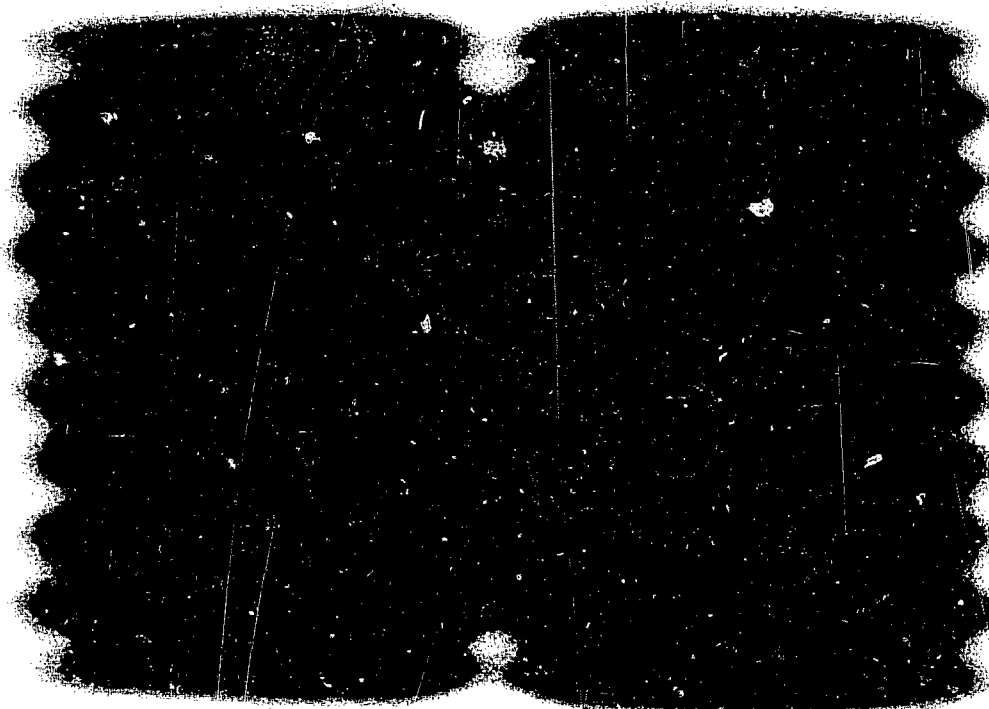


Fig. 10. An autoradiograph of a storage canister with 24 4-in. fuel plates inside. All 24 images are readily apparent, with the outer fuel plates producing fainter images due to reduced scattering effects at the edges of the canister.

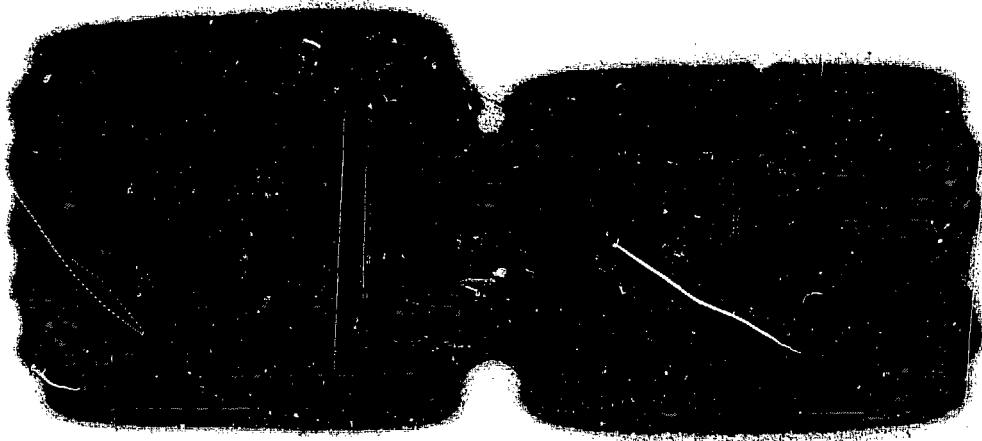


Fig. 11. An autoradiograph of a partially filled fuel plate storage canister in which one plate is missing (third position from bottom on the right side).

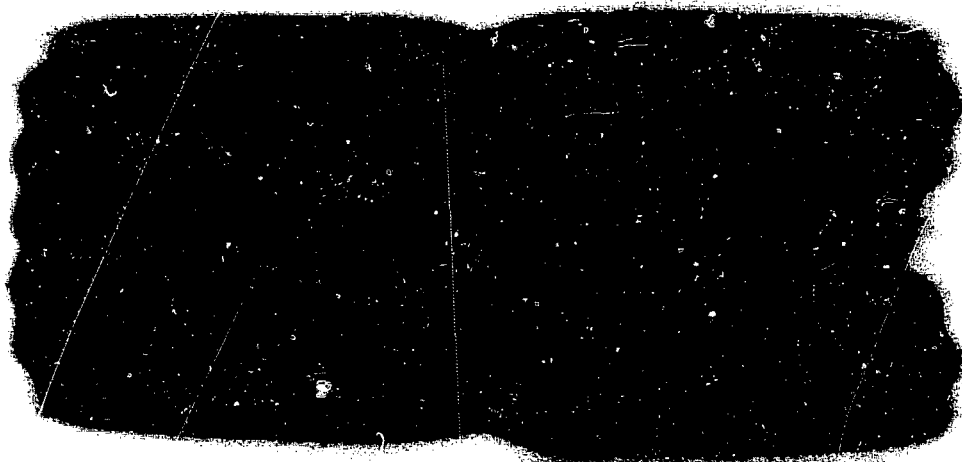


Fig. 12. An autoradiograph of a filled fuel-plate storage canister in which one plate has been replaced with a "dummy" plate (center position, right side).

radiographs of canisters are not quite as clear or as simple to interpret as radiographs of fuel drawers, a trained inspector should be able to determine the number of fuel plates in a canister.

Autoradiography provides a relatively simple and rapid method for identifying plutonium fuel plates. This is an attribute check (nonquantitative) for plutonium contents, and requires a minimum of interference with normal operation. The technique, however, has some limitations. The need for processing the film before it can be read may pose a problem to the inspector. Off-site

processing would seriously reduce the timeliness of this technique. Another disadvantage of autoradiography is that it is essentially a surface effect, and therefore it does not verify the total plutonium content. Furthermore, it does not uniquely identify plutonium but only the presence of a correspondingly intense source of radiation. These disadvantages should not negate the use of autoradiography, but they should serve to emphasize that it must be used in conjunction with other independent verification techniques. However, its demonstrated usefulness in selected measurement situations has led to its recommended inclusion in the safeguards activities at critical facilities.^{15,16}

C. Autoradiography of Uranium-Bearing Fuels¹⁴

At present there are several nondestructive assay techniques that can verify the ^{235}U content of isolated individual LWR fuel rods or of the rods only on the outer periphery of fuel assemblies. This limitation is dictated by the exceedingly low penetrability of the primary gamma radiation from ^{235}U used in gamma-ray-based nondestructive assay (185.7 keV in energy). What is needed, however, is a nondestructive technique for verifying ^{235}U content of the rods in the interior of a fuel assembly. Ease of use and portability in the field are prime concerns in the selection of such a technique.

Two inspection techniques have been investigated for application to LWR fuel assemblies. One technique uses active neutron interrogation of a portion of a fuel assembly and will be discussed in detail in the next section of the course manual (Sec. 11). Application of the second technique, autoradiography, is discussed in this section.

The most serious difficulty in obtaining an autoradiographic image of a single, isolated ^{235}U -containing fuel element is the exposure time required for useful image density. The methods used for obtaining images of plutonium-containing fuel elements require only 15- to 45-min exposures, depending on the source-film distance and the absorber thickness.¹² If these same methods are used for low-enriched ^{235}U -containing fuel elements, these elements would take 200 times longer to produce a given image density than would identical fuel elements containing equal concentrations of plutonium. In addition, the high-energy beta radiation from the ^{234}Pa alpha-decay daughter of ^{238}U contributes to the image density and results in poor sensitivity to ^{235}U concentration.

The problem of lengthy exposures (caused by the low gamma-ray specific activity for ^{235}U) can be overcome by using image-intensifying screens on either side of the high-speed film. Each incident gamma ray can cause fluorescence in the screen's coating material, producing several secondary photons of lower energy than the incident gamma ray. The screens are then used with film having enhanced sensitivity for the lower-energy photons.

The high-beta background problem can be ameliorated by using absorber layers to stop the beta radiation. Thus, the film response is dominated by ^{235}U sensitivity, making the resulting autoradiograph a useful measure of fissile content of the fuel material. Shown in Fig. 13 is a schematic of an autoradiographic film package for uranium-bearing fuel, showing the relative positions of the film, image-intensification screens, and beta-absorbing material. In addition to the precautions taken inside the film packet, it is also advisable to provide some shielding of the film packet from radiation by the adjacent rows of fuel rods in the fuel assembly. To examine only one row at a time, a gamma absorber, such as lead foil, placed between the film and the row of rods not under examination will allow the row of the rods of interest to make a greater contribution to the final image density. Figure 14 shows the configuration of a film cassette and the "row-isolation" shielding in a model fuel assembly.

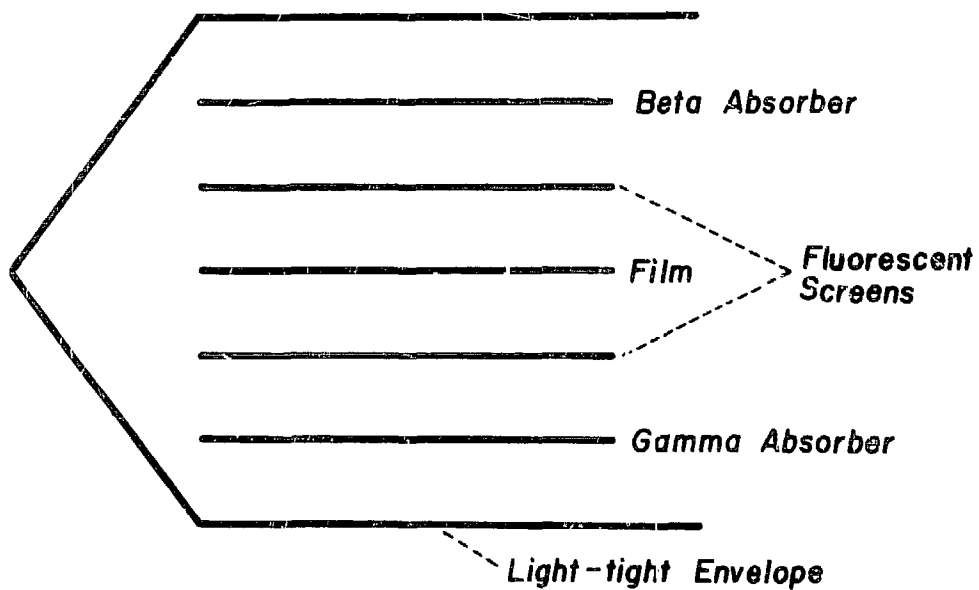


Fig. 13. Schematic diagram of the contents of a film package for the auto-radiography of uranium.



Fig. 14. A uranium autoradiography film package inserted in a model LWR fuel assembly. Note that, in addition to the film package, there is also a packet of lead foil placed to the right of the film. This means that the row of rods behind the film is being measured.

The primary goal in autoradiographic examination of fuel assemblies is to verify that fuel rods have not been removed or replaced with dummy rods, because this would be a symptom of a possible diversion of material. In the model assembly, voids (that is, missing rods) are easily detected. Substitution of "dummy" rods is also readily detected. The substitution of rods containing depleted uranium is also detectable, but only marginally by visual inspection of the developed radiograph. This is shown in Fig. 15, where an autoradiograph of each of the three cases above is shown.

A slightly more quantitative analysis of an autoradiograph can be accomplished with a scanning densitometer, which measures the optical transmission of the exposed radiograph negative. This type of analysis of autoradiographs is depicted in Figs. 16-18. In Fig. 16, an autoradiograph of a pressurized-water reactor (PWR) fuel assembly is shown. The radiographic images of all of the fuel rods in the assembly row next to the film are apparent. The features of the optical densitometry scan (also shown in the figure) clearly point out the location of the individual fuel rods. In addition, one can infer (at least qualitatively) that the enrichment of the rods appears to be relatively uniform, with the caveat that the images of the rods near the end can be fainter than normal as a result of the diminished scattering effects from lack of neighboring rods on both sides. In Fig. 17, an autoradiograph and scanning densitometer trace were shown for another PWR fuel assembly in which "dummy" rods (containing no uranium) have been substituted in three locations in the row being imaged. In Fig. 18, a similar analysis of a BWR assembly is shown, with the features that one rod is missing and one other rod possesses a lower enrichment than nominally present in the row. The scanning densitometer trace demonstrates that the first rod on the left has a lower ^{235}U content than any of the other rods, the effect being in excess of even the expected image deficiency for a rod at the end of the row.

D. Summary

Autoradiography of plutonium-bearing fuels provides a simultaneous piece count and attribute check for fuel elements. The technique is not quantitative, but it does provide assurance that an item of the appropriate size,

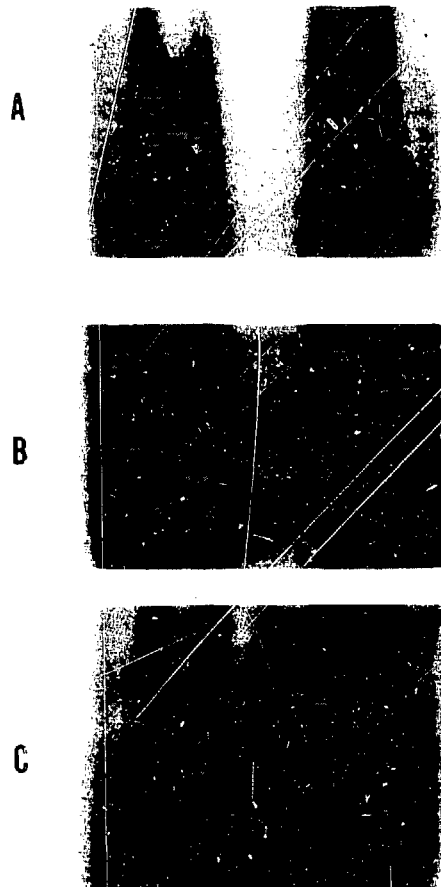


Fig. 15. Autoradiographs of a row of rods in a model LWR fuel assembly: (A) center position void, (B) center position contains "dummy" rod, (C) center position contains depleted uranium rod.

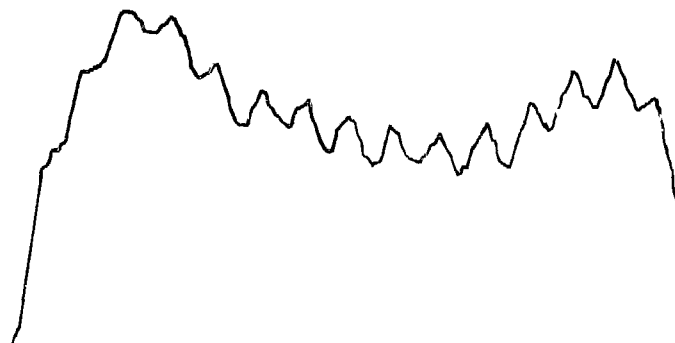
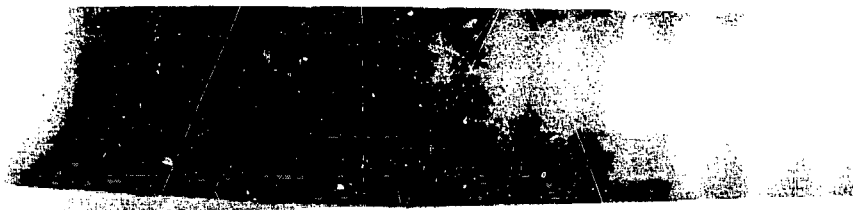


Fig. 16. Autoradiograph and scanning densitometer trace of a row of rods in a complete PWR fuel assembly. Individual rod positions can be clearly discerned, and the less intense outer rod images are caused by reduced scattering effects at the ends of the rows.

C I C

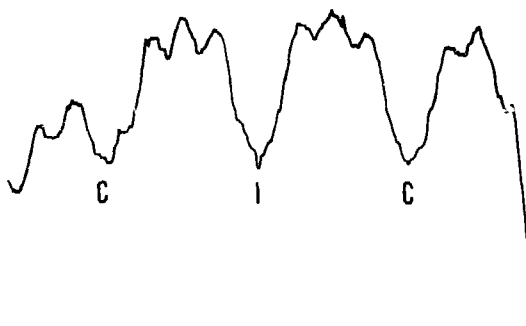
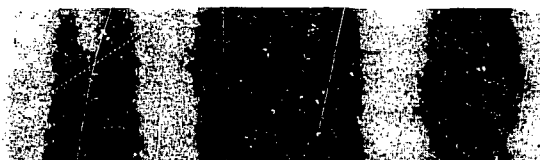


Fig. 17. Autoradiograph and scanning densitometer trace of a row of rods in a PWR fuel assembly. The positions labeled "I" and "C" contain "dummy" rods.

2.14 2.87 2.87 2.87 w 2.87 2.87 2.87



Fig. 18. Autoradiograph and scanning densitometer trace of a row of rods in a BWR fuel assembly. The per cent enrichment of each rod is indicated above the autoradiograph, and row W contains no uranium.



shape, and location (in storage or even in the reactor) contains radioactive material with the qualitatively correct gamma-ray energy profile and specific activity.

Autoradiography can also provide a means of verifying that rods in the interior of uranium-bearing unirradiated LWR fuel assembly do contain ^{235}U . The technique provides a relative indication of ^{235}U content and must be accompanied by an absolute-enrichment measurement for external rods. The autoradiographic technique has the advantage of being simple to implement and is capable of detecting defects such as dummy rods, missing rods, and rods containing depleted uranium. These defects can be detected by visual inspection of the autoradiographs.

Autoradiography should be supported by quantitative nondestructive assay measurements on selected fuel components in any complete inventory or inspection procedure.

V. CONCLUSION

From the discussion presented in this section, it should be clear that gamma-ray-based nondestructive assay/verification techniques for fresh nuclear

fuel provide useful inventory and fuel attribute information. Generally, however, the available gamma-ray techniques cannot provide quantitative information from the full volume of a fuel assembly and so are limited to delivery of qualitative information on the fuel material nearest the detection site. Thus, complete inspections and inventories for fuel management and materials control purposes must also involve other nondestructive measurements such as those involving neutron-based assay techniques.

Efforts continue to develop more compact photon detectors (such as CdTe and HgI₂), which can be used at room temperature and could be employed in quantitative measurements of the interior regions of fuel assemblies.¹⁷ The successes of research efforts in these areas should enhance the capability to obtain the detailed information characteristic of gamma-ray measurements, but over a more extensive and more representative portion of a fresh fuel inventory.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 9, PART II: NEUTRON TECHNIQUES FOR ASSAY/VERIFICATION OF UNIRRADIATED URANIUM AND MOX FUEL

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I. INTRODUCTION

Quantitative nondestructive assay (NDA) methods are necessary for the verification of special nuclear material (SNM) inventories of unirradiated fuel components, including fuel rods, trays, plates, and assemblies. International Atomic Energy Agency (IAEA) inspectors require NDA instruments to measure fuel materials in product form. These instruments must be adapted to the specific fuel geometries and conditions found in the fabrication facilities. The specific NDA measurement techniques described here quantitatively determine the composition of uranium and plutonium fuels by detecting and recording neutrons emanating from the fuel material.

The preceding paper by H. Smith covered the use of gamma-ray techniques and thus this paper will focus on neutron techniques for verification.

NDA methods are grouped into two major categories, passive and active. Passive assays use naturally occurring gamma or neutron radiation as direct signatures of the nuclear materials being assayed. Active assays first interrogate the material with a neutron or gamma-ray source to induce fissions and then measure the resulting prompt or delayed radiation. During the past decade, these techniques have been developed and demonstrated under field conditions so that a variety of instruments exist that are sufficiently reliable and accurate to satisfy safeguards needs. In many cases, the instruments are portable or, at least, transportable.

This paper deals with the various measurement problems listed in Table I and the NDA techniques that have been used to determine the SNM content. Most of these techniques have been developed to the stage where commercial equipment is now being used in the fabrication facilities. More recently, IAEA inspectors have been using portable equipment to verify the operator's declaration on the SNM content. The fuel categories to be considered include pressurized water reactors (PWR), boiling water reactors (BWR), material test reactors (MTR), high-temperature gas-cooled reactors (HTGR), fast breeder reactors (FBR), light-water reactors with mixed oxide fuel (LWR-MOX), and fast critical assemblies (FCA).

II. ASSAY SYSTEMS AND IMPLEMENTATION

For all of the fuel categories, I will describe the measurement system in terms of the method, typical hardware, and operating experience.

A. LWR Fuel Rods

The measurement systems for both PWR and BWR fuel rods are basically the same.

TABLE I
FUEL MATERIALS WITH NDA APPLICATIONS

<u>Fuel Type</u>	<u>Rods</u>	<u>Trays</u>	<u>Plates</u>	<u>Assemblies</u>
PWR	x			x
BWR	x			x
MTR			x	x
HTGR		x		
FBR	x	x		
LWR-MOX	x	x		x
FCA		x	x	

1. Measurement Method. Most of the fabrication plants that produce LWR fuel measure the rods for both total fissile content and pellet-to-pellet uniformity using active neutron techniques. Typically, a ^{252}Cf neutron source irradiates the fuel rod with thermal neutrons, and the resulting induced fission activity is counted using NaI detectors or plastic scintillators for the gamma rays and ^3He or ^4He detectors for the neutrons. The measured signal rate as the rod is passed through the system is proportional to the ^{235}U content per unit length.

2. Assay Equipment. Figure 1 shows a schematic diagram of the Pin and Pellet Assay System (PAPAS)^{1,2} that was developed at Los Alamos in 1970 for the assay of LWR fuel rods. This was a forerunner of many of the commercial units that came into use during the mid and late 70s. The unit contained a ^{252}Cf source ($\sim 500 \mu\text{g}$ of ^{252}Cf), moderator materials including D_2O , Pb, C, and CH_2 , and NaI and ^4He detectors. Several fuel-rod channels were used to increase the throughput to handle the entire output of the plant. The complete system is shown in Fig. 2. Newer versions of the system used fewer fuel-rod channels with multiple (2-4) gamma-ray detectors on each channel to obtain equivalent throughput. This simplified the rod handling, calibration, and crosstalk between channels.

For the total ^{235}U content measurement with the PAPAS, the fission neutrons are counted by two ^4He detectors (per channel) biased above the majority of the ^{252}Cf background neutrons and all of the associated gamma rays. Four 4-m rods require only 37 s to pass through the assay unit simultaneously. After specially prepared standard rods are used to calibrate the fission neutron response, the ^{235}U content of production rods may be assayed directly by comparing the measured response to a calibration curve.

3. Application. A great deal of effort has been put into the calibration of fuel-rod scanners at the different fabrication facilities. The magnitude of this effort is increased because of the variables that influence the response from a thermal-neutron interrogation system. For high-accuracy work (better than 1%), it is necessary to match the rod diameters, density, and

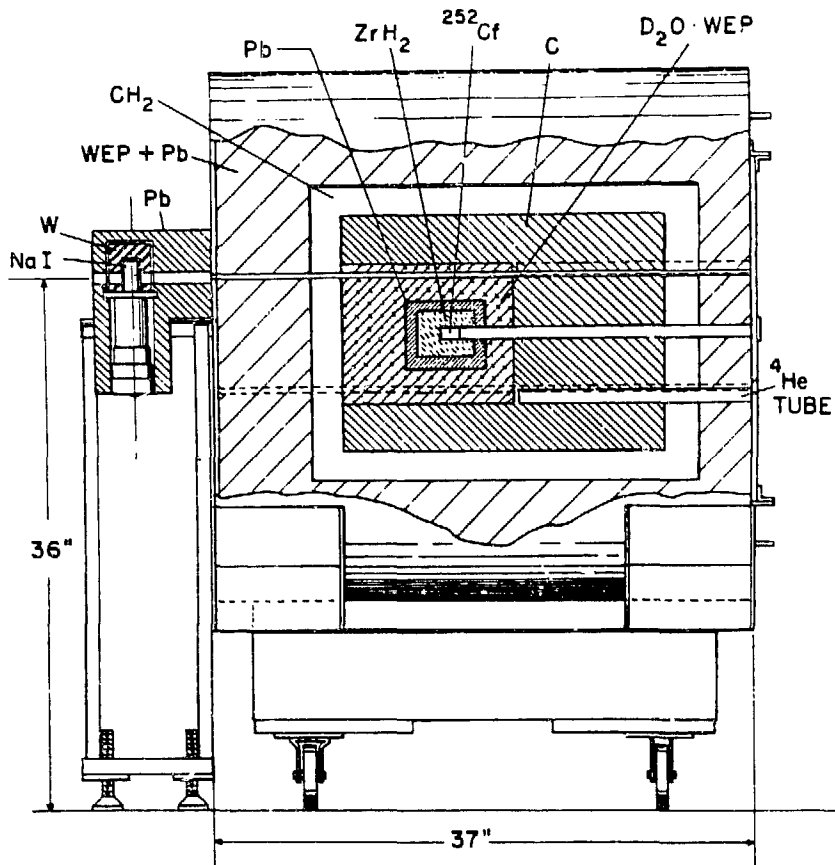


Fig. 1. Schematic of the PAPAS. The direction of rod travel is from right to left.

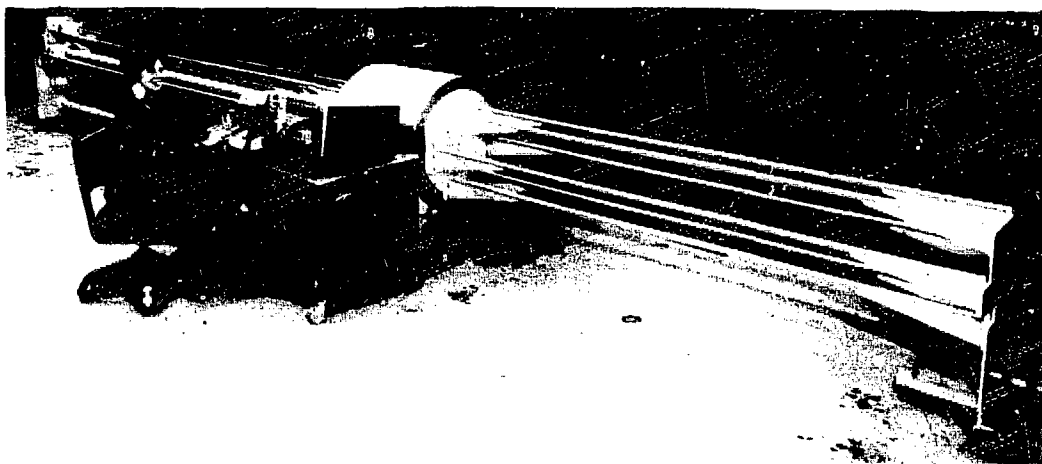


Fig. 2. Complete ^{252}Cf fuel-rod assay system with six-channel rod-handling mechanism and automated data acquisition, reduction, and printout. Assay accuracy $<1\%$ for fissile content of power reactor fuel rods; assay time ~ 37 s for six simultaneous channels.

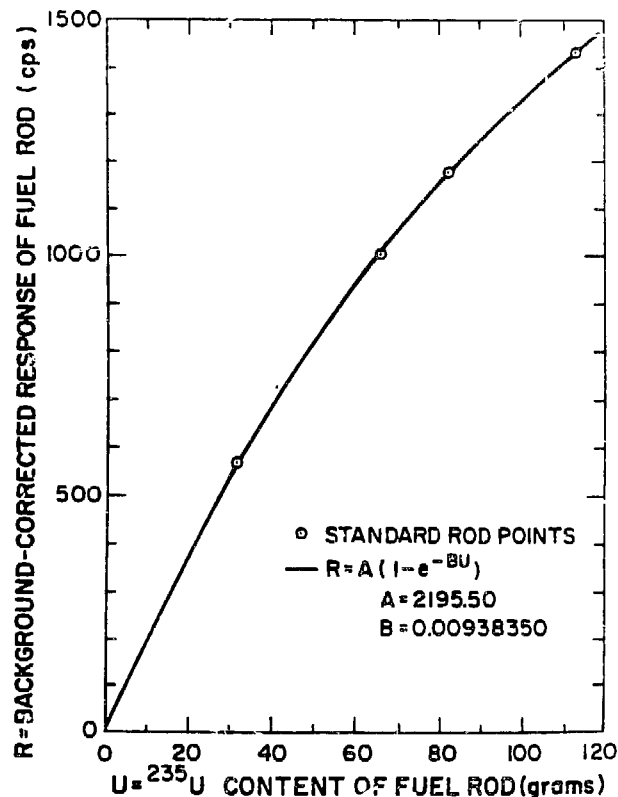
cladding with the standard rods. This is normally done and the resulting assays are good to better than 0.5%. A typical calibration curve for ^{235}U enrichment vs induced fission response is shown in Fig. 3. The ^{235}U content can be obtained using either integral neutron counts or gamma-ray counts with better accuracy being obtained with the neutron method.

For pellet-to-pellet uniformity measurement, a small detector with high counting rate is needed to obtain the high throughput and spatial resolution needed for assaying individual pellets. This is normally accomplished using a through-hole NaI(Tl) detector to monitor the delayed gamma rays. When the fuel rod is moved through the interrogation and counting system (see Fig. 1) with a constant velocity, the delayed gamma-ray counting rate in the NaI is proportional to the ^{235}U loading in the pellets. Figure 4 shows a measured profile for a test rod that had out-of-specification pellets as indicated in the diagram. The scan was for a 3.3% PWR rod moving at 4 cm/s past a 100- μg ^{252}Cf source. Each point represents the total counts accumulated in a 0.4-s interval, and the error bars represent 2 σ statistical uncertainties.

Almost all of the LWR fuel fabrication plants in the world now employ fuel rod scanners based on the technique illustrated in Fig. 1. The primary purpose is fuel quality control, with safeguards being a by-product. With a properly operating system, the accuracies are in the range of 0.2-0.5% (1 σ) for ^{235}U content and $\pm 10\%$ (3 σ) for single-pellet fissile uniformity.

The IAEA is now investigating methods to authenticate fuel-rod scanners for the verification of SNM inventories in fabrication facilities. The normal approach for the authentication is to check the enrichment of a small subgroup

Fig. 3. Typical PWR calibration curve with standard rod points for the ^{252}Cf fuel-rod assay.



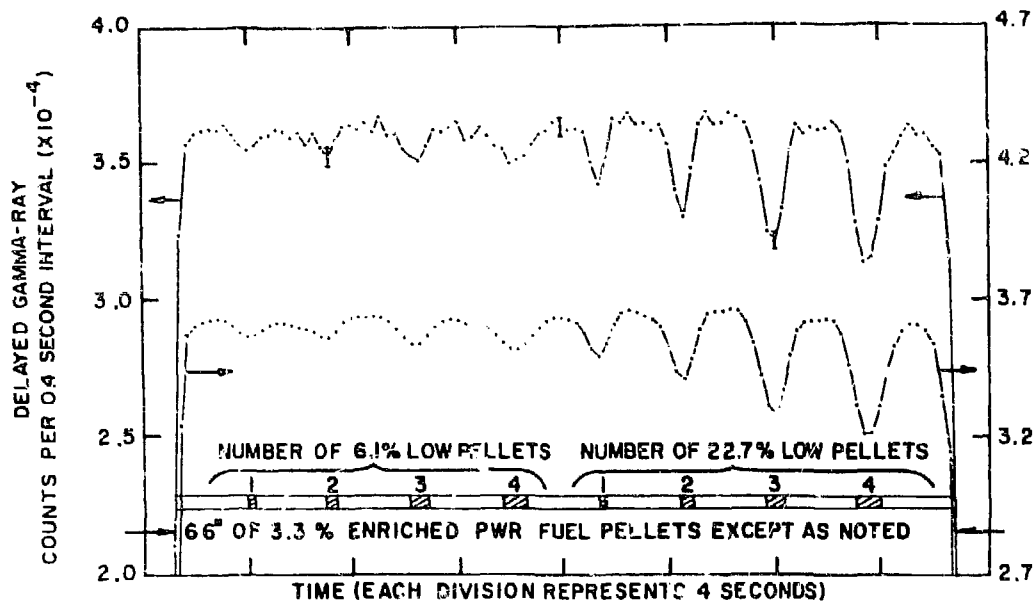


Fig. 4. A typical delayed gamma-ray scan of a 66-in.-long 3.3% PWR fuel rod with pellets of lower enrichments interspersed as shown above. The lower curve is a smoothed version of the raw data in the upper curve (the error bars represent 2σ uncertainties). Each point represents the total counts accumulated in 0.4 s for a rod feed rate of 8 ft/min.

of the rods that have been previously measured in the rod scanner. This can be done using the SAM-11 (or B-SAM),³ or with newer inspection equipment such as the portable mini-MCA⁴ or the Active Well Coincidence Counter (AWCC)⁵ with a special fuel-rod support insert (Fig. 5).

B. LWR Fuel Assemblies

For safeguards purposes it is of high interest to measure the full fuel assemblies because that is the output product from the plant and the input to the reactors. Often the enriched uranium will be transferred from one installation or country to another in the form of fuel assemblies.

1. Measurement Method--Coincidence Collar. An active neutron interrogation technique (the Coincidence Collar)⁶ has been developed for the measurement of the ^{235}U content in fresh fuel assemblies. The method employs an AmLi neutron source to induce fission reactions in the fuel assembly and coincidence counting of the resulting fission reaction neutrons. This coincidence counting eliminates the undesired neutron counts from the random AmLi interrogation source and room background. When no interrogation source is present, the passive neutron coincidence rate gives a measure of the ^{238}U through the spontaneous fission reactions. The system can be applied to the fissile content determination in BWR, LWR, and other type fuel assemblies for accountability, criticality control, and safeguards purposes.

Active neutron systems using thermal neutron interrogation, such as the Coincidence Collar, have neutron self-shielding problems that limit the sensitivity in the interior of the assembly, but the present Coincidence Collar gets



Fig. 5. AWCC system that was modified with a central CH_2 moderator and fuel-rod support fixture to permit the active assay of up to three LWR fuel rods.

The AmLi interrogation source produces random neutrons (not time correlated) with an average energy of ~ 300 keV. The Coincidence Collar uses a source fabricated by Monsanto Research Corp. with a yield of $\sim 4 \times 10^4$ n/s.

The electronics used with the Coincidence Collar are the same as that used with the High-Level Neutron Coincidence Counter (HLNCC).⁷ This unit contains the high- and low-voltage power supplies, six amplifier-discriminator lines, a microprocessor, and the shift-register coincidence logic. The electronic unit is directly interfaced to an HP-97 programmable calculator. The microprocessor is used to read out the run time, total counts, reals plus accidental counts, and accidental counts to the HP-97. The HP-97 is used to reduce the data using the software program selected by the operator.

The complete assay system is shown in Fig. 7, including the detector body, the electronic unit, the HP-97 calculator, and a support cart. For applications, the cart is moved next to a fuel assembly as shown in Fig. 7. The back detector bank of the unit is hinged to aid in positioning the system around the fuel assembly.

3. Application. Tests and evaluations of the Coincidence Collar have been performed at both PWR and BWR reactor fabrication facilities. Active-mode interrogation to measure ^{235}U content and passive-mode coincidence counting to

around this limitation by fast-neutron multiplication, which is higher in the central region. The multiplication effect is enhanced by the coincidence counting because of the increase in the effective number of time-correlated neutrons emitted by the sample when multiplication occurs. In effect, the system works like a reactivity gage for the fuel assembly, and the removal of fissile material from the assembly lowers the neutron reactivity and thus the coincidence response.

2. Assay Equipment. The Coincidence Collar consists of three banks of ^3He tubes and an AmLi source imbedded in a high-density polyethylene (CH_2) body as shown in Fig. 6. The 18 ^3He neutron detector tubes (4 atm pressure) are 2.54 cm in diameter and 33 cm long (active length).

The CH_2 body performs three basic functions in the system: (1) general mechanical support, (2) interrogation source neutron moderation, and (3) slowing down of induced fission neutrons prior to their detection in the ^3He tubes. For inspection applications, it is desirable to make the system portable. The weight of the detector system is ~ 24 kg.

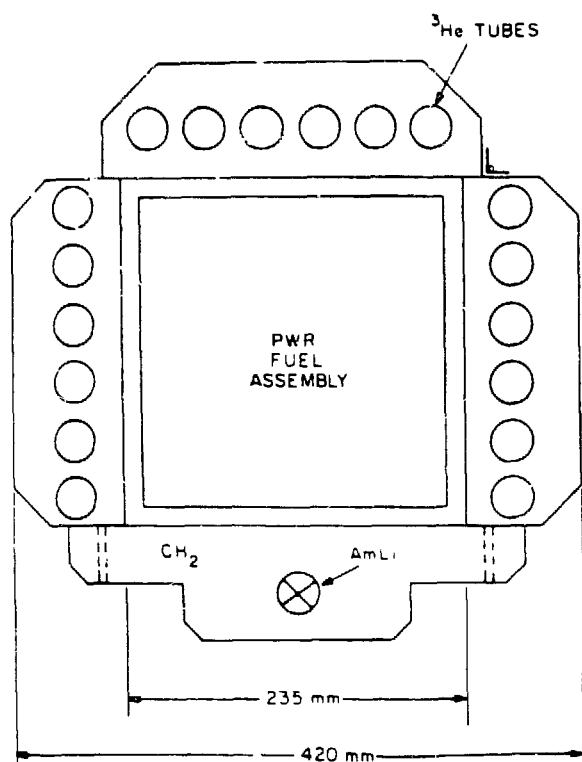


Fig. 6. Schematic diagram of Coincidence Collar showing the AmLi neutron source and the ^3He detector banks. The top detector bank neutron source pivots open to accommodate PWR, BWR, or HWR fuel assemblies.

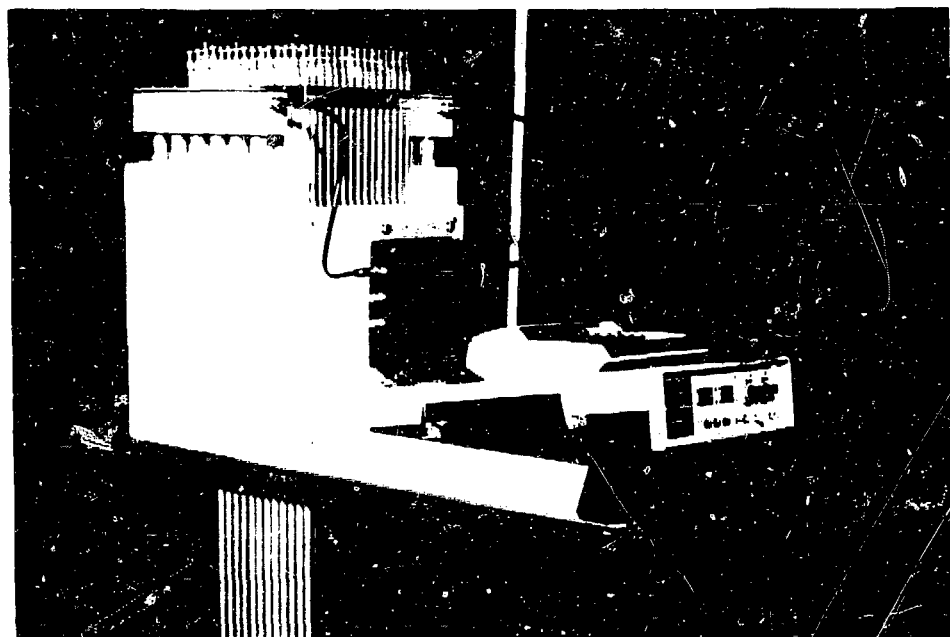


Fig. 7. Coincidence Collar system positioned around a mockup PWR fuel assembly.

determine ^{238}U content were performed. The Coincidence Collar measures the ^{235}U or ^{238}U content per unit length, which is proportional to the enrichment for a given type of assembly. The sampled region is 400 mm long, centered in the midplane of the detector body.

When an overhead crane is available for scanning the fuel assembly through the detector, then the entire assembly can be sampled. The measurement time is the same for the scanning or stationary modes for equivalent statistical precision. If the scanning mode is used, the calibration curve should be obtained in the same manner. This will take into account the end losses as the assembly enters and leaves the detector. The counting rate in the midsection of the assembly is the same for both the scanning and stationary modes of operation.

A series of measurements were performed using full-size (17 by 17 rods) PWR assemblies with enrichments ranging from 1.8 to 3.4% ^{235}U . The thermal-neutron interrogation is saturated for all of the fuel assemblies; however, the measured response continues to increase as a function of enrichment, as shown in Fig. 8, because the fast-neutron multiplication increases with increasing enrichment.

Measurements were performed where rods from different sections of the assembly were removed to determine the position sensitivity to rod substitution. The regions selected for the measurements are shown in Fig. 9, which is a cross-section of the rod positions in the PWR array. For each rod configuration, several 1000-s measurements were performed and the coincidence rate was compared with the full array count rate to determine the perturbation caused

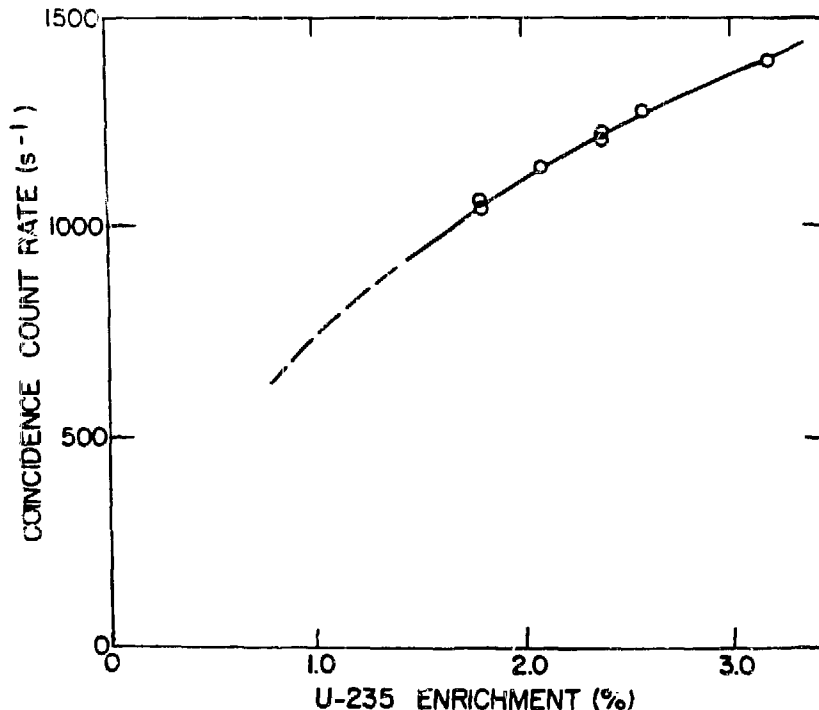


Fig. 8. Active neutron coincidence response as a function of enrichment for 17 by 17 PWR assemblies at the FBFC fabrication facility.

by the substitution. The measured perturbation was then divided by the number of rods in the substitution region to obtain the perturbation per rod.

Table II gives the results of the sensitivity measurements for the empty rod and iron rod substitution cases. The sensitivity limit is defined as the minimum number of rod substitutions that can be detected in a 1000-s measurement time at the 95% (2 σ) confidence level. The detection limit for empty rod substitution has an average value of 2.8 rods and is rather uniform at all rod positions including the central region. The iron rod substitution sensitivity is somewhat better (2.2 rod average) and there is high sensitivity on the front rows because of thermal-neutron absorption.

In summary, the statistical precision for a 1000-s run varied from 0.6-0.9% (1 σ), depending on the type assembly. For longer counting periods, the ultimate precision was about 0.1% for repeat runs with a fixed geometry.

The counting rates for the passive measurements are much lower than for the active case. The statistical standard deviation was $\sim$ 1.4% (1000 s), but the results were influenced by neutron-background variations.

The response curve is not saturated and continues to increase as the enrichment increases through the normal range of LWR fuel. Relative loading variations as small as 1.9% can be detected in a measurement time of 1000 s. Longer measurements can further reduce the statistical uncertainties.

The Coincidence Collar can be applied to the fissile content determination in BWR and LWR fuel assemblies as well as highly enriched uranium (HEU) fuel assemblies for accountability, criticality control, and safeguards purposes.

C. FBR Fuel Pins

1. Measurement Method. Fuel pins for FBRs are different than LWR rods in that there is PuO₂ mixed with UO₂, the diameters are smaller ($\sim$ 5 mm), and

TABLE II

FUEL-ROD REMOVAL DETECTION SENSITIVITY FOR A 15-BY-15 PWR ASSEMBLY

Rod Location ^a	Per Cent Change/Rod		Rod Detection Limit (2 σ) ^b	
	Empty	Iron	Empty	Iron
1	0.35	1.68	3.6	1.0
2	0.64	1.25	2.0	1.0
3	0.41	0.71	3.1	1.8
4	0.41	0.62	3.1	2.0
5	0.55	0.53	2.3	2.4
6	0.57	0.51	2.2	2.5
7	0.53	0.44	2.4	2.9
8	0.36	0.38	3.5	3.3
9	0.48	0.60	2.6	2.1
10	0.46	0.47	2.7	2.7
Average =	0.48	0.71	2.8 Rods	2.2 Rods

^aRod removal locations are shown in Fig. 9.

^bRod detection limit corresponds to the perturbation per rod removal being equal to twice the standard deviation for a 1000-s measurement.

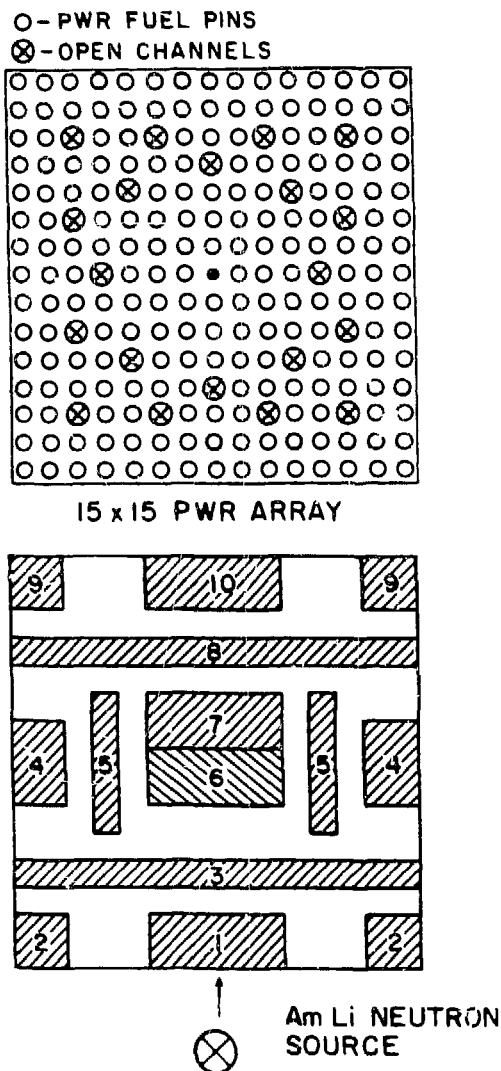


Fig. 9. Cross section of the mockup PWR fuel assembly showing rod locations and regions selected for rod substitutions.

2. Assay Equipment. A schematic diagram of the FBR fuel pin assay system is shown in Fig. 11. The fuel pins are activated by the ^{252}Cf source ($\sim 600 \mu\text{g}$) prior to counting by the gamma-ray detectors. The 12.7- by 12.7-cm NaI detectors have through holes with collimators to give single-pellet resolution. Figure 12 shows the complete scanning system that includes the detectors, source and shield, fuel pin transfer system, and electronics for data collection and analysis.

The neutron moderator for the ^{252}Cf source is designed to give a fast-neutron irradiation (the irradiation channel is lined with cadmium), as well as a high fissile/fertile ratio (400/1).

3. Applications. The FBR pin scanning system shown in Fig. 12 has been used at Hanford Engineering Development Laboratory (HEDL) since 1972. All of

the fissile density is about an order of magnitude higher ($\sim 25\%$ vs 2.5% enrichment). Because of this high fissile concentration, thermal-neutron interrogation does not penetrate to the interior of the FBR pins and LWR fuel-rod scanners cannot be directly applied. The plutonium self-shielding of the thermal neutrons causes the response function to saturate at about 20% enrichment.

To deal with this problem, fast-neutron interrogation systems have been developed specifically for FBR pins. The fast neutrons penetrate the entire pin giving an induced fission rate that is proportional to the fissile content. For the assay system⁹ developed for the Fast Flux Test Facility (FFTF), the induced fission reactions are measured by counting delayed gamma-rays that have energies above 1.2 MeV . This eliminates most of the plutonium passive gamma rays and also ensures low attenuation of the gamma-ray signal. This high-energy gamma-ray signal is used to get the total fissile content in the pin; and in addition, the low-energy ($60\text{--}500 \text{ keV}$) gamma rays are used to monitor the pellet-to-pellet uniformity for quality control purposes. Figure 10 shows the gamma-ray profile collected with the NaI(Tl) detector. Both the active (total fissile) and passive (pellet uniformity) measurements are performed simultaneously.

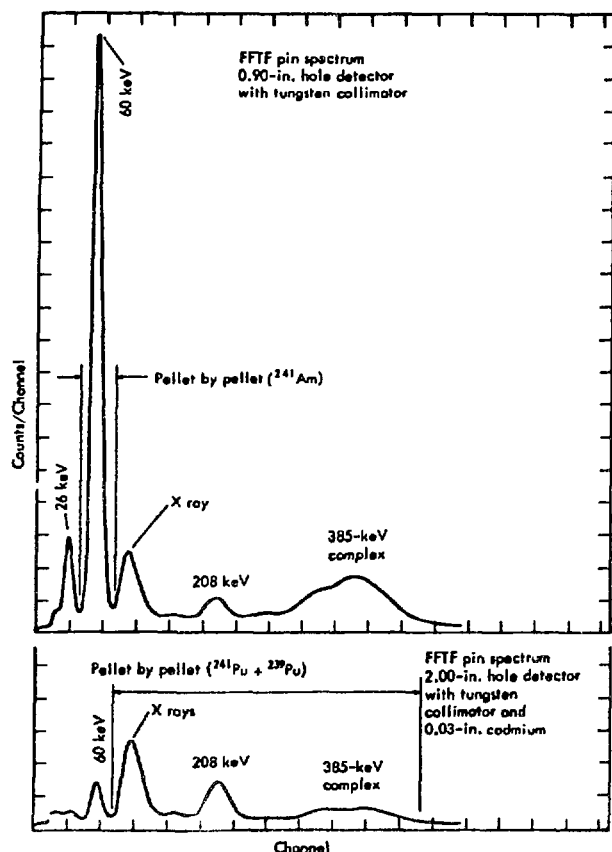


Fig. 10. Gamma-ray spectra from unirradiated FFTF fuel pins positioned in tungsten collimators in 5- by 5-in. NaI through-hole detectors.

the fuel pins (several hundred thousand) for the FFTF core have been measured with the instrument.

During operation, up to ~30 fuel rods are placed in the loading magazine of the fuel-rod translator. The automated translator picks up the rod to be assayed and moves it through the NaI crystals and the ^{252}Cf assembly at a rate of 6.4 cm/s in order to take a background count of the unirradiated rod. The direction of travel is then reversed and the rod is withdrawn at 1.0 cm/s, during which time the delayed fission gamma-ray data for the total fissile assay are acquired and the pellet-to-pellet scans are made. After the irradiation and scan of the fuel column, the rod is unloaded in the tray directly below the loading magazine and the cycle is repeated for the next rod.

A typical calibration response curve is shown in Fig. 13. The scan time for a single pin is less than 5 min and a statistical precision of 0.3% is obtained. Using calibration standards prepared by HEDL, the instrument has been used to measure all of the fuel pins received by the reactor facility. The accuracy of the plutonium assay for a batch of pins is better than 0.2%, which was determined by comparison with chemical analyses.

To determine the pellet-to-pellet uniformity, a test pin was fabricated by HEDL to help evaluate the accuracy of the scanning system. This pin contains various combinations of different plutonium pellet enrichments, as shown in the bottom of Fig. 14. The top curve in Fig. 14 corresponds to a recorder trace of the 60-keV energy window, and the bottom curve corresponds to the 100- to 500-keV energy window. The small enrichment variations (1.7 to 3.1% relative) can be easily observed in the top curve but not so readily in the bottom curve.

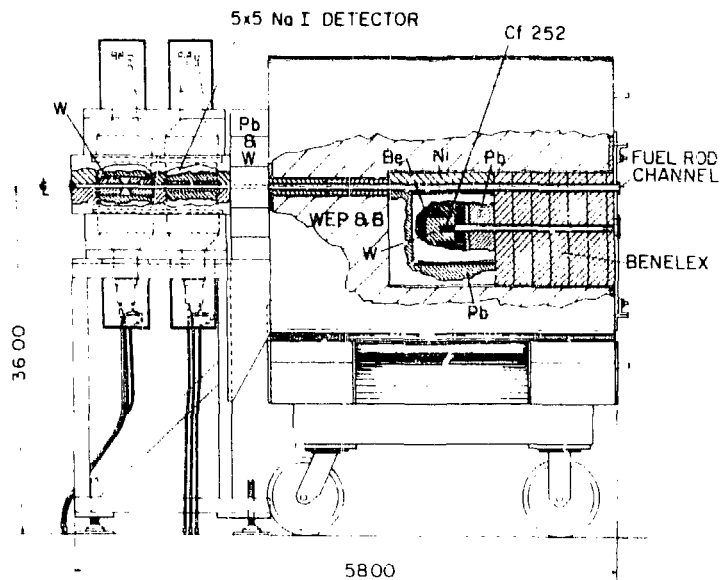


Fig. 11. Schematic diagram of the ^{252}Cf fast-neutron assay system for the FBR-type fuel rods. The delayed gamma rays induced by the fast-neutron irradiation are subsequently counted with the two NaI detectors, which also measure the passive gamma rays to determine pellet-to-pellet uniformity.

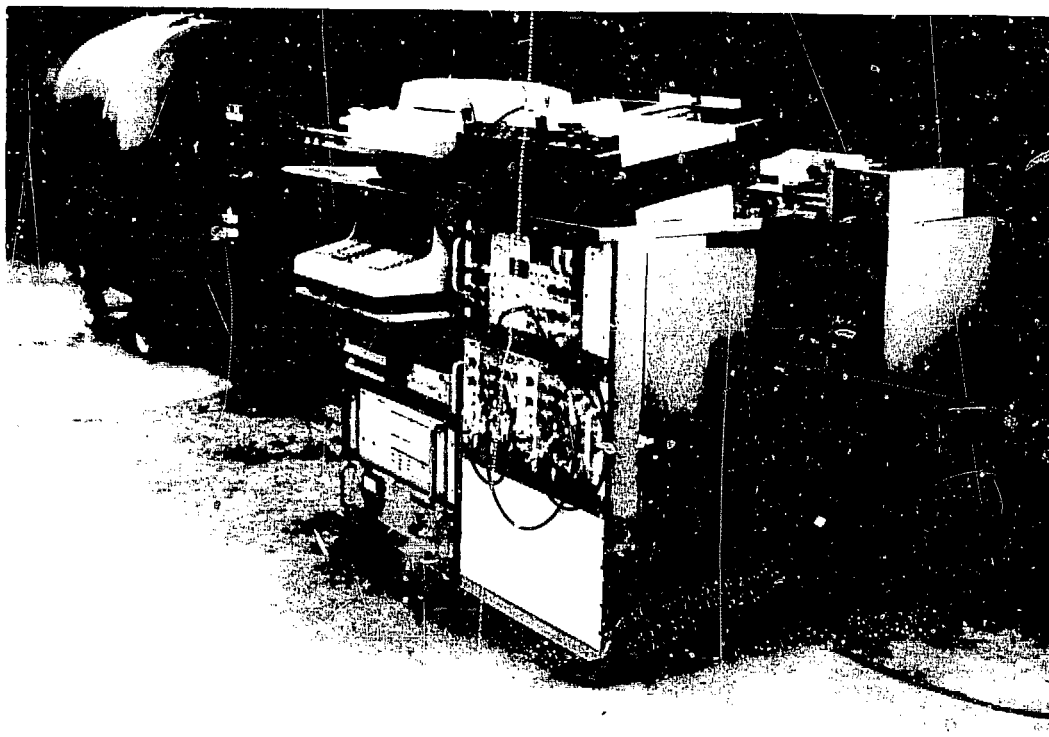


Fig. 12. Fast-neutron ^{252}Cf assay system for FFTF fuel rods. System includes 600- μg ^{252}Cf source and shield, two 5- by 5-in. NaI detectors to count the delayed gamma rays, automated fuel-rod handling, and data-reduction system.

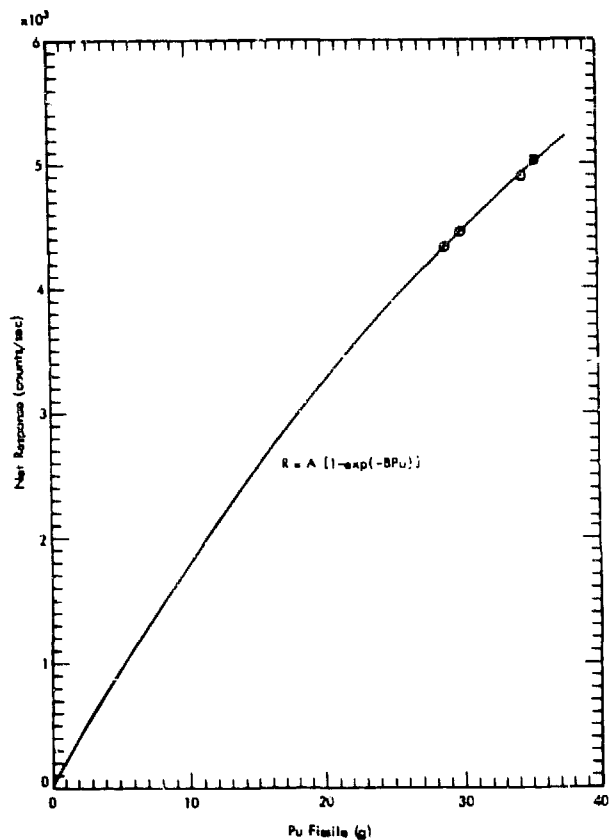
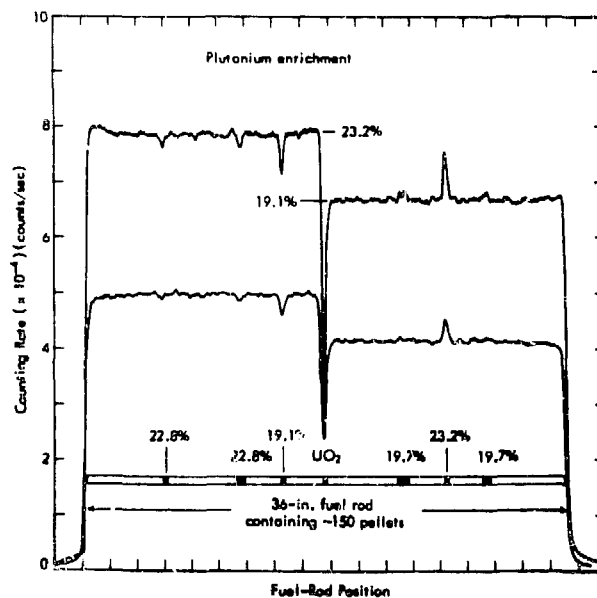


Fig. 13. Calibration curve for fast-neutron ^{252}Cf assay system using standard FFTF fuel pins at HEDL.

Fig. 14. FFTF fuel pin with various combinations of plutonium enrichments for pellet-to-pellet scanning. Top curve corresponds to 100- to 500-keV window.



Plutonium recycle fuel rods for LWR reactors have approximately the same plutonium fissile loading/unit length as the FFTF fuel rods. Thus, the signal and background rates for plutonium recycle rods in the present system would be roughly the same as for FBR-type fuel rods.

D. FBR Fuel Subassemblies

1. Method. Recently, a technique has been developed to assay complete FBR subassemblies using passive neutron coincidence detectors. This passive neutron measurement gives the ^{240}Pu -effective⁷ content in the assembly. The even isotopes of plutonium spontaneously fission and the time correlated neutrons are counted using the coincidence electronics that was developed for the HLNCC.

More recently, a Passive Coincidence Collar¹⁰ has been developed for the passive/active assay of fuel assemblies containing plutonium. In the passive mode, the neutrons originating from spontaneous fission reactions are measured using normal neutron coincidence counting to determine the ^{240}Pu -effective. In the active mode, the passive neutrons are reflected back into the assembly to induce fission reactions in the fissile component of the fuel. To determine the fraction of the neutrons due to the reflection process, the albedo of the boundary surrounding the assembly is changed by inserting and removing a cadmium liner.

Both the coincidence rate (R) and totals rate (T) are measured with and without the cadmium absorber. The normal passive-mode calibration curve corresponds to R vs ^{240}Pu -effective, and it is generally necessary to make corrections for the multiplication from the fissile component. Various techniques have been used to make this correction, but the present cadmium ratio determination gives a more direct measure of the fissile component and multiplication than have past procedures.¹¹

The induced fission rate from the reflected neutrons is related to the quantity

$$R(\text{no Cd}) - R(\text{Cd}) \equiv \Delta R ;$$

however, the value of the induced coincidence counts is also proportional to the neutron source strength, which is different for each subassembly. To remove the source strength from the response function, one divides by T to obtain the quantity $\Delta R/T$, which is proportional to the fissile content independent of the source strength.

2. Assay Equipment. Figure 15 is a photograph of the cylindrical coincidence counter that has been used for the measurement of FFTF fuel subassemblies. The unit consists of 12 ^3He tubes placed in a CH_2 annulus for neutron moderation. The active length of the detectors is 1.21 m so that the entire plutonium region is contained inside the counter. The absolute efficiency of the counter is 8%.

The electronics for the system is the same as that used for the HLNCC⁷ and Coincidence Collar⁶ mentioned previously.

For both the passive and active measurements for MOX fuel, the Passive Coincidence Collar is used. Figure 16 shows a schematic diagram of the unit, which is similar to the Coincidence Collar, except the side containing the AmLi neutron source has been replaced by a fourth detector bank, and removable cadmium liners have been placed between the detector and the fuel assembly. A detector head is shown in Fig. 17. The return of thermal neutrons into the



Fig. 15. Passive neutron coincidence counter used for the measurement of FBR fuel sub-assemblies.

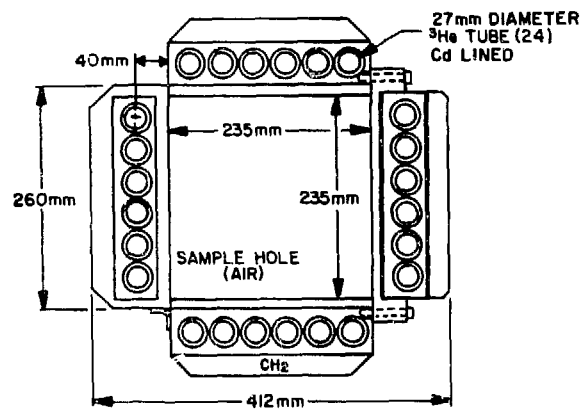
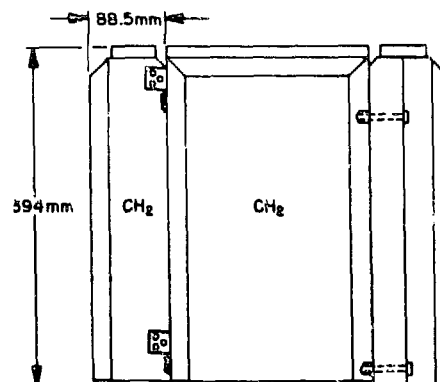


Fig. 16. Schematic diagram of Passive Coincidence Collar used for the passive/active assay of FBR and LWR-MOX fuel assemblies.



assembly is prevented by the insertion of the cadmium liners (0.4 mm thick) between the wall and the assembly.

3. Applications. The cylindrical coincidence counter (Fig. 15) has been evaluated¹ for the assay of FFTF fuel subassemblies. The counter was placed on a vertical stand and the subassemblies were lowered into the detector using an overhead crane. The statistical precision for a 1000-s count was 1% for the ^{240}Pu content.

An estimate of the statistical error for the Passive Coincidence Collar is plotted as a function of the count time for a FBR subassembly in Fig. 18. A 200-s count should give a standard deviation of 1.0% for the ^{240}Pu -effective content. To obtain the same statistical precision in the active mode will require approximately 1000 s.

In summary, the passive/active methods give the reactivity of the fuel assembly, which is primarily a function of the fissile content for a given sample-moderator configuration. For FBR subassemblies, the combination of the active fissile determination with the passive mode $^{240}\text{Pu} + ^{242}\text{Pu} + ^{238}\text{Pu}$ measurement gives a more complete verification of the total plutonium content.

This general technique can be applied to any high-mass sample of PuO_2 and UF_6 as well as for MOX and spent-fuel assemblies. The primary requirements are that the sample must have a self-emitting source of neutrons, well-defined geometry, and a reasonably large solid angle for the reflected neutrons. Test and evaluation of the passive/active technique will be performed during the next year for different types of FBR and LWR-MOX fuel assemblies. The technique has particular promise for FBR fuel assemblies because the total plutonium content can be verified both before and after irradiation in the reactor.

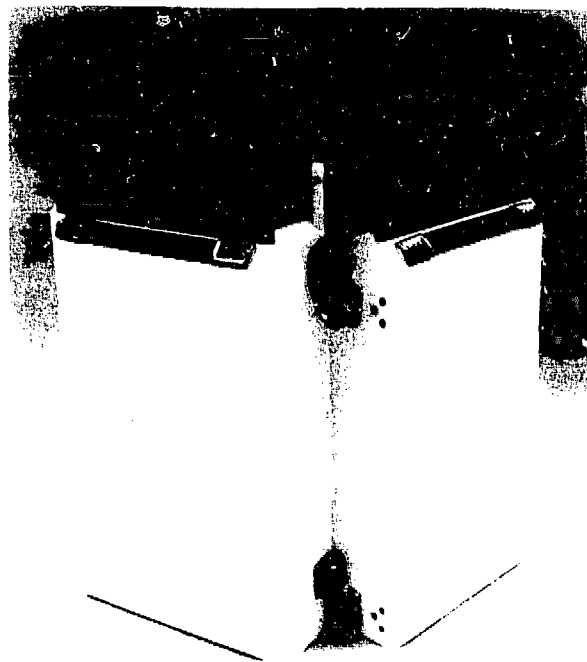


Fig. 17. Passive Coincidence Collar detector head.

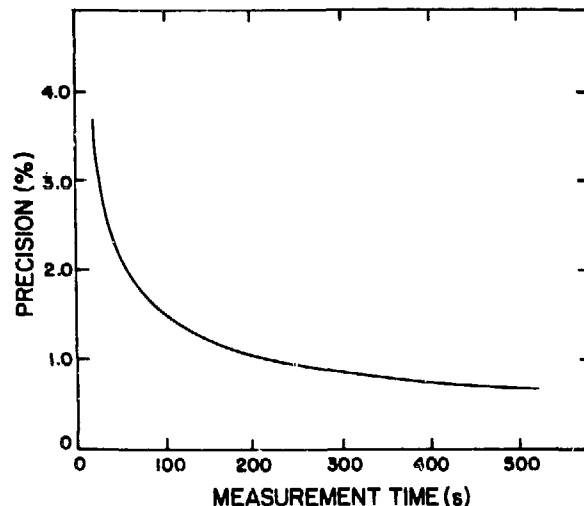


Fig. 18. An estimate of the counting precision (standard deviation) vs time for a FBR subassembly in the Passive Coincidence Collar.

E. Fast-Critical Assembly (FCA) Facilities

Quantitative NDA methods are necessary for verification of SNM inventories in fast-critical facilities.¹² IAEA inspection teams require NDA instruments to measure individual fuel pieces, reactor fuel drawers, and vault storage canisters. These instruments must be adapted to the specific geometries and fuel compositions found in each critical facility under IAEA safeguards.

1. Measurement Method. The use of gamma-ray spectroscopy and autoradiography for FCA fuel plates was described in the previous paper by H. Smith. This paper focuses on neutron methods that have been used for the quantitative verification of FCA fuel.

Passive neutron verification of plutonium fuel plates can be performed by counting the total number of neutrons emitted by the fuel or by counting the neutrons emitted in coincidence by the spontaneous fission of the even plutonium isotopes and by the induced fissions in plutonium or uranium. The total number of neutrons emitted by the fuel depends on the mass of the even isotopes of plutonium (spontaneous fission neutrons), the chemical composition of the fuel [(α, n) reaction neutrons], and the total fissile mass and geometry of the fuel (neutron multiplication). On the other hand, the neutron coincidence counting rate does not depend on chemical composition (except for small self-multiplication effects) or room neutron background fluctuations.

Measurements have been made with simultaneous totals and coincidence counts of plutonium fuel plates in a fuel drawer and in a vault canister to study the usefulness of passive neutron counting for quantitative inventory verification.

2. Assay Equipment. Passive neutron coincidence counters have been designed for both fuel drawers and storage canisters. The fuel drawers are rectangular (5 by 5 by 40 cm) and are used to position the plutonium metal plates in the reactor core. The storage canisters are cylindrical containers with the plutonium plates configured like the spokes of a wheel.

a. Fuel Trays. The Channel Coincidence Counter¹³ was designed for the assay of FCA fuel drawers and MOX rods. The principal feature of the detector is a 7- by 7- by 97-cm sample channel that provides a uniform neutron detection efficiency of 16% along the central 40 cm of the channel. The electronics system is identical to that used with the HLNCC. An isometric view of the detector is shown in Fig. 19. The three top plugs can be removed to provide slots for gamma-ray measurements of the sample.

b. Storage Canisters. The Bird Cage Counter¹⁴ was designed for the measurement of plutonium storage canisters positioned in the transfer "bird cages." A photograph of the counter is shown in Fig. 20. The detector consists of ^3He tubes in a CH_2 matrix covered with cadmium sheet (0.4 mm thick). The detector is placed inside the bird cage around the sealed storage canister.

3. Application. The Channel Coincidence Counter and the Bird Cage Counter are being used for international inspection and verification of FCA facilities. A typical calibration curve for the Channel Coincidence Counter is shown in Fig. 21. The response shows some multiplication increase for the higher mass loadings, but this can be corrected for by using the coincidence/totals ratio.¹¹

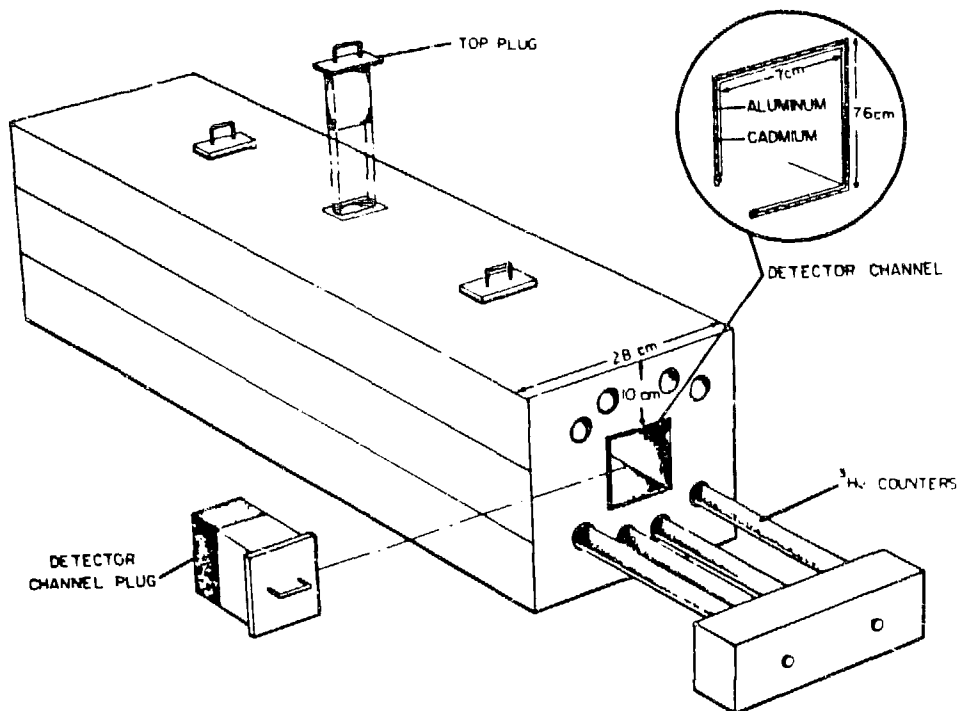


Fig. 19. Isometric diagram of Channel Coincidence Counter used for the assay of FCA fuel trays and MOX fuel rods.



Fig. 20. Bird Cage Counter used for the verification of storage canisters for FCA plutonium plates.

Precisions and accuracy of μ l₃ can be obtained in counting times of roughly 1000 s.

F. HTGR and MTR Fuel

High-temperature gas-cooled reactors (HTGR) and materials test reactors (MTR) generally contain high-enrichment uranium. Because the passive neutron intensity is very low, passive coincidence counting cannot be used, and active neutron techniques are used for the assay of this type fuel.

1. Measurement Method. Both Random Drivers¹⁵ and AWCCs have been used for the active assay of bulk HEU samples. During the past several years, the Random Driver has been used for the measurement of HTGR fuel rods. The Random Driver is an active, fast-neutron interrogation system that employs a ²⁴¹AmLi neutron source to induce fissions in the fissile material within a sample; hence, it "drives" the sample. The AmLi source energy ($E_n = 525$ keV) is below the fission threshold of ²³⁸U and ²³²Th, yet it is sufficiently energetic to achieve the penetrability required for assaying HTGR particle samples of high fissile mass; that is, self-shielding effects within a fuel particle and shadowing of one particle by another both are minimized.

2. Assay Equipment. Figure 22 is a schematic diagram of the random driver showing the AmLi neutron sources, the plastic scintillator, the lead, boral, and nickel shielding materials. The relatively thick lead shielding (5 cm) separating the scintillators from the sample cavity serves three purposes:

- (1) It reduces the passive gamma-ray background from the sample and the AmLi source, thereby decreasing the accidental coincidence rate from uncorrelated singles events.
- (2) It serves as a fast-neutron "pile" that provides for better time discrimination between $\gamma - \gamma$ and $n - n$ coincidences, and also increases neutron detection efficiency.
- (3) It tends to dampen out perturbations of the fast-fission neutrons that are caused by variations in the heavy-element matrix from sample to sample.

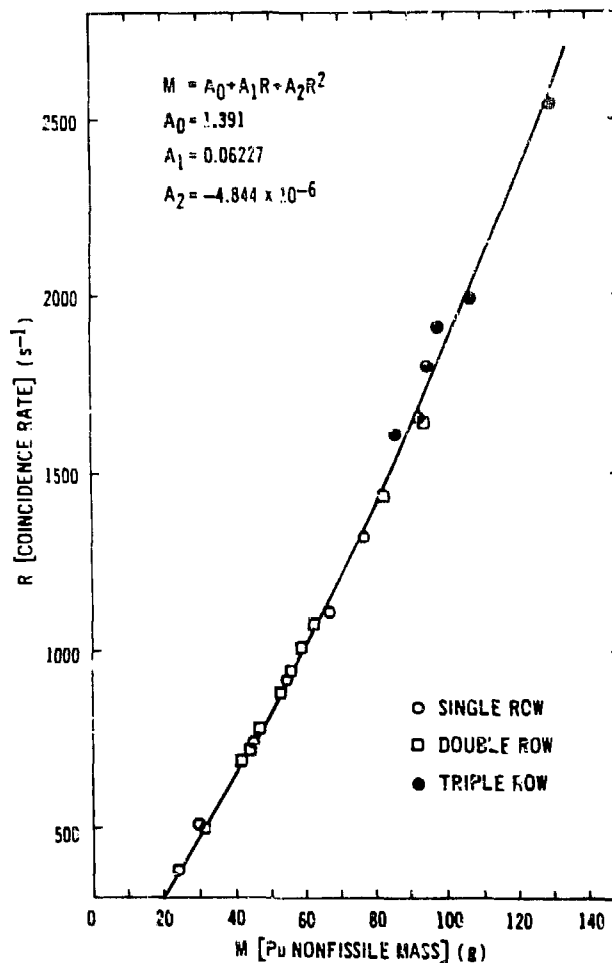


Fig. 21. Calibration curve for plutonium plates in FCA fuel drawers.

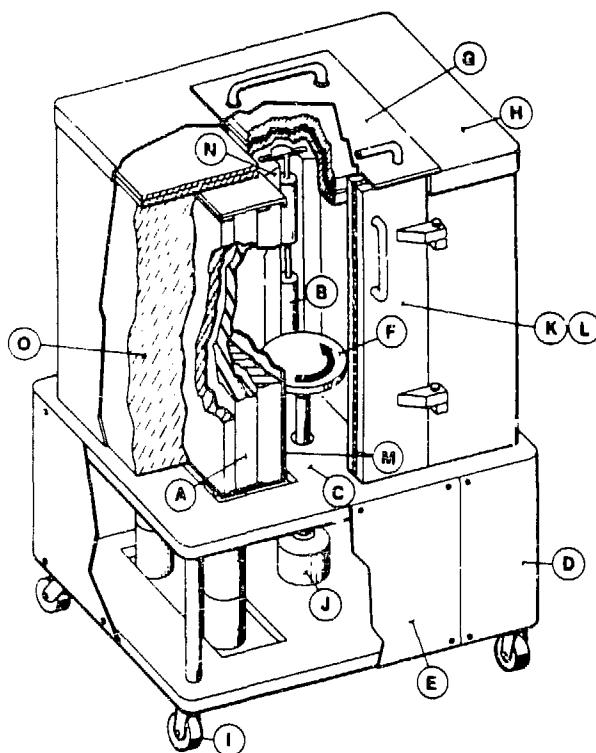


Fig. 22. Random Driver counting chamber - legend: (A) scintillator assembly, (B) source, (C) base, (D) base skirting, (E) front maintenance panel, (F) sample platen, (G) lid, (H) top cover, (I) casters, (J) weighing mechanism, (K) door, (L) steel shielding, (M) boral shielding, (N) source tailoring container.

3. Applications. The Random Driver has been used at the General Atomic Company (GAC) Sorento Valley Plant since November 1974. Figure 23 is a graphical presentation of the Random Driver assay results and the chemical assay results for the total of 62 different batches of material, including the 7 standards. The total mass assayed was 17.7 kg of ^{235}U , and the Random Driver results were only 30 g (-0.17%) lower than the chemical assay results.

III. SUMMARY

Neutron-based measurement techniques have found wide application in the control and safeguarding of nuclear fuel. Both uranium and plutonium in the form of pellets, rods, plates, and subassemblies can be measured with accuracies in the range of 0.2 to 1.0%. The accuracies are better than for most NDA applications because the product material is well defined in composition and geometry.

Applications to fuel pins and pellets have been under way for roughly a decade and the measurement equipment and procedures are now an integral part of the plant operation. Nondestructive measurements for fully assembled LWR

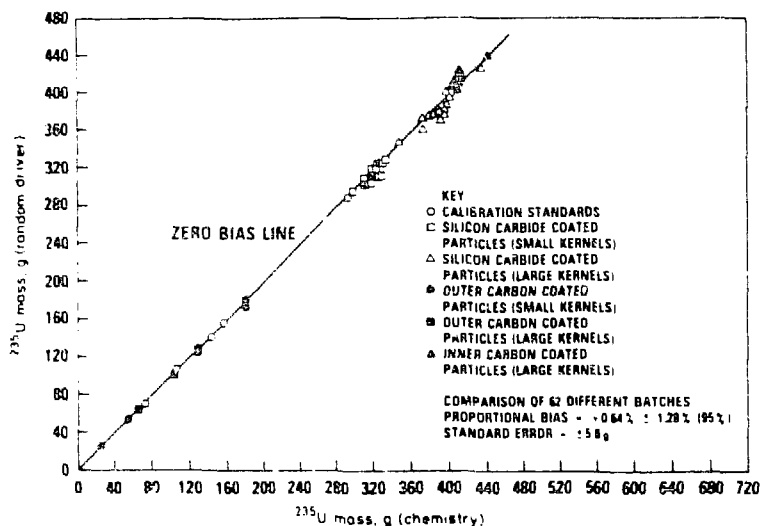


Fig. 23. Comparison of Random Driver NDA and chemical analysis for HEU fuel in the form of HTGR-coated particles.

and FBR fuel subassemblies is rather new and just coming into use. International inspection of fuel materials at both the fabrication and reactor facilities gives added emphasis to the assay of the complete fuel assemblies.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 10: TOUR OF LOS ALAMOS SAFEGUARDS RESEARCH AND DEVELOPMENT
LABORATORY: DEMONSTRATION AND USE OF NDA INSTRUMENTS AND
METHODS FOR ASSAY/VERIFICATION OF UNIRRADIATED FUELS

The group will visit the Los Alamos National Laboratory and tour the nuclear safeguards laboratory. Many of the instruments described in sessions 8-10 will be available for the participants' examination and "hands on" use. These will include:

- Portable single channel and multichannel pulse height analysers for field measurement of uranium enrichment.
- Neutron coincidence collar for LWR fuel assembly assay.
- High-level neutron coincidence counter for measurement of mixed oxide fuel pins and complete fuel assemblies.
- Active well coincidence counter for MTR fuel material assay.

The NDA instruments will be set up to perform simple nuclear fuel measurement experiments. Participants will use the instruments to perform the indicated nondestructive assay. Attendants will be at all instruments to answer questions and guide the measurements.

After the session participants will be able to:

- Describe the range of nondestructive assay instrumentation available to the reactor operator and safeguards inspector for unirradiated fuel measurement.
- Discuss the range of accuracy and precision of NDA instruments for unirradiated fuel measurement.
- Understand the characteristics and operation of NDA instrumentation.



Mike Baker demonstrates measurement of LWR fuel assembly with neutron coincidence collar.



Bob Likes explains operation of portable multichannel analyzer so that participants can use it to measure ^{235}U enrichment in case of uranium oxide.



Merlyn Krick explains operation of active well coincidence counter to measure samples of highly enriched uranium.



Mary Judy Roybal shows participants how to use the real time materials accounting systems simulator.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 10: TOUR OF LOS ALAMOS SAFEGUARDS R&D LABORATORIES:
DEMONSTRATION AND USE OF NDA INSTRUMENTS AND MATERIAL
CONTROL AND ACCOUNTING SIMULATION

Los Alamos Safeguards Staff

I. INTRODUCTION

This session will be devoted to a "hands-on" tour of nondestructive assay techniques and instrumentation that can be used to measure the fissile content of unirradiated nuclear fuel assemblies and fuel components. In addition, time will be allocated for all students to experience a computer simulation of a materials accounting exercise during attempted diversions of material in a nuclear process. There are stations set up on the tour to allow the exploration of five different exercises:

<u>Station</u>	<u>Title</u>
1	Measurement of Uranium Enrichment: The Portable, Intelligent Multichannel Analyzer
2	Measurement of LWR Fuel Assemblies in the Field: The Neutron Coincidence Collar
3	Active Neutron Coincidence Assay of Uranium-Bearing Fuel: The Active-Well Coincidence Counter (AWCC)
4	Passive Neutron Coincidence Assay of Mixed-Oxide Fuel: The High-Level Neutron Coincidence Counter (HLNCC)
5	Real-Time Materials Accounting Systems Simulator (RTMASS).

At each station, there will be approximately 1 hour available during which members of the Los Alamos National Laboratory Safeguards Assay Group and Safeguards Systems Group will explain the exercises and encourage the course participants to take part in the activities themselves. There should be ample time for all participants to sample the measurements and simulator and to discuss their uses and significance with the staff.

The tour participants will be divided into four groups, and the groups will circulate among the four demonstration stations according to the following schedule:

TOUR SCHEDULE

	Station 1	Station 2	Station 3	Station 4	Station 5
9:00					
	Introductory Lecture and Coffee Break				
10:00	Group 1	Group 2	Group 3	Group 4	Group 5
11:00	Group 2	Group 3	Group 4	Group 5	Group 1
12:00	LUNCH at LANL Cafeteria				
13:00	Group 3	Group 4	Group 5	Group 1	Group 2
14:00	Group 4	Group 5	Group 1	Group 2	Group 3
15:00		• • • Coffee Break • • •			
15:30	Group 5	Group 1	Group 2	Group 3	Group 4
16:30					

Stations 1-4 are located in the Safeguards Assay Group instructional center (Bldg. 110); station 5 is in the conference room of Bldg. 27. Participants will be guided to the station outside Bldg. 110.

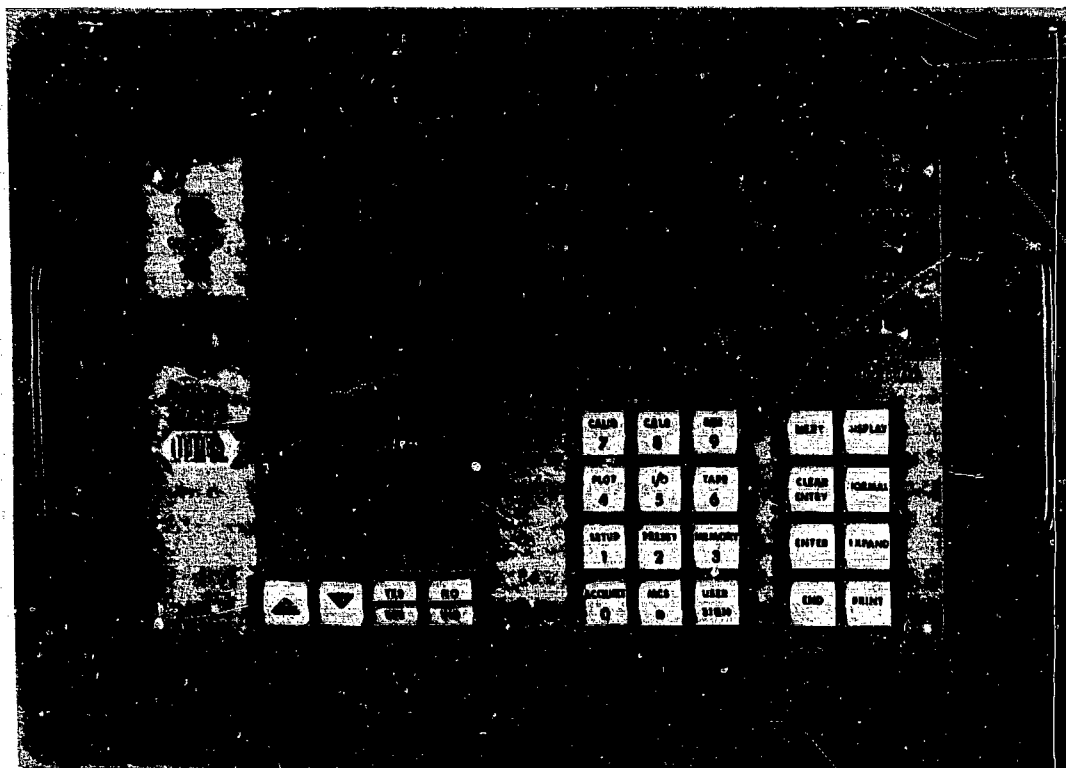
II. STATION 1: MEASUREMENT OF URANIUM ENRICHMENT: THE PORTABLE, INTELLIGENT MULTICHANNEL ANALYZER

Description

The goal of this exercise is to measure the ^{235}U enrichment of uranium oxide. The measurement is based on the fact that the emission rate of the 185.7-keV gamma ray from an infinitely thick sample (4.0 g/cm² or 4 mm of sintered oxide) is directly proportional to the atom fraction of ^{235}U in the sample. The equipment used for this measurement is a 1.9- by 1.9-cm NaI detector and a battery-powered, 4096-channel analyzer. Advances in electronic technology have made it possible to package a quality MCA into a package 18 by 28 by 20 cm. There are three such analyzers commercially available. All use microprocessors to reduce hardware and to implement basic MCA functions. In addition, the power of the microprocessor can be applied to accomplish a friendly user-instrument interface and to implement automated measurement procedures. A software system developed at Los Alamos for the IAEA has been implemented in the Davidson 2056-4K portable MCA (Fig. 1). This analyzer features a 100-MHz, 4096-channel analog-to-digital converter and memory with 16 777 215 counts/channel capacity. It has a built-in high-resolution pre-amplifier with 3 μs , and 1 μs fixed time constants that are used with solid-state and NaI detectors, respectively. The amplifier has separate pole zero adjustments for each type of detector and has a baseline restorer and pulse pileup rejection. The unit can power itself and a detector for more than 8 h from an internal rechargeable battery. The spectrum is displayed on a 7-cm diagonal cathode-ray tube. The detector has a ^{241}Am alpha source that provides a reference peak in the output spectrum that is used to stabilize the instrument. A two-window peak area determination is made using one window set over the 185.7-keV peak and another set just above this to sample the Compton background produced in the detector by higher energy gamma rays from ^{238}U and its daughter products. The ^{235}U enrichment is determined from the equation:

$$\% ^{235}\text{U} = A \cdot C_1 + B \cdot C_2 ,$$

where C_1 and C_2 are the measured activities in the two windows and A and B are calibration constants. The calibration constants are determined by measuring two samples of known enrichment. The PMCA ENRH software function guides the user through the calibration and subsequent measurement of unknown samples.



Front-panel controls on the Davidson Model 2056-4K multichannel analyzer.
Fig. 1.

Procedure

1. Turn Power Switch to On.
2. Press Memory--Set to 1 of 8.
3. Press Setup--Set parameters as follows:
 Det = NaI
 Polarity = POS
 Gain = 1000
 HVPS = 700 V, on
 LLD = 6
 ULD = 511
 Zero level = 0
 Threshold = 50
 Pu - Rejector = On
4. Press Preset, Set Livetime to 100 s. Place the 10% uranium oxide can on the detector and collect a 100-s spectrum by pressing Acquire. The MCA will collect the spectrum and stop after 100 s. Observe the spectrum and identify the 185.7-keV peak from ^{235}U .

5. Press Preset, set livetime to 300 s. Press "USER" and follow instructions. Use preset windows.
6. Place 17.5% can on detector, enter 17.5 as enrichment for standard 1, and press NEXT. At end of the 300-s measurement of first standard, record C_1 and C_2 at the top of the attached data sheet.
7. Place 1.96% can on detector, enter 1.96 as enrichment for standard 2, and press NEXT. Record C_1 and C_2 and the calibration constants A and B on the data sheet.
8. The system is now calibrated and ready to measure unknown samples. Measure other cans as time allows and record the results on the lower part of the data sheet. During the assay the system displays a running estimate of the enrichment based on the accumulated counts.

DATA SHEET FOR ENRICHMENT MEASUREMENT WITH NaI AND PORTABLE MCA

CALIBRATION:

<u>%²³⁵U</u>	<u>C₁</u>	<u>C₂</u>	<u>Time</u>
17.5			300 s
1.96			300 s

A = _____

B = _____

MEASUREMENT OF UNKNOWN CANS:

<u>Can No.</u>	<u>%²³⁵U</u>	<u>±</u>	<u>Error</u>	<u>Known %²³⁵U</u>

III. STATION 2: MEASUREMENT OF LWR FUEL ASSEMBLIES IN THE FIELD: THE NEUTRON COINCIDENCE COLLAR

Description

The goal of this exercise is to verify the ^{235}U loading of a fresh BWR fuel assembly mockup. An active assay technique will be used where random neutrons from an Am-Li (α, n) source induce fissions in the ^{235}U contained in the assembly. The induced fissions are characterized by the emission of time-correlated neutrons that are counted using the same coincidence circuitry used by the high-level neutron coincidence counter (HLNCC). The system has four slabs of polyethylene surrounding the assembly; three contain ^3He gas proportional neutron detectors and the fourth contains the Am-Li interrogation source. A simulated BWR fuel assembly containing 36 rods enriched to 2.34% ^{235}U will be used for the test. The assembly will be measured full and with several rods removed to measure the response to rod removal or substitution of dummy rods. A measurement will be made with several rods replaced by others of lower enrichment (1.77%) in order to demonstrate the sensitivity to rod substitution. (The neutron collar is also used to verify the fissile content of fresh PWR assemblies.)

The collar measures the ^{235}U in the region between the slabs; its response is proportional to the total quantity of ^{235}U per unit length. Its use by a safeguards inspector would require that a previous calibration be made on fuel assemblies of known composition and enrichment. The inspector then checks that the assemblies to be tested do not differ significantly from the expected response. If the Am-Li interrogation source is removed, the detector measures the time-correlated neutrons from ^{238}U spontaneous fission decay. This passive measurement is an additional verification for the inspector.

Procedure

1. Check that front-panel controls are set as follows:

HV = 7.5 (1500 V)
DISC V = 3.0 (1.5 V)
GATE(μs) = 64
TIME(s) = 1×10^2 (100 s)
MANUAL/RECYCLE = MANUAL

2. Turn PWR on (up).
Turn HV on (up).
3. Read the magnetic program card DATA-2 into the HP-97.
4. Take a 100-s background measurement with collar off assembly and the source out. At end of measurement, press READOUT; the program will yield the following information on the printer:

Count time in seconds
Totals
R+A
A

$\dot{R}$, totals rate (deadtime-corrected)
 $\dot{R}_0$, coincidence rate (deadtime-corrected)
 $\sigma(\dot{R}_0)$, uncertainty in $\dot{R}_0$
 $\%$ uncertainty in $\dot{R}_0$

Record the data on the sheet provided. The coincidence rate for this measurement should be very low; is it? _____

5. Place the collar around the full assembly, do a 300-s measurement, and record the results. The collar is now measuring the spontaneous fission neutrons from ^{238}U .
6. Now pull the collar away from the assembly, insert the Am-Li interrogation source, and make a 100-s count. The coincidence rate should be zero; is it? _____
7. Place the collar around the full assembly, make a 300-s measurement, and record the results.
8. Remove two rods from the center of the assembly, make a 300-s measurement, and record the results. Compute the following:

$$\dot{R}_0(\text{full}) - \dot{R}_0(-2) = \underline{\hspace{4cm}}$$

$$\sqrt{\sigma^2(\dot{R}_0 f) + \sigma^2(\dot{R}_0 - 2)} = \underline{\hspace{4cm}}$$

In 300 s, can you detect this removal or the equivalent substitution of dummy rods? _____

9. Remove two more rods (four total) from the center of the assembly, make a 300-s measurement, and record the results. Compute the following:

$$\dot{R}_0(\text{full}) - \dot{R}_0(-4) = \underline{\hspace{4cm}}$$

$$\sqrt{\sigma^2(\dot{R}_0 f) + \sigma^2(\dot{R}_0 - 4)} = \underline{\hspace{4cm}}$$

10. Put four rods enriched to 1.77% in the empty positions, make a 300-s measurement, and record the results. Compute the following:

$$\dot{R}_0(\text{full}) - \dot{R}_0(\text{sub}) = \underline{\hspace{4cm}}$$

$$\sqrt{\sigma^2(\dot{R}_0 f) + \sigma^2(\dot{R}_0 \text{ sub})} = \underline{\hspace{4cm}}$$

Can you detect this rod substitution in a 300-s measurement?

11. Turn HV off (down).
Turn PWR off (down).

DATA SHEET FOR NEUTRON COLLAR MEASUREMENT OF BWR FUEL ASSEMBLY

[illegible]

IV. STATION 3: ACTIVE NEUTRON COINCIDENCE ASSAY OF URANIUM-BEARING FUEL: THE ACTIVE WELL NEUTRON COINCIDENCE COUNTER

Description

The goal of this exercise is to measure the ^{235}U content of uranium metal discs similar to material produced in the early stages of MTR fuel fabrication. The AWCC measures uranium by inducing fissions with $^{241}\text{Am-Li}$ neutron sources and by counting coincident neutrons from the induced fissions. Neutrons from the Am-Li source yield a background of accidental coincidences. The passive neutron shift-register unit is used to determine the "real" coincidence rate by providing the real-plus-accidental count and the accidental count. The HP-97 calculator is used to calculate the actual real coincidence rate, appropriate correction factors, and the ^{235}U assay based on the calibration.

The AWCC is an instrument that is compact compared to other active neutron systems. The difference between a passive coincidence counter and the AWCC is the addition of neutron sources located in the end caps. The detector is identical or at least similar to passive systems used to assay plutonium by spontaneous fission detection. The shift-register package is identical to those electronic packages used in passive coincidence counters.

The AWCC measurement principle works best when applied to uranium. The AWCC assay technique has found its application in verifying the fissile uranium content of well-characterized material such as finished fuel.

Procedure

1. Turn on the electronics package.
 - a. Check that the front panel controls are set as follows:
 - (1) HV = 7.5 (1500 V)
 - (2) DISC V = 3.0 (1.5 V)
 - (3) GATE = 64 (64 μs)
 - (4) TIME = $3 \times (10)^2$ (300 s)
 - (5) MANUAL/RECYCLE = MANUAL (use recycle with repeat assays)
 - b. Turn on electrical power.
 - (1) Turn on HP-97 calculator (use appropriate power adapter).
 - (2) Check line voltage switch on back of shift register.
 - (3) Turn on shift-register circuit power (PWR switch up).
 - (4) Turn HV on.
 - c. Load program into HP-97 calculator (use DATA-1).
2. Check the shift register, calculator, and neutron detectors.
 - a. Insert end caps with the neutron sources.
 - b. Take a 100-s background measurement.
 - c. At the end of the measurement, push READOUT.
 - The following information appears on the printer:
Count time in seconds (t)
Total neutron counts (T)
Real + accidental coincidence counts (R+A)
Accidental counts (A)

Coincidence deadtime correction
Deadtime-corrected totals rate (T_0)
Deadtime-corrected real coincidence rate (R_0)

One standard deviation error (σR_0)
Percentage error ($100\% \times \sigma R_0 / R_0$)

- o Check that t , T , $R+A$, and A agree with shift register.
- o Check that T_0 agrees with known value.
- o Check that R_0 is consistent with zero.
- o Record data on attached sheet.

3. Demonstration Measurements

- a. Load a 500-g ^{235}U standard into the assay chamber.
 - o Make a 300-s measurement.
 - o Push READOUT at the end of assay.
 - o Record data on attached sheet.
- b. Load a 250-g ^{235}U standard into the assay chamber.
 - o Make a 300-s measurement.
 - o Push READOUT at the end of assay.
 - o Record data on attached sheet.
- c. Load the 500-g and one 250-g disk into the assay chamber, make a 300-s measurement, and record the data.
- d. If time permits, load the 500-g and two 250-g disks into the chamber, make a 300-s measurement, and record the data.
- e. Plot the corrected coincidence count rate, R_0 , as a function of ^{235}U mass on the graph paper provided. Discuss the shape of the resulting calibration curve.
- f. Turn the HV off (down).
Turn the PWR off (down).

DATA SHEET FOR MEASUREMENT OF URANIUM WITH THE AWCC

[illegible]

ACTIVE NEUTRON COINCIDENCE ASSAY OF URANIUM-

BEARING FUEL: THE ACTIVE WELL NEUTRON

COINCIDENCE COUNTER (AWCC):

PLOT OF CORRECTED COINCIDENCE COUNT RATE VS
URANIUM-235 MASS

Corrected Coincidence Count Rate

Uranium-235 Mass (Grams)

V. STATION 4: PASSIVE NEUTRON COINCIDENCE ASSAY OF MIXED-OXIDE FUEL:
THE HIGH-LEVEL NEUTRON COINCIDENCE COUNTER

Description

The goal of this exercise is to verify the $^{240}\text{Pu}(\text{eff})$ content of mixed oxide fuel rods. The fuel rods to be measured are simulated BWR rods, which are similar to the uranium rods in the BWR assembly used with the neutron coincidence collar. In this measurement neutrons are detected from the spontaneous fission of the even-numbered isotopes of plutonium. Plutonium-240 is the major contributor for most common isotopic compositions; the instrument response is proportional to the quantity:

$$m_{\text{eff}}^{240} = 2.49 m^{238} + m^{240} + 1.57 m^{242},$$

where m^A is the mass of the isotope with mass number A and the coefficients, 2.49 and 1.57, are the ratios of the spontaneous fission yields per gram for ^{238}Pu and ^{242}Pu to ^{240}Pu . The coincidence electronics statistically separates fission neutrons from (α, n) neutrons by time correlation analysis of the pulse train from the detector.

Procedure

1. Check that front panel controls are set as follows:
HV = 7.5 (1500 V)
DISC V = 3.0 (1.5 V)
GATE(μs) = 32
TIME(s) = $1(\times 10)^2$ (100 s)
MANUAL/RECYCLE = MANUAL
2. Turn PWR on (up).
Turn HV on (up).
3. Turn the HP-97 calculator on and read the magnetic program card DATA-1 into the calculator memory.
4. Take a 100-s background measurement with no sample in the counter. At the end of measurement press READOUT; the program will output the following information on the printer:
Count time in seconds
Totals
R+A
A

Coincidence deadtime correction

Deadtime corrected totals rate

$\dot{R}_0$, deadtime corrected coincidence rate

$\sigma(\dot{R}_0)$, uncertainty in $\dot{R}_0$

% uncertainty in $\dot{R}_0$

Record the data on the sheet provided. Check that the scaler readings agree with what is printed on the tape. There should be less than 100 counts in R+A or A.

5. Place two rods in the holder provided inside the assembly, make a 100-s measurement and record the data.
6. Place two more rods in the detector, make another measurement and record the data. Continue this until 12 rods are in the detector.
7. Plot the corrected response ($\dot{R}_0$) vs number of rods on the graph paper provided.
8. Now remove six of the rods and insert six uranium rods (enriched to 1.77% or 2.34% ^{235}U). Make another 100-s measurement and record the results. Compute the corrected response and compare it with that measured above. Discuss the significance of this result.
9. Turn HV off (down).
Turn PWR off (down).

DATA SHEET FOR HLNCC MEASUREMENT OF MOX RODS

[illegible]

MEASUREMENT OF MIXED-OXIDE FUEL RODS WITH THE
HIGH-LEVEL NEUTRON COINCIDENCE COUNTER (HLMCC):
PLOT OF CORRECTED RESPONSE PER ROD VS

NUMBER OF RODS

Corrected Response per Fuel Rod

Number of Fuel Rods

VI. STATION 5: REAL TIME MATERIALS ACCOUNTING SYSTEMS SIMULATOR

Introduction

The Real Time Materials Accounting Systems Simulator (RTMASS) has been developed primarily as an educational tool to simulate in real time the operation of a nuclear materials accounting system. In essence the RTMASS is a computer game with two principal players, the Diverter and the Nuclear Material Control Officer (NMCO). The Diverter attempts to steal a goal quantity of nuclear material while the NMCO tries to detect the missing material. Each of the players has his or her own computer terminal for interfacing with the RTMASS.

The RTMASS encompasses the three major codes, MODEL, MEASIM, and DECANAL, that have been developed primarily for Safeguards Systems Studies at Los Alamos. The MODEL code is used for the simulation modeling of the process. This code consists of the SLAM II¹ Simulation Language along with some auxiliary input and output subroutines. The MEASIM² code simulates the process measurements and computes the measurement error variances. DECANAL,³ the DECision ANALYSIS code, consists of a series of detection algorithms that can be used to determine the presence of a diversion.

A block diagram showing the flow of information in the simulator is shown in Fig. 1. The Process Model block simulates the process. Output from this block consists of the arrays of the process variables that serve as the input to the Measurement Model. The measured values of the process variables are computed by the Measurement Model. These measured values are then input to the Decision Analysis block where calculations are made to determine if a diversion has taken place. As can be seen in Fig. 1, the Diverter interfaces with the process model in the theft of material, while the NMCO monitors the outputs from the decision analysis algorithms. After this analysis the NMCO makes a decision of either "diversion" or "no diversion." If the decision is "diversion," the simulation continues. In addition to modeling the process, SLAM II is also used to control the flow of information for the complete RTMASS system as represented in Fig. 1.

Model Description

The RTMASS has been designed to allow for flexibility with regard to the particular process being simulated. For purposes of education on the simulator, however, a simple process serves just as well as a more complicated process and also improves the response time because of the reduced computational requirements. A simple tank with a single input and single output as shown in Fig. 2 is used for the process in this exercise. The tank has an inventory of 100 kg, with daily batch input and output transfers of 10. Materials balances are drawn around the process at a frequency of once per day. The process measured values are assumed to behave according to the following equations.

$$I_m = I(1 + \epsilon_I) \quad (1)$$

$$T_m = T(1 + \epsilon_T + \eta_T) \quad (2)$$

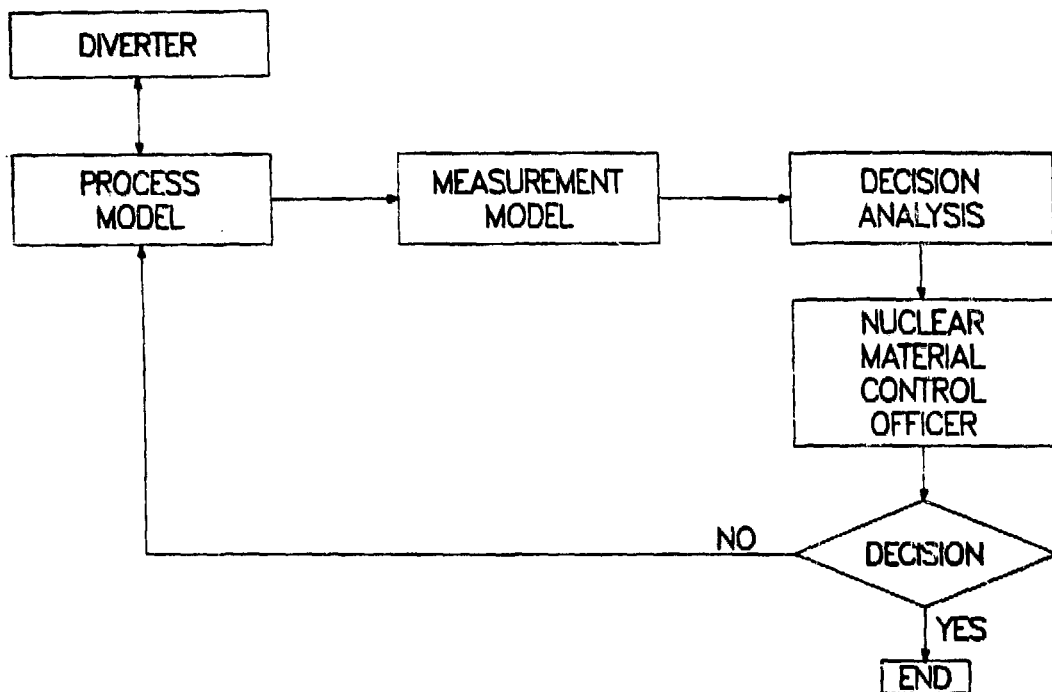


Fig. 1. Information flow diagram for the RTMASS.

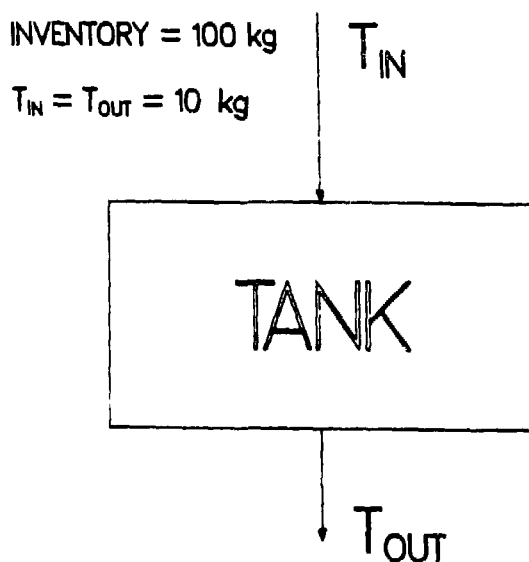


Fig. 2. Process model for the RTMASS.

where

I_m = measured value of the inventory,
 I = actual true value of the inventory,
 ϵ_I = inventory random error,
 T_m = measured value of the transfer,
 T = actual true value of the transfer,
 ϵ_T = transfer random error, and
 η_T = transfer systematic (correlated) error.

Both the input and output transfer measurements are assumed to behave according to the same equation. All the measurement errors are assumed to have a standard deviation of 0.6868%, that is,

$$\sigma_{\epsilon_I} = \sigma_{\epsilon_T} = \sigma_{\eta_T} = 0.006868 \quad . \quad (3)$$

Detection Algorithms

As indicated above, the game is played by the Diverter trying to steal material and the NMCO attempting to detect this loss of material. If there were no measurement errors, the NMCO would be able to detect the loss of material without fail. With measurement errors present, however, the NMCO cannot be sure if the signals he observes from the decision analysis tests are due to a real diversion or to measurement errors. The NMCO has at his disposal a number of tests that he can use to detect a diversion. These include the Shewhart, CUMUF, Residual MUF, and Page's CUMUF tests. The Shewhart test is merely a control chart with each materials balance and its associated error variance being plotted. The CUMUF, Residual MUF, and Page's tests employ threshold boundary crossings to indicate the presence of a diversion.

Diversion Sensitivity

The simulation is designed to run for a maximum of 50 days with a materials balance taken once per day. The goal quantity of material to be taken by the Diverter over this time period is calculated for this exercise from the "diversion sensitivity."

"Diversion sensitivity" is an important consideration for this system from a safeguards standpoint. For the purposes of this discussion "diversion sensitivity" is that quantity of material diverted over a given period of time that can be detected as missing by the NMCO with some reasonable probability of success and with a relatively small false-alarm probability. One way of calculating "diversion sensitivity" is via the standard deviation for the CUMUF over the time interval being considered. If N is equal to the number of balances and σ is the CUMUF standard deviation for N balances, then a uniform diversion of $2\sigma/N$ over these N balances can be detected via the sequential CUMUF test ~60% of the time with a false-alarm probability of 5%. Hence, the 2σ value is one way of measuring "diversion sensitivity." In this simulator exercise the 2σ value will be used to define the goal quantity of material to be diverted.

For the simple process of Fig. 2 and the measurement error models of Eq. (1) and (2), it can easily be shown that the CUMUF variance for N balances is given by

$$\sigma^2 = 2I^2 \sigma_{\epsilon_I}^2 + 2NT^2 (\sigma_{\epsilon_T}^2 + N\sigma_{\eta_T}^2) \quad (4)$$

where

σ^2 = CUMUF variance for N balances,

N = number of balances,

$\sigma_{\epsilon_I}^2$ = variance of the inventory random error,

$\sigma_{\epsilon_T}^2$ = variance of the transfer random error, and

$\sigma_{\eta_T}^2$ = variance of the transfer systematic error.

From Eq. 3 and with N = 50 it follows that

$$\sigma = 5.0 \quad .$$

Hence, the "diversion sensitivity," or 2σ as it has been defined for the purposes of this exercise, is equal to 10 kg. This is the amount of material that the Diverter will attempt to steal over N balances without being detected by the NMCO.

Operational Procedures

As indicated above, the RTMASS is designed to simulate 50 days of the process with materials balances drawn once per day. Control of the system alternates between the Diverter and the NMCO. For the first 10 days of the simulation, the Diverter can divert material but the NMCO cannot observe any of the data. At the end of 10 days the NMCO enters the game to observe the outputs from the decision analysis tests. He now has the opportunity to observe the results from the following decision analysis tests:

1. Shewhart,
2. CUMUF,
3. Residual MUF test,
4. Page's test.

The NMCO enters a carriage return to bring up another test on the screen. After viewing the resultant graphical output from the decision analysis tests, the NMCO makes the decision "diversion" or "no diversion." If the NMCO concludes that a diversion has taken place, he terminates the simulation by entering a "0" at the terminal. If the NMCO is uncertain at this point about "diversion," he continues the simulation by entering a "1" and then enters the number of days to which the simulation is to continue to before more data are

analyzed. The Diverter is then given another chance to divert material, and the cycle repeats itself. If the simulation continues for 50 days without the NMCO detecting a diversion, then by default the decision is "no diversion."

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 11: THE CHEMICAL ASSAY OF UNIRRADIATED FUEL MATERIALS

Destructive assay techniques to measure unirradiated fuel materials will be discussed in detail. The discussion will be organized around the key measurement points in low-enriched uranium fuel fabrication facilities. Major emphasis will be placed on methods for the elemental and isotopic analysis of uranium; however, plutonium assay methods will be described briefly. The precision and accuracy of the analysis techniques will be discussed in detail.

After the sessions, participants will be able to:

- List the major destructive assay techniques for the elemental and isotopic analysis of uranium.
- Discuss how the measurement techniques are used for accountability and quality control of LWR fuel.
- Understand the precision and accuracy of the measurement techniques and the factors which influence measurement performance.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 11: SURVEY OF CHEMICAL AND BULK MEASUREMENT METHODS

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I. INTRODUCTION

Determination of the quantity of uranium and plutonium in nuclear materials generally requires two types of measurements: a bulk measurement, such as weight or volume, and a concentration measurement. Concentration may be determined by a variety of destructive and nondestructive techniques. This chapter surveys bulk and chemical measurement methods that are used for either conventional or near-real-time materials accounting as well as for process control in nuclear facilities, with emphasis on fuel fabrication, conversion, and reprocessing. Nondestructive methods of analysis that rely on a nuclear signature are discussed in another chapter.

The sequence of measurements usually involves three steps. First, the mass or volume and density of the material must be determined. Next, samples representative of the material must be obtained and analyzed, and, finally, the total quantity of uranium or plutonium must be computed from the mass or volume and the analysis results.

II. MASS, VOLUME, DENSITY, AND FLOW MEASUREMENTS

The mass of solids and gases is generally determined by weighing. Solution measurements include mass, volume, and density determinations and in some cases flow measurements integrated over time.

A. Mass Measurement

A wide variety of weighing devices is available. Some commonly used devices include pan balances, beam balances, hydraulic load cells, and mechanical load cells. Selection of a weighing device depends on the application. Load cells have application where high capacity is required, where weights of highly radioactive materials must be determined, or where the environmental conditions are harsh. Pan and beam balances are generally selected where small quantities are to be measured and where environmental conditions can be controlled (i.e., in a noncorrosive atmosphere). All of these devices require frequent calibration. For accurate work, the measured weights should be corrected for buoyancy.

B. Volume and Density Measurement

The volume of solution in a tank can be measured by weight methods or by measuring liquid level and density of the solution. The use of load cells on a tank is potentially able to give the most accurate measurement of the amount of solution. However, in practice it is difficult to isolate the tank effectively from the remainder of the process, and vibrations transmitted through connecting pipes impair measurement accuracy.¹ Accordingly, methods now in use rely on measurements of liquid level and density.

The liquid level and density of solutions in process and storage tanks are most commonly measured with a pneumatic bubbler system, also called dip-tube manometers or pneumerators. The pressure differences between ends of purged dip-tubes are measured with precision equipment. Three pressure probes are required: one probe extends to near the bottom of the vessel, the second ends at a known elevation above the lower probe, and the third terminates in the vapor space of the tank (see Fig. 1). The dip-tubes are connected to differential pressure transmitters and are purged continuously to prevent the entry of process solution and vapors into the tubes. Rotameters are adjusted to give equal flow of dry air or nitrogen in each probe. Rotameters are adjusted to give equal flow of dry air or nitrogen in each probe.

The differential pressure between either of the two lower probes and the vapor space probe is proportional to the liquid level, whereas the differential pressure between the two lower probes is proportional to the solution density. For differential pressure measurements an electromanometer or a precision pressure transducer with digital output is used. Direct computer-compatible output can be obtained from these instruments. Given values for the liquid level and density, one may calculate the solution volume from an experimentally--determined relationship between liquid level and volume for each tank.

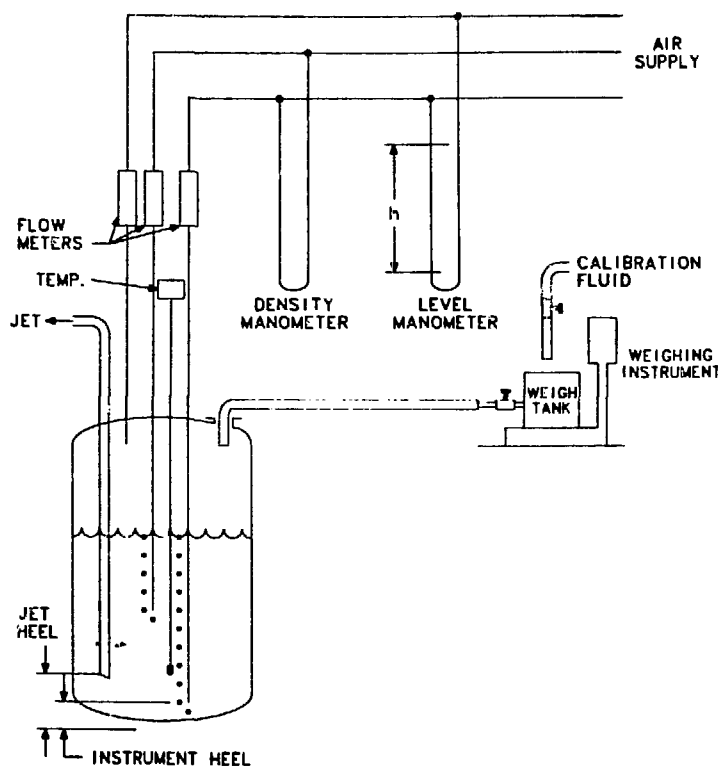


Fig. 1. Schematic of instrumentation for volume measurement with a pneumatic bubbler system.

Tests of high-precision electromanometers at the Allied General Nuclear Services plant suggest that relative accuracies of 0.1% in level measurement and 0.2% in density measurement can be achieved.² The Paroscientific "Digiquartz" transducer and the Ruska Corporation model DDR-6000 transducer were compared in a calibration exercise at Tokai, Japan. A precision piston gauge was the pressure source. Uncertainties of <1 mm of water were found for both transducers.³ The Paroscientific transducer uses a quartz crystal resonator whose frequency of oscillation is changed by the applied pressure difference. The Ruska instrument is a precision electromanometer that uses a hollow helix quartz Bourdon tube as the pressure sensor. Experiments have in general been made with water or non-radioactive nitrate solutions. If these transducers are used in a highly radioactive environment, special hardening may be required.

Although good results have been achieved with pneumatic bubbler systems, there are problems associated with them. Because the dip tubes contact the solutions, gradual plugging of the tubes or probe growth because of salt crystallization can occur. Variations or imbalances in bubbler air flow will affect measurement accuracy, and bubbling of air through the solution may change the concentration. Other problems are caused by leaks or restrictions in pressure sensing lines.⁴

Other types of level detectors have been tested. These detectors have the advantage that the density of the solution need not be known. The problem of finding suitable detectors is compounded by the variety of tanks found in nuclear materials processing plants. Obviously, more than one type of detector is needed to meet different applications.

The time domain reflectometer (TDR) consists of a coaxial probe suspended from the top of the tank. Liquid level is inferred from the time delay after an electrical pulse has been initiated in the probe until the reflected electrical signal has been received from the impedance mismatch at the liquid surface. TDR systems have been installed in a number of tanks at the Itrec and Eurex plants in Italy. Laboratory tests of the technique gave an accuracy of ± 0.2 cm in the measurement of liquid level.⁵ The accumulation of solids in the tank does not affect the TDR technique, and an air supply is not needed.

An ultrasonic level probe that can also detect the aqueous-organic interface in solvent extraction contactors was developed at Harwell, UK.⁵ Trials showed that the position of an interface or the liquid level could be measured with a precision of 0.3 cm over a range of 15-500 cm.

A pulsed sonic liquid level indicator, the Bartex Aquatrak, has been extensively studied⁶ at Rockwell International, Rocky Flats Plant and at Savannah River Plant. A calibration accuracy of 0.025% of measurement and a resolution of 0.3 cm throughout a 762 cm (25 ft) range were verified. One of the Aquatrak systems was recently recalibrated after being in continuous operation for approximately 1 year. The difference between the original calibration data and the recalibration data was negligible.

Conductivity probes have been found unsuitable. They malfunctioned regularly either because they collected a coating of solids that prevented them from contacting the liquid, they dissolved in the solution, or they shorted out.⁷

Large tanks filled with Raschig rings and horizontal pencil tanks present special problems. Raschig rings break and settle in the tank with time, so that the volume they occupy changes. The liquid level in the

tank is usually determined with a sight glass. Horizontal pencil tanks are only 15 cm in diameter and often have precipitated material on the bottom. There is at present no satisfactory technique for measuring the volume of liquid in such tanks.

Density measurements also have been used to estimate uranium or plutonium concentrations in process tanks for process and criticality control. Empirical relationships between the uranium or plutonium concentration and the density, temperature, and hydrogen ion concentration have been developed.⁸⁻¹⁰ Density measurements on flowing streams could be useful for process control and near-real-time materials accounting for continuous processes. Such measurements would be used in conjunction with flow measurements and would require a knowledge of the solution acidity. A prototype in-line free acid monitor was developed at Oak Ridge.¹¹

On-line density instruments that use a vibrating U-tube or a vibrating vane are commercially available. These instruments have not been tested under the conditions encountered in nuclear processing plants.

An on-line instrument that relates transmission time of an ultrasonic signal to the concentration of uranium and plutonium in solution was developed in the UK. Under ideal conditions, the relative standard deviation on solutions containing 60 g U + Pu/L was ~1%. The instrument is calibrated and standardized during operation against material that is sampled and analyzed chemically.⁵ Acoustic measurements may also be useful and deserve investigation.

Density measurements are nonspecific and require complementary information from other measurement techniques for accountability purposes. The advantages of on-line density measurements are that they can be made continuously and that the results are immediately available.

C. Flow Measurement

Flow measurement instruments in nuclear materials processing plants are used primarily for process control where high precision is not a requirement. Monitoring of process streams is held to a minimum, and only those instruments essential for plant operation generally are provided. These instruments have not been designed for use in accountability systems.

Near-real-time materials accountability requires measurements or estimates of the in-process inventory of major process vessels and columns. For continuous processes, such as solvent extraction where volume measurements and sampling are not possible, the requirement can be met by flow measurements and on-line concentration (or density) measurements on streams entering and leaving the vessels and columns. Flow and on-line concentration measurements are also valuable in batch processes in situations where the volume of solution in a tank is difficult to measure and the solution is difficult to homogenize.

Flow rates are generally low. Some streams, such as the output from the plutonium concentrator in a reprocessing plant, have flow rates below 10 L/h. In other streams, flow rates of 400 L/h (1.7 gpm) must be measured with a precision of 1% and an accuracy of 0.5%.

Flowmeters with no moving parts are preferred. It is often desirable that the flowmeter does not contact the process fluid nor intrude into the process piping. An ideal flowmeter would have the above properties and be totally unaffected by viscosity, density, pressure, temperature, air bubbles, or suspended solids. The meter's calibration would be unaffected by

material buildup in the flow sensor. The ideal flowmeter would be accurate to better than 0.5% of rate at low flow rates and be easy to calibrate and maintain remotely.

Flowmeters that have been used in radioactive environments include differential pressure devices (orifice plates and weir meters), positive displacement flowmeters, electromagnetic flowmeters, and electronic rotameters. Thermal, gyroscopic deflection (Coriolis effect),¹² and ultrasonic¹³ flowmeters offer promise of approaching the properties of an ideal flowmeter and deserve evaluation in a corrosive, radioactive environment.

III. SAMPLING PROBLEMS AND TECHNIQUES

The purpose of a sampling program is to obtain material for characterization that is representative of the bulk material in analysis; no analytical method can provide accuracy or precision that is better than that associated with obtaining an increment that is truly representative of the total population. In addition to the normal operations of selecting, splitting, and compositing the material, sampling also may encompass ancillary operations such as grinding, evaporating, drying, and screening. For on-line or in-line measurements, sampling may be as simple an operation as installing the measuring device directly in the process line or into a bypass line that samples a fraction of the process stream.

Samples may be in the form of solids, powders, or liquids (and occasionally gases). Samples derived from reprocessing operations may be associated with fission product activity of thousands of curies per kilogram. Each type of material presents its own unique sampling problems. This section reviews some of the basics of sampling; more detailed descriptions are found in Ref. 14, pp. 25-56 and Ref. 15.

A. Sampling of Liquids

Liquids in process materials normally are sampled using an air-lift sampling system. The air-lift sampling system should be designed to permit extensive recirculation of solutions through the sample lines and the sample bottle. Vessel sparging, mixing time, and sample-circulation time should be considered in establishing proper sampling procedures for replicate analyses. Any solids must be dissolved after sampling and be included in the total analysis. For highly radioactive samples the possibility of bubble formation must be considered in volume measurements, and temperature corrections should be applied.

The main sources of sampling error for solutions are expected to be (1) concentration or dilution of the sample by the air-lift system, (2) incomplete mixing of the solution in the tank, (3) contamination of the sample in the sample lines, and (4) sample heterogeneity caused by suspended solids. Where analytical precision better than 0.5% is required, all sampling should be done on a weight basis rather than a volume basis. Remote volume samplers seldom can provide routine precision better than 0.5%, and even normal volumetric measurements are no better than 0.2% on a routine basis.

A sampling apparatus capable of providing pipetting accuracy of 0.1% under hot-cell conditions has been described.¹⁶ All steps, including

pipet rinsing, are performed remotely, and the Teflon piston surface does not touch the solution being pipetted.

B. Sampling of Powders

Sampling of process powders to obtain a representative sample generally is more difficult than sampling liquids. Sample heterogeneity may be a function not only of such factors as mixing but also of particle size and proximity to the surface. Sampling plans must be properly designed to assure representative samples. Flowing streams generally are easier to sample than bulk containers. Powder samples are obtained using dippers, scoops, core samplers, powder thief probes, or rifflers.

Samples may be further subdivided using a splitting or riffing technique. Alternatively, the total sample may be dissolved and suitable aliquots removed for analysis.

C. Problems Associated with Sampling

Obtaining a homogeneous sample does not necessarily assure a representative sample. Care must be exercised that the sample form or composition is not altered after sampling. In the case of liquids, evaporation or radiolytic decomposition of the solution is a concern if subsamples are obtained subsequently. Dilution of plutonium nitrate solutions with water leads to formation of plutonium polymers that could precipitate or adsorb on container surfaces.¹⁷ Therefore, plutonium nitrate solution should be diluted with some concentration of nitric acid rather than with water.

Powders of UO_3 and PuO_2 are hygroscopic and gain weight rapidly, particularly in humid atmospheres. Powders of UO_2 may change mass through air oxidation as well as by moisture adsorption.

IV. MEASUREMENT UNCERTAINTIES

The topic of measurement statistics will not be discussed here, but it is necessary to introduce the concept of measurements relative to true values.

For any analytical or physical measurement, one is never absolutely certain that the measured value represents the true value of the measured parameter. Two basic types of errors contribute to this uncertainty.

First, if we perform multiple analyses of the same material, we cannot exactly duplicate the sample either in quantity or quality (sampling error) or perform the measurement in exactly the same manner (measurement error). These errors are commonly referred to as random error of sampling and random error of measurement. The combination of these two errors, the random error of the analysis, is a measure of how precisely we can reproduce the same value for the measurement, or the precision of the analysis.

Secondly, after we have performed the analyses and obtained the best estimate of the quantity we are measuring, we cannot be sure that the measured value equals the true value. Contributing factors to this uncertainty could result from uncertainties (biases) in reference materials used in calibrating instruments or procedures or unknown biases in the instruments or procedures. These errors are referred to as systematic errors and are a measure of how accurately we can reproduce the true value, or the accuracy of the analysis.

V. ELEMENTAL METHODS OF ANALYSIS

A. Gravimetric Methods

Gravimetric methods involve separating a compound of the element to be determined and igniting it to a constant-weight stoichiometric compound. The technique is applicable only to relatively pure materials; impurities must be determined by spectrographic or other procedures and the final weight corrected by difference. When clean separations from impurities can be obtained, precisions better than 0.1% often can be realized.

1. Uranium. Gravimetric methods can be used for the determination of uranium in pure product streams such as uranyl nitrate, U_3O_8 or UO_2 , and UF_6 . Nitrates are evaporated to dryness; UF_6 is hydrolyzed in H_2O and evaporated to dryness. Ignition is at 850-900°C to U_3O_8 and the final weighing provides a relative standard deviation (RSD) of 0.02 to 0.5% (Ref. 14, pp. 70-74). Ignition at temperatures >946°C leads to formation of UO_2 ; temperatures <800°C yield slow decomposition reactions.¹⁸ Presence of NO_3^- , F^- , Cl^- , or SO_4^{2-} does not interfere, but the presence of PO_4^{3-} leads to formation of uranyl pyrophosphate rather than U_3O_8 .¹⁹

2. Plutonium. Of the plutonium precipitates, only PuO_2 is reported to form a compound suitable for gravimetric analysis (Ref. 20, pp. 297-298). Even with PuO_2 the stoichiometry of the compound depends on initial composition and ignition temperatures, and even when the compound is heated to 1200°C the final composition is variable. In addition, PuO_2 is hygroscopic, with the amount and rate of H_2O adsorption dependent upon the ignition temperature. The suitability of gravimetry for determination of plutonium is questionable.

B. Spectrophotometric Methods

Spectrophotometric methods rely on the principle that a compound or complex in solution will absorb light of a specific wavelength in a quantity proportional to the concentration of the measured species. Generally, the concentration-absorbance function is a simple proportional relationship expressed by Beer's law, but variations may result at high concentrations or when other competing reactions occur. The RSD attainable by direct spectrophotometric methods usually is 0.5% or more and is seldom better than 0.2%. However, it can be improved to 0.05% with differential techniques that compare the absorbance of the unknown to a reference that is known precisely.

Specificity in spectrophotometry is a complex function of factors such as sharpness of absorption bands, specificity of reagents, other elements or compounds present, and the quality of the monochromators. Specificity often can be improved by using preliminary separations, masking agents, and pH control. Because of the nonspecificity of reagents for uranium and plutonium in the presence of each other and because of the nuclear reaction products (americium, neptunium, fission products), time-consuming separations often are involved. These additional separations may limit the precision in major element analysis. For these reasons, spectrophotometric methods are of limited use for measuring uranium and plutonium in fuel cycle materials and are usually restricted to process control applications and analysis of waste streams.

The sensitivity of many spectrophotometric reagents to high radiation levels necessitates several supplemental separations during analysis of dissolver or waste-stream solution, even in the absence of elemental interference.

1. Uranium. Spectrophotometric methods for determining uranium in reprocessing streams generally are limited to measurement of trace concentrations of uranium in waste streams and possibly in the final product plutonium. Reagents of interest are listed in Table I.

The method for determination of uranium with 2-(2-pyridylazo)-5-diethylaminophenol (PADAP) has been modified specifically for determining uranium in reprocessing-plant waste streams²⁵ and in plutonium nitrate and oxide products.²⁶ The method for determination of uranium as the tetrapropylammonium complex has been automated for determining uranium and plutonium simultaneously in scrap materials.³²

The uranyl-nitrate method^{21,22} used for in-line process control is sensitive to variations in nitric acid concentration and temperature and lacks the precision required for safeguards applications. Browning of the optics can be a serious problem in process streams containing fission products. However, by measuring absorbance at two wavelengths with an on-stream detector, the method has been adapted to measure uranium concentration in-line with a precision of 3%.⁴⁰ Instruments that use fiber optics for light transmission and narrow-band interference filters for wavelength isolation have been described.⁴¹⁻⁴³ Only the process flow cell and the fiber optics are exposed to radioactivity.

2. Plutonium. Most chromogenic reagents that react with plutonium also yield intensely colored species with uranium and many fission products; therefore, if plutonium is to be determined in reprocessing samples, a separation is required. For low plutonium concentrations, methods such as x-ray fluorescence, alpha, or gamma spectrometry may be preferred.

The tetrapropylammonium method³⁶ is of interest because uranium and plutonium can be determined sequentially in the same sample. The method has been automated.³² A more sensitive method for determination of plutonium with chlorophosphonazo III in n-pentanol was developed especially for use with an automated spectrophotometer. The relative standard deviation was 1.5% for the range of 2.5-17.5 $\mu\text{g Pu}$.³³

The differential spectrophotometric method of determining plutonium³⁷ can be used for plutonium-nitrate product with precision and accuracy equivalent to that obtainable by the best electrometric methods for the analysis of scrap material.³⁸

A rapid off-line spectrophotometric method has been described for determining plutonium in dissolver and other reprocessing samples.⁴⁴ Plutonium is oxidized to the hexavalent state using AgO ; neodymium is added as an internal standard; and plutonium and neodymium absorbances are measured in the near infrared region. Fission-product activity to 500 Ci/L and uranium concentration to 400 g/L do not interfere for determining 1-5 g/L of plutonium. Precision of better than 1.5% (1 σ) and an accuracy of 0.5% is claimed for an analysis time of 1 h.

TABLE I
SPECTROPHOTOMETRIC METHODS FOR DETERMINATION OF URANIUM AND PLUTONIUM

Element	Compound	Concentration Range or Amount/Sample	RSD (%)	Interferences; Comments ^a	Ref. No.
U	Uranyl nitrate	10-300 g/L	1-5	Temperature-sensitive	21,22
U	Dibenzoylmethane	30-750 µg	1.7	Ce, Th, W, Fe(III), Cr(IV); Pu at Pu/U = 2	14, ^b 23
U	Peroxide	1-25 mg	0.4-4	Cr, Cu, Fe, Mn, Ni	14, ^c 24
U	PADAP	1-100 µg	5	As ⁵⁺ , Cr ³⁺ , V ⁵⁺ , PO ₄ ³⁻ , SiO ₄ ²⁻ , Pu, Pd; Pu removed by extraction	25,26
U	PAR	1-100 µg	1	R.E.'s, Th(IV), Pu(IV)?	27
U	5-Me-2-PAN	10-80 µg	7	F ⁻ ; extraction removes most interfering cations	28
U	Bromo-PADAP	0.5-100 µg	2	Cr ³⁺ , V ⁵⁺ , PO ₄ ³⁻	29
U	Tetrapropylammonia	0.12-14 mg	0.3-10	Ce, Th	14, ^d 30-32
U	Arsenazo-III	1-50 µg	1-10	Th, Pu, R.E.'s, others	34,35
Pu	Tetrapropylammonia	1-30 mg	0.2-2	Cr ⁶⁺ , Th	32,36
Pu	Pu(III) nitrate	12-18 g/L	0.05	Differential spectrometry	37,38
Pu	Arsenazo-I	5-100 µg	2-6	U, Zr, others	39

^aR.E. = rare earth; fp = fission product.

^bRef. 14, pp. 337-343.

^cRef. 14, pp. 222-225.

^dRef. 14, pp. 343-353.

C. Electrometric Titration Methods

Both uranium and plutonium can be determined with high precision and accuracy by the use of titrations involving oxidation-reduction reactions. Electrometric methods are classified by the technique used to detect the titration end point, such as potentiometric, amperometric, or coulometric. All of these methods can provide RSDs of better than 0.1%. To attain the highest precision, weight burets should be used in all critical volume measurements.

Potentiometric titrations are based on measurement of the change in potential of the system as a component is removed by oxidation or reduction. Because the change in potential is the critical measurement, the accuracy of potentiometric titrations is limited by the sharpness of the potential break at the equivalence point. An RSD of 0.02 to 0.5% is attainable; therefore, the method is used for measurement of final product such as U_3O_8 , UF_6 , PuO_2 , or $\text{Pu}(\text{NO}_3)_4$.

Amperometric titrations involve measurement of the change in current between two electrodes that are maintained at a constant potential as titrant is added. The precision of amperometric titrations is limited by the accuracy of the inflection point measurement in the current-titration curve; hence the potential is selected such that the limiting current for either reactant, titrant, or product can be measured, depending on which yields the sharpest end point.⁴⁵

Controlled-potential and constant-current coulometry are well established methods for determining uranium and plutonium in solutions. Both methods are based on the principle that the weight of a substance oxidized or reduced at an electrode is proportional to the quantity of electric charge passed through the electrode.⁴⁵ In controlled-potential coulometry the potential of the electrode is maintained at a constant value relative to a reference electrode to minimize the number of reactions that can take place. Interference from the reactions occurring at lower potentials can be eliminated by performing a preliminary coulometric titration at a potential such that those reactions have occurred before measurements are started. Reactions requiring higher potentials cannot occur. Both uranium and plutonium can be titrated in the same sample without separation by performing successive titrations at different potentials. In constant-current coulometry an electrogenerated reagent reacts with the substance to be determined.

Coulometric methods can provide RSDs of 0.1% without the use of chemical standards. Generally, smaller samples are required than for potentiometric or amperometric measurements. Coulometric methods have gained increased popularity in recent years, but the determination of plutonium is subject to the following considerations (Ref. 46).

- The method is applicable to the determination of plutonium in nitrate product solutions and in dissolver solutions following an anion-exchange separation of fission products.
- The titration can be performed in 1N HClO_4 , 1N HNO_3 , 1N H_2SO_4 , or 1N HCl electrolytes. However, any Pu^{6+} , if present, is only partially reduced to Pu^{3+} in HNO_3 , HCl , or HClO_4 ; therefore, 1.0N H_2SO_4 is preferred if the sample is believed to contain appreciable Pu^{6+} . Titration in 1.0N H_2SO_4 must be performed in an oxygen-free atmosphere.

- If iron is present, it is titrated quantitatively in H_2SO_4 and a correction must be applied. Iron does not interfere in 1N HClO_4 if the plutonium-iron ratio is >50 .
- If 1N HNO_3 is used as electrolyte, sulfamic acid must be added to reduce nitrites.
- Polymeric plutonium does not titrate in any case and must be destroyed. Fluorides, used to destroy polymeric plutonium, and organics interfere in all media and must be removed.
- Although electrical calibration can be used, for optimum accuracy the instrument should be calibrated using NBS-SRM-949 and a correction for isotopic composition of the sample should be applied.
- Replacement of the platinum electrodes with gold can result in a lower and more reproducible background and an approximately five-fold improvement in precision.⁴⁷
- Coulometers may be subject to serious interference from nearby RF generators, such as induction heaters.

1. Uranium. The electrometric methods for determining uranium rely on the reaction $\text{U}^{4+} - \text{U}^{6+}$ after quantitative reduction of the uranium to the tetravalent state (Ref. 1, pp. 74-83 and 86-88; Refs. 48-51). Differences between methods result from the reductants and oxidants used. Reducing agents used to obtain U^{4+} include zinc amalgam, the Jones reductor, Ti^{3+} , the lead reductor, Fe^{2+} in phosphoric acid, and electrical reduction. The Jones reductor can generate a mixture of U^{3+} and U^{4+} , and the trivalent uranium must be air-oxidized to the tetravalent state before titration. The reduction with an excess of ferrous ion in concentrated phosphoric acid followed by titration with dichromate can be performed in the presence of Pu^{4+} without interference. Some of the standard electrometric techniques for determination of uranium are summarized in Table II. Of these methods for determining uranium, the Davies-Gray method⁴⁸⁻⁴⁹ is the most versatile and has been automated for rapid analysis of process samples (Ref. 14, pp. 77-83 and 86-88; Refs. 49-52).

2. Plutonium. For the electrometric determination of plutonium, the plutonium may be oxidized quantitatively to Pu^{6+} , then titrated to Pu^{4+} . This couple is preferred if uranium or iron is present. Oxidants for the first step include AgO and HClO_4 . If AgO is used, the excess is destroyed by gentle heating. The oxidizing power of HClO_4 is destroyed by diluting the sample after the plutonium has been oxidized. Errors can be introduced into the plutonium measurement if conditions for the dilution are not followed properly. The reduction to Pu^{4+} usually is carried out with Fe^{2+} , and can be performed directly or, by using a potentiometric end point, can be sharpened by adding a measured excess of Fe^{2+} , the excess being titrated with Ce^{4+} . Methods involving the Pu^{4+} - Pu^{6+} couple are summarized in Table III.

Alternatively, plutonium can be determined by quantitative reduction to Pu^{3+} and subsequent titration to Pu^{4+} (Ref. 14, pp. 268-299 and 385-388; Refs. 55, 57, 61-65). Common reducing agents are liquid zinc amalgam, the Jones reductor, the lead reductor, Ti^{3+} , Cr^{2+} , and electrical methods. The liquid-zinc-amalgam method suffers in that an extra transfer is required to remove the amalgam before titration. Small amounts

TABLE II
ELECTROMETRIC METHODS FOR DETERMINING URANIUM

<u>Reductant</u>	<u>Oxidant</u>	<u>End Point</u>	<u>Interferences</u>	<u>RSD^a (%)</u>	<u>Sample Size</u>	<u>Ref. No.</u>
Zn amalgam	K ₂ Cr ₂ O ₇	Colorimetric	None	0.62	200-300 mg	14, pp. 74-77
Fe ²⁺ H ₃ PO ₄	K ₂ Cr ₂ O ₇	Potentiometric	Ag, Hg	0.07	200-300 mg	48,49
Electrolytic	Excess K ₂ Cr ₂ O ₇ back-titrated with Fe ²⁺	Potentiometric	Cr, Fe, Mo, Ti, V	0.02	1-10 g	14, pp. 77-83
Electrolytic	Electrolytic	Coulometric	Fe, Hg, Ag, V	0.06-0.44	2-20 mg	14, pp. 86-88; 50
Pb reductor	Ce ⁴⁺	Potentiometric	Mo	0.03	0.5-1 g	51

^aGenerally, the optimum obtainable for fuel process or product samples.

TABLE III
ELECTROMETRIC METHODS FOR PLUTONIUM DETERMINATION USING THE Pu(VI)-Pu(IV) COUPLE

Reductant	Oxidant	End Point	Interferences	RSD ^a (%)	Sample Size	Ref. No.
Fe ²⁺	AgO	Amperometric	Ce, Cr, V, Mn Ru, Au, Rh, Pt Pb, Am, Np	0.03	10-20 mg	53,54 ^b 55,57
Excess Fe ²⁺ Ce ⁴⁺	AgO	Potentiometric	Ce, Cr, V, Mn	0.1-0.19	4 mg	14, pp. 261-268
Excess Fe ²⁺ K ₂ Cr ₂ O ₇	AgO	Potentiometric	Cr, Ce, Mn, V, Ru	0.17	20-80 mg	58
Excess Fe ²⁺ Ce ⁴⁺	HClO ₄	Potentiometric	Cr, Au, Mn, Pt,	0.03	200-250 mg	59 ^c
Electrically generated Fe ²⁺	KMnO ₄	Coulometric	Ag, Hg, Ru, Pd, Os, Ce, Au, Sn, (U)	1	0.003-10 mg	60
Electrical	Electrical	Coulometric		0.1	1-10 mg	46,56

^aGenerally, the optimum RSD obtainable for fuel process or product sample.

^bSee also Ref. 14, pp. 254-261.

^cSee also Ref. 14, pp. 274-285.

of titratable material can be leached from the Jones reductor. Fewer elements are reduced with TiCl_3 than with the Jones or lead reductors; therefore, there may be fewer potential interferences.⁶² However, Ti^{3+} is readily destroyed by contact with air.

The oxidation titration to Pu^{4+} usually is performed electrolytically or with Ce^{4+} or $\text{K}_2\text{Cr}_2\text{O}_7$. Dichromate has the advantage of being available as a primary standard and is more stable than Ce^{4+} over extended storage periods. Uranium and iron interfere with methods involving the Pu^{3+} - Pu^{4+} couple. An exception is the controlled-potential coulometric technique in the presence of uranium, and with this method, the two elements can be measured successively in the same sample.⁵⁵ A measurement technique for plutonium that incorporates a concurrent correction for iron or neptunium has been described.⁵⁶ For the controlled-potential coulometric determination of plutonium in dissolver solutions, zirconium must be removed completely to prevent fouling of the electrode through adsorption of hydrous zirconia.⁴⁶ Methods involving the electrochemical determination of plutonium using the Pu^{3+} - Pu^{4+} couple are summarized in Table IV.

D. Fluorometry

The fluorometric determination of low concentrations of uranium is based on the principle that uranium fluoresces when excited by ultraviolet light; the method is not applicable for determining plutonium. The fluorescent spectrum for uranium consists of four bands in the visible region, with the most intense band at 554.6 nm. If 365-nm excitation from a high-intensity mercury lamp is used, few other elements will interfere by fluorescence.

Samples generally are evaporated and fused in a carbonate flux containing 10% NaF (Ref. 66) or a NaF flux, sometimes containing LiF (Ref. 67) or some carbonate. Carbonate fluxes are preferred for ease of handling the fusion mass and for providing better analytical precision. Fluoride fluxes provide better uranium sensitivity but are more sensitive to variations in flux temperature and cooling conditions.

Although few elements or compounds interfere with the method by overlapping fluorescent spectra, a number of elements interfere through quenching or enhancement. Quenching can result from absorption of the incident light, absorption of the fluorescent light, interference with the energy transfer paths at phosphor centers, or precipitation of uranium.⁶⁸

Most of the transition elements interfere by quenching to some degree. With a flux of 98% NaF-2% LiF, serious quenching is observed in the presence of small amounts of cadmium, chromium, cobalt, copper, iron, magnesium, manganese, nickel, lead, platinum, silicon, thorium, and zinc. Many of the interferences can be removed by a combination of precipitation and solvent extraction of the uranium into methyl isobutyl ketone containing tetrapropylammonium nitrate (Ref. 14, pp. 398-403) or into ethyl acetate (Ref. 14, pp. 232-237). Alternatively, quenching interference can be minimized by measuring the fluorescence from a fused button of the sample and a similar button containing a known added amount of uranium (spiking).

Fluorometry has evolved as a standard method for determining small amounts (1-100 ng) of uranium. High-throughput, off-line measurements of uranium in waste streams can be made by processing samples in the analytical laboratory. The RSD for measuring uranium in simulated reprocessing

TABLE IV
ELECTROMETRIC METHODS FOR PLUTONIUM DETERMINATION USING THE Pu(III)-Pu(IV) COUPLE

<u>Reductant</u>	<u>Oxidant</u>	<u>End Point</u>	<u>Interferences</u>	<u>RSD (%)</u> ^a	<u>Sample Size</u>	<u>Ref. No.</u>
Zn amalgam	Ce ⁴⁺	Potentiometric	Fe, Mo, Ti, W	0.02-0.02	600 mg	14, pp. 274-285
Zn amalgam	K ₂ Cr ₂ O ₇	Potentiometric	Cr, Fe, Mo, Ti, U, V, W	0.03	250-400 mg	61 ^d
Pb or Jones reductor	Ce ⁴⁺	Spectrophotometric	Fe, Ga, Mo, Ti, U, W, V, Cr, NO ₃	0.07	500 mg	14, pp. 268-274
Ti ³⁺	Ce ⁴⁺	Potentiometric	Cu, Fe, U, W NO ₃	0.1	25 mg	62 ^e
Cr ²⁺	Ce ⁴⁺	Potentiometric	Fe, U, Cu, W	0.4	20-50 mg	57
Electrical	Electrical	Controlled potential coulometric	Dependent on electrolyte	0.04 ^b 0.63 ^c	1-10 mg	47, 63-65 ^f

^aGenerally, the optimum RSD obtainable for fuel process or product samples.

^bFor nitrates.

^cFor dissolver solutions.

^dSee also Ref. 14, pp. 285-292.

^eSee also Ref. 14, pp. 293-299.

^fSee also Ref. 14, pp. 385-388.

plant waste streams is reported to be 13%.⁶⁹ In addition, an automated fluorimeter capable of processing sixteen samples per 1.5 h has been designed and tested at ORNL;⁷⁰ sensitivity is 0.2 $\mu\text{g/L}$ and precision is 6% at the 10- $\mu\text{g/L}$ level.

Recently, a laser fluorometric method has been described that is simpler, more sensitive, and more rapid than the pellet fusion method.^{71,72} The method is direct and requires no separations, extractions, or fusions. The commercially available laser fluorometer employs a pulsed nitrogen laser that emits at 337 nm. The uranium sample is added to a proprietary buffered pyrophosphate reagent, which causes the uranium to emit a green luminescence when exposed to the light from the laser. A green filter between the sample cell and the photomultiplier tube removes the blue fluorescence from organics. Longer wavelengths will still be transmitted, however. The instrument has electronic gating circuitry that waits for other fluorescence from organics to decay before the longer-lived uranium phosphorescence is measured. A standard addition technique is used to overcome sample matrix effects.

E. X-Ray Fluorescence

X-ray fluorescence techniques have been applied to the analysis of actinide solutions for over 25 yr. They are sensitive and accurate and are capable of measuring microgram quantities of material to relative accuracies of 1% or less. Typical analysis times are short (0.5 h or less). Currently, x-ray fluorescence is used primarily to measure solutions from the reprocessing of spent fuels. However, the technique has been applied to solutions at all stages of the fuel cycle.

The fundamentals of x-ray fluorescence analysis are reviewed in detail in the literature.⁷³⁻⁷⁶ The analyst has considerable latitude in selecting a technique, including choices of excitation source, x-ray line, and detector.

For uranium and plutonium measurements, lines in the K, L, or M series can be used; the more intense lines in these series are summarized in Table A-III of Ref. 77. The main considerations in the choice of a line to use in an x-ray fluorescence measurement are line intensity above background and interferences from adjacent lines. The most common choice is one or more of the L lines. They provide good intensities in an air path for solution or solid analyses, and can be used with any type of excitation or detection system.

The most commonly used excitation source is the x-ray generator. The use of radioisotopes is becoming more common because solid-state detectors required to measure the resulting lower intensity x rays are readily available. Both wavelength- and energy-dispersive detection systems have been used to measure uranium and plutonium in solution. Wavelength dispersion offers high resolution but at low efficiencies. Energy dispersion is generally more efficient but has poorer resolution at energies >20 keV. At energies >20-25 keV, a Si(Li) energy-dispersive spectrometer provides better resolution as well as efficiency. Energy dispersion using Ge(Li) or intrinsic germanium detectors is required for measuring actinide K-series x rays.

Although x-ray fluorescence measurements of nuclear material solutions have had wide exposure in the literature, current safeguards applications are limited because of competition from other less complex and less matrix-dependent methods. To reduce the matrix sensitivity, samples are often

evaporated as thin films. That technique reduces the scattered background compared to the sample signal, thus increasing sensitivity and reducing counting times.⁷⁸ Measurements of thin films are linear with the concentration of nuclear material over a small range, above which saturation effects become important.

Several systems have been developed to assay highly radioactive spent-fuel solutions. One of these systems⁷⁹⁻⁸¹ uses the $L\alpha_1$ x rays measured with a wavelength-dispersive spectrometer to assay solutions having uranium-plutonium ratios of 50:1 to 300:1 with accuracies of ~1% (1 σ) in 2-5 min. Solutions having activities of <1 Ci/mL are handled routinely by evaporating small samples onto a filter paper. Samples having low beta-gamma activities can be analyzed directly in solution after addition of an internal standard. The fission products cause no serious line interference. An automatic sample-preparation system is being developed to allow on-line analysis.

Uranium and plutonium solutions that have beta-gamma activities to 1 Ci/L have been measured directly through a Plexiglas window.⁸² An automatic sampler removes solutions from the sample line and yttrium is added to serve as an internal standard. Solution transfers are made by pneumatic tube. A minicomputer performs data reduction, and overall precision and accuracy of better than 1% are claimed for the technique.

In another system,⁸³ dissolver solutions containing uranium and plutonium at ratios up to 400:1 and uranium concentrations of ~50 g/L are measured with an x-ray tube in combination with a Si(Li) detector. Preliminary tests indicate that accuracies of 1-2% are possible with 10-min analysis times. However, accuracies of only 3% were obtained because of problems in sample preparation.

A proposed wavelength-dispersive system may be able to measure uranium and plutonium concentrations in irradiated samples to a precision of better than 1%.⁸⁴ The instrument should be adaptable to off-line analysis of dissolver-type samples (as slurries or solution) with an analysis time of 4-8 h. For samples of lower activity, analysis time is significantly shorter.

A system also has been developed for automatic sampling and sample preparation of dissolver solutions from the reprocessing of thorium-uranium fuels⁷⁸ and could be applied to uranium-plutonium fuels. For solutions emitting up to 2000 Ci/L, samples containing 1.0 mL of solution are automatically aliquoted and mixed with an internal standard. The aliquots are evaporated onto a filter paper, which is transferred to a shielded x-ray spectrograph. All operations are performed remotely under computer control.

The introduction of stable x-ray generators⁸⁵ that can operate to 200 KeV and the use of ^{57}Co gamma-ray sources^{86,87} have increased interest in K x-ray fluorescence analysis of uranium and plutonium. X-ray generators provide a significant improvement in peak-to-background intensity compared to radioactive sources such as ^{57}Co . The primary advantage in using the K x rays results from the ability to use normal process materials such as stainless steel for cells; thus, in-line analysis is possible. The method is proposed for analysis of samples at any step of reprocessing from dissolver solutions to final product. Tests of an on-line monitor for uranium concentrations in process streams showed that 300-sec data collection times were sufficient to measure concentrations of 20-150 g/L with less than 2% error.⁸⁸ Energy-dispersive K x-ray fluorescence and ^{57}Co sources were used.

The high sample throughput for analytical laboratories at large re-processing plants may dictate automated analytical methods such as x-ray fluorescence to provide timely, accurate information for both process control and safeguards.

F. X-Ray Absorption Edge Densitometry

X-ray absorption edge densitometry is an element-specific analytical method that can be applied in-line, at-line, or off-line to many process measurement needs. In this technique the transmitted intensity through the sample is measured for two x rays or gamma rays selected above and below an absorption edge for the element determined. For determining uranium and plutonium both K and L_{III} absorption edges can be used. The L_{III} edge is useful for uranium and plutonium concentrations below ~100 g/L. The K edge allows greater versatility in selecting cell materials and cell thickness and, thus, is applicable over a wider concentration range in process-type cells. With proper selection of cell path length and absorption edge, the method is applicable to concentrations in the range 1-500 g/L.

Instruments using K and L_{III} edges have been developed at the Los Alamos National Laboratory for determining uranium and plutonium in a variety of process samples.⁸⁹ A radioactive ^{169}Yb source is used for routinely determining uranium in the 100-400 g/L range. The precision is better than 0.5%. For the determination of plutonium, a radioactive ^{75}Se - ^{57}Co source is employed. Field tests at Tokai-mura, Japan, have demonstrated a precision of 0.3% over the concentration range of 150-300 g/L.^{90,91} A K-edge instrument designed by Lawrence Livermore National Laboratory and evaluated at Allied General Nuclear Services gave a precision of 0.5% for the design concentration range of 150-300 g/L.⁹²⁻⁹⁴ An on-line K-edge densitometer designed for lower concentration, typically 30 g/L, is being evaluated at the Savannah River Plant. Solution is circulated from process tanks through a 7 cm path length cell. Well characterized control samples having concentrations in the range 25-40 g/L were analyzed with a precision of 0.3%.⁹⁵

An L_{III} edge instrument using an x-ray generator as a source to permit simultaneous determinations of uranium and plutonium was designed at Los Alamos for the Savannah River Laboratory.⁹⁶ Precision of 0.25% was reported at the optimum concentration of 50 g/L.

A versatile densitometer that can be applied to K or L edge analysis using x-ray tube excitation has been designed and evaluated at Karlsruhe.⁹⁷ Proposed applications include determination of uranium in accountability tank solutions and both uranium and plutonium in intermediate process and final product solutions. The x-ray tube excitation approach to absorption edge densitometry also is being developed in the UK.⁹⁸

K-edge densitometry has a limited dynamic range and is unable to provide precise plutonium assays in the presence of a large excess of uranium. For instance, plutonium in LWR dissolver accountability solutions, where the ratio of uranium to plutonium concentration is 100:1, cannot be determined by K-edge densitometry. An instrument that allows simultaneous K x-ray fluorescence measurements and K-edge densitometry may provide the solution to this problem.⁹⁹

G. Alpha-Spectrometric Methods

Quantitative alpha-particle spectrometry is based on measurement of the alpha-radiation intensity of the sample. The alpha particles are ejected with discrete energies, and for the uranium, neptunium, plutonium, and americium isotopes found in the nuclear fuel cycle these energies range from 4 to 5.5 MeV. Alpha-particle energies are rapidly degraded by interactions with bound electrons, and complete attenuation is effected by ~5 cm of air. Absorption by H₂O and self-absorption by the sample are even more rapid, and normal quantitative alpha spectrometry is limited to analysis of thin, dry films. The method has been applied to determination of plutonium in dissolver solutions following solvent-extraction separation of the plutonium (Ref. 100; Ref. 14, pp. 392-398). The RSD for dissolver samples is in the range 3-5% (1 σ). The method also is used for determining plutonium in process waste and uranium product streams.¹⁰¹

Detectors used for alpha spectrometry include standard radiation instruments such as proportional counters, scintillation detectors, and solid-state devices. Special techniques have been developed for alpha-counting of Purex process streams, where the detector must be in direct contact with the alpha-radiation sources.¹⁰²⁻¹⁰⁶ Performance requirements are severe. The detector must operate in a flowing-liquid environment that is often of high acidity; it must be corrosion- and abrasion-resistant; it should discriminate alpha signals in the presence of high beta-gamma fluxes from fission products; and it should be insoluble in organics. Originally, ZnS(Ag) scintillation detectors were used, but were too fragile and difficult to fabricate. Organic scintillators in general cannot withstand the liquid environments. A cerium-activated Vycor glass detector has been developed¹⁰³ that will operate for months in a Purex process stream. The stopping distance for 5-MeV alpha particles in Vycor glass is estimated to be 25 μ m, whereas beta particles of energies >0.5 MeV have penetration thicknesses in excess of 900 μ m.¹⁰⁵ Thus, by reducing the cerium-activated layer to the alpha-particle penetration thickness, the alpha particles can be stopped in the active layer while most of the high-energy betas will pass through undetected. A further decrease in beta interference is attained by using thin sample cells¹⁰⁷ and electronic pulse-shaping circuitry.¹⁰⁸

The instruments can measure plutonium solutions in the concentration range between 10⁻⁴ and 1 g/L in the presence of beta-gamma fields up to 10⁸ dis/min-mL.¹⁰⁹ Deviations from linearity at low plutonium concentrations (<10⁻³ g/L) are caused by plating on the detector window and at high plutonium concentrations (>0.1 g/L) are caused by pulse pile up. Interference results from large variations in sample density that cause self-attenuation problems and from the presence of other alpha emitters such as neptunium, americium, or uranium. Calibration is performed empirically for each stream by chemical analysis. Modification of the electronics system was reported to provide a range of 0.1 to 30 g/L.¹¹⁰

In-line alpha counters using a scintillation (ZnS) probe and a rotating drum to present the sample to the probe were developed in France for routine use in reprocessing plants.¹¹¹ The relative standard deviation is 3.2% for the aqueous phase and 4.6% for the solvent phase in the first cycle extraction and partition banks. An improved version of this alpha monitor was compared at Karlsruhe with an instrument that uses Ce-activated Vycor glass.¹¹² It was concluded that the drum cell is more likely to fail because of its moving parts and is less sensitive, but can be used in

the concentration range 1-10 mg Pu/L. The other instrument can be used for concentrations below 1 mg Pu/L, but the detector surface is easily contaminated.

Diamond detectors were investigated in the Soviet Union as in-line alpha monitors, but data for process operation in the presence of high beta-gamma fluxes are not available.¹¹³

H. Mass-Spectrometric Methods

In most nuclear fuel cycle facilities thermal-ionization mass spectrometry is used to determine the amount of each isotope of uranium and plutonium and, subsequently, to determine the effective atomic weights used in calculating the total uranium and plutonium from the chemical analysis of accountability-tank samples. Isotope-dilution mass spectrometry can be used for high precision-high accuracy measurement of the plutonium and uranium concentrations in tanks.

The instrumentation for mass spectrometry is complex and expensive, consisting basically of a source, which is either a single or triple filament; a mass spectrometer; a detector, which is either a Faraday cup or an electron multiplier; and associated control and readout electronics.^{114,115} Much of the instrumentation can be computer controlled.¹¹⁶ Single-filament ion sources require precarburization to provide high metal ion yields, and triple filaments are preferred. The mass spectrometer should have an abundance sensitivity of 100 000 or better and a resolution, expressed as $M/\Delta M$, of 400 or greater, where ΔM is the peak width at 5% of the height above background. The Faraday cup detector provides better measurement accuracy but the electron multiplier yields better sensitivity.

The methods require elaborate sample preparation (Ref. 14, pp. 353-368; Refs. 117-119). A typical mass-spectrometric procedure involves (1) dissolving and diluting the sample with HNO_3 , (2) for isotope-dilution mass spectrometry, spiking part of the diluted sample with accurately known amounts of ^{233}U and ^{242}Pu or ^{244}Pu , (3) oxidizing plutonium to the hexavalent state, (4) sequentially separating the fission products, plutonium, and uranium by an anion-exchange procedure or by solvent extraction; and (5) transferring aliquots of the separated uranium and plutonium fractions to separate mass spectrometer filaments.

Because small amounts of uranium and plutonium (<100 μg , and generally <1 μg for plutonium) are to be analyzed, extreme care must be taken in sample preparation and filament loading to avoid contamination by environmental uranium. Determination of ^{238}Pu is particularly sensitive to the presence of ^{238}U , and for ^{238}Pu concentrations <0.7%, alpha spectrometry is preferred. For the measurement of ^{241}Pu , ^{241}Am must be completely separated. For the isotope-dilution mass-spectrometric determination of plutonium, both spiked and unspiked samples must be analyzed to correct for ^{242}Pu in the sample, if ^{242}Pu is added as the spike.

Chemical errors may arise also from the presence of high concentrations of alkali elements. Potassium, in particular, interferes with the measurement of ^{234}U and ^{236}U because of overlap of K_6 polymers. Filaments must be checked to ensure that they are free of organics. If organics are detected at ion currents greater than background, the analyses are suspect.

In thermal ionization sources, the light isotopes are evaporated and ionized preferentially relative to the heavier isotopes of an element.¹¹⁵ This effect is temperature dependent and is a function of the

mass, sample size, and time. It is referred to as the filament bias and must be measured for the instrument every 6 months, and must be controlled as accurately as possible between samples and standards in a series. Temperature can be controlled by direct pyrometer measurement or by adjusting to a constant signal intensity of a filament ion. Double spiking with ^{233}U and ^{236}U has been proposed to increase the accuracy of uranium isotopic measurements.^{120, 121} This technique should cancel almost completely the effects of mass fractionation during the analysis.

The abundances of major isotopes can be determined with RSDs of 0.01 to 0.02%.¹²² The precisions obtained for well-characterized material (a solution of a standard reference material) are significantly better than for process and product solutions, particularly at isotopic concentrations >1%. Studies show that improvements in precision and accuracy may be obtained if proper care is taken to minimize instrumental and operating errors. In an interlaboratory exchange program an overall RSD of 0.1% was obtained in measuring total plutonium in four samples having different isotopic compositions.¹²³ In terms of these data an RSD of 0.1 to 0.2% should be attainable for measuring total plutonium in fuel cycle samples if careful attention is paid to details of separation chemistry and instrument operation.

High gamma-radiation levels limit the mass-spectrometric analysis of dissolver solutions to extremely small samples or to samples that have been purified previously in shielded facilities. Microsamples of uranium and plutonium can be adsorbed on ion-exchange resin beads for subsequent analysis.¹²⁴ The analytical precision is better than 1% (RSD). Alternatively, the mass spectrometer can be modified for analysis of radioactive samples by enclosing the sparking chamber and the initial stage of the flight tube with 10 cm (4 in.) of lead shielding.¹²⁵ Samples emitting up to 100 R/h of gamma activity can be analyzed in this fashion.

Mass spectrometry currently provides the most accurate isotopic analysis, even in the presence of fission products. The isotope-dilution method provides overall accuracy of 0.3-1% for the measurement of total uranium and plutonium concentrations in dissolver solutions. However, it is believed that improvements in sampling and analysis techniques could improve accuracy to better than 0.3%. The control of sample preparation is crucial to the accuracy, particularly where analyses are performed on nanogram amounts of uranium or plutonium.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 12: SAFEGUARDING NUCLEAR POWER STATIONS

The basic features of nuclear fuel accounting and control in present-day power reactors are considered. Emphasis is placed on reactor operations and spent-fuel characteristics for light water reactors (LWRs) and heavy water reactors (HWRs).

After the session, participants will be able to:

- Describe the basic fuel characteristics in LWRs and HWRs.
- Describe typical fuel quantities (fresh, in-core, and spent) at LWR and HWR facilities.
- Describe the movement of fuel and basic fuel management practices within each facility.
- Describe the basic fuel accounting and inventory verification procedures for both LWR and HWR facilities.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



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SESSION 12: SAFEGUARDING NUCLEAR POWER STATIONS

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I. INTRODUCTION

Most of the world's inventory of plutonium is contained in the spent-fuel assemblies that reside in the spent-fuel ponds of nuclear power stations. Because reprocessing of these spent-fuel assemblies is occurring at a very low rate and because away-from-reactor storage has not yet occurred to any significant extent, the world's inventory of plutonium will by necessity remain at nuclear power stations for many years. Thus, the nuclear power station is of significant nuclear safeguards interest.

The discussion in this paper focuses on the single facility--the nuclear power station with its inventories of fresh fuel, in-core fuel, and spent fuel. The focus is on the off-load refueled, light-water-cooled power reactor (LWR) because it is found in the greatest numbers in the world; however, attention is also given to the on-line refueled, heavy-water-moderated and cooled reactor (HWR).

The nuclear power station has several characteristics, which are unique to safeguards, in the nuclear fuel cycle. The nuclear material is almost always found in discrete, encapsulated units (called fuel assemblies or fuel bundles), and it remains in the same physical form during its entire residence time at the power station. It arrives at the power station in the form of fuel assemblies, it resides in the reactor core as fuel assemblies, and it is stored in the spent-fuel pond as fuel assemblies. The integrity of the assemblies is therefore maintained. Fuel assemblies are rarely disassembled at nuclear power stations; however, this could change in the future and introduce new safeguards problems. At all other facilities in the nuclear fuel cycle--except the away-from-reactor storage pond--the nuclear material can change both physical and chemical form.

The nuclear power station is the only facility in the entire fuel cycle where large quantities of fissile materials [uranium-235 (^{235}U) and plutonium-239 (^{239}Pu)] are consumed and produced. Nuclear material is not conserved. The ultimate result of this consumption and production of fissile materials is, of course, the generation of electrical energy.

Because the integrity of the fuel assemblies is maintained and because the nuclear material content of the fuel assemblies is not conserved, safeguarding at nuclear power stations is primarily done by item accountability, containment, and surveillance, which will be addressed later in detail.

II. CHARACTERISTICS OF NUCLEAR POWER STATIONS^{1,2}

A nuclear fission power reactor provides an environment in which fission reactions are initiated, sustained, and controlled; and it provides for removal of heat for power production. Shown in Fig. 1 are certain components

common to all reactors: the core, coolant, control rods, and shielding. Most current power reactors also include a moderator. The coolant, either liquid or gas, flows over the fuel rods and removes the fission heat from the fuel. The control rods, usually made of materials that readily absorb neutrons, are positioned inside the fuel assembly (core) to regulate the fission chain reaction. The shielding consists of special materials that surround different portions of the reactor system to prevent harmful radiation from escaping into the local environment.

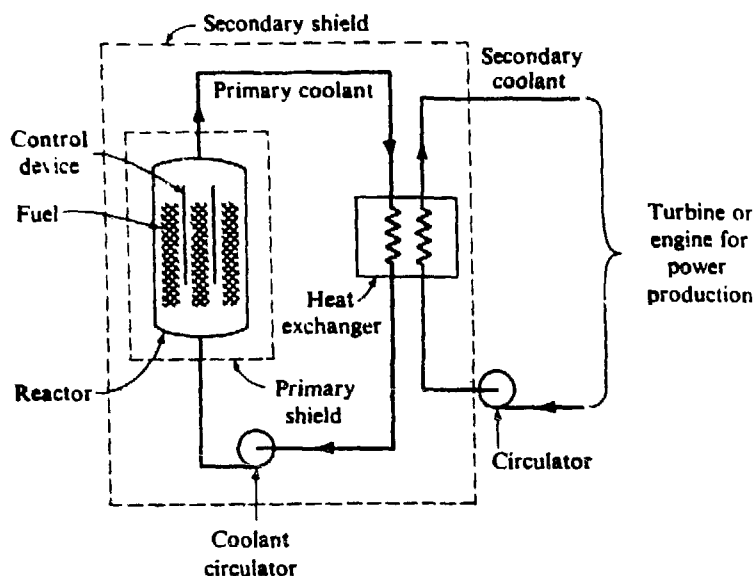


Fig. 1. Schematic of a nuclear power reactor.

The core of the nuclear reactor is that volume that contains the fission fuel. The fissile fuels used in nuclear reactors are ^{235}U , uranium-233 (^{233}U), or plutonium-239 (^{239}Pu). Uranium-238 (^{238}U) is not a fissile fuel but it is a fertile material that leads to the production of fissile ^{239}Pu . Enrichment of the fuel refers to the amount of fissionable material in the fuel. In the case of uranium it means the isotopic percentage of ^{235}U in the fuel (0.7% is contained in natural uranium). Typical water-cooled power reactors contain 2-4% ^{235}U .

The kinetic energy of the neutrons inducing fission also constitutes an important characteristic of nuclear reactors. Thus, a thermal reactor is one in which fission is induced by slow neutrons (neutrons in thermal energy equilibrium with the reactor core material). Most of today's power reactors are indeed thermal. A moderator is put into the reactor core to slow the neutrons to thermal energies by scattering collisions. Fast reactors make no attempt to slow the high-energy neutrons produced in the fission chain reaction and thus they contain no moderator. Fast neutron fission favors breeding and opens the possibility of converting vast supplies of ^{238}U and thorium-232 (^{232}Th) to fissile nuclear fuel.

Of the many possible types of nuclear fission power reactors, only two have so far attained worldwide use in electric power systems. These are the light-water-cooled, light-water-moderated reactors developed initially in the United States and the heavy-water-cooled, heavy-water-moderated reactors developed initially in Canada. Gas-cooled, graphite-moderated, thermal reactors,

developed initially by Great Britain, have seen only limited application. Although fast breeder reactors hold promise for the future, they are still under development.

As the name implies, LWRs use ordinary water (H_2O , as opposed to heavy water D_2O) as both coolant and moderator. The two common versions of LWR are the boiling-water reactor (BWR) and the pressurized-water reactor (PWR). The major difference between the PWR and the BWR is in the operating pressure. In the PWR, typical operating conditions are 2200 psia (pounds per square inch absolute--compared to 14.7 psia for normal atmospheric pressure) and $330^\circ C$, a greatly subcooled condition. At this very high pressure, the water in the reactor cannot boil, even at a $330^\circ C$ temperature. The high-pressure water removes fission heat from the reactor core and is circulated through a steam generator to produce $290^\circ C$, 1000 psia steam. The steam then drives a turbine-generator to produce electrical energy.

In the BWR, the operating pressure is reduced to about 1000 psia; the cooling water at this lower pressure boils into $290^\circ C$ steam directly within the reactor vessel. After passing through moisture separators and steam dryers, the steam then goes directly to the turbine. Thus, the BWR does not require an intermediate heat exchanger (steam generator).

One disadvantage in the use of LWRs for electrical power generation is the relatively low steam temperature ($290^\circ C$) and resulting low thermal-conversion efficiency. An LWR generates electricity with about 32-33% efficiency as compared with 36-38% net efficiencies achieved with modern fossil-fueled plants.

The use of natural uranium, heavy-water, and on-line refueling are the basic elements in the HWR design philosophy. The most common version of HWR is the CANDU reactor manufactured and marketed worldwide by Atomic Energy of Canada Limited (AECL). CANDU systems employ D_2O as both the moderator and the coolant. The term CANDU is essentially synonymous with CANDU-PHW, the Canadian Deuterium Uranium - Pressurized Heavy Water reactor.

The CANDU-600 reactor, the most common design today, can produce more than 600 MWe with a thermal efficiency in the range of 29-30%. The reactor vessel is a large, horizontally oriented cylindrical tank that contains the low-pressure heavy-water moderator. This tank is penetrated by a number of horizontal fuel channels that contain the natural-uranium fuel and the pressurized, heavy-water coolant. The coolant is pumped through the fuel channels and then through heat exchangers (steam generators) to produce light-water steam at 400 psia and $260^\circ C$, which is then fed to the turbine.

III. "TYPICAL" FUEL CHARACTERISTICS AT NUCLEAR POWER STATIONS

The fuel assemblies for the two types of LWRs are very similar. Both use uranium dioxide as the fuel material; the uranium is enriched to 2-4% ^{235}U (Refs. 1,2). The slightly enriched uranium dioxide is fabricated into the form of cylindrical fuel pellets (~8-10 mm diam and ~10 mm long). The pellets are then loaded into long zirconium-alloy cladding tubes to produce fuel pins or fuel rods (~4 m long). A rectangular array of pins forms the final fuel assembly, or fuel bundle. The fuel assemblies are then placed in a steel pressure vessel in a right circular cylinder array. The number of assemblies required depends on the geometrical arrangement, the reactor type, the uranium enrichment, and the operating power level of the reactor.

A shortened cutaway version of a BWR fuel assembly is shown in Fig. 2. The 8 x 8 square pin array typically contains approximately 62 fuel pins per assembly.^{1,3} The PWR pin array, 16 x 16 or 17 x 17, is larger than that of the BWR and the fuel assembly shown in Fig. 3 typically contains 236-264 fuel pins.^{1,4}

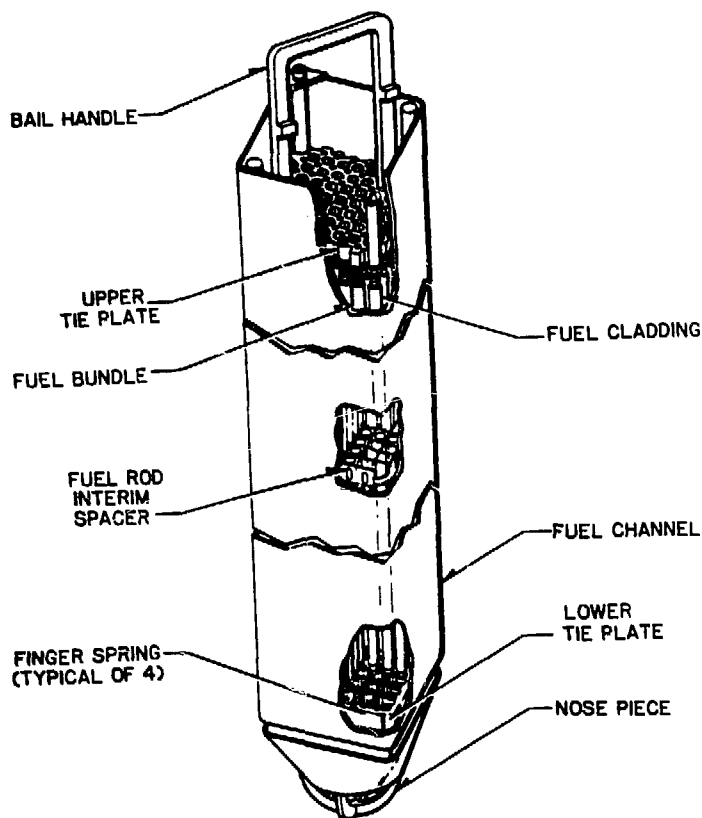


Fig. 2. BWR fuel assembly - shortened cutaway view.³

Both the BWR and PWR fuel assemblies have unique serial numbers engraved on the top plate for positive identification.

In the HWR each fuel bundle (about 500 mm long and 100 mm in diameter) is made up of 37 Zircaloy-clad fuel pins containing natural UO_2 pellets (~12 mm diam and ~16 mm long), Fig. 4 (Refs. 1,5). Each fuel bundle contains 20-21 kg of natural uranium dioxide. The heavy-water moderator is contained in a horizontal reactor vessel (calandria), which is about 7.6 m in diameter by 4 m long. The vessel is penetrated by 380 calandria pressure tubes. Twelve fuel bundles are placed end-to-end in each of the pressure tubes such that the heavy-water coolant flows through the tubes and the fuel bundles simultaneously.

The characteristics of the PWR fuel listed in Table I correspond to a typical 1000-MWe station that operates on a once-through fuel cycle. The reactor is refueled off-load once a year; that is, the reactor is shut down for refueling. About one-third of the core (approximately 65 fuel assemblies) is replaced during the refueling. The uranium in the fuel is enriched to 2-4%.

The characteristics of the BWR fuel in Table I also correspond to a 1000-MWe plant operating on a once-through fuel cycle. This reactor is refueled off-load once a year during which about one-fourth of the core (or approximately 190 fuel assemblies) is replaced. The uranium in the fuel is enriched to 2-3%.

In both of these LWRs, the fuel in the reactor is inaccessible during periods of operation. The top of the reactor pressure vessel must be removed before the refueling can take place. The fuel assemblies at a LWR are therefore basically stationary during most of the year.

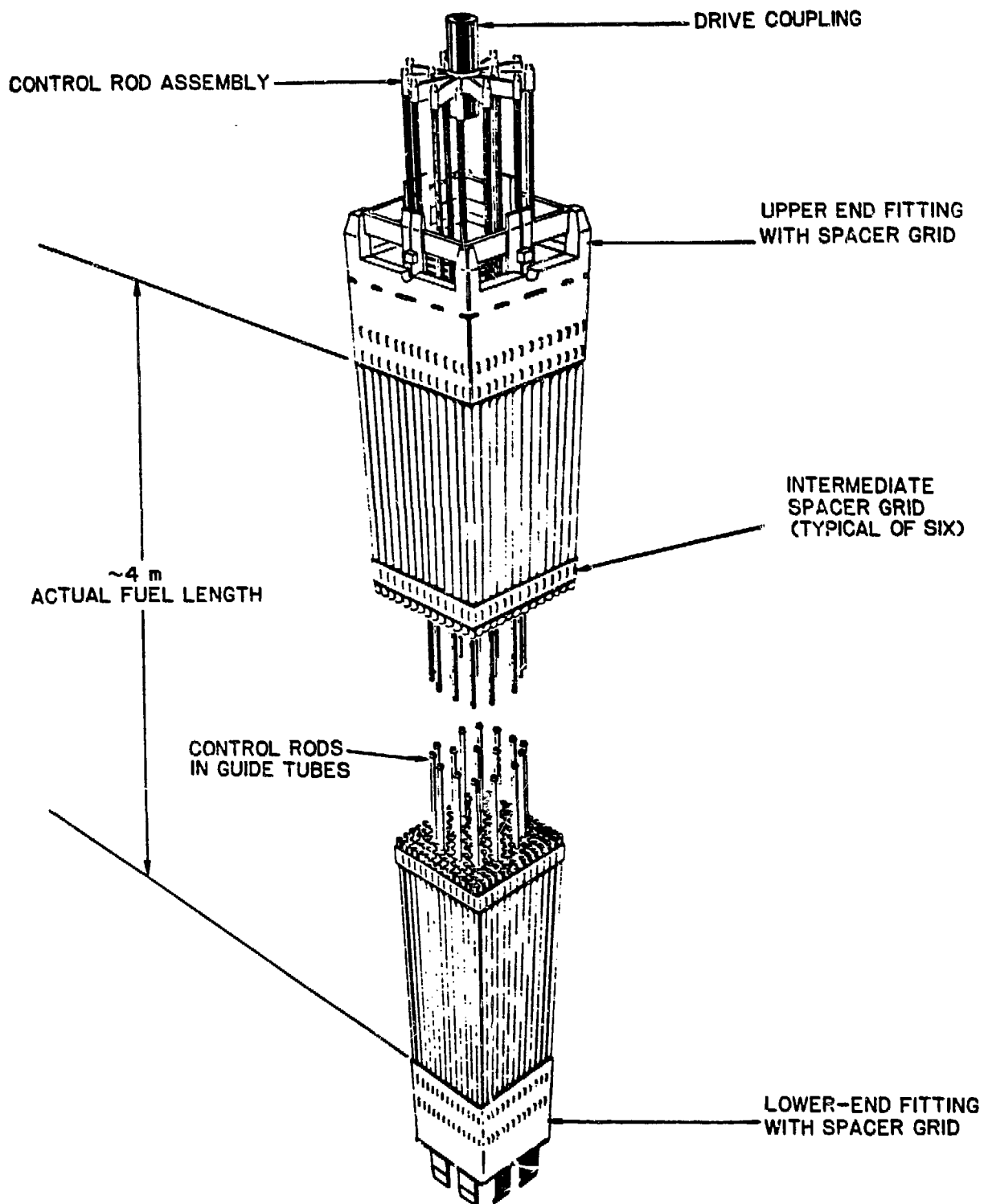


Fig. 3. PWR fuel assembly - cutaway showing partially inserted control rod assembly.⁴

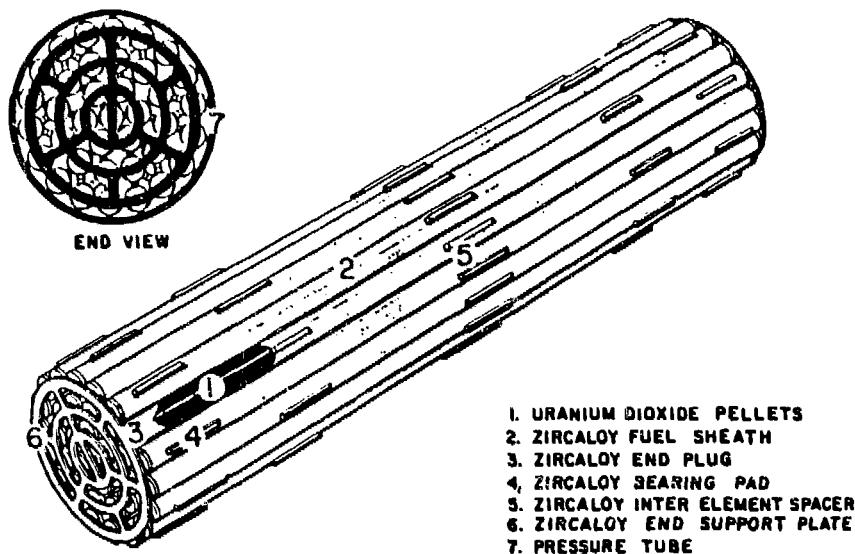


Fig. 4. CANDU fuel bundle.⁵

The characteristics of the HWR fuel listed in Table I correspond to a 600-MWe unit, characterized by the CANDU-600.⁵ This reactor is refueled on-line; that is, refueling is done while the reactor is running. The reactor contains approximately 4600 fuel bundles. About 16 fuel bundles are replaced each day while the reactor is operating at full power.

IV. "TYPICAL" FUEL INVENTORIES AT NUCLEAR POWER STATIONS

As illustrated in Fig. 5, the inventories of nuclear material at nuclear power stations can be grouped as:

- Fresh fuel
- In-core fuel
- Spent fuel.

The amount of nuclear material in the power station depends on the reactor type, the reactor power level, and the operating history of the station. Estimates of the nuclear material characteristics and flow for "typical" LWRs and HWRs are given in Table I.

Typical fuel assembly inventories for the three types of nuclear power stations are shown in Table II. The PWR has the smallest number of fuel assemblies in the inventory and the HWR has the largest. Spent-fuel ponds are usually designed to hold spent-fuel assemblies from several years of refueling.

Because the "back end" of the fuel cycle is developing slowly, many spent-fuel ponds throughout the world are being reconfigured to hold even more spent-fuel assemblies. Thus, the number of spent-fuel assemblies remaining at the spent-fuel ponds of nuclear power stations is increasing.

The large throughput of fuel bundles in the HWR causes the number of items in the spent-fuel pond to be very large. More than 20 000 spent-fuel bundles will accumulate at a single 600-MWe power station in 5 years.

TABLE 1

TYPICAL POWER REACTOR FUEL CHARACTERISTICS*

Typical PWR (1000 MWe)

Off-load refueling	1 year
Fuel enrichment	2-4%
Core inventory	~200 fuel assemblies 100 000 kg
Reload	~65 fuel assemblies
Spent fuel	~65 fuel assemblies
Pu production	~200 kg/year
Pu content spent fuel	~3 kg/assembly

Typical BWR (1000 MWe)

Off-load refueling	1 year
Fuel enrichment	2-3%
Core inventory	~750 fuel assemblies (150 000 kg)
Reload	~190 fuel assemblies
Spent fuel	~190 fuel assemblies
Pu production	~200 kg/year
Pu content spent fuel	~1 kg/assembly

Typical HWR (600 MWe)

On-line refueling	16 fuel bundles/day
Fuel enrichment	Natural uranium (0.72%)
Core inventory	4600 fuel bundles (380 pressure tubes x 12 fuel bundles per pressure tube) 98 000 kg
Reload	On-line 4500 fuel bundles/year
Pu production	~300 kg/year
Pu content spent fuel	~0.07 kg/bundle

*Adapted from Refs. 1, 2, 3, 4, 5, and 13.

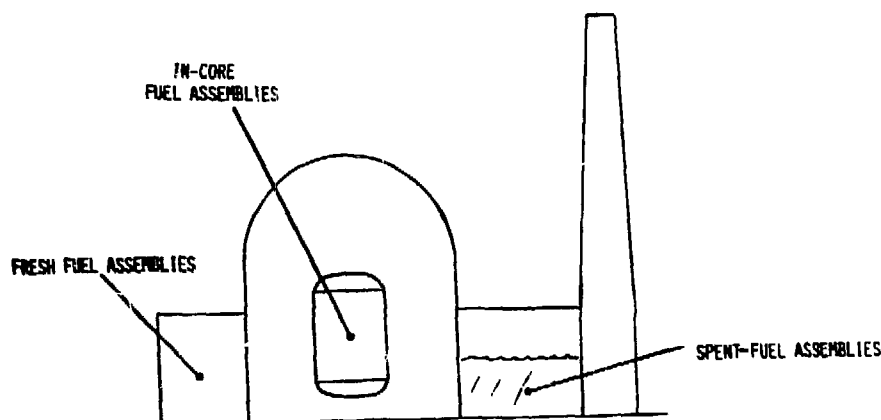


Fig. 5. Inventories of nuclear fuel at a nuclear power station.

TABLE II

TYPICAL FUEL INVENTORIES AT NUCLEAR POWER STATIONS¹³
(Number of Fuel Assemblies/Bundles)

<u>Reactor Type</u>	<u>Fresh-Fuel Storage</u>	<u>In Core</u>	<u>Spent-Fuel Storage</u>
PWR	75	200	Few hundred
BWR	200	750	Several hundred
HWR	3000	4600	Several thousands

V. SAFEGUARDING NUCLEAR POWER REACTORS⁷

Since the fuel in a nuclear power reactor is packaged in large discrete fuel assemblies or bundles, the reactor is classified as an "item facility;" that is, the nuclear fuel is contained in identifiable items, the integrity of which is usually preserved during their presence in the plant. Such items can be followed from the fuel fabrication plant, through the reactor, to reprocessing or long-term storage. Safeguards inspectors can count the fuel bundles, identify them, and verify the fuel composition by nondestructive measurements.

Safeguarding by item accountability requires that the safeguarding authorities, which are both the national authority (including EURATOM) and the international authority (IAEA), be able to verify the identity of the items. This is generally done by item counting and identification of serial numbers. Seals and surveillance cameras (both film and video) are used to complement item accountability to reduce the effort required during physical inventory verification.⁶

A. Safeguards Systems

The main emphasis for safeguarding nuclear reactors is on accounting for each individual fuel assembly, and in this way verifying the quantity of nuclear material present. The quantity of material in these units is not measured by the reactor operator, but instead is based on data supplied by the fuel fabricator and upon theoretical calculations of production and loss resulting from fuel burnup. Physical inventory is established by counting assemblies and identifying serial numbers. Seals and surveillance cameras are used to maintain continuity of safeguards knowledge during time intervals between inspections.⁷

The major components of the safeguards system can be summarized as follows:

- auditing of facility records and comparison with reports submitted to the IAEA,
- periodic closing of the materials balance by the operator (usually annually) by taking a physical inventory,
- independent verification of nuclear material by the IAEA, usually by item counting and serial number identification, and
- application of containment and surveillance measures to maintain knowledge of nuclear material.

B. MBA Structure and Key Measurement Points

Reactor facilities are usually considered as one materials balance area (MBA) subdivided into several key measurement points (KMPs) to determine material flow and inventory. Typical material flow and material inventory KMPs are shown in Table III.

TABLE III

TYPICAL KEY MEASUREMENT POINTS AT A REACTOR FACILITY

Material Flow KMPs:

KMP1	Receipt of fresh fuel
KMP2	Nuclear loss and production
KMP3	Shipment of irradiation fuel

Material Inventory KMPs:

KMPA	Fresh-fuel storage
KMPB	Fuel in reactor vessel
KMPC	Spent-fuel storage
KMPD	Other locations

Figure 6 schematically illustrates the typical KMP structure at a reactor facility.⁸ It depicts the current IAEA policy of having loss and production reported at the time of discharge from the reactor core.

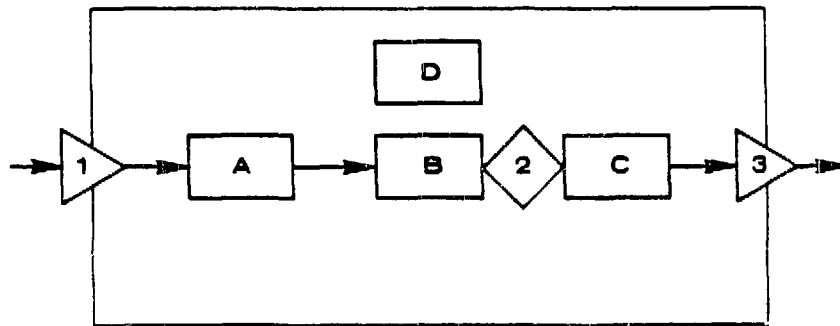


Fig. 6. Typical key measurement points in a power reactor. Inventory key measurement points are: A--fresh fuel, B--in core, C--spent fuel, D--other. Flow key measurement points are: 1--receipt, 2--production, 3--shipment.

C. Inspection Activities

The IAEA generally performs one physical inventory verification (PIV) per year to verify the entire fuel inventory and three to five additional interim inspections to audit records, review surveillance pictures, check seals, and service cameras. These inspections normally involve a total of 10-15 man-days/year.

The objective of the annual physical inventory verification is to establish that the station's declared inventory is correct. For LWRs, inventory verification generally occurs at the end of the annual refueling, but before the top of the reactor vessel is replaced so that the inventory of items inside the reactor vessel can also be verified. For HWRs, the PIV coincides with the state's annual material balances report and physical inventory list.

1. Auditing of Records and Reports. The facility accounting records are examined to ensure that:

- (a) adequate records are being kept,
- (b) they are arithmetically correct and internally consistent, and
- (c) they agree with reports submitted to the IAEA from the state safeguards authority.

Based on the records examination, the inspector is able to establish the "book" inventory of nuclear material at the facility, that is, the amount of material to be accounted for. In addition to accounting records, operating records may be examined to confirm that records of core changes and fuel movements are consistent with accounting records.

2. Inspection Reports. Since the reactor is classified as an item facility, quantitative measurements of nuclear materials are generally not performed. Inspectors' reports, therefore, are based on verification of the major components of the safeguards system described on this page, that is, audit of records, physical inventory, and containment and surveillance (C/S) measures.

VI. SAFEGUARDING LIGHT-WATER REACTORS¹³

A. Fuel Inventory

The items (fuel assemblies) being safeguarded are located at three places in the LWR power station: fresh-fuel storage, reactor vessel, and spent-fuel storage. Upon arrival at the reactor site, fuel assemblies are stored in a fresh-fuel storage area (usually dry, vertical racks) where they are accessible for inspection. The number of fresh-fuel assemblies in storage may range from 5-10 shortly after refueling to approximately 120-200 shortly before a scheduled refueling shutdown. The total annual flow of fresh fuel at a large LWR is ~35 000 kg of low-enriched uranium or 800 kg of ^{235}U (Refs. 1-4).

The reactor core contains 50 000-150 000 kg of uranium, depending upon reactor size. Refueling occurs annually with about one-third (PWR) or one-fourth (BWR) of the spent fuel replaced with fresh fuel. The irradiated elements remaining in the core may be shuffled to new locations to optimize power and fuel utilization. Fuel assemblies inside the core are accessible only during the refueling shutdown.

Irradiated fuel elements removed from the core are stored underwater in a storage pond. Most reactors are designed with a spent-fuel storage capacity of two to four complete cores, but safety procedures require that sufficient capacity always be available to permit a total core unloading in case of an emergency.

From a safeguards viewpoint, the irradiated fuel is strategically important because it contains roughly 1 and 3 kg of plutonium in BWR and PWR assemblies, respectively (see Table I). To ship spent fuel, a massive shielded shipping cask is lowered onto a special pad in the spent-fuel pond and loaded underwater with up to seven irradiated assemblies. Although it was initially envisioned that spent fuel be stored at reactor sites only temporarily (~1 year), the lack of reprocessing has resulted in large accumulations of irradiated fuel in reactor storage ponds.

B. Detection Targets

Based on material type and form, the technical objectives of IAEA safeguards at LWRs should be the detection of diversion of (1) 75 kg of ^{235}U contained in low-enriched fuel within 1 year or (2) 8 kg of plutonium within 1-3 months (see Appendix).¹⁰ At 3% ^{235}U , which is a typical midvalue for LWRs, 75 kg contained ^{235}U is equal to 2500 kg of uranium, which corresponds to ~6 PWR or ~13 BWR assemblies. Similarly, 8 kg plutonium is contained in ~3 PWR and ~8 BWR assemblies.

However, because of the widely accepted safeguards approach at LWRs, namely item accounting for discrete fuel assemblies, the IAEA has adopted the goal of (a) detecting the absence of one or more spent-fuel assemblies within 2-3 months and (b) the absence of one or more fresh-fuel assemblies within 1 year.¹¹

C. Diversion Possibilities

Table IV shows individual diversion possibilities, associated concealment methods, and corresponding safeguards measures.⁸ The basic diversion threats at power reactors fall into two categories: (1) removal of discrete assemblies (either fresh or irradiated, with or without the substitution of dummies) and (2) irradiation of undeclared fertile material.

D. Inspection Activities

Typical activities that occur during inspections at an LWR are shown in Table V, which is an abbreviated version of a table in Ref. 12.¹³

TABLE IV

SUMMARY OF DIVERSION POSSIBILITIES FOR LIGHT-WATER REACTORS^{8,13}

<u>Diversion Possibilities</u>	<u>Concealment Methods</u>	<u>Safeguards Measures</u>
Removal of fuel elements from the fresh-fuel storage	Substitution with dummies	Item counting and identification Application of seals NDA measurements
Removal of fuel elements from the core	Substitution with dummies	Item counting and identification Seals Optical surveillance
Irradiation of undeclared fuel elements in the core	Undeclared shutdowns	Seals Optical surveillance
Removal of fuel elements from the spent-fuel pond	Substitution with dummies	Item counting and identification Optical surveillance NDA measurements
Removal of fuel elements from consignment when or after they leave the facility	Substitution with dummies or consignment. Understating of number of elements shipped and substitution with dummies in spent-fuel pond	Sealing of shipping container before shipment and verification of content at recipient facility, if possible

E. Verification of Nuclear Material¹³

Typical safeguards verification activities that take place at an LWR are shown in Fig. 7. The inventory of fresh-fuel assemblies is verified by identification of the serial numbers that are stamped on the top of the fuel assemblies. The fresh-fuel assemblies are either verified while they are in their storage containers (where they are dry) or in the storage pool just prior to being transferred to the reactor building for insertion into the reactor. Inspection of serial numbers of assemblies in the storage pool where they are located under several meters of water requires the use of an optical magnifier, such as binoculars.

During refueling when the reactor vessel is open, the in-core inventory is verified by counting and identification of serial numbers. The in-core fuel assemblies are located under about 10 m of water, and binoculars are required for the identification of the serial numbers of these fuel assemblies. After the refueling is complete and the reactor vessel is closed, seals are applied to the shielding blocks above the reactor vessel.^{14,15} Since these seals must be broken prior to the removal of the top of the reactor vessel, they provide

TABLE V

SAFEGUARDING ACTIVITIES AT LIGHT-WATER REACTORS¹³

Event	Task
After receipt of fresh fuel (one visit)	Removal of seals at assemblies and identification, records audit, check of seal at vessel, routine identification of irradiated assemblies, maintenance of camera.
After shutdown but before refueling	Removal of seal at vessel; identification and counting of fuel at reactor vessel, fresh-fuel storage and spent-fuel storage; records audit; maintenance of camera.
After refueling but before start-up	Identification and counting of fuel vessel and storages, fixing of seal to vessel, maintenance of camera, and records audit.
Intermediate inspections (three visits)	Identification and counting of fuel at storages, check of seal at vessel, re-records audit, maintenance of camera.
After completion of shipment of irradiated fuel	Identification and counting of fuel at storages, check of seal at vessel, records audit, isotopic data acquisition, maintenance of camera.

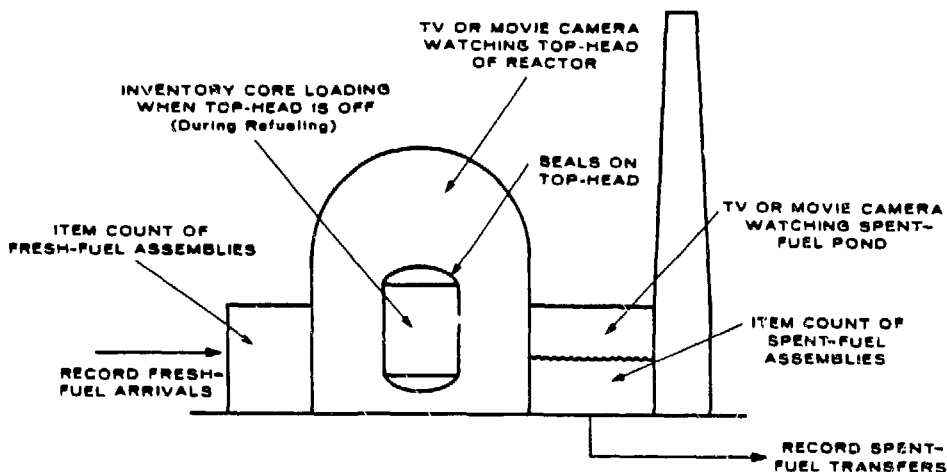


Fig. 7. Safeguards verification activities at an LWR power station.¹³

verification that the in-core inventory was not changed during the absence of the inspector. Also, a surveillance camera is installed inside the reactor vessel as a backup to the seals. Because removal of both the shielding blocks above the reactor vessel and the head of the reactor vessel takes considerable time (days), this surveillance camera needs to take frames only infrequently; as few as 400 frames in a period of 6 months will ensure that an unreported opening of the reactor vessel will be detected.¹⁴

As spent-fuel assemblies are removed from the reactor vessel and are transferred to the storage pond, a map giving the grid location of each assembly is made. In the storage pond, as in the reactor vessel, the assemblies are located under about 10 m of water. During a physical inventory the inspector verifies with binoculars that the spent-fuel assemblies are in their proper locations. Also, surveillance cameras (film or video) are installed inside the spent-fuel bay to record the movement of the crane, fuel assemblies, spent-fuel cask, and the entrance and exit doors. The surveillance pictures are reviewed by the inspector to verify that the station's record of activities since the past inspections is correct. The surveillance cameras ensure the authenticity of the assemblies in the spent-fuel pond and reduce the effort required to complete the inspection. If the surveillance equipment fails, the integrity of the pond must be re-established by visual verification of all the assemblies.

Inspectors also collect data from the operators of the station that are related to calculated burnup, and nuclear consumption and production, which will be useful for safeguards of reprocessing plants.¹⁶

VII. SAFEGUARDING HEAVY-WATER REACTORS

The CANDU-600 reactor, the most common design today, will serve as the model system in this description. The CANDU-600 falls into that class of nuclear facility characterized by a continual flow of nuclear material. But since the fuel is packaged in discrete fuel bundles, this reactor is still classified as an "item facility."

The CANDU safeguards system also follows the principle of establishing item inventories and flows within the MBA, whereby measurements are made at KMPs for the determination of flow or inventory. These KMPs are the new fuel storage room, the reactor core, and the irradiated-fuel storage areas. Examples of item flows are deliveries of new fuel to the storage room, transfer of new fuel from storage into the reactor core, and discharges of irradiated fuel from the core into the spent-fuel storage bays.

Safeguarding an HWR is characterized by keeping track of a very large number of items that are primarily inaccessible. A PIV of bundles inside the reactor is not possible because the fuel bundles are not visible inside the pressure tubes. Verification of the bundles in the spent-fuel pond is difficult because most of the bundles, stacked vertically on trays, cannot be seen. Only the fresh-fuel bundles are readily accessible.

A. Fuel Inventory

In the CANDU-600 each fuel bundle contains 20-21 kg of natural uranium dioxide, giving a total core loading of roughly 98 000 kg in 4600 fuel bundles or ~700 kg of ²³⁵U (Ref. 1). Fresh-fuel assemblies are stored in dry boxes (32 bundles per box) where they can be available for inspection. The number of fresh-fuel bundles in storage may vary from a few hundred to a few thousand, but generally at least a 9-month supply is on hand. Under normal operating conditions, a CANDU-600 reactor charges 16 fresh-fuel bundles and discharges 16

irradiated-fuel bundles each day, which contributes to an annual inventory of 4000-5000 spent-fuel bundles.⁵ The basic features of the fuel-handling facilities and sequences are shown in Fig. 8.

Fresh fuel is transferred from its storage area through an equipment lock into the reactor containment building. It is then loaded into the "charge" portion of the fueling machine. The charge-machine and the accept-machine attach to opposite ends of the same pressure tube without disrupting the flow of heavy-water coolant or reactor energy production. As the charge-machine inserts a fresh-fuel bundle into the channel, a spent bundle is pushed through to the accept-machine. The accept-machine then moves to the spent-fuel discharge ports and the bundles are pushed through the containment onto the ladders of the discharge elevators, where the spent fuel travels underwater through the transfer canal to the storage pond. There the irradiated bundles are loaded onto trays, each holding 24 or more bundles, which are then stacked in modular assemblies consisting of two stacks 18-20 trays high piled on a common base.¹⁷ After 10 years of operation the spent-fuel storage pond may contain 40 000-50 000 bundles, with a total inventory of 3000-4000 kg of plutonium.

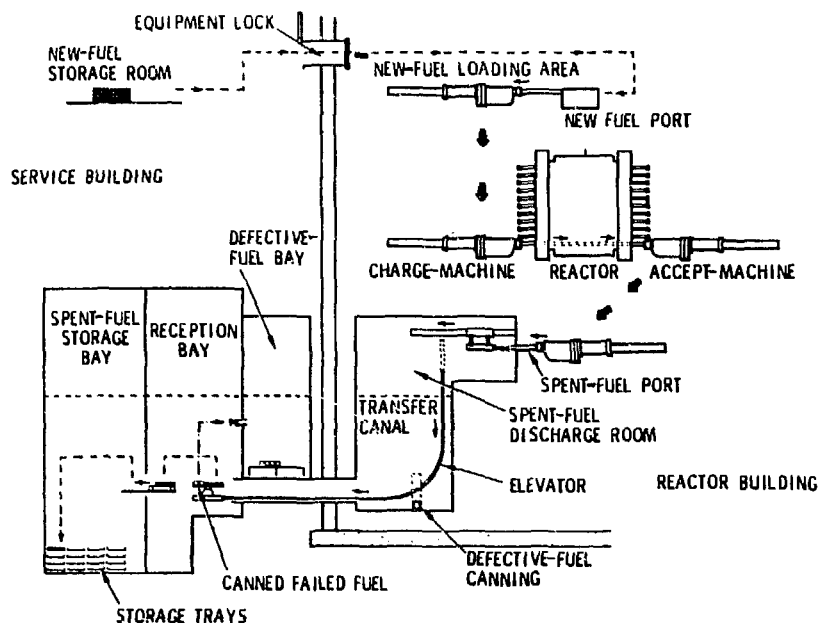


Fig. 8. CANDU fuel-handling sequence.^{1,5}

B. Detection Targets

The HWR safeguards system is designed to detect unreported removal of (1) fresh fuel from the facility, (2) irradiated fuel from the core, and (3) spent fuel from the irradiated-fuel storage bays. IAEA report STR-90 gives the following estimates of minimum removal times for one significant quantity of nuclear material from a CANDU reactor:¹⁷

Fresh fuel	1/2 day
Reactor vessel	2-30 days
Spent-fuel storage	<1 day.

The design of a CANDU safeguards system is therefore based on the following detection considerations:

1. The natural-uranium fuel bundles receive minor safeguards emphasis before they are loaded into the reactor. Natural uranium has a high significant quantity (10.5 tonnes), long conversion time (10-18 months), and is considered "indirect-use" nuclear material. Indirect-use material is defined as nuclear material that must undergo enrichment or transmutation in a nuclear reactor prior to its use in a nuclear explosive device.⁹
2. A direct inventory verification of the core is avoided.¹⁸ Due to the inaccessibility of the reactor core and the on-line refueling, direct verification would be difficult and highly intrusive to the operator of the facility. Emphasis is therefore placed on safeguarding irradiated fuel.
3. Irradiated bundles are discharged at burnups in the range of 165-250 MWh/kg uranium, with an average of 196 MWh/kg uranium (~8000 Mwd/tonne uranium). Since the conversion ratio of HWRs is ~0.8, somewhat higher than that of LWRs, the bundles contain typically 40-90 g of plutonium with an average of ~70 g/bundle.^{17,19} Thus, approximately 115 spent-fuel bundles contain a significant quantity of plutonium.
4. Although the absence of one or more spent-fuel bundles throws a cloud of uncertainty over the entire accounting and inventory procedures, only the absence of a relatively large number of bundles (perhaps 10-20) within 2 or 3 months would have safeguards significance.

An instrumented CANDU detection system typically consists of the following items:¹⁸

1. Core input monitors--to count the number of unirradiated fuel bundles loaded into the core.
2. Closed-circuit television or film cameras--in areas where the unirradiated or irradiated fuel moves or is stored, to record anomalies that might indicate diversion.
3. Irradiated-fuel bundle counters--to count the number of irradiated-fuel bundles discarded from the reactor and to verify that the objects counted are highly radioactive.
4. Yes/no monitors--to indicate when irradiated fuel has been abnormally moved through penetrations or routes that do not have KMPs.
5. Irradiated fuel verifiers--to verify authenticity of irradiated fuel in the storage bay.
6. Tamper-indicating containers and seals--to hold irradiated fuel in large batches. A check of the integrity of the seal and container shows that no fuel has been removed.

C. Diversion Possibilities

Considering that the fresh-fuel bundles are indirect-use nuclear material and that the core is never really "open" to diversion, the undeclared removal of irradiated fuel from the spent-fuel storage bay is considered to be the most attractive diversion possibility for HWRs. Casks and transfer equipment are readily available. Furthermore, since the fuel in the bay has decayed significantly, it would be feasible to remove one significant quantity in one or two operations. The diversion possibilities are summarized in Table VI.

D. Inspection Activities

As with LWRs, two types of inspections are required each year: three to five routine (interim) inspections and an annual (PIV) inspection. During routine (interim) inspections, newly transferred fuel bundles must be sealed and safeguards/equipment must be serviced. The minimum activities that need to be performed at the facility are the following:

- (1) Audit operating records and reports.
- (2) Service each film camera system and complete appropriate forms. Films will be developed and reviewed off-site.
- (3) Service all fuel bundle counters and fuel bundle counters/attribute verifiers.
- (4) Examine all yes/no monitors and convenience seals.
- (5) Verify that spent-fuel stacking frames to be sealed are full and verify the sealing of such stacking frames.
- (6) Verify fresh-fuel boxes received (box serial numbers) if possible.
- (7) Proceed with follow-up activities when required.

Review of appropriate film and bundle counter printouts for agreement with the operator's records is usually performed off-site. Verification of off-site shipments, which requires advance notification, would be in addition to the above activities. To perform these activities the inspector requires (inter alia) access to the spent-fuel bays and fresh-fuel storage and loading rooms.

The prime purpose of the annual inspection is to verify the ending physical inventory entries provided in the state's annual Material Balance Report (MBR) and Physical Inventory List. Verification of physical inventory may be accomplished by direct measurement (bundle count and random attribute test), or by accepted substitute techniques such as checking sealing systems. In addition, newly transferred fuel bundles are required to be sealed and all safeguards equipment is required to be serviced. The minimum activities that need to be performed during the annual inspection are as follows:

- (1), (2), (3), (4), and (5) Same as for routine (interim) inspections.
- (6) Inspect the seals and covers of randomly selected sealed stacks in all spent-fuel bays.
- (7) Perform the fresh-fuel inventory verification activities.

TABLE VI

SUMMARY OF DIVERSION POSSIBILITIES FOR HEAVY-WATER REACTORS

<u>Diversion Possibilities</u>	<u>Concealment Methods</u>	<u>Safeguards Measures</u>
Removal of fuel bundles from fresh-fuel storage	Substitution with dummies	Item counting and identification Container (box) counting NDA techniques (box verifier)
Removal of fuel bundles from the core	Substitution with dummies	Core input counters Irradiated-fuel bundle counters Closed-circuit TV or film cameras Yes/no monitors
Irradiation of undeclared fuel bundles in the core	Substitution with dummies	Core input counters Irradiated-fuel bundle counters Closed-circuit TV or film cameras Yes/no monitors
Removal of fuel bundles from spent-fuel pond	Substitution with dummies	Closed-circuit TV or film cameras Yes/no monitors Irradiated-fuel verifiers Containers and seals
Removal of fuel elements from consignment when or after they leave the facility	Substitution with dummies or consignment. Understating of number of bundles shipped and substitution with dummies in spent-fuel pond	Sealing of shipping container before shipment and verification of content at recipient facility, if possible

(8) Perform off-site (at IAEA Headquarters or a Field Office) inspection activities as follows:

- (a) Review the appropriate film and compare with operations records including crane operations list and large object removal list.
- (b) Check for discrepancies among the approximate bundle counter printouts and for discrepancies in the operator's records.

- (c) List the verified physical inventories for all categories of nuclear material or verified book inventory.
- (d) Compare the state's accounting report (ICR, MBR, PIL) to the facility records to ascertain that the state's report is consistent, accurate, and complete.
- (e) Calculate material unaccounted for (MUF) based on difference between ending book inventory and ending physical inventory.
- (f) Proceed with follow-up activities when required.
- (g) Update IAEA records and prepare inspection report on the basis of the inspection activities.

E. Verification of Nuclear Material

Typical safeguards verification activities that take place at an HWR are shown in Fig. 9. Fresh-fuel inventory verification consists of a count of containers (boxes) plus a count of loose (unpacked) bundles in combination with a random check of containers to assure that these contain the stated number of natural-uranium fuel bundles. Such testing is accomplished by NDA techniques (box verifier), which do not require opening of boxes.

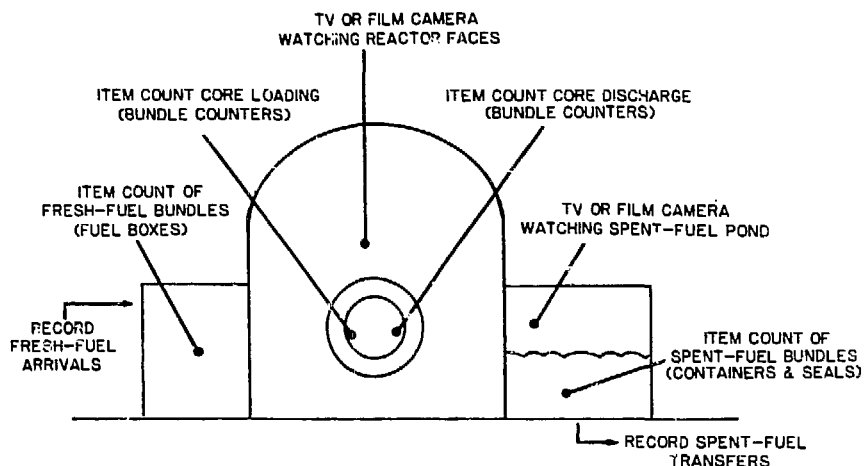


Fig. 9. Safeguards verification activities at an HWR power station.

The inspector can verify the inaccessible inventory in the core and fueling machines by the use of flow difference and C/S techniques. Flows can be independently measured by using core input counters and core discharge counters. The structural containment of the reactor building and additional C/S devices such as surveillance cameras, yes/no monitors, and seals provide a containment boundary and assure that undeclared removal of irradiated fuel by the more credible diversion routes will not occur without triggering anomalies.

Verification of the inventory in the spent-fuel bay makes use of the simplifications afforded by sealing the irradiated fuel and also uses flow difference and C/S techniques. Bundles discharged into the spent-fuel bay are

counted by the core discharge counters. These counters also incorporate an attribute test (fuel verifier) to provide assurance that all bundles counted were irradiated bundles. Any bundles shipped off-site (out of the bay) can be verified by an inspector being present. The structural containment of the walls of the storage bay and the use of C/S devices such as surveillance cameras, sealing system for spent fuel, as well as the discharge counters' reverse flow counting capability, are used to provide the containment boundary needed for the flow difference technique to be a valid means of verifying inventory. These C/S devices provide assurance that transfers other than through-flow KMPs did not occur. At each inspection the inspector would verify that stacks of irradiated fuel to be sealed are full, assuming the bundle counter and surveillance have indicated no anomalies. Verification of sealed fuel in future inspections will then consist of verifying the integrity of the sealing system.

APPENDIX⁹

SIGNIFICANT QUANTITIES

	<u>Material</u>	<u>Significant Quantity</u>	<u>Safeguards Apply To</u>
Direct-use nuclear material	Pu* U-233 U(U-235 $\geq$ 20%)	8 kg 8 kg 25 kg	Total element Total isotope U-235 contained
- Plus rules for mixtures where appropriate -			
Indirect- use nuclear material	U(U-235 < 20%)** Th	75 kg 20 t	U-235 contained Total element
- Plus rules for mixtures where appropriate -			

*For Pu containing less than 80% Pu-238.

**Including natural and depleted uranium.

ESTIMATED MATERIAL CONVERSION TO FINISHED Pu AND U METAL COMPONENTS

<u>Beginning Material Form</u>	<u>Conversion time</u>
Pu, HEU or U-233 metal	Order of days (7-10)
PuO ₂ , Pu(NO ₃) ₄ , or other pure Pu compounds; HEU or U-233 oxide or other pure compounds; MOX or other nonirradiated pure mixtures containing Pu, U[(U-233 + U-235) $\geq$ 20%]; Pu, HEU and/or U-233 in scrap or other miscellaneous impure compounds	Order of weeks (1-3)*
Pu, HEU or U-233 in irradiated fuel**	Order of months (1-3)
U containing <20% U-235 and U-233; Th	Order of one year

*This range is not determined by any single factor, but the pure Pu and U compounds will tend to be at the lower end of the range and the mixtures and scrap at the higher end.

**Criteria for establishing the irradiation to which this classification refers are under review.

POWER REACTOR FUEL
DETECTION TARGETS

SIGNIFICANT QUANTITIES & TIMES

<u>Material</u>	<u>Mass</u>	<u>Time</u>	<u>No. PWR Assemblies</u>	<u>No. BWR Assemblies</u>	<u>No. HWR Bundles</u>
U-235 in LEU	75 kg	1 Year	~6	~13	>500
Pu	8 kg	1-3 Months	~3	~8	>100

INSPECTION TARGETS

1	1	~10-20*

*Note the discussion of this number in the text, item B-4, page 13-17. Although only a relatively large number of HWR bundles would represent a significant diversion, the IAEA inspection goal is still the detection of the absence of one or more fuel bundles.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 13: GAMMA-RAY ASSAY OF IRRADIATED FUEL

The fission-product and activation-product sources of gamma rays in irradiated fuel materials will be discussed as well as the nondestructive techniques available to measure the emitted gamma rays. Each measurement technique will be related to the specific level of verification it provides for irradiated fuel assemblies.

After the session, participants will be able to:

- Understand the safeguards motivations to measure irradiated fuel materials.
- List the major gamma-ray signatures for irradiated fuel assay.
- Describe each of the gamma-ray measurement techniques including the instrumentation and appropriate data analysis.
- Discuss the advantages and limitations of each technique with respect to the desired level of verification.
- Discuss the present applications and experience with irradiated fuel measurement instruments.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 13: PASSIVE GAMMA-RAY METHODS FOR ASSAY/VERIFICATION OF SPENT FUEL

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I. INTRODUCTION

The national stockpiling of spent-fuel assemblies, which is occurring at an increasing rate because of delays in nuclear fuel reprocessing, is a potential source of very large and growing quantities of plutonium that can have undesirable international consequences.¹ The primary objective of international safeguards is the

"timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices, or for purposes unknown, and deterrence of such diversion by the risk of early detection."²

The International Atomic Energy Agency (IAEA) is committed to the use of material accountancy as the primary technique for detection of a diversion with containment and surveillance as an important complementary measure.

The plutonium content of spent-fuel assemblies is of primary importance, with the residual ²³⁵U content being less significant. The threshold amount (approximate quantity of special nuclear fissionable material required for a single explosive device) for plutonium has been generally accepted as 8 kg.³ This threshold amount is equivalent to the plutonium present in two pressurized-water reactor (PWR) or five boiling-water reactor (BWR) fuel assemblies discharged from typical power reactors. Another factor that must be included in evaluating the requirements for safeguarding spent-fuel assemblies is the time required for conversion of the plutonium present in the fuel assembly to a form compatible with the fabrication of a nuclear explosive. Time estimates for this conversion range from 1 to 3 months.⁴ At the present time, the IAEA has established a detection time of a few months for irradiated materials with a desired detection probability of 95%.⁵

To verify the inventory of spent-fuel assemblies in a storage facility, a straightforward sampling plan could be used. An approximate formula⁶ for the detection probability provided by the attribute measurement of fuel assemblies is

$$DP = 1 - (1 - f)^D, \quad (1)$$

where DP is the desired detection probability, f is the fraction of the total number of fuel assemblies that must be sampled, and D is the number of missing or altered fuel assemblies to be detected. To attain a 95% detection probability where D is 2 for PWR fuel assemblies and 5 for BWR fuel assemblies, the inspector would be required to select randomly and to measure approximately 80% of the PWR and 45% of the BWR fuel assemblies.

A variety of nondestructive measurement techniques are available for verifying spent-fuel assemblies, depending upon the level of verification required. Table I lists the various levels of verification and the associated nondestructive measurements divided between gamma-ray and neutron techniques.⁷ The primary purpose of this paper is to discuss the nondestructive gamma-ray techniques for the assay and verification of spent fuel. However, this table shows the relationship between the various gamma-ray and neutron techniques and the specific levels of verification. An inspector would not necessarily limit himself to only gamma-ray or neutron techniques, but would rather use a combination of the two techniques.

For reactors, the maximum time allocated for routine inspection is set at one-sixth of a man-year for each facility per calendar year.² An inspector cannot devote the entire inspection time to the verification of the spent-fuel assemblies. He must review the auditing records and compare them with the IAEA reports, examine operating records and compare them to the accounting records, verify the fresh fuel assemblies prior to core loading, and verify the number of fuel assemblies in the reactor core following refueling and before the reactor vessel is closed.⁴ Therefore an inspector selects a level of verification consistent with the constraint of available time. Each of the specific levels of verification will be discussed with respect to the particular gamma-ray measurement techniques.

Before we begin this detailed discussion, let us examine a typical gamma-ray spectrum of a spent-fuel assembly. Figure 1 shows the gamma-ray spectrum obtained from a PWR fuel assembly with an exposure of 30 GWd/tU following a 1-yr cooling time. The principal gamma rays are labeled in Fig. 1 and listed in Table II. As can be readily seen, there are only about a dozen isotopes that contribute to the gamma-ray spectrum. It is also important to note the absence of any ^{235}U and ^{239}Pu gamma-ray lines in the measured spectrum. For irradiated reactor fuel with 30 GWd/tU exposure, the prominent 414-keV line of ^{239}Pu has an intensity of only 1392 gamma rays/cm³/s, while the fission product ^{137}Cs emits 4.38×10^{10} gamma rays/cm³/s. For each ^{239}Pu 414-keV gamma ray emitted, more than 30 000 000 gamma rays are emitted from the single fission product ^{137}Cs . Any gamma ray from the fissile isotopes is completely overwhelmed by the fission product gamma rays and their associated Compton backgrounds. Therefore, the concentration of fissile isotopes in spent fuel cannot be measured directly but rather must be measured indirectly by relating the buildup of specific fission products to the depletion of the ^{235}U and the buildup of plutonium.

II. LEVELS OF VERIFICATION

A. Indication of Irradiation Exposure

The presence of gamma rays being emitted from a fuel assembly is a low-level verification. Two gamma-ray measurement techniques are available: (1) Cerenkov glow measurement^{8,9} and (2) ion chamber or scintillator¹⁰ measurements. The Cerenkov technique has the distinct advantage that the verification can be performed without having to place any instrumentation into the storage pond water. But it also has the disadvantage that most of the artificial lighting has to be eliminated. This requires the inspector to move around in a darkened environment and increases the safety hazard.

The gross gamma-ray signature of fuel assemblies can also be measured using ion chambers or scintillators or thermoluminescent detectors (TLD). The

TABLE I
LEVELS OF VERIFICATION FOR SPENT-FUEL ASSEMBLIES

<u>Specific Level of Verification</u>	<u>Measurement Technique</u>		<u>Instrumentation</u>
	<u>Gamma Ray</u>	<u>Neutron</u>	
Physical characteristics			Visual inspection
Indication of irradiation exposure	Cerenkov		Cerenkov viewing device
	Presence of gamma radiation		Ion chambers Thermoluminescence detectors Scintillators
Physical integrity of fuel assembly		Presence of neutron radiation	Fission chambers ^{10}B detectors
	Cerenkov		Cerenkov viewing device
	Relative intensities of high-energy gamma rays		Germanium detector Be(γ ,n) detector
		Relative values of neutron emission rates	Fission chambers ^{10}B detectors
Presence of fission products and actinides	Qualitative identification of specific gamma-ray lines		Germanium detector Be(γ ,n) detector
		Relative values of neutron emission rates	Fission chambers ^{10}B detectors
Relative concentrations of fission products and actinides	Quantitative measurement of ^{137}Cs , $^{134}\text{Cs}/^{137}\text{Cs}$, and $^{154}\text{Eu}/^{137}\text{Cs}$. Correlation with operator-declared information		Germanium detectors
		Quantitative measurement of neutron emission rate. Correlation with operator-declared information	Fission chambers ^{10}B detectors
Direct measurement of special nuclear material	Indirectly through correlations between NDA measurements and destructive analyses		Germanium detectors
		Quantitative measurement of induced fissions in special nuclear material	Neutron source Fission chambers ^{10}B detectors

TABLE II
LISTING OF MEASURABLE ISOTOPES IN
TYPICAL LWR FUEL ASSEMBLY

<u>Fission Products</u>		<u>Fission Yield (%)</u>		<u>Gamma-Ray Energy in keV (Branching Ratio)</u>
<u>Isotope</u>	<u>Half-Life</u>	<u>²³⁵U</u>	<u>²³⁹Pu</u>	
¹³⁷ Cs	30.17 yr	6.222	6.689	661.6 (0.851)
¹⁵⁴ Eu ^a	8.5 yr	2.69-6 ^b	9.22-5	1274.4 (0.355), 1004.8 (0.174), 996.3 (0.103)
¹²⁵ Sb	2.71 yr	0.0294	0.1110	427.9 (0.30), 600.8 (0.18), 686.2 (0.12), 463.5 (0.11)
¹³⁴ Cs ^a	2.062 yr	1.27-5	9.89-4	604.7 (0.976), 795.8 (0.854), 801.8 (0.087), 1365.1 (0.0304), 1167.9 (0.018)
¹⁰⁶ Ru-Rh	366.4 days	0.4018	4.280	622.2 (0.098), 1050.5 (0.016)
¹⁴⁴ Ce-P.	284.5 days	5.475	3.736	696.5 (0.0134), 2185.6 (0.0066), 1489.2 (0.0026)
⁹⁵ Zr	63.98 days	6.496	4.892	756.7 (0.546), 724.2 (0.431)
¹⁰³ Ru	39.35 days	3.043	6.948	497.1 (0.864), 610.3 (0.054)
⁹⁵ Nb	34.97 days	6.496	4.893	765.8 (0.9982)
<u>Activation Products</u>				
⁶⁰ Co	5.27 yr			1173.2 (1.00), 1332.5 (1.00)
⁵⁸ Co	70.3 days			811.1 (0.99), 511.0 (β^+ annihilation)
⁵⁴ Mn	312.2 days			834.8 (1.00)

^aEuropium-154 and ¹³⁴Cs values are given only for direct production of the isotope from the fission process. Each isotope is produced primarily through neutron absorption. For PWR fuel material irradiated to 25 GWd/tU, the "accumulated fission yields" of ¹⁵⁴Eu and ¹³⁴Cs were calculated as 0.15 and 0.46% for the total fissions, respectively.

^b2.69-6 should be read as 2.69×10^{-6} .

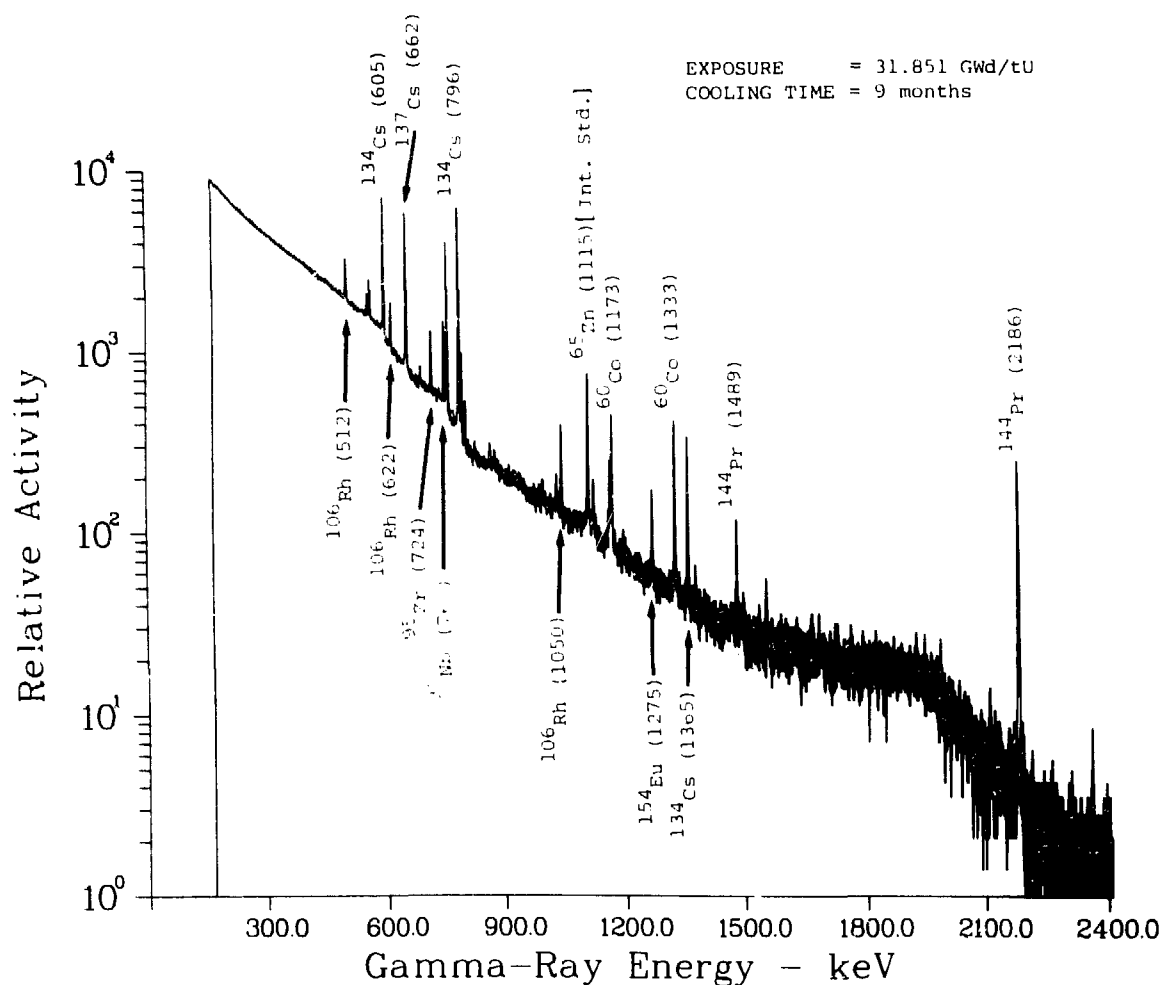


Fig. 1. Measured gamma-ray spectrum of a PWR fuel assembly with an exposure of 31.9 Gwd/tU and a cooling time of 9 months.

signature is primarily from the activation products, ^{60}Co , ^{58}Co , and ^{54}Mn , in the structural material at the top of the fuel assemblies. Ion chambers and scintillators provide the information immediately while the TLDs must be removed from the water and subsequently read using a special instrument.

1. Cerenkov Glow Intensity Measurement.^{8,9} Electromagnetic Cerenkov radiation is emitted whenever a charged particle passes through a medium with a velocity exceeding the phase velocity of light in that medium. In water, the phase velocity of light is about 75% of the value in vacuum. Any electron passing through water and having $\geq 0.26\text{-MeV}$ kinetic energy is thus a source of Cerenkov radiation. Irradiated fuel assemblies are a prolific source of beta particles, gamma rays, and neutrons. All three types of emissions can produce Cerenkov light, directly or indirectly.

Considering the normal fuel rod cladding (0.6-0.8 mm Zircaloy) few beta particles originating in the fuel material will escape directly into the pond water, therefore they are not a significant source of Cerenkov light. It is

possible that electron bremsstrahlung from energetic beta particles interacting in the fuel rods could be a significant source of gamma radiation escaping into the pond water. In the cooling water, energetic fission product gamma rays can undergo pair production or Compton scattering to produce ≥ 0.26 -MeV electrons. Neutrons may undergo $H(n,\gamma)$ reactions in the water and produce Cerenkov light through interactions of the 2.23-MeV capture gamma rays with water.

The most significant production of Cerenkov light is from high-energy fission fragment-decay gamma rays that interact with the cladding or water. The number of Cerenkov photons generated from gamma rays of any energy passing through water can be calculated. These calculations indicate that Cerenkov light production in the visible range of 4000-6000 Å is negligible for gamma rays with $E \leq 0.6$ MeV and rises steeply with greater gamma-ray energy. A gamma ray with $E = 2$ MeV produces more than 500 times the Cerenkov photons that are produced by a gamma ray with $E = 0.6$ MeV.

The absolute Cerenkov light level and its decay with time are related to burnup. Hiding a diversion either by substitution of dummy fuel assemblies or by incorrectly stating burnup would be difficult as long as the Cerenkov light intensity can be measured accurately.

The spatial extent of the Cerenkov glow surrounding an isolated irradiated assembly in water probably will be determined by the gamma radiation from the assembly's outer pins. The one-tenth-value-layer thickness of water for 1.0-MeV gamma rays is 36 cm, which is a reasonable estimate of the Cerenkov "halo" around an isolated point source. Fission product radiation from an irradiated fuel assembly's inner pins, however, must penetrate a much denser composite of fuel cladding, and interstitial water, which greatly reduces crosstalk among assemblies in either regular or high-density storage racks. The problem of crosstalk becomes a significant problem when the gamma-ray field is measured above the fuel assemblies using ion chambers or scintillators, as will be discussed in the next section.

A series of measurements have been performed at spent-fuel storage facilities to demonstrate the applicability of the technique to the rapid verification of spent-fuel assemblies. Figure 2 shows the Cerenkov image of a PWR fuel assembly in a storage rack at a reactor facility. The 20 bright spots correspond to the guide tube positions of a 15 x 15 fuel assembly. As can be seen in Fig. 2, the adjacent fuel assemblies do not exhibit similar sets of bright spots because the observer is not aligned with the axes of the assemblies.

The results of relative Cerenkov intensity measurements of the PWR irradiated assemblies are shown in Fig. 3. The solid curve in the figure is the calculated intensity vs cooling time for an exposure of 30 000 MWd/tU. The Cerenkov photon intensities from these assemblies depend on the cooling times as predicted.

The Cerenkov glow intensity measurement provides the inspector with a rapid nondestructive technique for verifying the presence of a gamma-ray source distributed within the fuel assembly. It has also been used to determine the removal of fuel pins from a fuel assembly. One of the most attractive aspects of the technique is that fuel assembly measurements can be performed without introducing any equipment into the pond water.

2. Gross Gamma-Ray Measurements.^{10,11} Ion chambers or scintillator detectors can also be used to verify the presence of gamma-ray radiation in spent-fuel assemblies in storage racks. Crosstalk between fuel assemblies

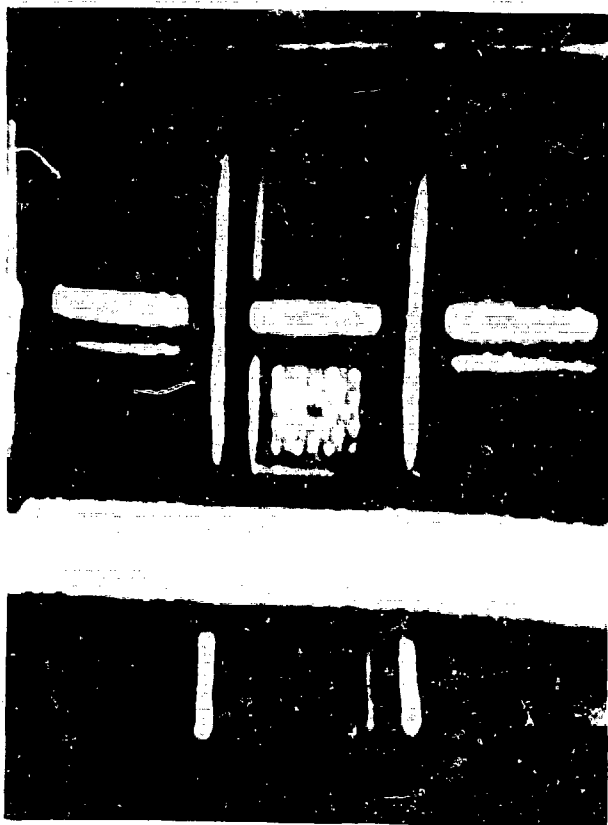
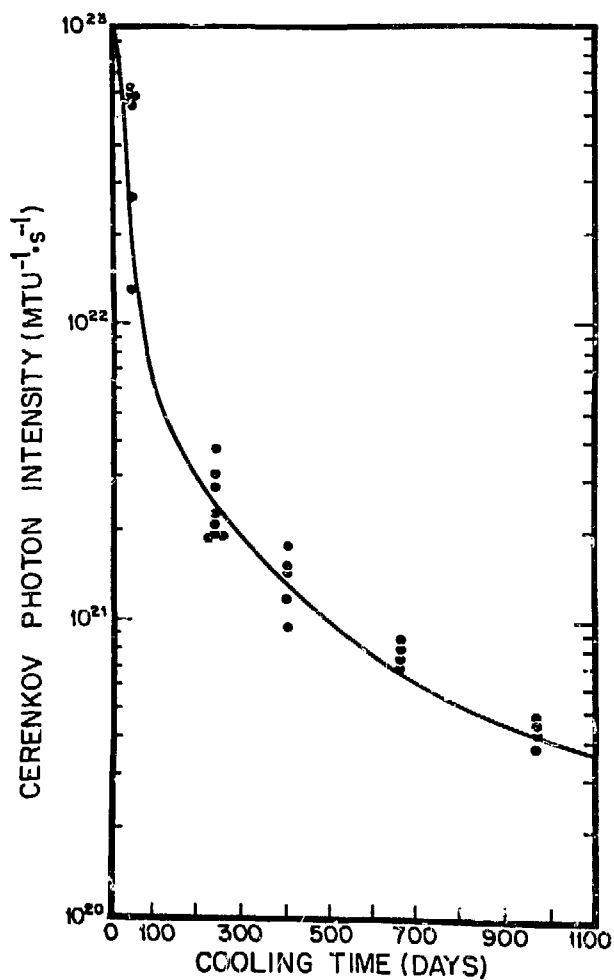


Fig. 3. Cerenkov intensity measurements of PWR fuel assemblies normalized to the calculated value at 270 days cooling time. Solid curve was calculated for a 30.0 GWd/tU exposure.

Fig. 2. Cerenkov image of a PWR fuel assembly showing the guide tube position.



located in adjacent storage positions must be reduced to a minimum to avoid the necessity of using unfolding techniques.¹² This measurement technique requires that the instrument (1) be lightweight and easy to position by hand, (2) give an immediate result, (3) not be damaged by high-radiation levels, and (4) have an accuracy ensuring a meaningful measurement of the fuel inventory.

We have measured the gamma-ray spectrum emitted from the top of two PWR fuel assemblies with an exposure of 40.1 and 26.9 GWD/tU and cooling times of 1575 and 2049 days, respectively. The prominent gamma-ray lines were from ^{60}Co , ^{137}Cs , and ^{134}Cs , with a very small contribution from the ^{144}Pr lines. The long cooling times would have significantly reduced the contribution from ^{144}Pr , which is the daughter of the 285-day fission product ^{144}Ce . Cesium-137 and ^{134}Cs were dissolved in the pond water; therefore it was not possible to determine quantitatively how much of the signals from these two isotopes were coming from the fuel assembly. Cobalt-60 existed as a contaminant in the pond water as well as coming from the structural material of the fuel assemblies. Based upon these measurements it is difficult to confirm how much of the signals measured from the collimated ion chamber and scintillator detector were due to the fission product inventory in the fuel assembly.

These two techniques have been demonstrated to distinguish between fuel assemblies having widely differing exposures. They are rapid measurement techniques requiring only 1-2 min for positioning the detectors and collecting the data. For fuel assemblies stored in canisters, this is the only technique for rapidly verifying the presence of radioactive material.

3. Neutron Measurements. It is not possible to measure the neutron signals of spent-fuel assemblies by placing neutron detectors on the top of the storage rack. The neutron flux is reduced by a factor of 10 for each 10 cm of water.¹³ The fuel material is located approximately 50 cm below the top of the storage racks, therefore the neutron source would be reduced below the detectability of the detectors. A more detailed discussion on neutron measurement techniques will be presented in the following paper.

B. Physical Integrity of a Fuel Assembly

1. Cerenkov Glow Intensity Measurement. As discussed in Sec. II.A, this measurement technique has been used to determine the removal of individual fuel rods by examining the image and detecting anomalies in the pattern. For fuel assemblies with top plates, it is not necessarily possible to see all of the fuel rods; therefore the technique is not fail safe.

2. Gross Gamma-Ray Measurements. Measurement of gross gamma-ray signals above fuel assemblies in a storage rack can only detect very gross changes in the number of fuel rods present. However, by removing the fuel assembly from the storage rack, that is, raising the assembly so measurements can be made from the side, the detectability of fuel pin removal can be significantly improved.

The gamma-ray measurements can be improved by using a $\text{Be}(\gamma, n)$ detector shown in Fig. 4, in which a ^{235}U fission chamber is surrounded by successive annuli of polyethylene and beryllium. Neutrons produced by a photoneutron reaction in the beryllium are thermalized in the polyethylene and then counted using the ^{235}U fission chamber. Because the threshold for photoneutron production in beryllium is 1.66 MeV, the only significant fission product with a gamma-ray energy above this energy is the 2.186-MeV gamma ray of ^{144}Pr . The ^{144}Pr isotope, with a short half-life of 17.3 min, is in secular equilibrium

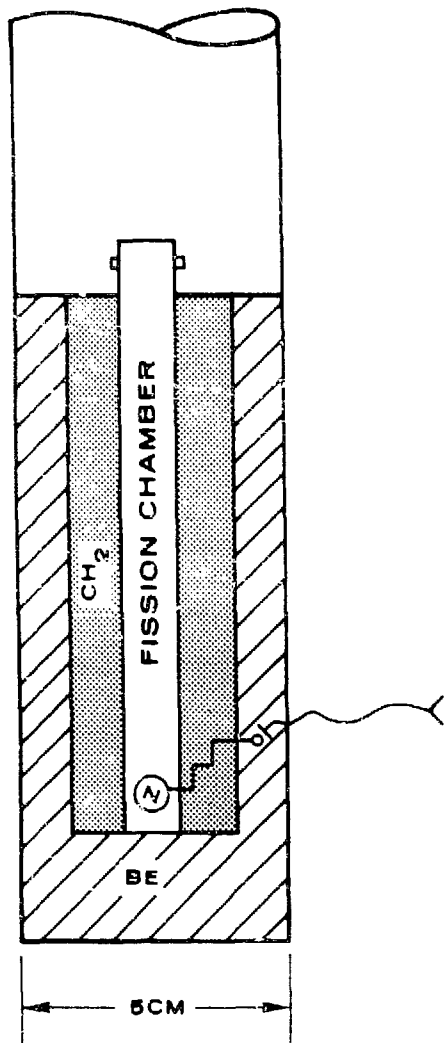


Fig. 4. Beryllium (γ,n) detector used to measure gamma rays with energies above 1.66 MeV.

with its parent ^{144}Ce (284.5 days). Therefore, the $\text{Be}(\gamma,n)$ detector measures the relative activity of ^{144}Ce .

Even the inner rods of a PWR fuel assembly can contribute to the 2.186-MeV source strength. For example, the center rods of a 15 x 15 fuel assay (row 7) can contribute 4% of the total flux at the surface of a fuel assembly. This measurement technique using a $\text{Be}(\gamma,n)$ detector is not fail proof, but it is better than just measuring the gross gamma-ray signal using an ion chamber or scintillator to determine the absence of fuel rods.

C. Presence of Fission Products and Actinides

The presence of fission products can only be detected by using instrumentation that allows some type of energy resolution. As discussed in the introduction, gamma-ray signatures of actinides cannot be measured directly because of the fission product background. A $\text{Be}(\gamma,n)$ detector can be used to detect the presence of gamma rays with energies above 1.66 MeV and to infer the presence of the fission product ^{144}Pr . High-resolution gamma spectrometry (HRGS)

using a germanium detector is the preferable technique for determining the presence of fission products. A spectrum similar to the one shown in Fig. 1 can be used to qualitatively identify the source material by comparing the relative intensities of specific fission products. For example, the relative intensities of ^{106}Ru and ^{137}Cs can be used to identify qualitatively if the material were ^{235}U or ^{239}Pu , because of the differences in the fission yields for ^{106}Ru -Rh, 0.4% for ^{235}U and 4.3% for ^{239}Pu . Similarly, information about the time since discharge from the reactor can be inferred by the presence or absence of fission products with short half-lives, for example, ^{95}Nb 34.97 days and ^{95}Zr 63.98 days.

This level of verification ensures that fission products are present and that a dummy fuel rod has not been substituted. Without a confirmation of fission products, an inspector cannot determine if the fuel rods have been replaced by ^{60}Co -activated rods or not. Therefore, this is a higher level of verification than the other previously described techniques.

D. Relative Concentrations of Fission Products and Actinides

For the gamma-ray measurement techniques that are routinely applied to spent-fuel assemblies, this is the most widely investigated and accepted technique.⁷ The exposures of spent-fuel assemblies are generally correlated with ^{137}Cs , $^{134}\text{Cs}/^{137}\text{Cs}$, and $^{154}\text{Eu}/^{137}\text{Cs}$ in the HRGS methods. Figure 5 shows the correlations of these fission products with exposure for four simulated irradiation histories. A detailed irradiation history is included for comparison.

The ^{137}Cs isotope has a long half-life, similar fission yields for ^{235}U and ^{239}Pu , and an easily resolvable gamma-ray spectrum. Changes in the scanning geometry or the distribution of the source can affect the measured results. Using an isotopic ratio like $^{134}\text{Cs}/^{137}\text{Cs}$ or $^{154}\text{Eu}/^{137}\text{Cs}$, changes in the source attenuation can be corrected using an intrinsic efficiency calibration.^{7,14} The initial ^{235}U enrichment can significantly influence the correlation of these two ratios with exposure as is shown in Fig. 6. Therefore, some information about the initial enrichment is necessary for the correct interpretation of the results.

Some experiments have attempted to establish correlations between the number of days since discharge and various isotopic concentrations and ratios.^{15,16} Knowledge of the cooling time is of particular importance for the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio because of the relatively short half-life of ^{134}Cs (2.062 yr). It is also important in the interpretation of the passive neutron measurements of spent-fuel assemblies. The primary sources of neutrons in irradiated fuel materials are from ^{242}Cm ($t_{1/2} = 163$ days) and ^{244}Cm (18.1 yr), and if the cooling time is greater than 3 yr the source is only related to the buildup of ^{244}Cm .

A word of caution must be raised about the interpretation of gamma-ray measurements and how much of the fuel assembly that they are representing. For a PWR fuel assembly, only the outer three or four rows contribute to the ^{134}Cs and ^{137}Cs signals. Table III gives the percentage of the total signal derived from each row of fuel pins. These values were calculated using a Monte Carlo technique to track each photon originating in the specific fuel rod positions.

For verification, HRGS is the most widely accepted technique. However, the inspector must be careful in the interpretation because of the source self-attenuation, as shown in Table III. The measurement technique has been applied to BWR fuel assemblies and compared with destructive measurements with reasonable good success.¹⁷

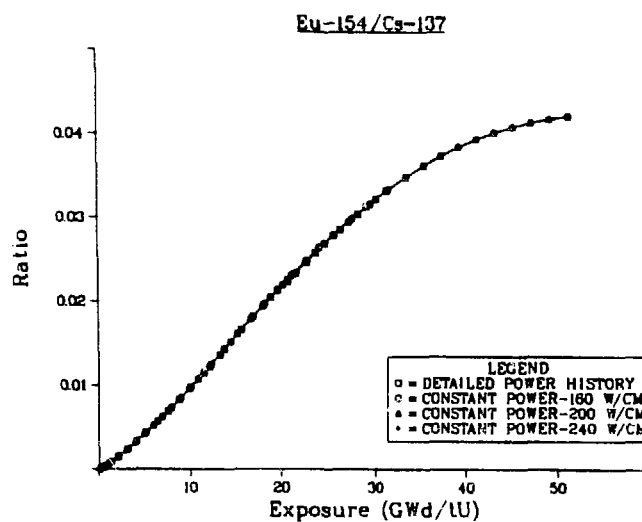
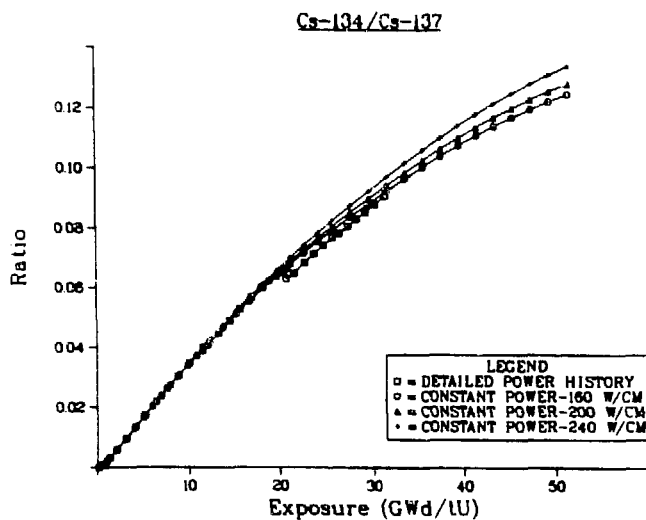
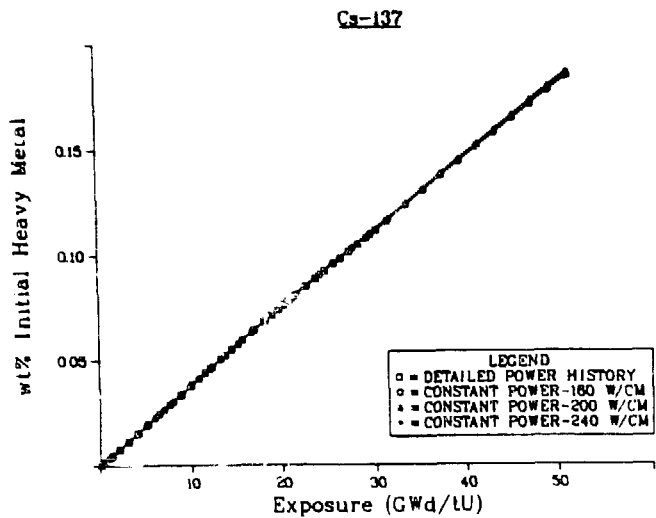


Fig. 5. Correlations between ^{137}Cs , $^{134}\text{Cs}/^{137}\text{Cs}$, and $^{154}\text{Eu}/^{137}\text{Cs}$ and exposure for four different irradiation histories. A detailed power history from a typical PWR fuel assembly as well as three constant power histories are shown.

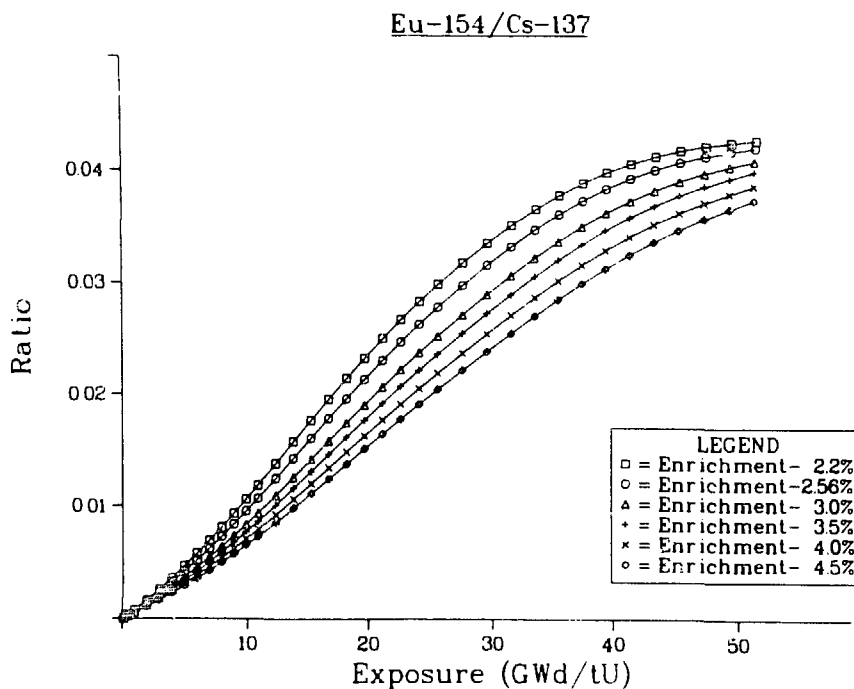
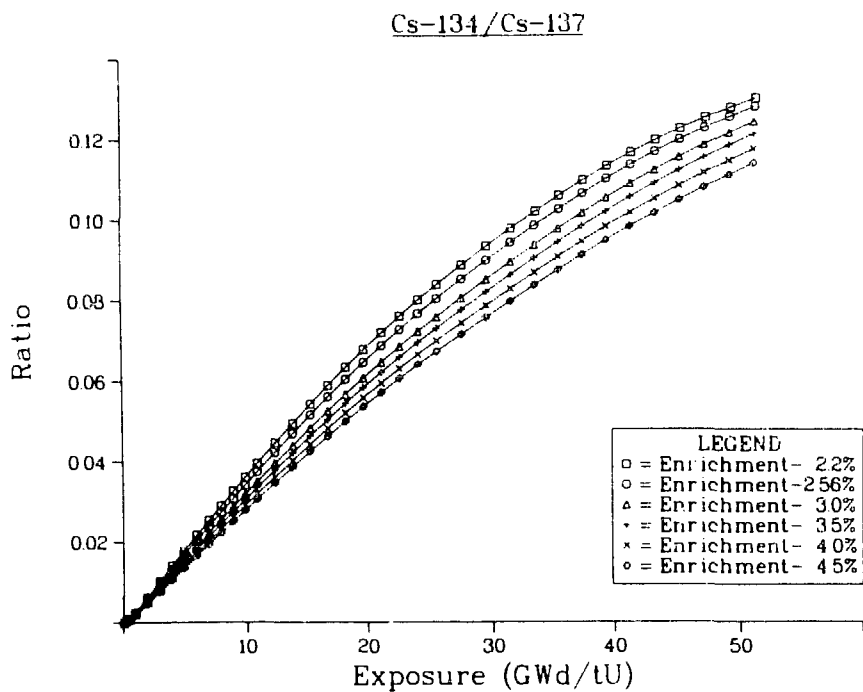


Fig. 6. Effects of changes in the initial ²³⁵U enrichment on the correlations between the ¹³⁴Cs/¹³⁷Cs and ¹⁵⁴Eu/¹³⁷Cs ratios and exposure are presented.

TABLE III

PERCENTAGE OF GAMMA RAYS FROM EACH ROW BASED ON THE TOTAL VALUE

Isotope	Energy (keV)	Row Number			
		1	2	3	4
^{134}Cs	605	67.6	20.6	8.6	3.2
^{134}Cs	796	54.0	22.4	11.9	5.2
^{137}Cs	662	61.3	20.7	9.5	3.5
^{154}Eu	1274	40.2	22.4	15.2	7.7

III. CONCLUSIONS

Gamma-ray measurement techniques can be applied to a wide range of levels of verification of spent-fuel assemblies. Cerenkov glow intensity technique is a rapid nonintrusive technique that provides information about the presence of gamma radiation and the integrity of the fuel assembly. Ion chambers and scintillators can provide similar results but require the placement of instrumentation in the storage pond water. The $\text{Be}(\gamma, n)$ detector provides a slightly higher level of verification in that it is only sensitive to high-energy gamma rays that primarily come from the ^{144}Pr fission product. Of the gamma-ray techniques, HRGS provides the highest level of verification; however, it is the most time consuming and complex of the gamma-ray techniques. In terms of manpower and time, the most cost effective level of verification is a combination of these gamma-ray techniques with the neutron measurements. An inspector should not limit his measurements to either gamma rays or neutrons alone.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 14: NEUTRON ASSAY OF IRRADIATED FUEL

The basic active and passive neutron techniques for measuring irradiated fuels will be considered. A survey of available techniques will be given with emphasis on the measurement capabilities and limitations.

After the session, participants will be able to:

- Discuss the characteristics of the passive neutron emissions from irradiated fuel.
- Discuss the basic principles that relate neutron measurements to fuel quantities.
- Describe the instrumentation available for the active and passive assay of irradiated fuel.
- Understand the nature of the conclusions that can be drawn from neutron measurements and the precision and accuracy of the conclusions.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 14: PASSIVE AND ACTIVE NEUTRON METHODS FOR ASSAY/VERIFICATION OF SPENT FUEL

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I. INTRODUCTION

This paper is a review of the neutron techniques available for assaying spent fuel. It is not intended to be a detailed technical description but merely an overview to provide the reader with information regarding the currently available techniques.

Passive neutron assay has been identified as a potentially useful inspection assay method of spent fuel in recent review^{1,2} papers and in an International Atomic Energy Agency advisory group meeting.³ 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 that the interior rods of pressurized-water reactor (PWR) assemblies contribute nearly the same amount to the total neutron emission rate as the exterior rods.⁴ 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 detectors. This simplicity can be a distinct advantage in the hostile environment of a spent-fuel storage pool.

Active neutron assay is a direct measurement of the fissile components in spent-fuel assemblies. It is more complicated than passive measurements but can overcome some of the difficulties that arise when the correlation of the passive neutron rate to burnup is poor.

II. PASSIVE NEUTRON MEASUREMENTS

A. Neutron Emissions

Neutrons from spent-fuel assemblies arise from three principal interactions:

- ⊙ Spontaneous fission of the transuranic isotopes
- ⊙ Interaction of alpha particles from the radioactive decay of plutonium, americium, and curium, with oxygen in the matrix
- ⊙ Induced fission of the fissile isotopes by neutrons from the above sources.

A list of the emission rates resulting from the (α, n) reaction and spontaneous fission for the transuranic isotopes is given in Table I. The principal sources of neutrons are the even isotopes of plutonium and curium. These isotopes are produced through the chains illustrated in Fig. 1. Although the curium isotope production rate is much lower than the plutonium isotope

TABLE I

PRINCIPAL SOURCES OF NEUTRONS IN IRRADIATED UO_2 MATERIALS

Isotope	Half-Life (Yr)	(α, n) Reaction ^a	Spontaneous Fission ^a	Total ^a
^{235}U	7.04×10^8	7.11×10^{-4}	2.99×10^{-4}	1.01×10^{-3}
^{238}U	4.47×10^9	8.26×10^{-5}	1.36×10^{-2}	1.37×10^{-2}
^{238}Pu	8.78×10^1	1.34×10^4	2.59×10^3	1.60×10^4
^{239}Pu	2.41×10^4	3.81×10^1	2.18×10^{-2}	3.82×10^1
^{240}Pu	6.55×10^3	1.41×10^2	9.09×10^2	1.05×10^3
^{241}Pu	1.47×10^1	1.27×10^0		1.27×10^1
^{242}Pu	3.76×10^5	2.05×10^0	1.72×10^3	1.71×10^3
^{241}Am	4.32×10^2	2.69×10^3	1.18×10^0	2.69×10^3
^{242}Cm	4.46×10^{-1}	3.76×10^6	2.10×10^7	2.48×10^7
^{244}Cm	1.81×10^1	7.73×10^4	1.08×10^7	1.09×10^7

^aNeutrons per gram per second

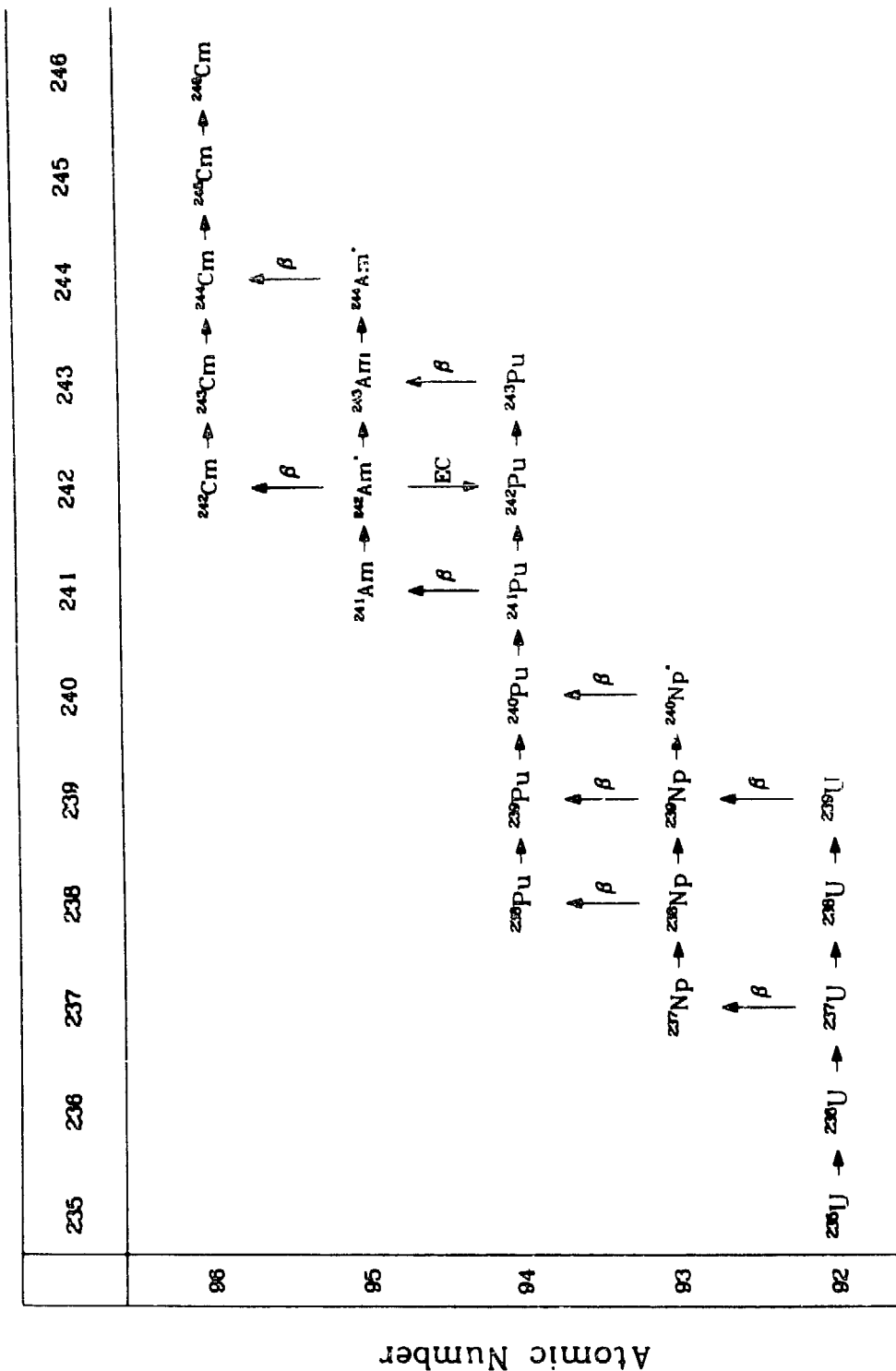
production rate, the curium isotopes are the dominant sources of neutrons within irradiated fuel assemblies with exposures above 10 000 MWd/tU because of the much larger neutron yield.⁵ The ^{242}Cm half-life is relatively short ($t_{1/2} = 4.46 \times 10^{-1}$ yr), and the decay of this isotope alters the total neutron source rate after discharge. The calculated neutron source rate as a function of cooling time for various exposures is shown in Fig. 2.⁶ As the exposure increases, the ^{244}Cm isotope becomes more important and the change in time of the total neutron yield attributed to the decay of the ^{242}Cm isotope is less pronounced.

A spent-fuel assembly is a multiplying medium with an effective multiplication constant ranging from ~ 0.9 for fresh fuel to ~ 0.6 for high exposure (~ 35 000 MWd/tU) spent fuel. By using the approximate relation⁷ that

$$M = \frac{1}{1 - k_{\text{eff}}} \quad (1)$$

where k_{eff} is the multiplication constant and M is the system multiplication, the system multiplication ranges from ~ 10 to ~ 3 . This means that the passive neutron rate will depend on both the system multiplication and the amount of the transuranic isotopes listed in Table I. The passive neutron rate can be written approximately⁷ as

Mass Number



Metastable states exist for ^{240}Np , ^{242}Am , and ^{244}Am

Fig. 1. Formation of actinide isotopes.

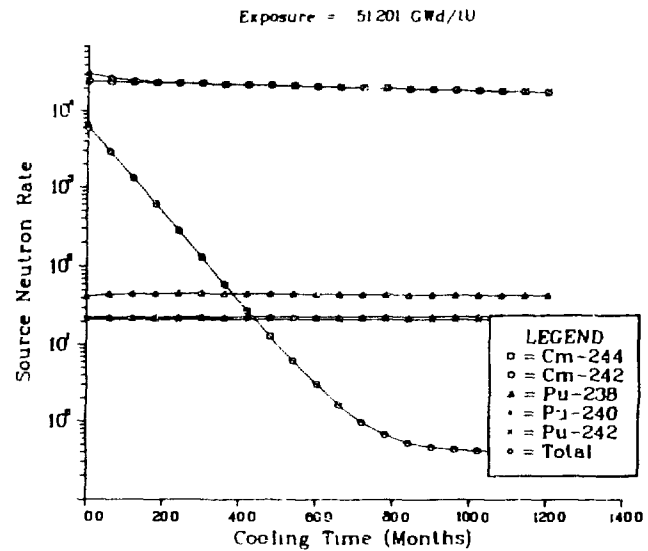
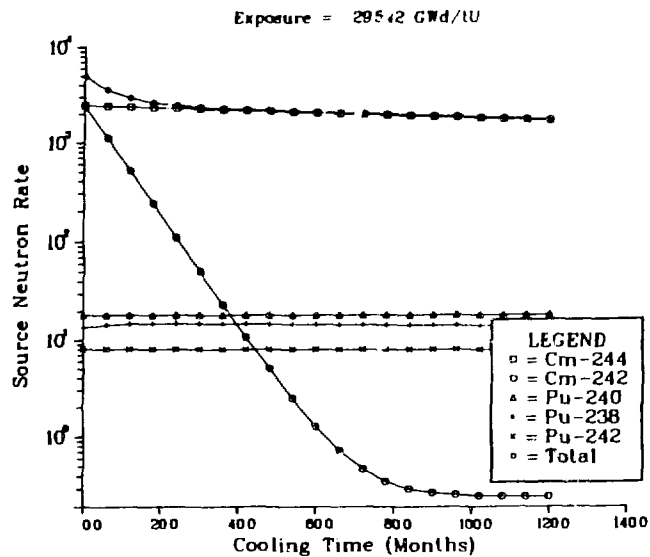
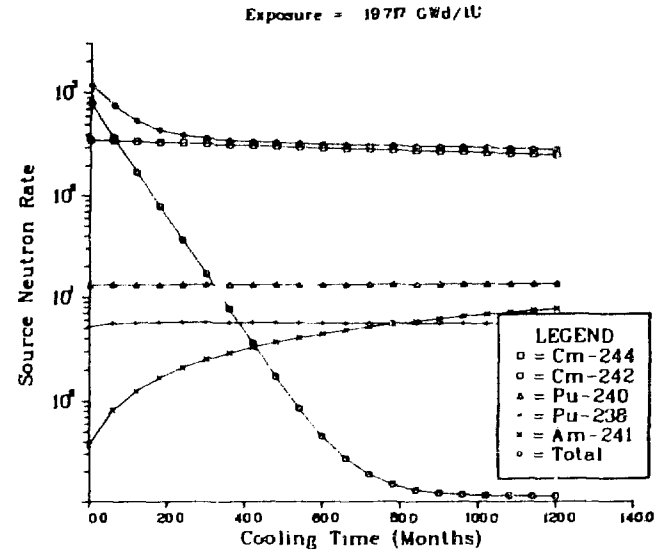
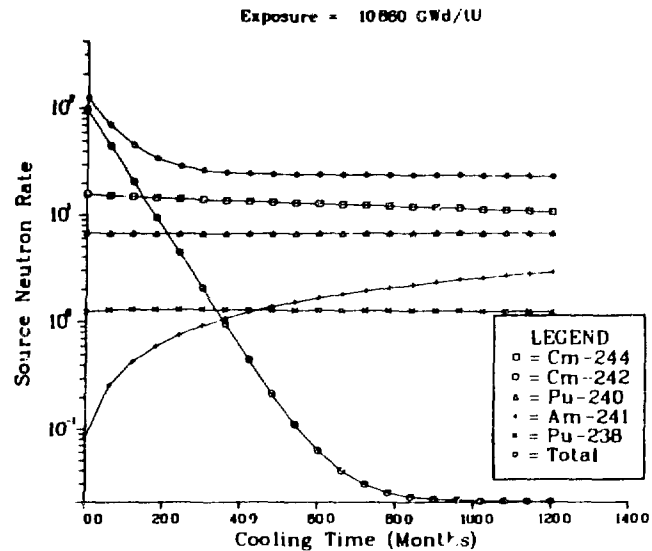


Fig. 2. Relative neutron source rates for each of the five largest contributions to the total neutron rate.

$$N_P = M S_1 \quad , \quad (2)$$

where the relationship to the multiplication, M , and the unmultiplied total neutron source strength, S_1 , is shown.

B. Passive Neutron Measurement Techniques

The measurement of the passive neutron rate from spent fuel is complicated by the very high gamma dose (50 000 to 100 000 R/h) originating from fission products. Thus, any technique used to measure the neutron rate must be insensitive to gamma-ray pileup. Foil activation and proportional counters are the two alternative methods for neutron measurements that have been successfully used.

1. Proportional Counter Techniques. Several different types of proportional counters have been considered for use in spent-fuel measurements.^{8,9} Table II lists these detectors. Because the spent-fuel element is typically stored in a water environment, thermal neutron detectors will provide the best sensitivity, and of those detectors listed in Table II, the ^{235}U fission chamber provides the best immunity from the high radiation fields. The low efficiency of the fission chamber is generally not a problem because the neutron emission rate from spent fuel is quite large (10^7 to 10^8 n/s). ^3He and BF_3 detectors can be used but because of their low gamma dose limit they require massive lead shielding. Small ^{10}B detectors can operate in gamma doses of over 10^4 R/h.^{8,9} One or two centimeters of lead shield can extend this range of operation to 10^5 R/h. Although fission chambers have been the most frequently used neutron detector for spent fuel, a ^{10}B detector might be a substitute when either transport or use of a detector with fissionable material presents a problem.

TABLE II
THERMAL-NEUTRON DETECTOR COMPARISON

Detector Type	Diam x Length (mm)	Fill Pressure (atm)	Operating Voltage (V)	Efficiency (cps/nv/cm)	Max γ Dose ^a (R/h)
^{235}U fission chamber	25 x 152	1	400	0.01	10^5
^3He detector	25 x 152	4	1500	1.1	2
BF_3 detector	25 x 152	1	1800	0.32	10
^{10}B detector	25 x 152	1	900	0.10	10^3
^{10}B detector	12 x 152	1	700	0.05	10^4
^{10}B detector	6 x 152	1	700	0.02	2×10^4

^aMaximum γ dose corresponds to unshielded condition. The addition of 1 or 2 cm of lead will increase this value 10 times.

Passive neutron methods have relied primarily on the use of fission chambers with either single or multiple measurements along the axis of the fuel assembly. One method of measurement is shown in Figs. 3 and 4.⁶ This system consists of a V-shaped positioning mechanism attached to a tube containing various gamma-ray and neutron detectors. Any number of neutron detectors can be placed inside the tube to measure either a single point or the axial neutron profile. Alternatively, the neutron detector could scan the fuel assembly or the fuel assembly could be scanned past a fixed neutron detector. The V-shaped measurement system measures only one corner of the fuel assembly, and it must be rotated if other corners are to be measured.

Another type of detector is shown in Fig. 5.⁵ This ring detector surrounds the fuel assembly and measures two or more sides simultaneously. It contains a circular ion chamber divided into quadrants and two fission chambers on opposite sides. Four fission chambers are possible. The ring

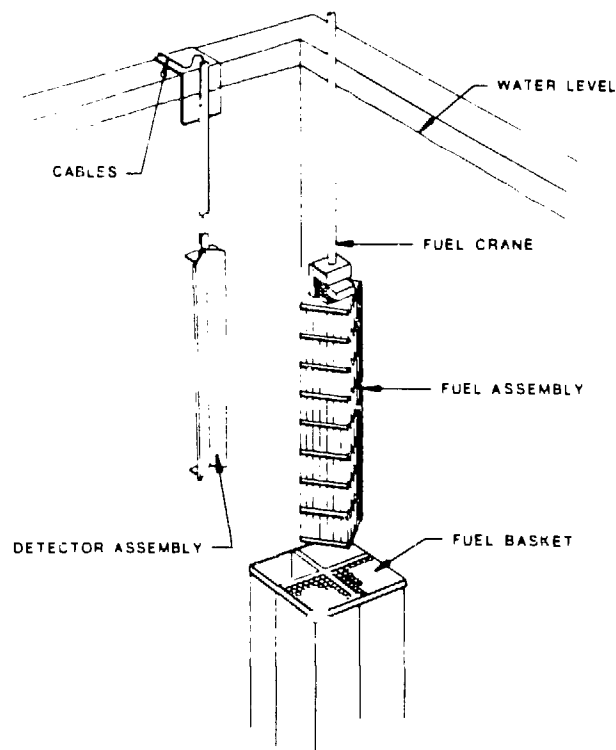


Fig. 3. Spent-fuel measurement system with V-shaped positioning mechanism.

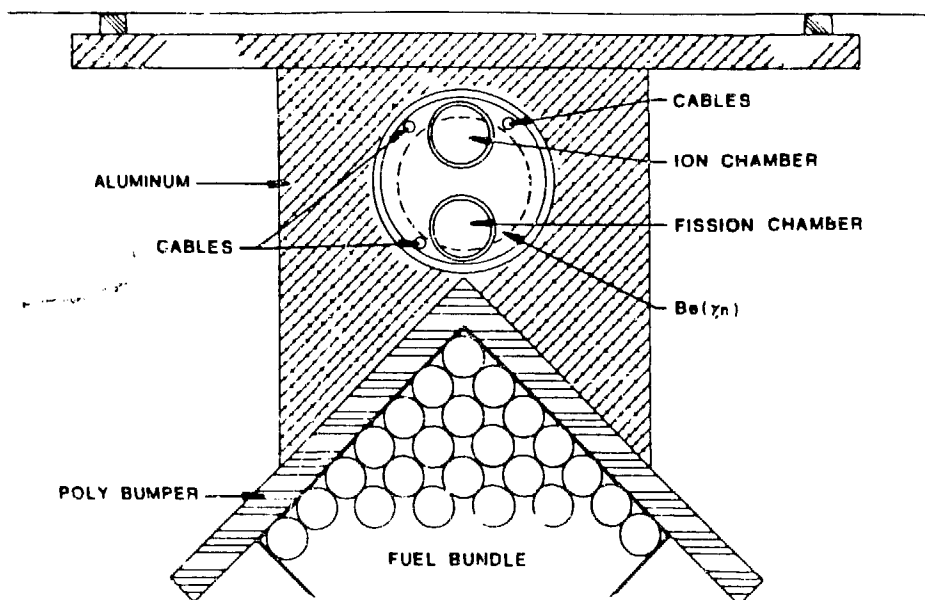


Fig. 4. Relative position of the fuel assembly and the detector tube.

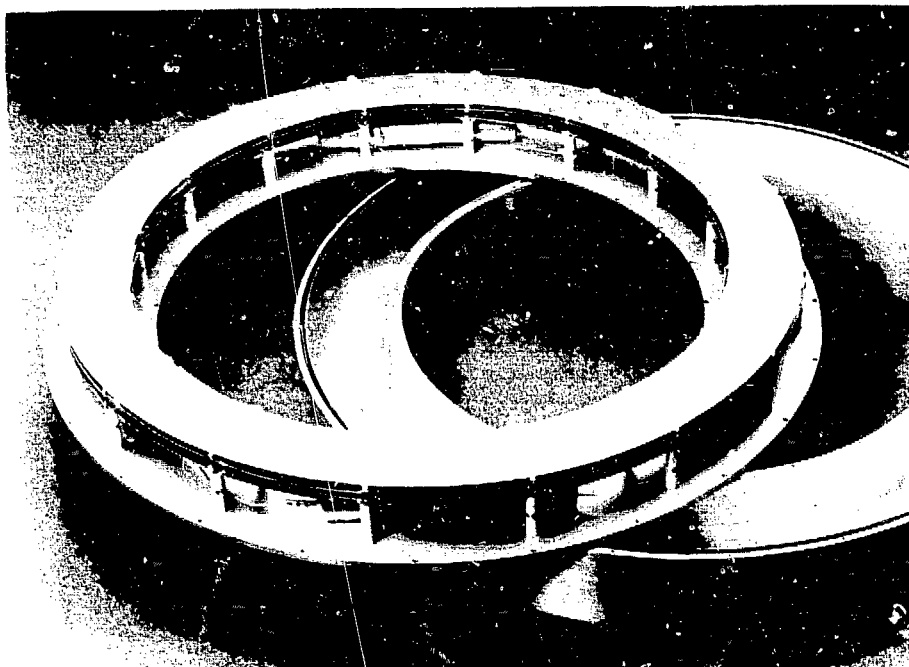


Fig. 5. Ring detector containing two fission chambers.

detector on top of a fuel storage rack with a fuel assembly in measurement position is shown in Fig. 6. This detector measures either the central position of the fuel assembly or multiple positions if the fuel assembly is moved. Whereas the V-shaped detector requires the fuel assembly to be removed from the storage rack location, the ring detector can be placed directly on the top of the storage rack so that only vertical motion of the fuel assembly is required. A variation of the ring detector is the "square" ring detector shown in Fig. 7.⁶ This detector operates in a manner identical to the "round" ring detector but is easier to position on the storage racks.

Generally, passive neutron assay and verification of spent fuel has concentrated on measuring the total neutron rate and comparing this quantity to the fissile content or burnup. Coincidence and higher multiplicity detection have not received much attention. However, such a system is possible. A modification of the neutron coincidence collar described in Session 11 (to allow underwater operation and the addition of lead shielding) would be suitable for coincidence measurements. A coincidence system would be sensitive to the spontaneous fission and induced fissions but less sensitive to the (α, n) reactions. However, total insensitivity to the (α, n) neutrons is not achieved because some of the induced fissions will always be attributed to the (α, n) neutrons. There is generally no real clear advantage of a coincidence counting system over a system relying on totals counting, and the simplicity of a totals counting system will make it the preferred method.

2. Foil Activation Techniques.¹⁰ Foil activation is an alternative to passive neutron counting with proportional counters. In this technique, metal foils with a high-neutron-absorption cross section are placed in close proximity to the spent fuel and are irradiated. After the irradiation, the foils are removed and the beta activities of the foils are measured with a suitable

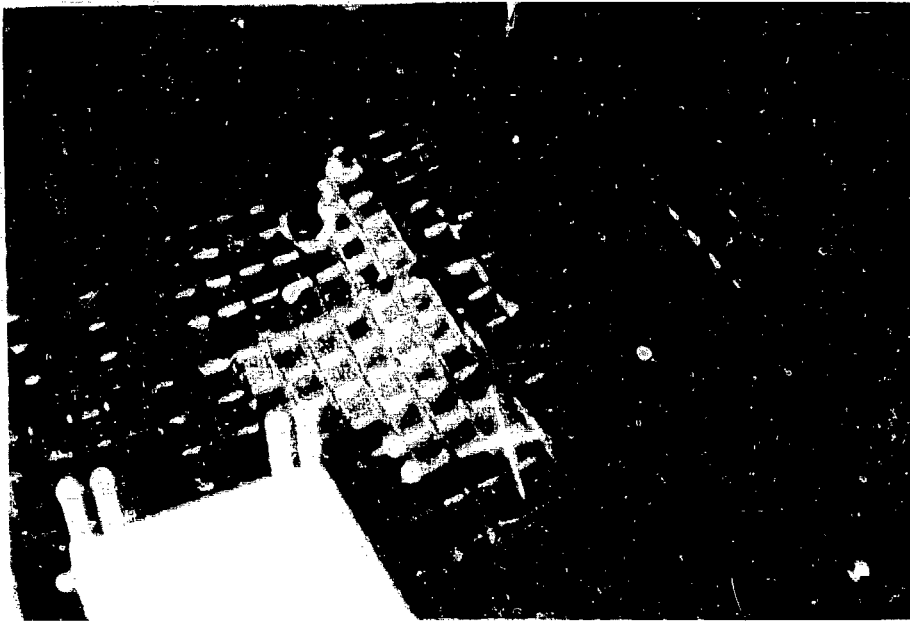


Fig. 6. Ring detector on top of fuel storage rack with fuel assembly in measurement position.

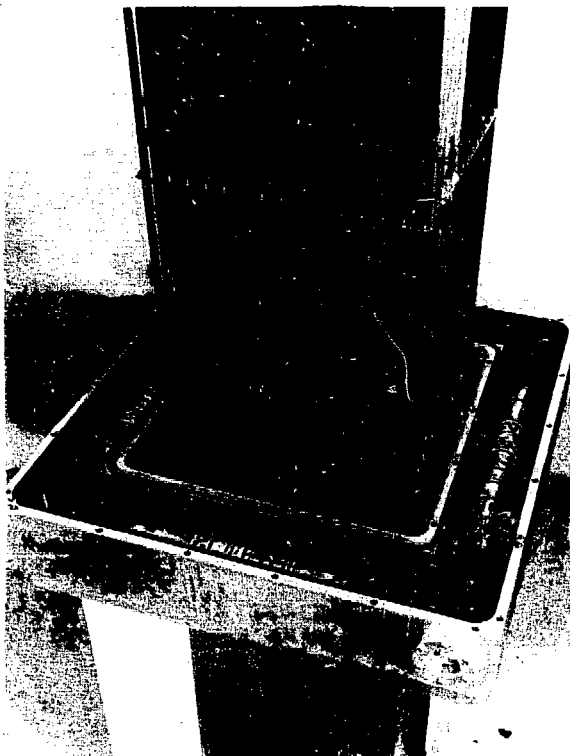


Fig. 7. Square detector placed around a 15 x 15 PWR fuel assembly.

detector such as a plastic scintillator. Metal foils such as gold ($t_{1/2} = 2.7$ days), indium ($t_{1/2} = 54$ min), and dysprosium ($t_{1/2} = 140$ min) are suitable foils, but the choice of foil depends on the passive neutron rate of the spent fuel. In a typical measurement on boiling-water reactor (BWR) fuel elements with exposures between 10 and 17 GWd/tU and cooling times of 247 to 711 days, the irradiation time of a gold foil was 12 to 72 h (Ref. 11). This can be compared to a total measurement time for fission chamber measurements of a fuel assembly of $\sim 1-1/2$ to 1 h.

The thermal neutron flux at a foil position is determined by the equation¹²

$$\sigma_{th} = \frac{(C/t_c - B/t_b)e^{\lambda t_{cool}}}{\{1 - e^{-\lambda T}\}(V \Sigma_{act} f_v f_\beta)} \cdot \frac{1 - 1/R_\beta}{\epsilon_\beta} \quad (2)$$

where C, B = total and background counts for counting times t_c and t_b
 λ = decay constant for the foil material
 T = irradiation time
 t_{cool} = cooling time
 V = foil volume
 Σ_{act} = macroscopic activation cross section for foil material
 ϵ_β = detector efficiency
 f_n = correction factor for foil perturbation
 f_β = beta self-shielding factor
 R_β = cadmium ratio, which is the activity of bare foil divided by the activity of cadmium-covered foil.

One observes that in addition to the time required for irradiation, additional time is spent scanning the foil and then calculating the thermal neutron flux. This large expenditure in time is a distinct disadvantage when many fuel assemblies must be measured.

D. Results of Passive Neutron Measurements

The ring detector was used to examine 36 PWR spent-fuel assemblies at a commercial reactor fuel storage facility.⁵ The results, shown in Fig. 8, are for measurements on the central axis with fission chambers on two sides. The solid line is a power function fit of the form

$$\text{count rate} = \alpha \cdot \text{exposure}^\beta ; \quad \beta = 3.00$$

where the fit does not include the 4-month cooling time data. No cooling time correction has been applied to the data. The short cooling time data do not

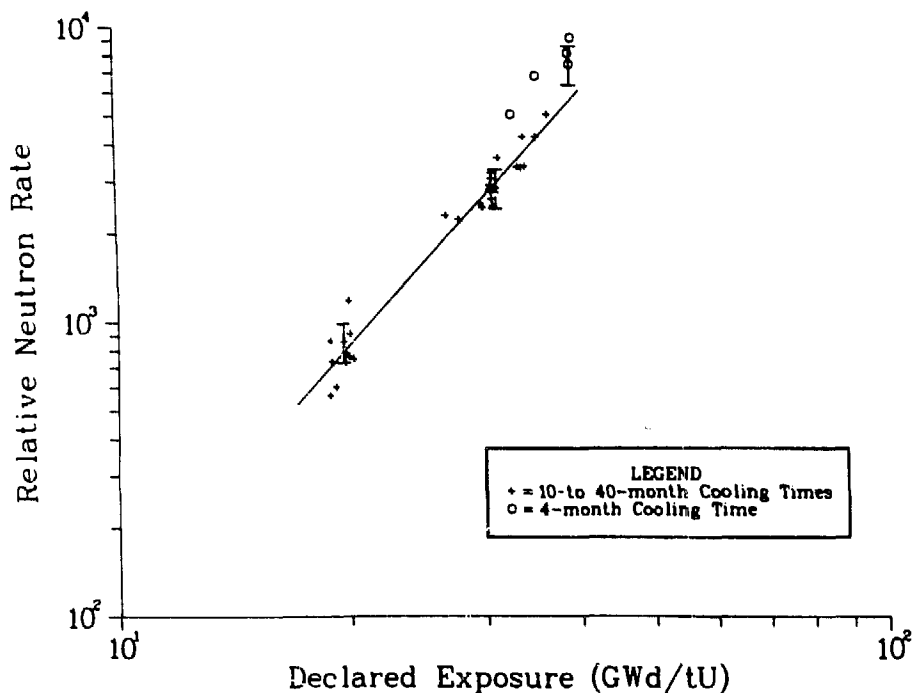


Fig. 8. Experimental results using ring detector to measure PWR fuel assemblies.

follow the power function relationship because the ^{242}Cm isotope has a short half-life (163 days). Curium-242 is an important neutron emitter for short cooling times but decreases in importance for cooling times >1 yr. If the 10-to 40-month data are corrected for the decay of ^{244}Cm , then β decreases to 2.88.

A power function relationship was also observed in measurements with the V-shaped measurement system at an away-from-reactor storage facility.⁶ The exponents for fuel assemblies from three different power reactors were 3.10, 3.68, and 3.91. The PWR fuel assemblies were 14 x 14, 15 x 15, and 14 x 14 rod arrays, respectively. A significant aspect of this series of experiments was the differences in counting rates observed for different corners of the fuel assemblies. Figures 9-11 show the variation in counting rates for different corners of the fuel assemblies. In all cases the corner with the lowest counting rates was on the periphery of the core, and this effect persists even if the fuel assembly is moved to an interior location of the core. Differences in counting rates of this magnitude make interpretation of one-sided measurements difficult and require that measurements on all sides (or corners) be made and the average or sum used for the assay.

Foil activation measurements on BWR spent-fuel elements have yielded some very interesting results.¹¹ Figures 12 and 13 show the neutron emission rate for various exposure values along the axis of the two fuel assemblies. The general shape of exposure vs neutron emission is a loop with the maximum normally centered approximately in the center of the fuel assembly. The fact that two parts of the fuel assembly with the same exposure can have two distinct neutron emission rates is due to the void fraction. The top of the fuel assembly with the higher void fraction has more transuranic nuclides

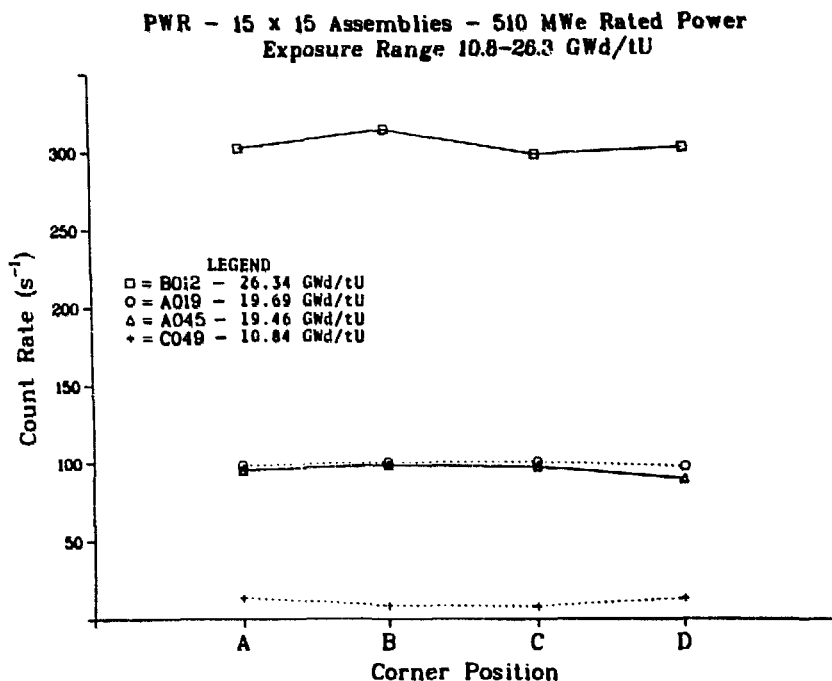


Fig. 9. Relative neutron emission rates as a function of the four corner positions for 15 x 15 PWR fuel assemblies.

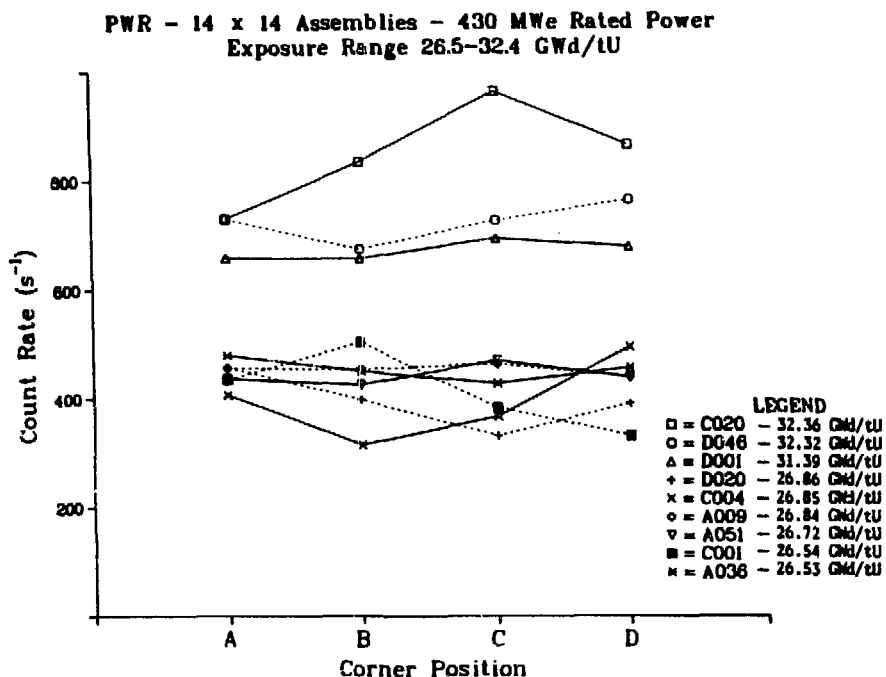


Fig. 10. Relative neutron emission rates as a function of the four corner positions for 14 x 14 PWR fuel assemblies.

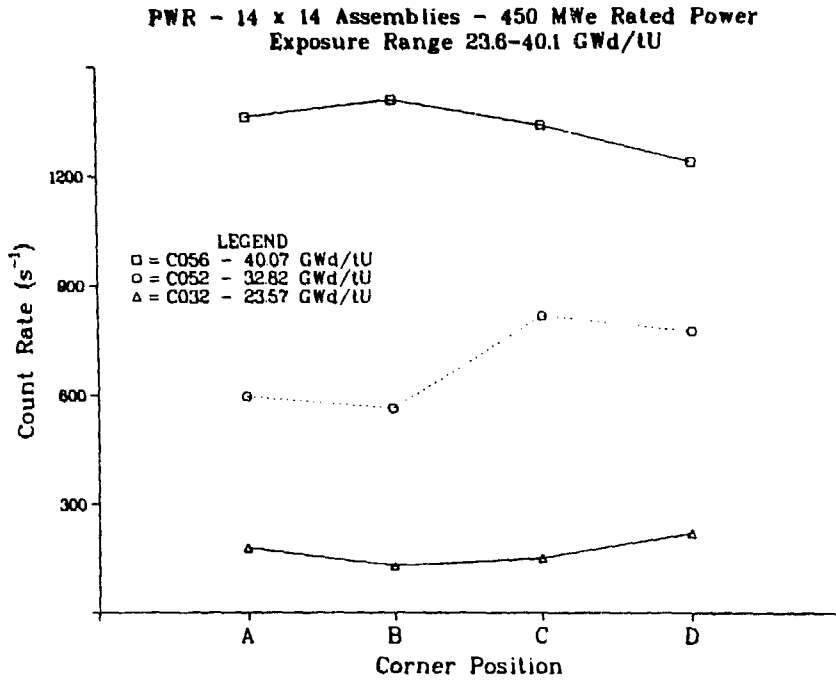


Fig. 11. Relative neutron emission rates as a function of the four corner positions for a 14 x 14 PWR fuel assembly.

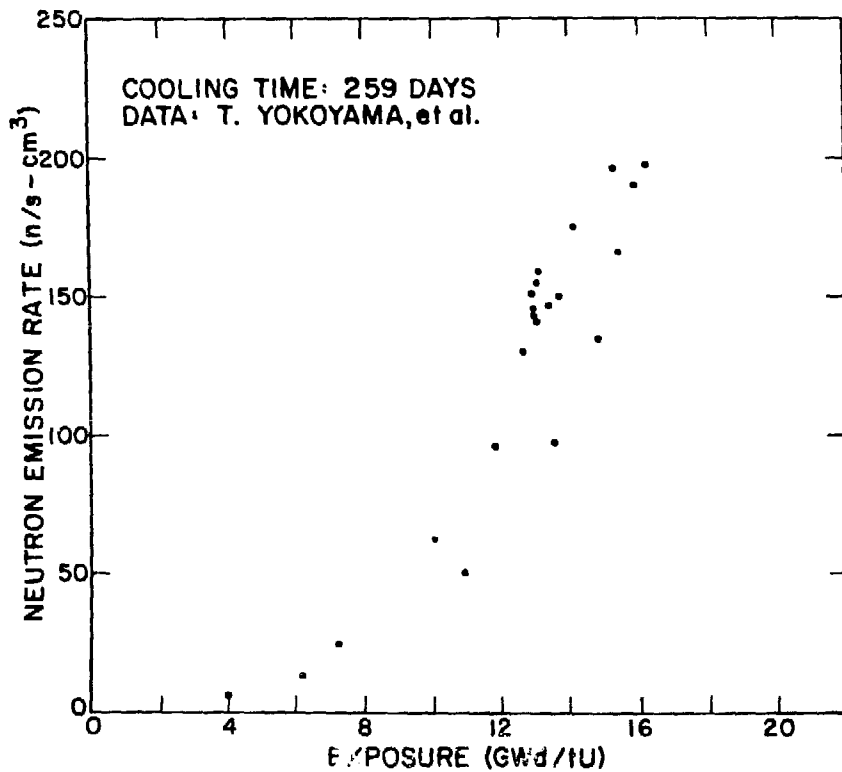


Fig. 12. Passive neutron results for a BWR fuel assembly.

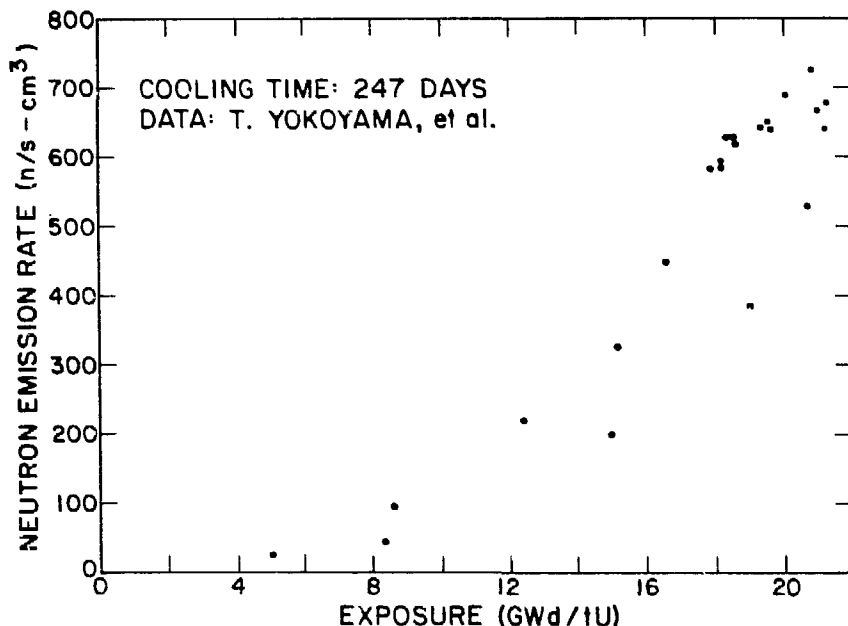


Fig. 13. Passive neutron results for a BWR fuel assembly.

produced than the bottom because the average neutron energy is higher on the top than on the bottom.

The results presented in Figs. 12 and 13 clearly demonstrate the problems that can arise when a correlation between neutron emission rate and exposure is attempted. This is due to the fact that the passive neutron rate depends on ^{242}Cm and ^{244}Cm . When problems such as this arise, a better assay and verification measurement would be one that is sensitive to the ^{235}U and plutonium isotopes alone. Active neutron interrogation accomplishes these requirements.

III. ACTIVE NEUTRON INTERROGATION--EXTERNAL SOURCES

Active assay has long been considered the best assay method for spent fuel. In this technique an external neutron source induces fission in the uranium and plutonium isotopes. The fission neutrons are detected and correlated to either burnup or fissile content. The interrogating sources can be either isotopic sources such as ^{252}Cf , $^{124}\text{Sb-Be}$, Amli, etc., or accelerator sources such as sealed-tube neutron generators. A generic active interrogation system is shown in Fig. 14. The neutron source of strength S_0 neutrons/second emits neutrons that interact in the fissile material of the fuel assembly and cause fissions. The neutron detector detects the neutrons from the induced fissions as well as the passive neutron background. The neutrons from the induced fissions are related to the amount of fissile material present in the fuel assembly.

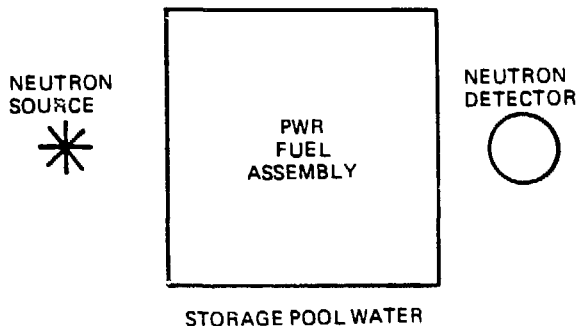


Fig. 14. A generic neutron interrogation system.

Mathematically, the detected count rate with active interrogation, N_A , is equal to the passive neutron count rate, N_p , plus the induced neutron count rate, N_i , so that

$$N_A = N_p + N_i \quad . \quad (3)$$

The induced count rate, N_i , depends on the neutron source strength and the amount of fissile material as well as geometrical factors. The induced rate, N_i , can be written as

$$N_i = g(x, y, z, U, Pu, \dots) S_0 \quad , \quad (4)$$

where the dependence on the source strength is explicitly shown. The undefined function $g(\dots)$ depends on all other variables of the system, including fissile content. Combining and rearranging Eqs. (3) and (4), the induced rate can be written as

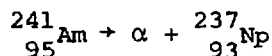
$$N_i = N_A - N_p = g(x, y, z, U, Pu, \dots) S_0 \quad . \quad (5)$$

An assay is accomplished by correlating the function g with the fissile content.

A. Neutron Sources

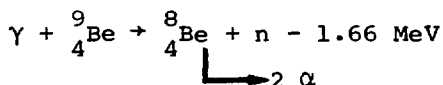
Neutron sources available for active interrogation systems include 14-MeV neutron generators, (α, n) isotopic sources, (γ, n) isotopic sources, ^{252}Cf spontaneous fission sources, nuclear reactors, and accelerator photoneutron sources. The first four types are generally considered the most useful sources for spent fuel. The last two are not generally useful for assaying spent fuel and will not be discussed further.

Isotopic sources based on the (α, n) reaction rely on the interaction of the alpha particle from the decay of an alpha emitter with a light element. For example, an $^{241}\text{Am-Li}$ (α, n) source undergoes the following reactions:



Light elements such as boron, lithium, beryllium, and fluorine and the alpha emitters plutonium, radium, polonium, and americium are typically used to make (α, n) sources. Approximate mean neutron energies for lithium-based (α, n) sources are ~ 0.3 MeV, while the approximate mean neutron energies for beryllium-based sources are ~ 4.7 MeV. Source strengths for reasonable size (α, n) sources are $\sim 10^6$ n/s.

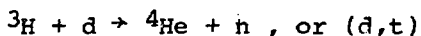
Isotopic photoneutron sources (γ, n) are based primarily on the light elements, beryllium and deuterium, because the energy necessary for the photoneutron reaction is quite small, 1.67 MeV and 2.23 MeV, respectively. A photoneutron source can be created by surrounding an appropriate gamma-ray-emitting material with a mantle of beryllium or heavy water (D_2O). The reaction for beryllium is



The typical gamma-ray emitters used are ${}^{124}\text{Sb}$ ($\gamma = 1.692$ MeV), ${}^{88}\text{Y}$ ($\gamma = 1.836$ MeV), and ${}^{226}\text{Ra}$ ($\gamma = 1.728$ MeV), with resulting photoneutron energies of 24 keV, 150 keV, and 55 keV, respectively. A moderate size (γ, n) source yields $\sim 10^7$ n/s.

Californium-252 is at present the only practical spontaneous fission source. The neutron energy distribution is that of a typical fission spectrum with an average energy of 2.35 MeV. The neutron yield is 2.3×10^{12} n/s/g, making it an intense neutron source. Californium sources with intensities between 10^5 n/s to more than 10^{10} n/s are readily available.

Small sealed-tube neutron generators are compact charged-particle accelerators that are used to produce neutrons. Both (d, d) and (d, t) reactions are possible:



The accelerating voltage is normally 100 to 150 keV and produces neutrons of ~ 2.4 MeV for the (d, d) reaction and ~ 14 MeV for the (d, t) reaction. Neutron yields of up to 10^{12} n/s can be achieved with the (d, t) reaction and 10^{10} n/s for the (d, d). Sealed tubes can be operated in a pulsed mode.

B. Measurement Techniques

The active assay of spent fuel can be accomplished by measuring either the prompt neutrons from the induced fissions or the delayed neutrons.

A system has been designed that utilizes a subthreshold interrogation and prompt- and delayed-neutron counting technique.¹³ In this system, four intense ${}^{124}\text{Sb}$ -Be sources (each with ~ 200 Ci ${}^{124}\text{Sb}$ and producing 8×10^8 n/s)

interrogate a spent-fuel subassembly as it enters a fuel reprocessing plant. The high-energy fission neutrons and delayed neutrons are discriminated from the lower energy (24 keV) interrogating neutrons by pulse-height discrimination applied to methane-filled proton recoil proportional counters. The assay system is shown in Fig. 15. A 20-cm-thick lead shield surrounds the proton recoil counters to shield them from the gamma radiation from the ^{124}Sb sources and spent fuel. The sensitivity of this system in a 20-min assay is estimated to be <5% in the total fissile content. One observes in Fig. 15 that assay systems not in a storage pool are large and massive.

An assay system based on a ^{252}Cf source and prompt neutron counting is shown in Fig. 16.¹³ The measurements are designed to be performed in the spent-fuel storage pool. Some lead shielding is used to shield the fission chamber, and a cadmium sheet limits the neutron response to epithermal neutrons. By limiting the response to epithermal neutrons, the assay sensitivity to fissile content is improved. The neutrons from the ^{252}Cf source are moderated in the water medium so that the primary neutron flux is thermal and epithermal. Some residual fast fission still occurs to the extent that fission of ^{238}U contributes approximately 7% to the total count rate. For PWR-type fuel assemblies with exposures up to 30 Gwd/tU, a ^{252}Cf source strength of $\sim 3 \times 10^8$ n/s was required. The precision for the assay was ± 1.2 Gwd/tU for PWR fuel and ± 1.5 Gwd/tU for BWR fuel.

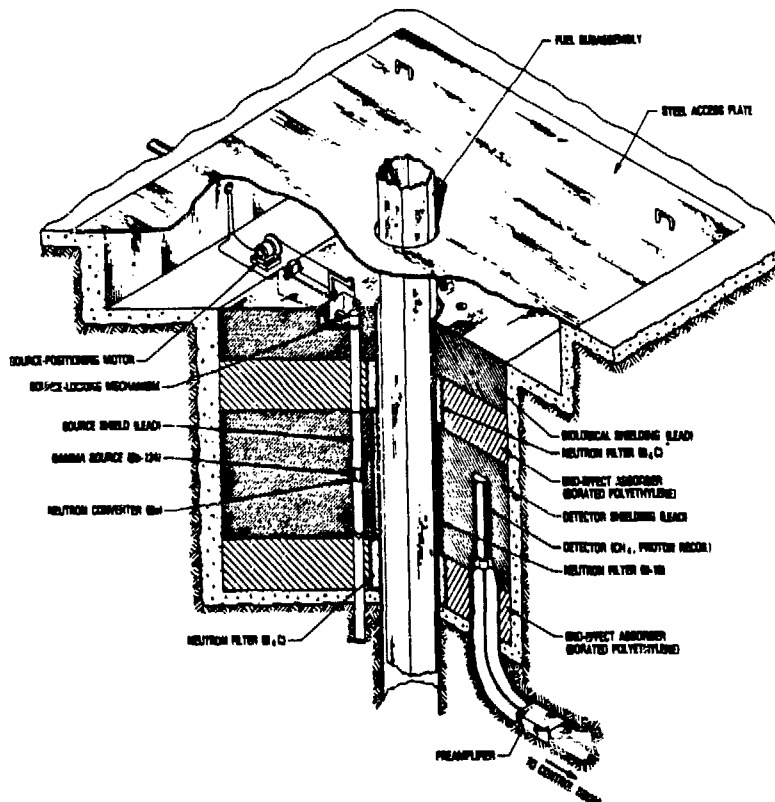


Fig. 15. ^{124}Sb -Be active assay system.

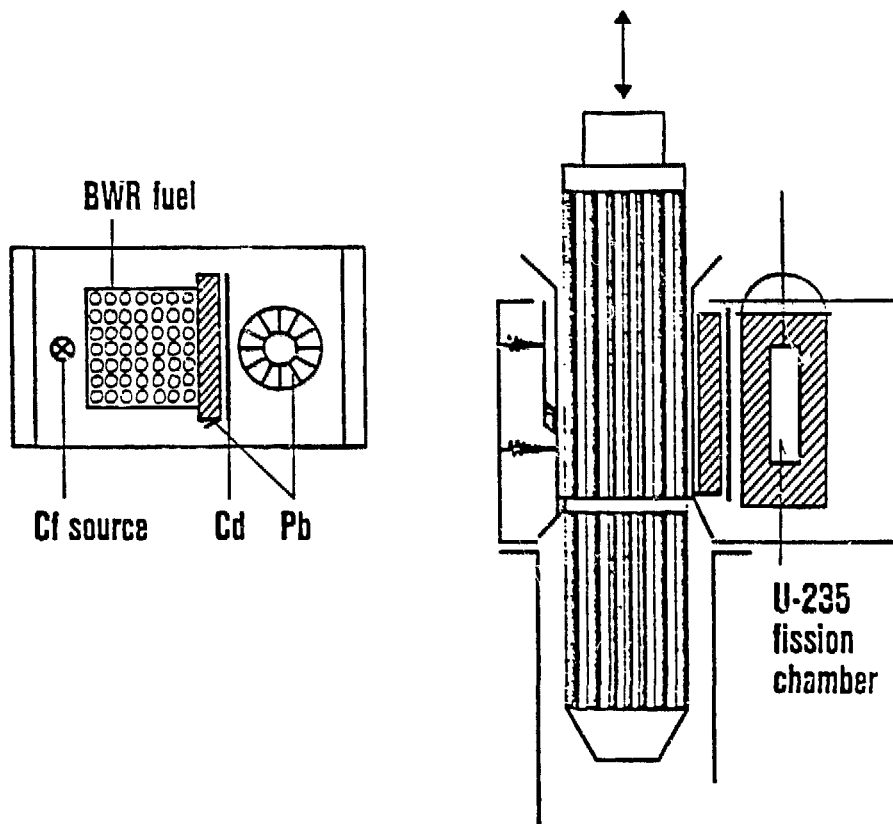


Fig. 16. ^{252}Cf active assay system.

An alternative to prompt neutron counting is delayed neutron counting. A system based on ^{252}Cf interrogation and delayed neutron counting is shown in Fig. 17.¹⁴ This system was designed to assay the ^{235}U content in high-enriched uranium spent fuel. Fast neutrons from the californium source irradiate the spent fuel for approximately 8 s. The source is removed to the lead shield with a transfer time of ~ 0.5 s, and then the ^3He detectors count the delayed neutrons from the induced fissions for ~ 3 s. This cycle is repeated until the required precision is attained. Because of the high gamma background, 15 cm of lead shielding is placed between the detectors and the assay tube. The ^{252}Cf source yields 10^9 - 10^{10} n/s, and the design goal is a ^{235}U sensitivity of 5% for a fuel loading of ~ 10 kg.

IV. SELF-INTERROGATION OF SPENT FUEL

While active assay is the one technique that measures directly the fissile content of spent fuel, the requirement for a high-intensity neutron source can at times be a serious disadvantage. A technique has recently been developed that eliminates the need for external isotopic sources, yet retains the advantages of an active assay system.¹⁵

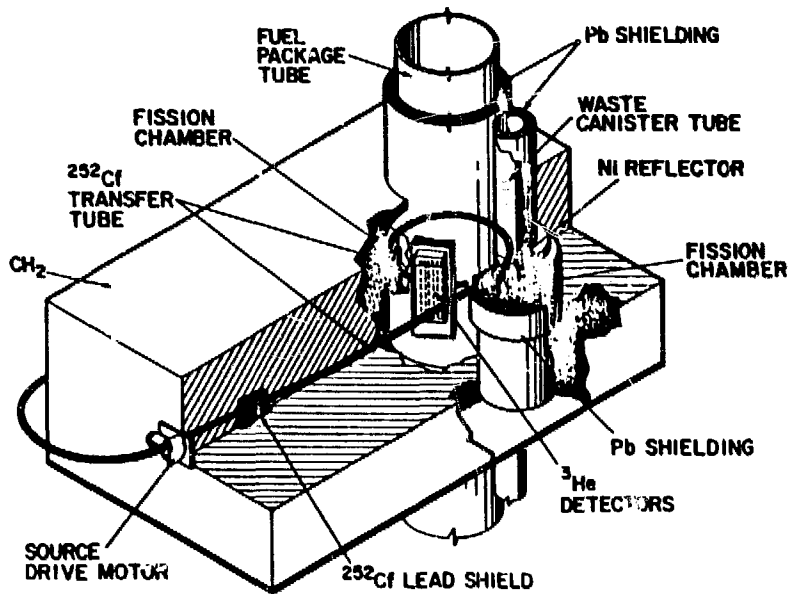


Fig. 17. ^{252}Cf delayed neutron interrogator.

A generic self-interrogation system is shown in Fig. 18. In this method, the assay is accomplished by making passive neutron measurements with and without the presence of a neutron-absorbing material near the assembly.

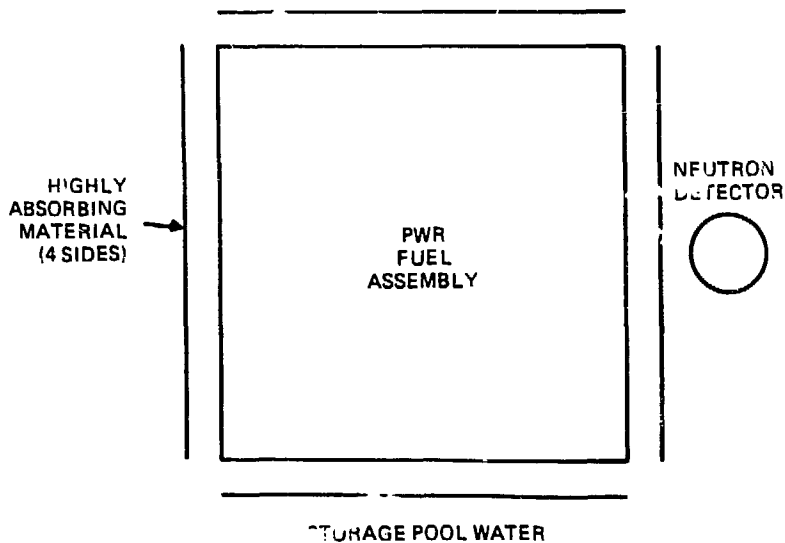


Fig. 18. A generic self-interrogation system.

Mathematically, the neutron count rate, N_R , with the absorber in place can be written in a manner similar to Eq. (3) as

$$N_R = N_p - N_j \quad , \quad (6)$$

where N_p is the passive neutron count rate previously defined, and N_j is the count rate reduction due to the absorber. N_j depends on the original passive rate, N_p , the fissile content, and geometrical factors. By analogy to Eq. (4), N_j can be expressed as

$$N_j = f(x, y, z, U, Pu, \dots) N_p \quad , \quad (7)$$

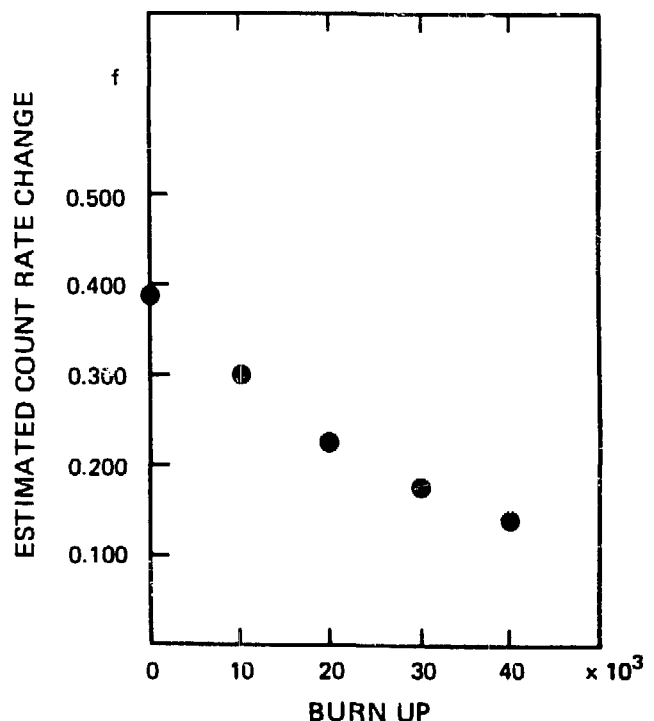
where the dependence on the passive neutron rate is explicitly shown. The undetermined function f contains all other variables. Combining and rearranging Eqs. (6) and (7), the rate N_j can be expressed as

$$-N_j = N_R - N_p = -f(x, y, z, Pu, U \dots) N_p \quad . \quad (8)$$

A comparison of Eqs. (5) and (8) reveals that one of the differences between the active-source interrogation technique and the self-interrogation technique is that the thermal neutron source in Eq. (5) is replaced by a thermal neutron sink in Eq. (8). The unspecified functions f and g , which are very complex, contain all parameters describing the fuel assembly, including the geometry, absorber, moderator, and fissile content. Because, in general, the fissile content is the only variable, a correlation of fissile content with f or g provides a means of assaying the fuel assembly.

The results of preliminary work on the correlation of the function f to burnup are shown in Fig. 19. For a $\sim 2.4\%$ precision on f , a 1000 MWd/tU precision on burnup is estimated for a burnup value of ~ 35 000 MWd/tU. The advantages of this method are that no extra uranium sources are required and any spent fuel assembly can be assayed, no matter how intense the passive neutron rate.

Fig. 19. Estimated fractional count rate change vs burnup.



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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 15: ISOTOPE CORRELATION AND MASS SPECTROMETRY TECHNIQUES FOR
IRRADIATED FUEL ASSAY

Isotope dilution mass spectrometry is used to measure the masses of uranium and plutonium isotopes present in solutions of irradiated nuclear fuel in reprocessing plants. The masses of isotopes expected to be present in such solutions can be estimated independently by isotope correlation techniques. The principles and applications of the two techniques will be discussed.

After the sessions, participants will be able to:

- Understand the techniques of isotope dilution mass spectrometry and discuss its application to the assay of irradiated fuel.
- Describe the sampling problems of irradiated fuel assay and the status of the resin bead mass spectrometric technique.
- Discuss the precision and accuracy of isotope dilution mass spectrometry.
- Understand the principles and practical aspects for applying isotope correlation techniques to verify irradiated low-enriched uranium fuel materials in reprocessing plants.

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 - 2.2. Standard Plant Accountability Procedure
 - 2.2. 1. Bulk Volume Measurement
 - 2.2. 2. Sampling
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 - 3.5. 3. Across the Plant Element and Isotope Balances
4. Conclusion



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SECTION 15: ISOTOPE CORRELATION AND MASS SPECTROMETRY TECHNIQUES FOR IRRADIATED FUEL ASSAY

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International Atomic Energy Agency

1. INTRODUCTION

The purpose of this presentation is to describe and discuss the methods used for the accountability of the fissile material in irradiated nuclear fuels entering reprocessing plants and for its verification.

Three to ten or more years elapse between the fabrication of the nuclear fuel, its irradiation in the reactor, post irradiation cooling and its reprocessing. At the reprocessing plant, the spent fuel assemblies are removed from the storage ponds and transferred into a hot cell where they are dismantled. The dismantled elements are then unclad, usually by mechanical chopping, and collected in a basket which is introduced into a dissolver tank. The fissile material is dissolved with a concentrated solution of nitric acid. Then the resulting solution, containing 200 - 400 g/l of uranium is usually filtered before it is transferred into an input accountability tank, where its volume is measured and samples are taken for analysis. The concentrated spent fuel solution enters next the separation plant where uranium and plutonium are decontaminated of the fission products and purified. The products of the purification are either separated as mixed oxides of uranium and plutonium, or solutions of nitrates of these elements.

For the sake of accountability and safeguards verifications, the plant is usually divided in a minimum of three material balance areas (Figure 1):

(a) The Head End includes the cells for dismantling, decladding, and dissolution.

(b) The Process Area starts at the Input Accountability Tank and covers the separation plant.

(c) The Low End refers to the output area where the products of the reprocessing are conditioned and measured again.

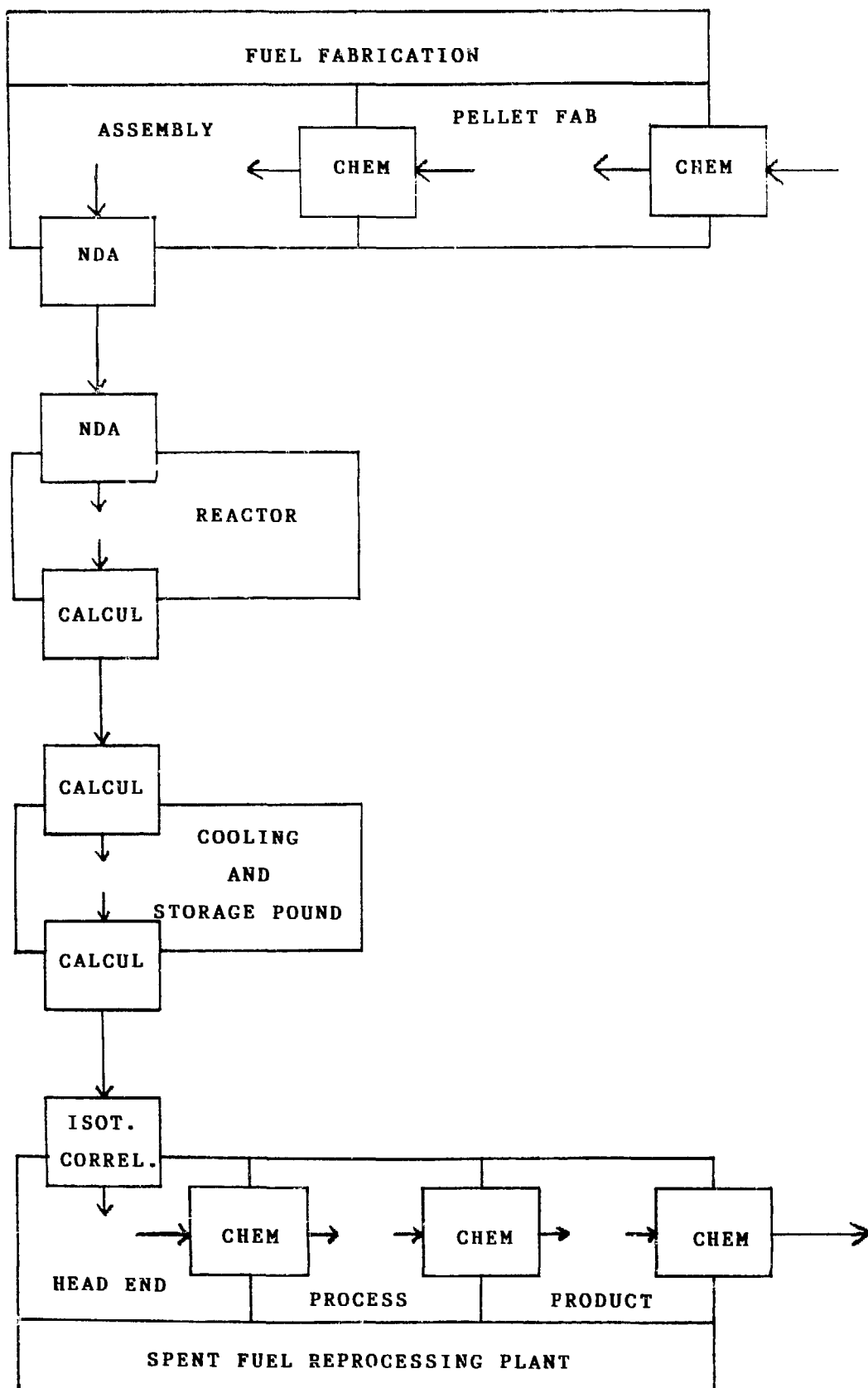


Figure 1. MATERIAL FLOW

The input accountability tank is a particularly sensitive point because the plutonium cycle starts there, but also because it is there that the fuel can be sampled and analyzed accurately for the first time since it left the fabrication plant many years earlier. Isotopic and Isotope Dilution Analyses by Mass Spectrometry are the basic methods for accountability and verification measurements at this point. These measurements establish how much fissile material actually enters the process area. They are to be compared with the declarations of the reactor operator, which are themselves based on the fuel fabricator declarations and the prediction of reactor physics. In turn, these predictions can, again according to reactor physics, be confirmed by measuring only the isotopic composition of the actinides or fission products present in the spent fuel. Isotope Correlation Techniques designate the methods used to obtain such confirmations.

Thus, at a spent fuel reprocessing plant, the isotope correlation techniques serve principally to estimate the Shipper/Receiver differences (SRD) between the reactor output declaration and the inventory change expected on receipt of the spent fuel at the storage pond of the reprocessing plant. They provide also an estimate of the differences between the expected and measured outputs out of the Head End of the reprocessing plant.

It is hoped that the following notes can help the reader to

- Understand the techniques of isotope dilution mass spectrometry and discuss its application to the assay of irradiated fuel.
- Describe the sampling problems of irradiated fuel assay and the status of verification techniques such as the resin bead mass spectrometric technique.
- Discuss the precision and accuracy of isotope dilution mass spectrometry.
- Understand the principles and practical aspects for applying isotope correlation techniques to verify irradiated low enriched uranium fuel materials in reprocessing plants.

2. ISOTOPE DILUTION MASS SPECTROMETRY

2.1. Principle of the Chemical Assay of Spent Fuel Dissolver Solutions

Concentrated spent fuel solutions contain activities of 1000 Ci/l (3.7×10^{10} MBq.m⁻³) or more of fission products (F.P.) emitting gamma rays of energies ranging up to 3-4 MeV. They must be handled remotely behind heavy biological protection shielding. Under these conditions it was not possible up to recently to apply conventional methods of wet chemical analysis for their assay. Isotope Dilution Mass Spectrometry is a tracer technique which enables us to perform accurate assays on small aliquots of dissolver solutions, which contain only about 10 mCi (370 MBq) of F.P. and can be handled behind a minimal shielding such as a 5 cm thick lead wall.

In order to introduce the principle of the technique, let us consider first a simple and ideal case (Figure 2): A sample solution contains an unknown number of atoms of isotopically pure ²³⁸U isotope, N_g, which is to be measured;

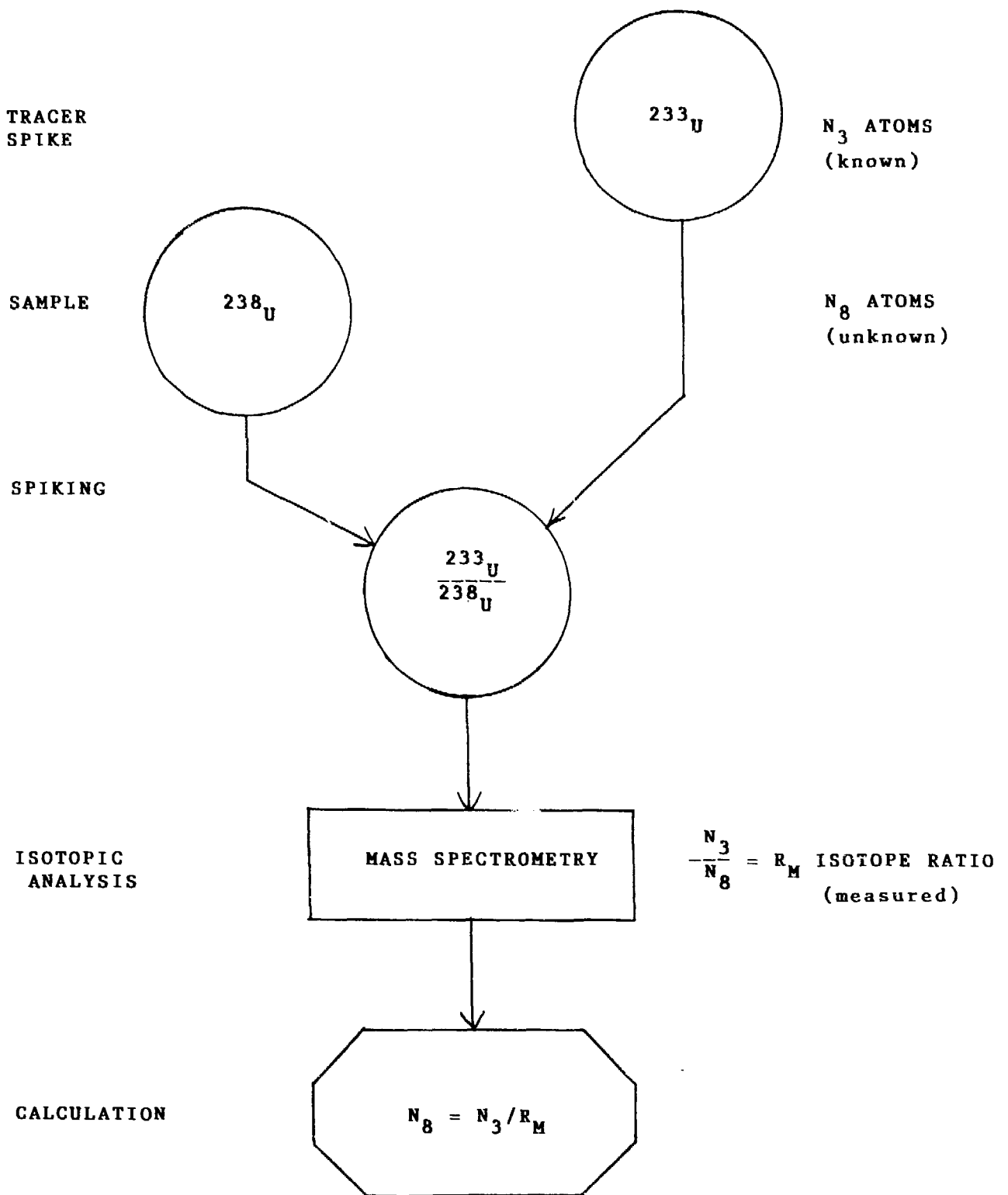


Figure 2. PRINCIPLE OF THE ISOTOPE DILUTION ASSAY OF URANIUM

a standardized solution of an isotopically pure tracer of ^{233}U isotope is available; an accurately measured aliquot of the tracer solution, containing a well known number of atoms of the tracer isotope ^{233}U , N_3 , is mixed with the sample; the tracer aliquot is called a "spike", its mixing with the sample is referred to as the "spiking"; an isotopic analysis of the mixture is performed by mass spectrometry to measure the ratio R_M of ^{233}U and ^{238}U isotopes in the mixture:

$$R_M = \frac{N_3}{N_8} \quad (1)$$

The calculation of the number of atoms N_8 of isotope ^{238}U present in the sample is straight forward:

$$N_8 = N_3 / R_M \quad (2)$$

When the sample, or the tracer, or both are not isotopically pure, one must also analyze the isotopic composition of the sample and/or of the tracer. The principle and the outline of the method remain otherwise the same, but the calculations are slightly more complex. In the most general case,

$$N_8 = N_3 \cdot \frac{1 - \frac{R_M}{R_T}}{R_M - R_S} \quad (3)$$

where R_T , R_S , R_M are the ratios of ^{233}U to ^{238}U isotopes respectively in the tracer, the sample and the mixture of both.

Light water reactor fuels however are not expected to contain any ^{233}U isotope ($R_S = 0$). If in addition the tracer is highly enriched in ^{233}U isotope (R_T is very large), it is clear that equation (3) becomes equal to equation (2)

The number of atoms N_8 and N_3 are themselves proportional to the concentrations of uranium element in the sample and in the tracer respectively, C_S and C_T :

$$N_8 = \frac{C_S \cdot V_S \cdot A_8}{A(238)} \cdot N \quad (4)$$

$$N_3 = \frac{C_T \cdot V_T \cdot A_3}{A(233)} \cdot N \quad (5)$$

where V_S and V_T are the volumes of sample and tracer which were mixed together

A_8 is the weight fraction of isotope 238-U in the sample

A_3 is the weight fraction of isotope 233-U in the tracer

$A(238) = 238.05$

$A(233) = 233.05$ are the atomic masses of isotope 238-U and 233-U

$N = 6.023 \times 10^{23}$ is the Avogadro Number

Equation (3) can thus be rewritten to calculate directly the uranium concentration.

$$C_S = \frac{C_T \cdot V_T}{V_S} \cdot \frac{A(238)}{A(233)} \cdot \frac{A_3}{A_8} \cdot \frac{1 - \frac{R_M}{R_T}}{R_M - R_S} \quad (6)$$

Plutonium is analyzed in a similar manner using usually an enriched 242-Pu tracer.

2.2. Standard Plant Accountability Procedure

Reprocessing plants usually base their accountability of the input of the process area on a combination of the measurements of the volume V of dissolver solution in the input accountability tank with mass spectrometric analyses of its content. The mass spectrometric analyses involve on one side an analysis of the isotopic composition of uranium and plutonium, and, on the other, a determination of the element concentrations, C , by isotope dilution assay (IDA). The inputs of uranium, $M(U)$, and plutonium, $M(Pu)$, are calculated accordingly:

$$M(U) = C(U) \cdot V \quad (7)$$

$$M(Pu) = C(Pu) \cdot V \quad (8)$$

$C(U)$ and $C(Pu)$ are the results of the determinations of the uranium and plutonium concentrations, respectively.

BULK VOLUME

INPUT TANK

SAMPLING

DILUTION 1/150

SUBSAMPLING

SPIKING

Pu VALENCE ADJUST

CHEMICAL SEPARATION

MASS SPECTROMETRY U

MASS SPECTROMETRY Pu

10

11

12

20

21

22

IC

IDA

IDA

IC = ISOTOPIC COMPOSITION

IDA = ISOTOPE DILUTION ASSAY OF U, Pu CONCENTRATION

Figure 3. TYPICAL OUTLINE OF ACCOUNTABILITY MEASUREMENTS AT THE INPUT ACCOUNTABILITY TANK OF A SPENT FUEL REPROCESSING PLANT

This procedure is frequently called the Volumetric Method. Its successive steps are listed in Figure 3.

The capacity of input accountability tanks varies between 500 and 3000 l. The tanks are fitted with a number of pipes and tubes:

- A feed pipe allows the transfer of the spent fuel solution from the dissolver tank into the input accountability tank,
- Another transfer line serves to empty the input tank into the process lines,
- Air sparging tubes or recirculation lines are used to stir and homogenize the solution,
- Dip tubes allow us to measure the pressure in and above the solution in the tank,
- And sampling pipes are provided to sample the solution.

2.2.1. Bulk Volume Measurement

The Dip Tube Technique is the standard method for measuring the volume of highly radioactive solutions. The three tubes T_1 , T_2 and T_3 are flushed with a slight stream of air (Figure 4). Differential manometers measure the difference of the air pressure, P_O , in tubes T_1 and T_2 , respectively equal to:

$$\Delta P_O = \rho H_O g \quad (9)$$

$$\Delta P = \rho H g \quad (10)$$

where ρ is the density of the solution
 g the acceleration of the gravitation field
 H_O the level differences between the tips of tubes T_1 and T_2
 H the level difference between the tip of tube T_1 and the free surface of the solution.

As H_O is known, it is possible to calculate

$$H = H_O \frac{\Delta P}{\Delta P_O} \quad (11)$$

and

$$\rho = \frac{\Delta P_O}{H_O g} \quad (12)$$

The temperature of the solution in the tank is recorded.

The height, H , is related to the volume of solution, V , held in the tank via a tank calibration curve (Figure 5), after appropriate corrections for thermal expansion effects.

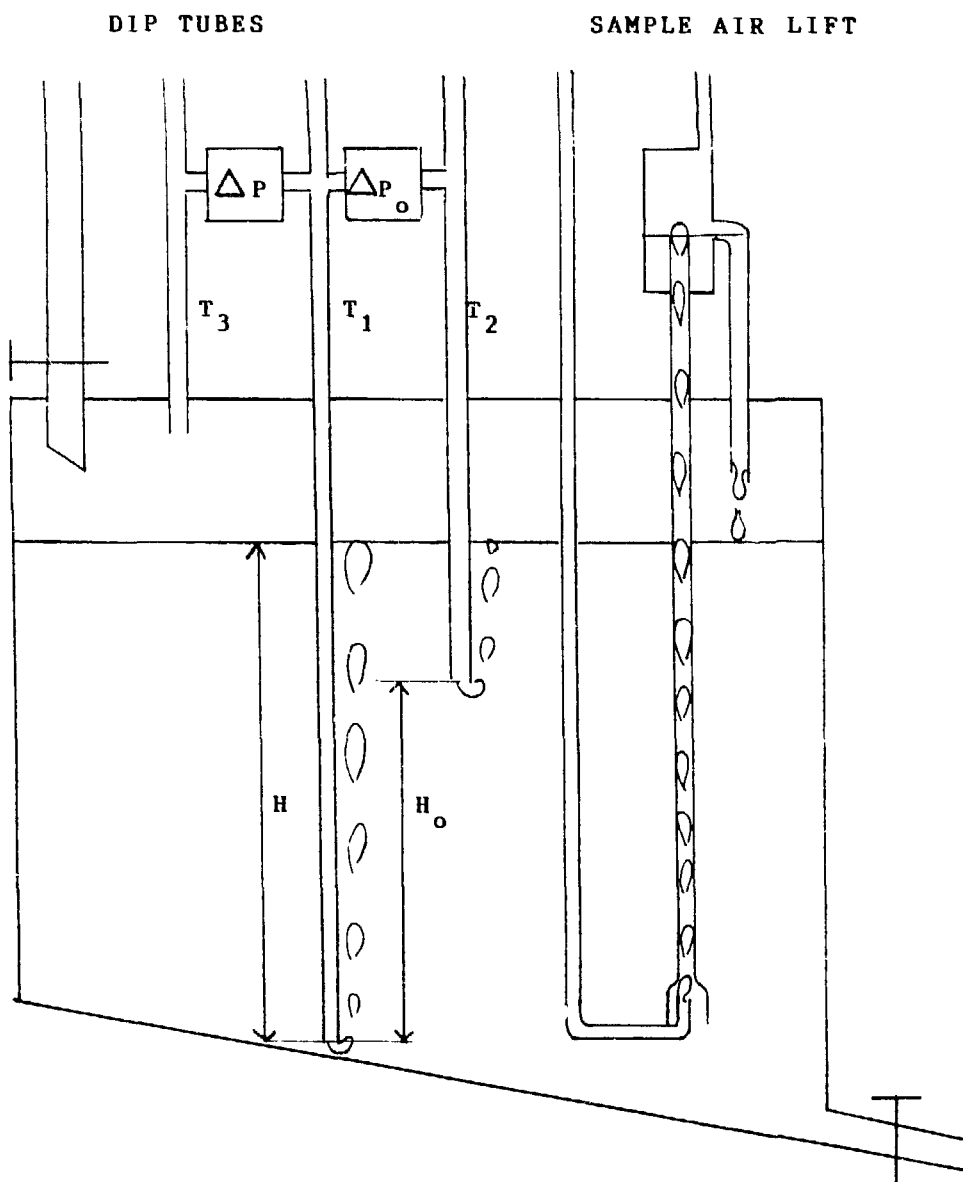


Figure 4. INPUT ACCOUNTABILITY TANK

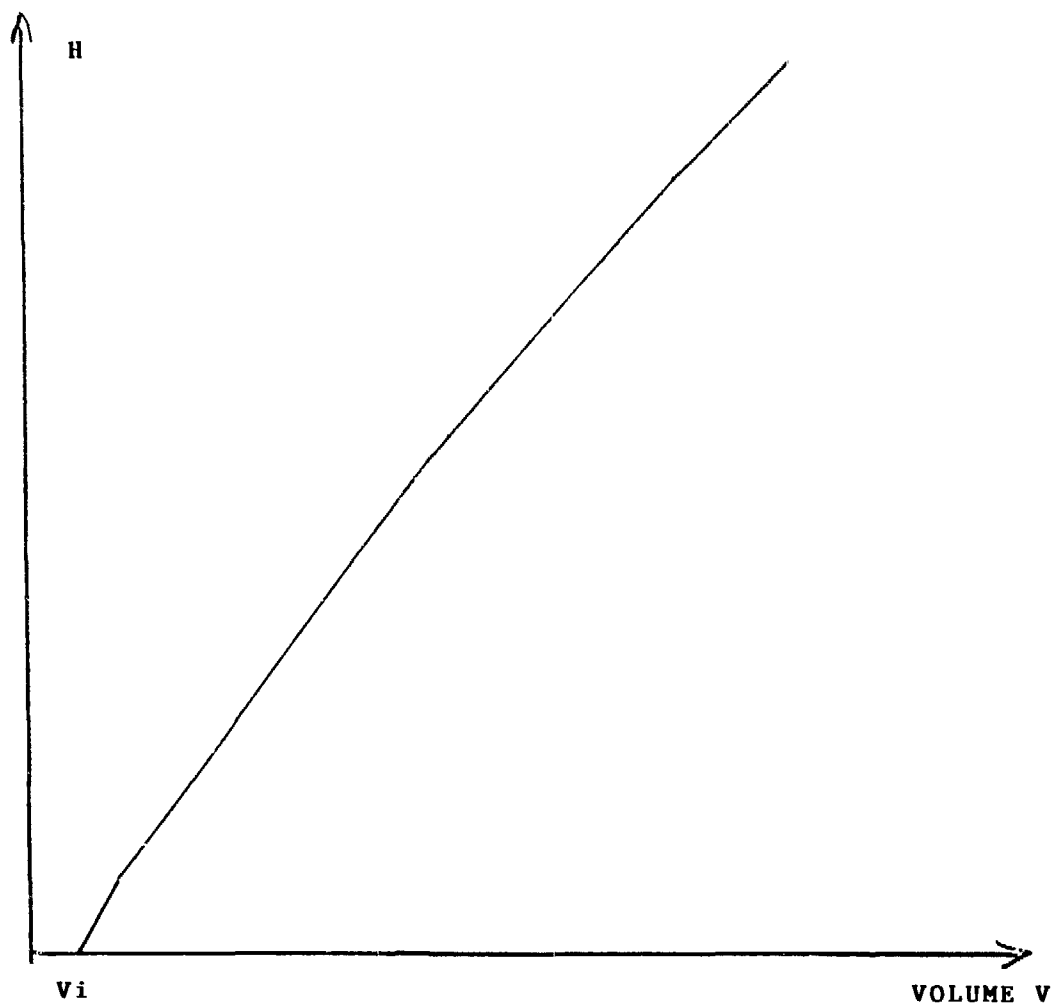


Figure 5. TANK CALIBRATION CURVE

An alternative and more direct method of measuring the content of the tank would be to weigh it. Research and development are being pursued to investigate the applicability of strain gauges for this purpose.

2.2.2. Sampling

Before a sample is to be taken, the solution must be homogenized, generally by air sparging, for a time varying between 30 minutes and several hours, depending upon the geometry of the tank and the composition of the solutions, which were introduced into it. At some plants, the spent fuel solutions are circulated in the sampling lines during the entire homogenization step.

The homogeneity of the solution is tested and the procedure is qualified by measuring the density of samples taken at successive intervals. This measurement must be made with a precision of $2 \cdot 10^{-7} \text{ kg} \cdot \text{m}^{-3}$ ($2 \times 10^{-10} \text{ g} \cdot \text{cm}^{-3}$) to verify that the concentrations of the solution vary no more than 0.1% within the tank.

When the solution is deemed to be homogeneous, a succession of 4 to 10 samples of 2 - 4 ml are drawn and transferred via a pneumatic tube to a hot cell of the high-activity analytical laboratory (Figure 6). The first sample is usually discarded. The other ones are used for the measurement of the density and for accountability and process control analyses. One or two are saved a few days so that measurements may be repeated if necessary.

2.2.3. Dilution

In an analytical hot cell (behind 1 meter of concrete or 15 cm of lead, typically), an aliquot of 1 ml is accurately pipetted with a calibrated pipet. The aliquot is diluted with 1 - 3M nitric acid by a factor 150 - 2000 using calibrated volumetric glassware.

Alternatively the aliquot of concentrated solution may be weighed on an electromagnetic analytical balance to $\pm 1 \text{ mg}$. In this case, the dilution factor is obtained by weighing the diluted solution.

Generally a duplicate sample is taken and diluted in parallel as shown in Figure 3.

2.2.4. Subsampling

The diluted solution, which contains 0.1 to 2 mg/ml of uranium and 1 to 15 $\mu\text{g/ml}$ of plutonium when analyzing light water reactor fuel, shaken or stirred to homogenize it. An aliquot of 5 - 20 ml is taken and transferred to a shielded glove box of the medium activity laboratory. A 5 cm lead shielding is usually sufficient at this stage.

According to Figure 3, one to three subsamples of 0.2 to 5 ml are drawn from each diluted solutions.

One subsample of one or the other diluted solution is used for the isotopic analysis of the fissile elements.

At least one subsample of each of the diluted solutions is measured accurately by volume or weighing and serves for the isotope dilution assay of the concentrations of uranium and plutonium.

2.2.5. Spiking

Separate calibrated solutions of ^{233}U and ^{242}Pu may be prepared. Many laboratories find it more practical to prepare and calibrate a single solution containing both tracers together. There are accordingly three modes of spiking and of performing the isotope dilution assays:

- (a) if the ^{233}U and ^{242}Pu tracers are prepared in separate solutions, one may spike one aliquot of the diluted solution with the ^{233}U tracer and a second one with the ^{242}Pu tracer, to perform the assays of uranium and plutonium on separate aliquots;
- (b) one may spike the same aliquot of diluted solution successively with a measured aliquot of both tracer solutions to perform both uranium and plutonium measurements on the same aliquot;
- (c) or the latter may also be done by spiking a sample aliquot with a single measured aliquot of a mixed tracer solution, as proposed on Figure 3.

The tracers are prepared in 3 - 6M nitric acid solutions to ensure good conservation and to avoid polymerization of plutonium species and their plating out on the walls of the storage vessel. Precautions are to be taken to guard also against evaporation of the tracer solutions on storage. If the titer of the tracer solutions varies in an uncontrolled manner during storage, the analyses will be biased unless the tracer is constantly recalibrated.

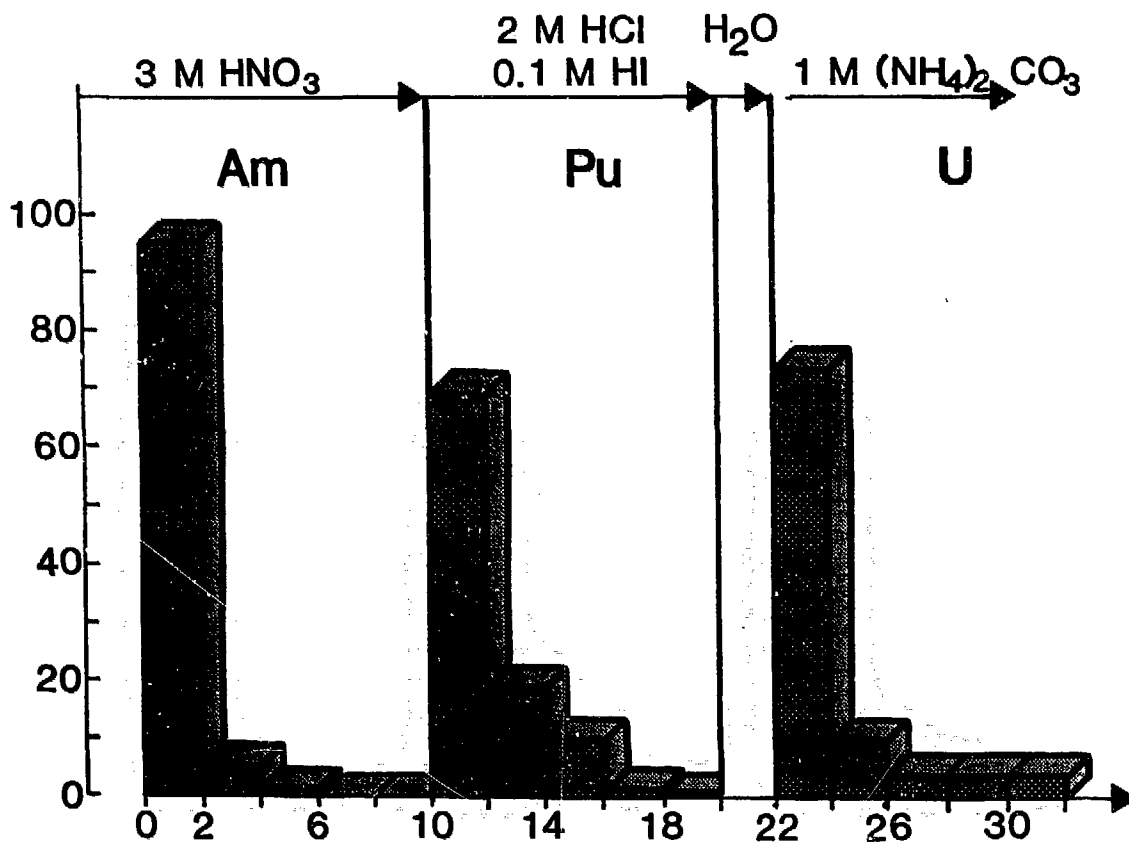
It is usually recommended to add the tracer isotopes in quantities equivalent to the amount of the isotope of interest, ^{238}U , ^{235}U or ^{239}Pu , present in the sample aliquot. The tracer solutions are, therefore, prepared to contain typically 0.1 to 1 mg/ml of ^{233}U isotope, and/or 1 - 10 $\mu\text{g}/\text{ml}$ of ^{242}Pu isotope.

2.2.6. Isotopic Exchanges

Uranium and plutonium are encountered in nitric acid solutions in various solvated forms. Plutonium may be present in several oxidation states, e.g. 4, 5 and 6, and in various chemical forms in varying proportions. Such varying species of the same element can differ notably in their chemical and kinetic properties. It is imperative that the tracer isotope be brought exactly to the same chemical state as the element to be analyzed, before making any chemical separation of the spiked mixture. Isotopic exchange is readily achieved with mixtures of uranium and ^{233}U tracer by a brief heating. Sample plutonium and ^{242}Pu tracer will best reach chemical equilibrium after undergoing a valency adjustment. A common and well proven procedure consists in reducing plutonium first to the trivalent state with ferrous sulfate and then reoxidizing it to the tetravalent state with sodium nitrite.



Figure 6. HOT CELLS IN HIGH ACTIVITY ANALYTICAL LABORATORY



Elution curves of Am, U, Pu using a TOPO/microthene (0.116 g/g) in the form of a dry powder

Figure 7

2.2.7. Chemical Separation

The purpose of the chemical separation, which is carried out next, is:

- to eliminate the fission products,
- and to separate uranium and plutonium from isotopes and species that may interfere in the analysis.

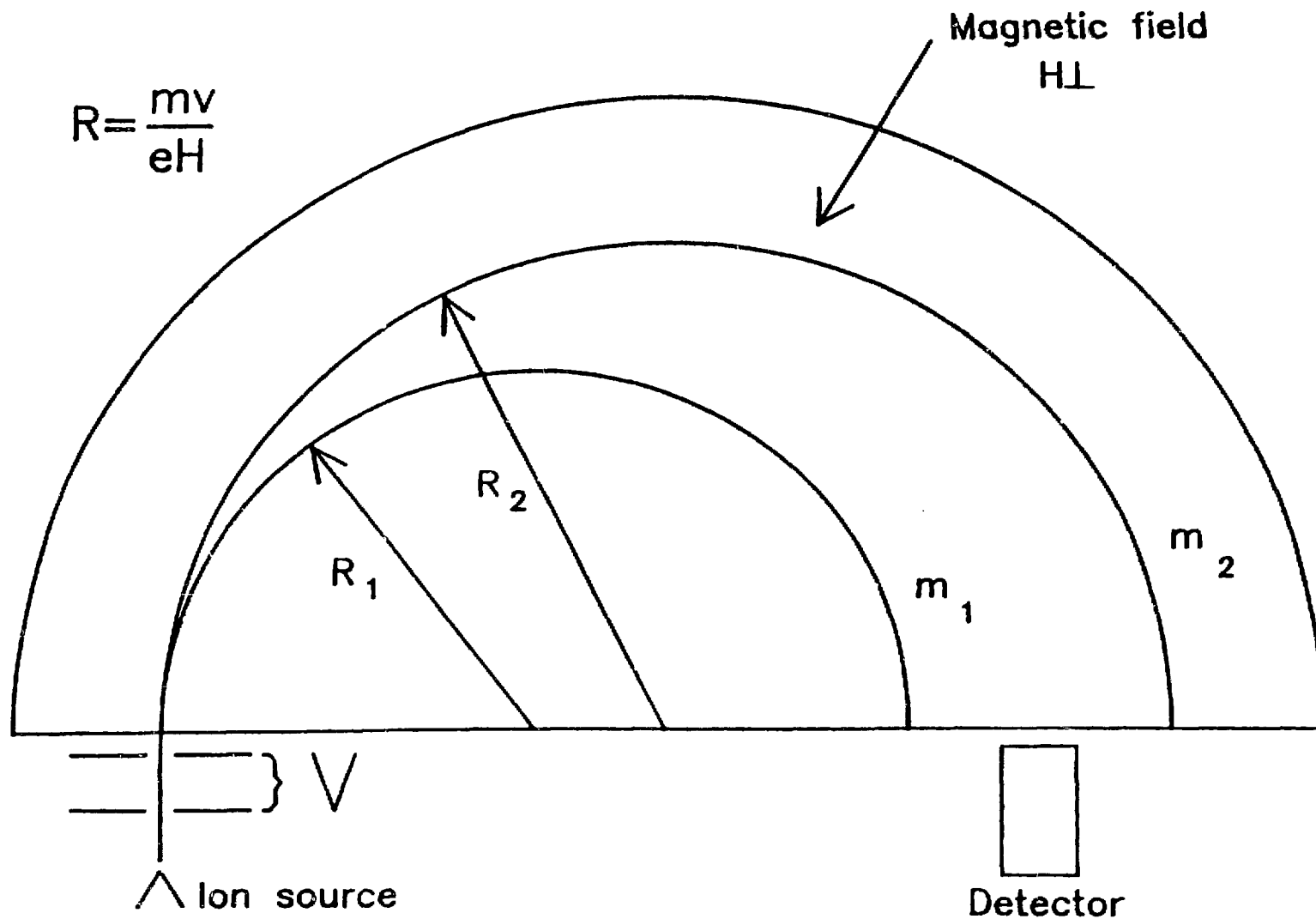
Solvent extraction, anion exchange chromatography, extraction chromatography, and capillary column chromatography are some of the popular and effective methods of separation used at this point. Figure 7 outlines an example of separation by extraction chromatography. Small beads of polyethylene (90 - 200 mesh) are coated with a uniform and thin film of trioctyl phosphineoxide (TOPO). Anionic nitrate complexes of uranium and plutonium are readily soluble in this solvent, which is almost totally immiscible in water. A small column, about 5 mm in diameter, is filled with the coated beads to a height of 5 - 6 cm. After isotopic exchange, the sample aliquot is reduced to a small volume and adjusted to be 3 M in nitric acid. It is drained slowly into the column. Plutonium and uranium are extracted into the layers of TOPO and remain fixed on the first centimeter at the top of the column. Americium, fission products, iron and sodium stay in the aqueous phase and are washed out with a pure solution of 3 M nitric acid. Plutonium is then reduced to the trivalent state with an iodide. In this state, plutonium is no longer soluble in TOPO: it passes into the aqueous phase, and can be washed out (eluted) with a mixed solution of 2 M hydrochloric acid and 0.1 M hydroiodic acid, while uranium remains fixed on the top of the column. The latter is retrieved by eluting it with water or with a solution of a complexing agent like ammonium carbonate or hydrofluoric acid.

Out of each initial sample aliquot, one collects a pure fraction of plutonium and a pure fraction of uranium. Both are practically free from fission products and can be transferred to the low activity analytical laboratory where the mass spectrometers and other analytical instruments are installed.

2.2.8. Mass Spectrometry

The decontaminated fractions are analyzed with thermal ionization mass spectrometers.

A drop of the decontaminated solution containing 0.01 to 10 µg of uranium or plutonium is evaporated on the middle of a tungsten, tantalum or rhenium filament, which is then mounted inside the ionization source of the spectrometer. The source is tightly closed and placed under a vacuum of 10^{-7} Bar or less. The filament is heated to vaporize the sample, which decomposes into its elemental atoms. U^+ , or Pu^+ ions are formed, accelerated by a strong electrostatic field and focused into a thin beam, which is injected into an evacuated tube called a fly tube. The fly tube is placed in the gap of a magnet, which deflects the ions onto circular paths. The curvature of the path, which is a function of the mass of the ion m , is altered by changing the accelerating electrical field V , or the deflecting magnetic field H , such that the ion of interest comes to hit a detector (Figure 8). The detector collects the ions of the isotopes of interest, ^{233}U , ^{234}U , ^{235}U , ^{236}U , and ^{238}U for



Principle of the Mass Spectrometer
Mass Spectrometric Measurement of Isotopic Abundance

Figure 8

uranium or ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , and ^{244}Pu for plutonium. The ion current is measured at each "mass position". It is proportional to the abundance of the corresponding isotope. One records the ratio of the ion current of a given isotope to that of a more abundant isotope, $^{234}\text{U}/^{235}\text{U}$, $^{235}\text{U}/^{238}\text{U}$, or $^{240}\text{Pu}/^{239}\text{Pu}$.

2.2.9. Calibration of the Mass Spectrometer

The mass spectrometer must be calibrated by analyzing certified reference materials in order to relate a measured ion ratio (R_{Obs}) to the actual isotope ratio (R)

$$R = K \times R_{\text{Obs}} \quad (13)$$

2.2.10 Calibration of the Tracers

Certified tracers may be purchased from organizations like the National Bureau of Standards (NBS) and the New Brunswick Laboratory (NBL) in the USA, or the Central Bureau for Nuclear Measurements (CBNM) of the European Communities. Most laboratories still prepare and calibrate their tracer solutions.

The isotopic composition of the tracer is measured by mass spectrometry. It is necessary to define the abundance of the tracer isotope (e.g. A_3) and the ratio R_T used in equation (6).

The concentration C_T of the tracer solution is measured by isotope dilution mass spectrometry after mixing a volume V'_T of the tracer with a volume V_{ST} of a solution of a chemical standard of natural or depleted uranium of concentration C_{ST} , abundance $A_8(ST)$, and $R_{ST} = ^{233}\text{U}/^{238}\text{U} = 0$. The isotope ratio of the mixture ($R'_M = ^{233}\text{U}/^{238}\text{U}$) is measured. The concentration of the tracer is calculated as follows:

$$C_T = \frac{C_{ST} \cdot V_{ST}}{V'_T} \cdot \frac{A(233)}{A(238)} \cdot \frac{A_8(ST)}{A_3} \cdot \frac{R'_M - R_{ST}}{1 - \frac{R'_M}{R_T}} \quad (14)$$

The ^{242}Pu tracer may similarly be calibrated against a solution of enriched ^{239}Pu chemical standard.

It is advisable to confirm these results by measurements done with another independent method of equivalent or better accuracy, like chemical titration or coulometry.

2.3. Safeguards Verification Procedures

Isotopic dilution mass spectrometry is the basic method of verification by safeguards authorities of the input accountability analyses.

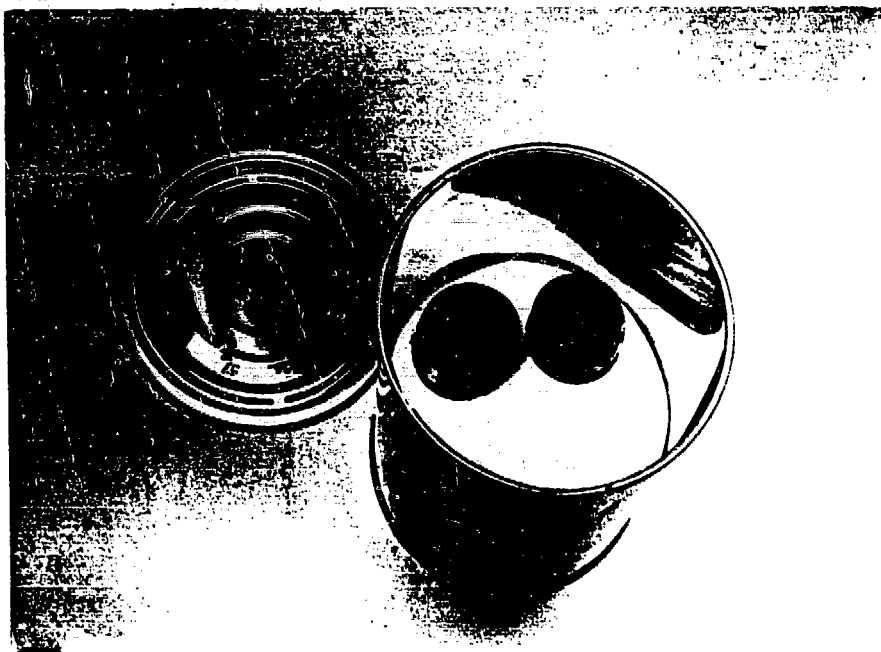


Figure 9. SAMPLE BOTTLES OF THE DRIED SPIKE TECHNIQUE

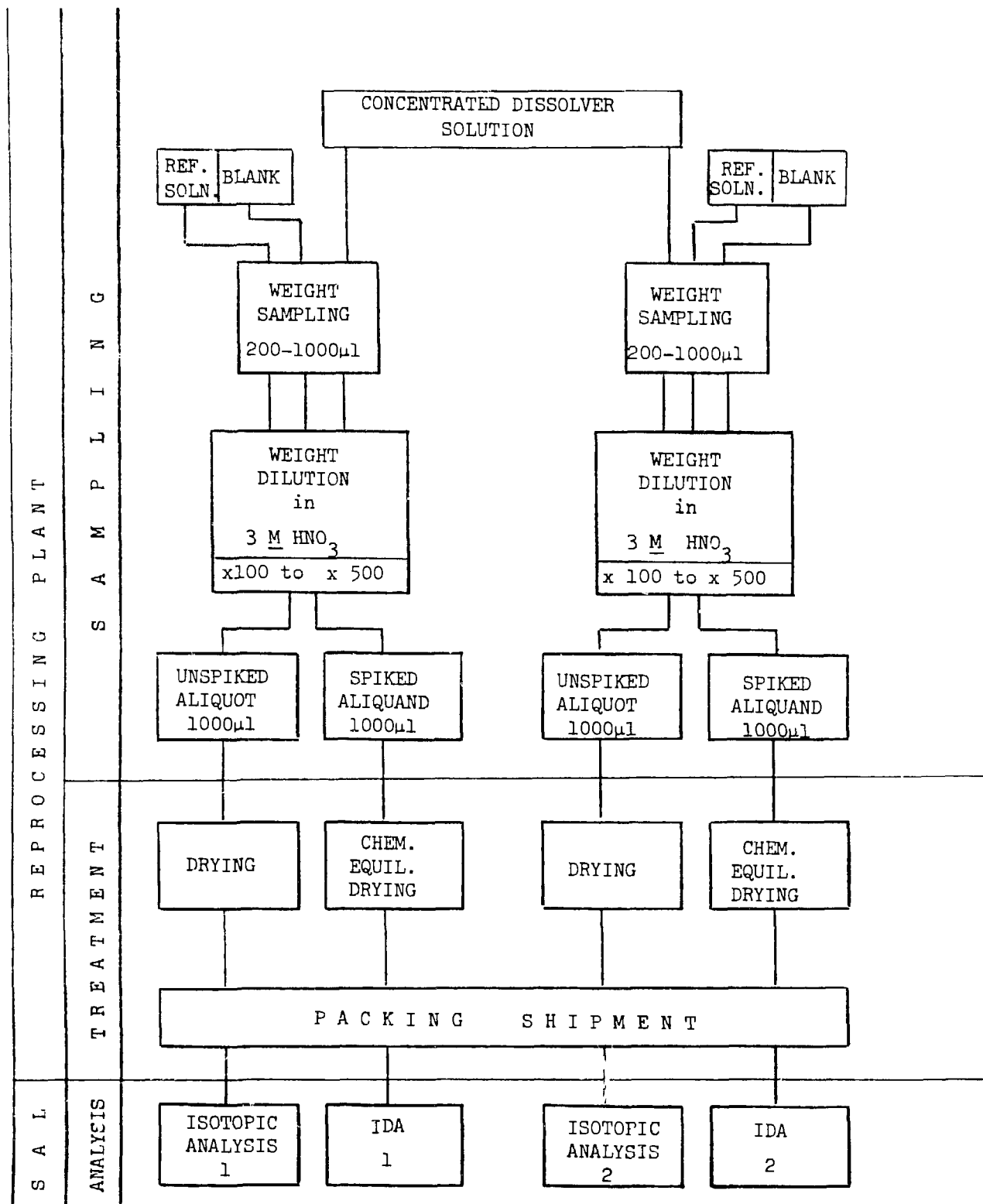


Figure 10. OUTLINE OF THE DRIED SPIKE TECHNIQUE

Samples of concentrated dissolver solutions are too radioactive and unstable to be shipped over long distances. We describe now briefly three techniques, which are or may be used to prepare safeguards inspection samples, so that they can be shipped practically and remain representative of the original dissolver solution even over long shipment times and distances.

2.3.1. Dried Spike Technique

In the dried spike technique, the verification laboratory provides the inspector with two pairs of sampling bottles (Figure 9). Each pair consists of one empty bottle and a second bottle carrying a well known quantity of ^{233}U and ^{242}Pu tracers. The latter has been prepared by introducing into the bottle an accurately weighed aliquot of a calibrated tracer solution and evaporating it cautiously to dryness. The inspector requests the operator to put about 1 ml of one of the diluted solutions prepared for accountability analyses into the empty bottle. A second aliquot, also about 1 ml, is accurately measured and introduced into the bottle carrying the dried spike. Both bottles are heated carefully to avoid projections and evaporated to dryness. In the course of this operation, the spike redissolves and isotopic exchange is partially achieved before the sample is dried again. The second pair is used to obtain duplicate samples preferably on a duplicate dilution of the dissolver solution (Figure 10). The dried samples containing about 10 mCi of fission products are air freighted to the verification laboratory. The unspiked samples are used there for the isotopic analysis, while the concentrations of uranium and plutonium are measured by isotope dilution assay of the spiked samples.

2.3.2. The Resin Bead Technique

The resin bead technique initially follows the same outline as the dried spike technique. However, after heating the samples to redissolve the spikes, the samples are not evaporated but are subjected to a valency adjustment of the plutonium with ferrous sulfate and nitrite. Then concentrated nitric acid is added to obtain about 4 ml of a solution in 8 M nitric acid, which is stirred vigorously for 5 - 10 minutes with about 1000 beads of an anion exchange resin (50 - 100 mesh). Each bead absorbs about 1 - 10 ng of uranium and plutonium, while the fission products and americium stay in the solution. About 50 beads are recovered by filtering 0.5 ml of the bead suspension on a small fritted crucible (Figure 11). The frits are sealed in plastic bags and tin cans. A one liter can containing the resin bead samples of up to 10 batches of dissolver may be air-mailed to the verification laboratory.

At the laboratory, a single bead loaded with an unspiked sample is mounted on a filament of the mass spectrometer and analyzed sequentially for plutonium and uranium isotopic composition. Another resin bead loaded with a spiked sample is mounted on a second filament to measure the element concentrations.

Thus, two minute resin beads, carrying only 10 ng of fissile elements each, can provide a complete verification of half a ton of spent nuclear fuel (Figure 12).

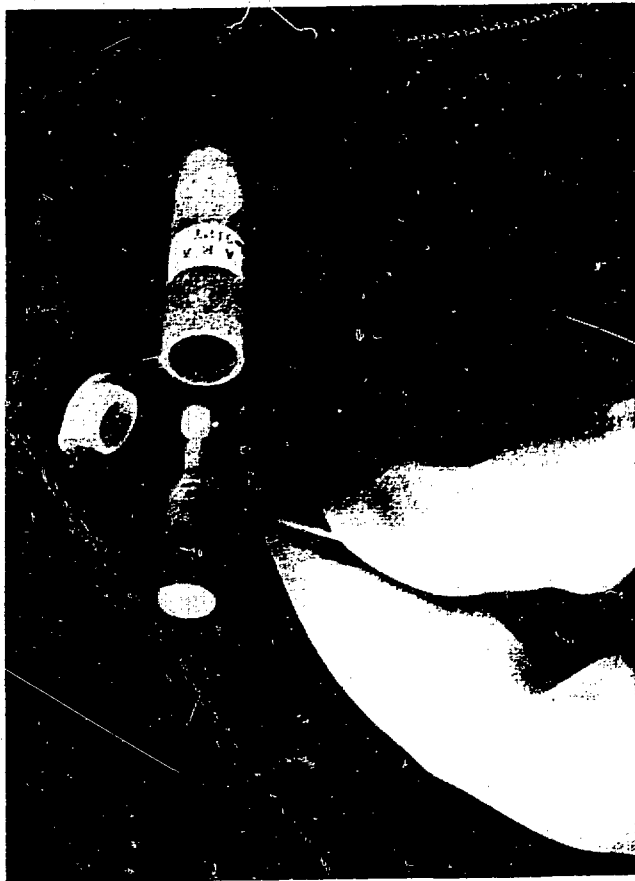


Figure 11. SAMPLE VIALS FOR THE RESIN BEAD TECHNIQUE

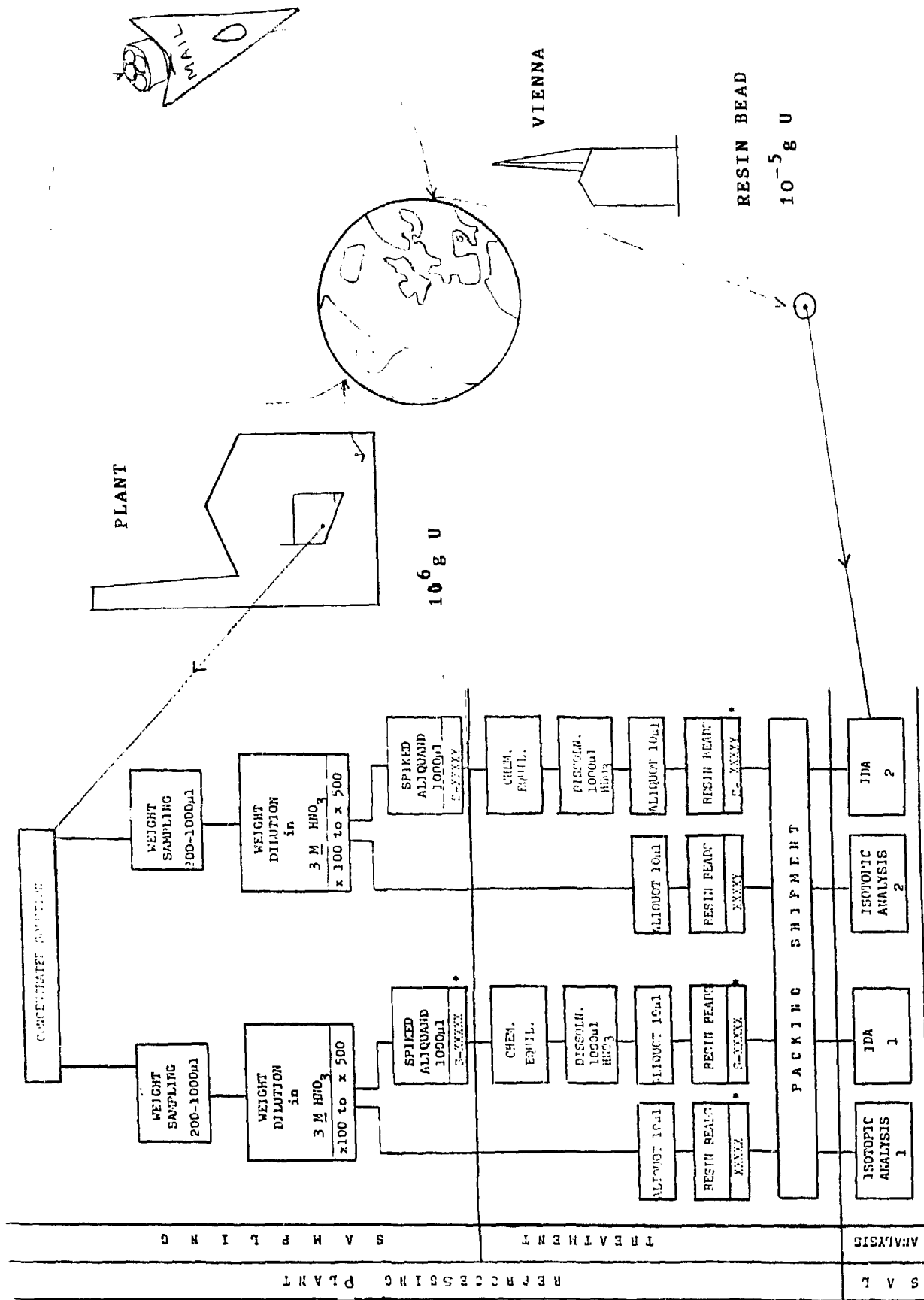


Figure 12. OUTLINE OF THE RESIN BEAD TECHNIQUE

Summary of the results of 2 PNC-IAEA resin bead experiments. Mean relative differences and standard deviations (%).

SAMPLE	MEASUREMENT	EXPERIMENT	
		No. 2	No. 3
UNSPIKED	U-234/U-238	7.6 (1)	+ 0.93 ± 5.5 (8)
	U-235/U-238	-0.35 (1)	- 0.38 ± 0.81 (9)
	U-236/U-238	-1.2	- 0.64 ± 0.88 (9)
	Pu-238/Pu-239	-7.3 (1)	+ 10 ± 7.2 (9)
	Pu-240/Pu-239	-0.10 (1)	+ 0.47 ± 0.22 (9)
	Pu-241/Pu-239	+ 0.97 (1)	+ 1.02 ± 0.53 (9)
	Pu-242/Pu-239	+ 0.12 (1)	+ 0.79 ± 1.32 (9)
SPIKED	U-233/U238	- 0.02 ± 0.52 (5)	- 0.52 ± 0.44 (9)
	U-conc.	- 0.01 ± 0.52 (5)	+ 0.49 ± 0.45 (9)
	Pu-242/Pu-239	+ 0.16 ± 0.34 (5)	± 0.36 ± 0.61 (9)
	Pu-conc.	- 0.60 ± 0.36 (5)	+ 0.11 ± 0.65 (9)
	Pu-239 conc.	- 0.54 ± 0.47 (5)	- 0.07 ± 0.61 (9)

The figures in parentheses are the number of samples measured.

2.3.3. The Metal Spike Technique

In both methods just described, the verification remains incomplete until the dilution step is also verified. This drawback would disappear if samples of the concentrated spent fuel solution could be spiked before dilution. The CBNM has proposed and tested a practical procedure to do this. The tracer is prepared as a very homogeneous metal alloy of uranium and plutonium by melting weighed portions of the pure metals in an inert atmosphere in a levitation furnace. The alloy is then divided into a number of pieces that are individually weighed and sealed in quartz ampoules. Each piece weighs approximately 200 mg and contains about 1 - 3 mg of plutonium.

In the plant, two ampoules are put into the hot cell where sample aliquots of concentrated spent fuel solution are taken and measured. An ampoule is opened and the metal spike is placed in a dissolution vessel that has received an accurately measured aliquot of the concentrated spent fuel solution. A duplicate sample aliquot is spiked with the second piece of metal spike. The metal spikes are carefully dissolved with nitric acid and a trace of hydrofluoric acid. The success of the procedure relies on a totally quantitative dissolution of the metal spikes.

Because the alloy is a strong reducer, the plutonium of the sample is reduced to the trivalent state during the dissolution, while the plutonium of the spike dissolves and passes also into solution in the trivalent state. When the spike is completely dissolved, further heating of the nitric acid solution oxidizes both sample and spike plutonium to the tetravalent state. The uranium sample and spike pass simultaneously through the valence states (III) and (IV) to yield hexavalent uranium. The dissolution yields a very efficient isotopic exchange and achieves complete isotopic equilibrium between the sample and the spike. Any further dilution and treatment will not alter the isotopic composition of the mixture.

The "nec plus ultra" is then to aliquot about 10 - 20 μ l of the resulting solution into 4 ml of 8 M nitric acid and to load anion exchange resin beads (Figure 13).

A disadvantage of the technique is that it consumes relatively large amounts of tracers. This is easily overcome for the uranium assay by using highly enriched ^{235}U metal tracer, which is available in very pure form and large quantities. When high burn-up fuel is analyzed, highly enriched ^{239}Pu metal, which is also readily available in the form of a primary chemical reference material, is a perfectly valid tracer to determine the concentration of plutonium, which contains only 50 - 60% of ^{239}Pu isotope.

This approach will be tested by the Agency in collaboration with the European Communities and the Federal Republic of Germany.

2.4. Performance and Error Factors

Tables 1 and 2 summarize various estimates of the performance of isotopic and isotope dilution assay that may be expected in the analysis of spent nu-

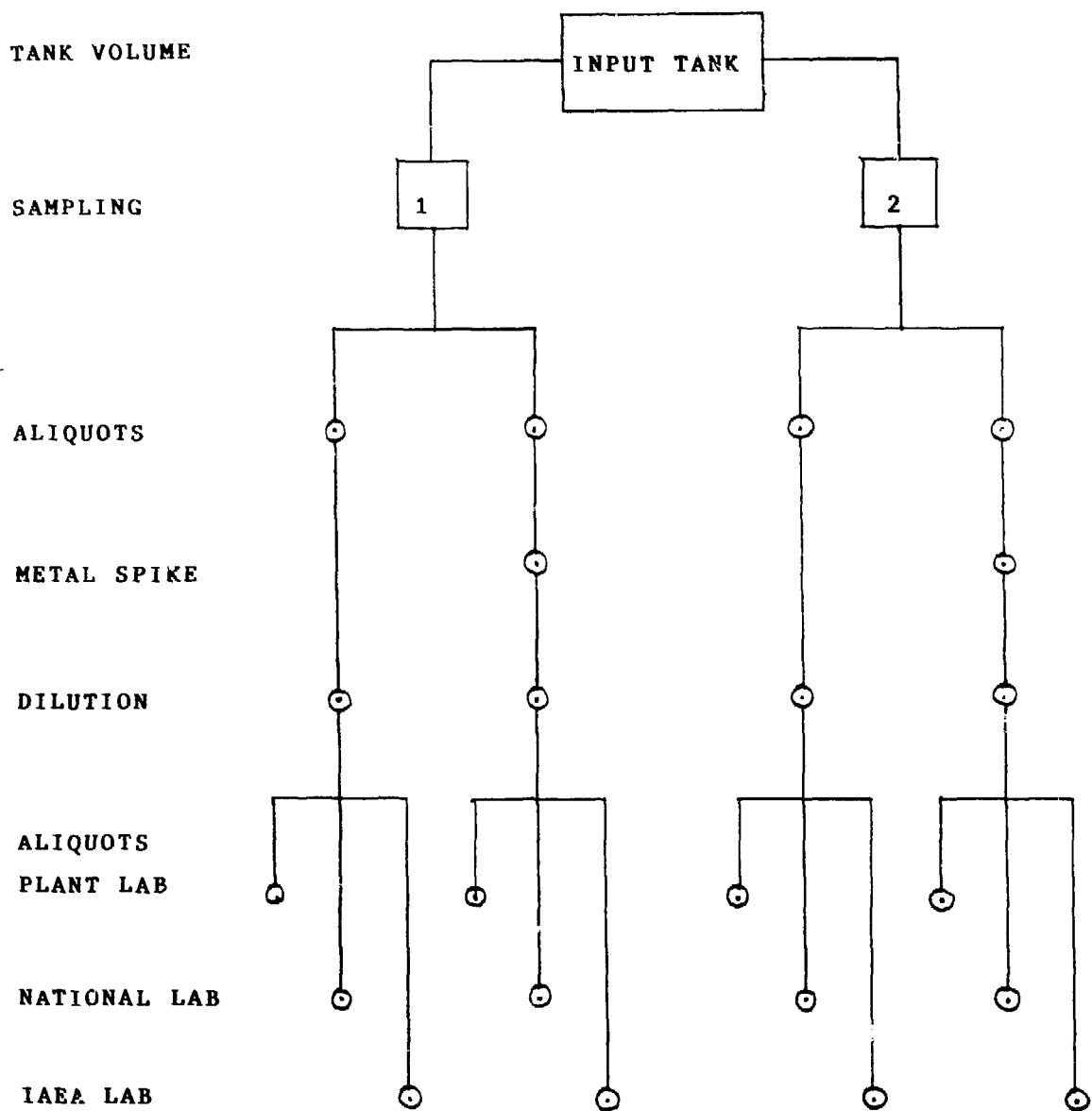


Figure 13. OUTLINE OF THE USE OF THE METAL SPIKE AS A COMMON SPIKE

clear fuel solutions. The random error is expressed by the coefficient of variation (CV) of individual replicate measurements performed at the same time and under the same conditions. The coefficients of variation of the systematic errors are calculated by a statistical evaluation (analysis of variance) of the reproducibility of replicate measurements done at different times and under different experimental conditions.

The state of the art expresses the demonstrated capability of internationally recognized certification laboratories like CBNM and NBS.

The target values are estimates of the coefficient of variation of differences between the analyses of one operator and one verification laboratory, which are believed to be achievable under actual industrial and inspection conditions. They are derived from the ESARDA-1983 target values proposed for the state of practice of a single laboratory.

Table 1

Performance of Isotopic and Isotope Dilution Analyses (IDA). State of the Art, Coefficients of Variations.

Measurement	Isotope or Element	Random Errors (%)	Systematic Errors (%)
Uranium Isotopic Composition	^{234}U	1	0.4
	^{235}U	0.07	0.03
	^{236}U	0.8	0.3
	^{238}U	0.002	0.001
Plutonium Isotopic Composition	^{238}Pu	0.8	0.3
	^{239}Pu	0.04	0.02
	^{240}Pu	0.08	0.03
	^{241}Pu	0.15	0.05
	^{242}Pu	0.25	0.08
IDA	U	0.1	0.03
	Pu	0.25	0.08

Table 2

Isotopic and Isotope Dilution Analyses of Spent Nuclear Fuel Solutions. Differences between Operator's and Inspector's Data.

Measurement	$\bar{d}$ (%)	SD (%)	SYS (%)
^{235}U abundance	0.17	0.34 (1.1)	0.53 (1.4)
U concentration	-0.34	1.50 (0.71)	
Pu concentration	-1.1	1.53 (0.71)	1.26 (0.71)

$\bar{d}$ mean relative difference in %

SD coefficient of variation of random errors of relative differences

SYS coefficient of variation of systematic errors of relative differences

The data in parentheses are derived from the 1983 ESARDA target values.

The performance values are the coefficients of variation of differences observed in the course of Agency inspections in 1982. They are derived from the statistical evaluation of paired comparisons between declared data and the results of inspectors' verification measurements (Figure 14). These values approach the target values except in isotope dilution assay, where progress is still needed. Performance is, of course, much worse than what would be ultimately achievable under state of the art conditions. Several factors contribute to this situation.

The measurement errors of mass spectrometric analyses depend upon the ability to control very precisely the temperature of the filaments, the amount of sample deposited on a filament, the nature of the chemical species produced on the filament and its chemical purity. If these parameters are not controlled, random and systematic errors of 0.5% CV will occur in the measurements of major isotope ratios. A CV of 0.05% can be obtained with very careful control of all parameters that influence isotope fractionation effects within the spectrometer (Figure 15).

Chemical components of the reagents used in the chemical separations may accompany the sample and determine part of these fractionation effects. It is advisable to submit the isotopic reference materials used in the calibration of the measurements to the same chemical treatment as the samples.

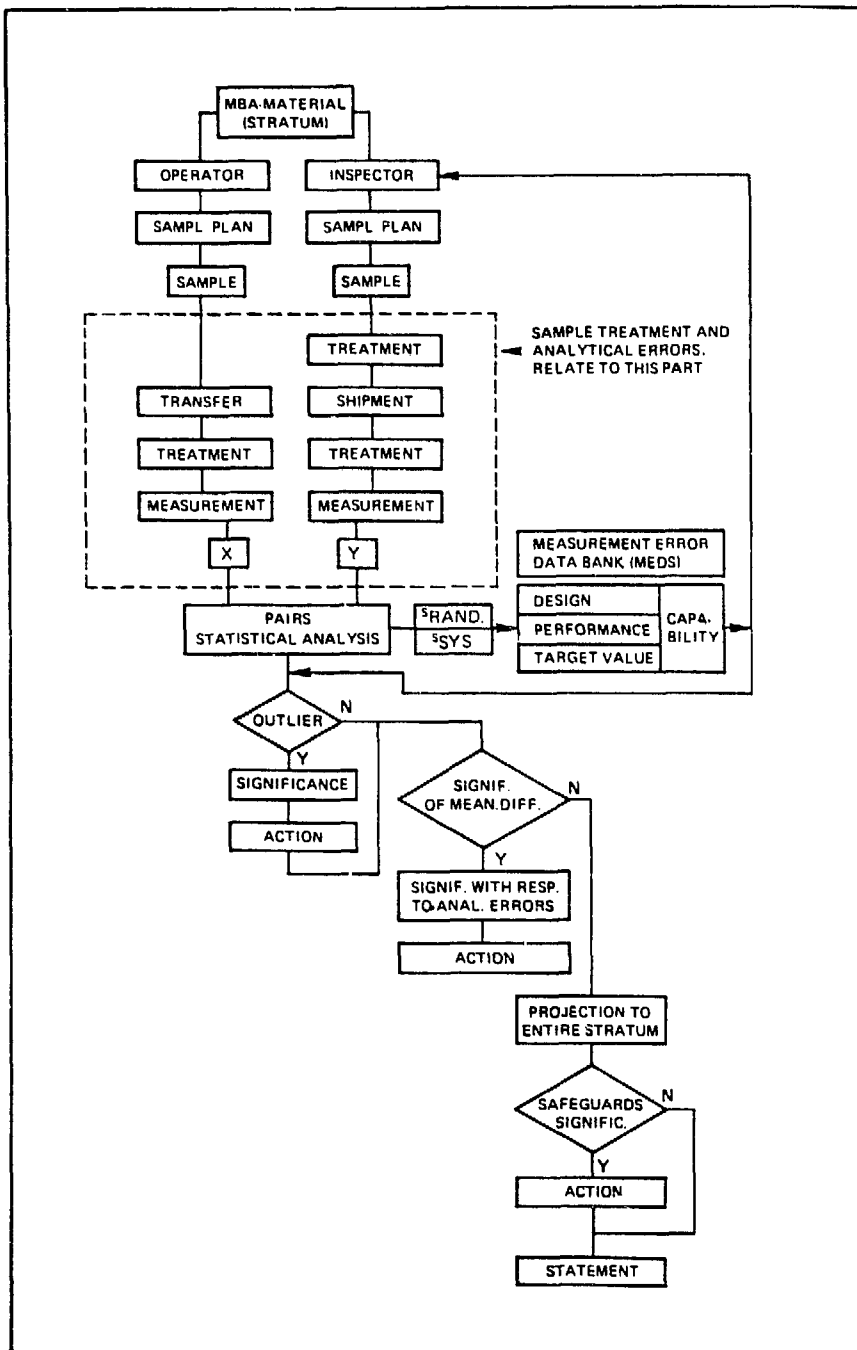


Figure 14. EVALUATION OF OPERATOR INSPECTOR DIFFERENCES

In isotope dilution assay, incomplete mixing of the sample and tracer or lack of isotopic exchange can be the source of large random and systematic errors (Figure 16). In routine operations, it is difficult to maintain sample and tracer aliquoting error below 0.1 - 0.2%.

Tracer calibration errors may have coefficients of variation of 0.1 - 0.3% and constitute a typical source of systematic errors.

Undetected transcription and calculation or rounding errors should also not be underestimated.

The performance values quoted in Tables 1 and 2 do not include dilution factor errors and sampling errors, which may both have CVs of 0.1 - 0.15%.

The magnitudes of these various error factors and the effect of their propagation are summarized in Table 3.

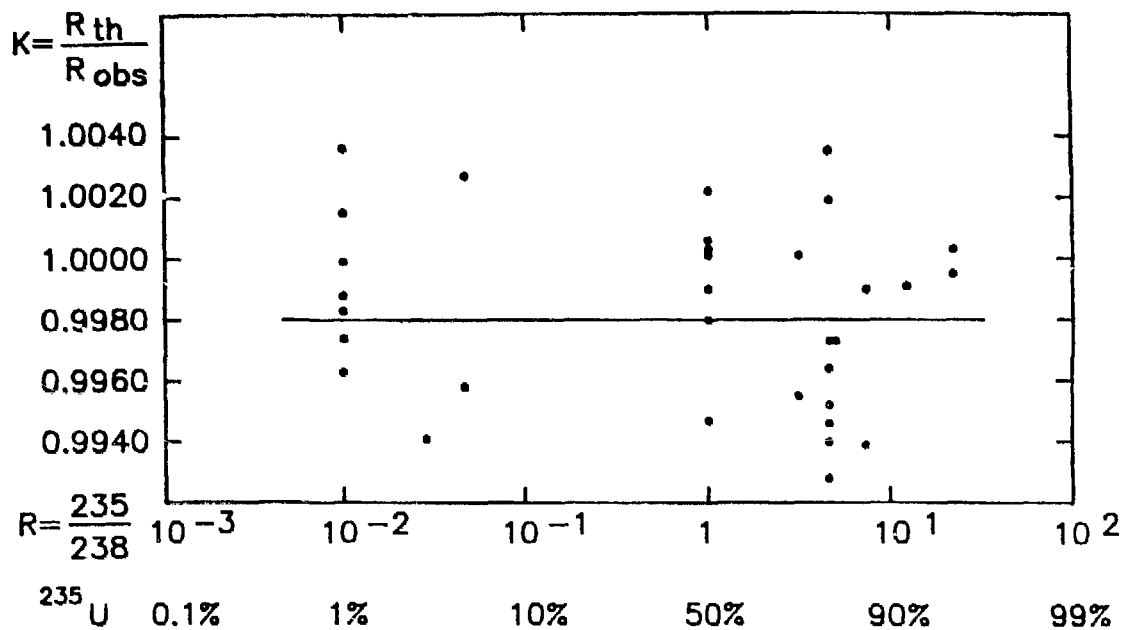
2.5. Use of Isotopic Tracers for the Direct Verification of the Volume and Content of Active Tanks

Tracer techniques have been used for the verification of the volume of spent nuclear solutions in active tanks. An element that is normally absent like Li or Lu or that may be present only in very low concentrations in the spent fuel solutions (e.g. Mg, Mn, Pb, Nd, etc...) is selected as a tracer. An accurately measured quantity of the tracer M_{TR} is added to the tank and carefully homogenized. A sample of the homogenized solution is drawn to measure the concentration of the tracer C_{TR} . The volume V of the solution in the tank is readily calculated:

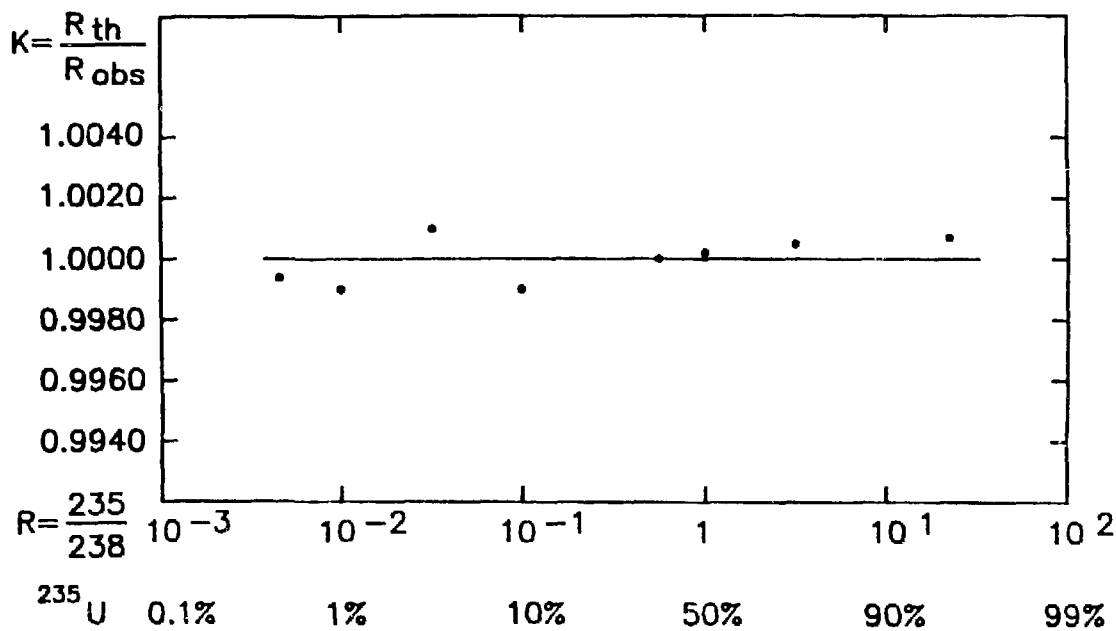
$$V = \frac{M_{TR}}{C_{TR}} \quad (15)$$

The concentration of the tracer C_{TR} is normally measured by isotope dilution analysis with an enriched isotope of the tracer element. In such a case, for example when ^{178}Lu is added as a tracer, the sample may be spiked with a calibrated mixture of ^{233}U , ^{242}Pu and ^{176}Lu to determine simultaneously the concentrations of uranium $C_S(\text{U})$, plutonium $C_S(\text{Pu})$, and lutetium ^{178}Lu C_{TR} , and also the volume of the solution in the tank V as well as the total masses of uranium $M(\text{U})$ and plutonium $M(\text{Pu})$.

The ideal tracer element must be compatible with the process and acceptable to the operator; its isotopes of interest must be readily available; its isotopic and isotope dilution analysis by mass spectrometry must be well established, precise and accurate; and the risks of contamination of the samples must be low.



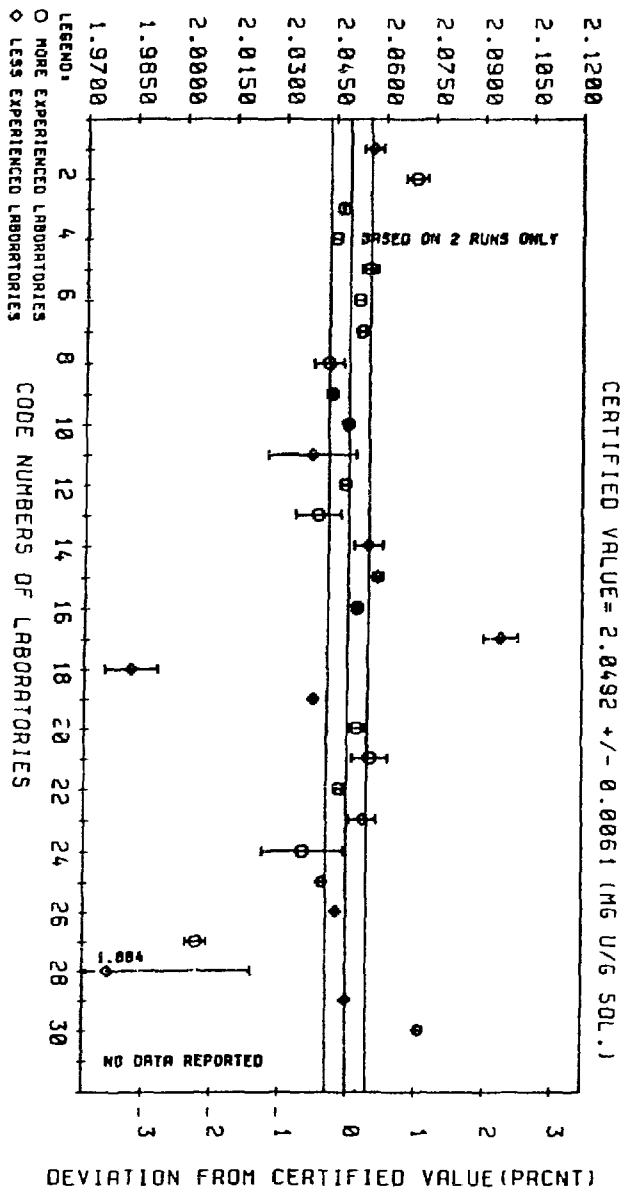
Calibration of large R range without control of thermal ionization parameters ($K=0.998 \pm 0.005$).



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

Figure 15

URANIUM CONCENTRATION OF SOLUTION B
CALCULATED FROM PART 1.11 (MG/G SOL.)



PLUTONIUM CONCENTRATION OF SOLUTION B
CALCULATED FROM PART 1.11 (UG/G SOL.)

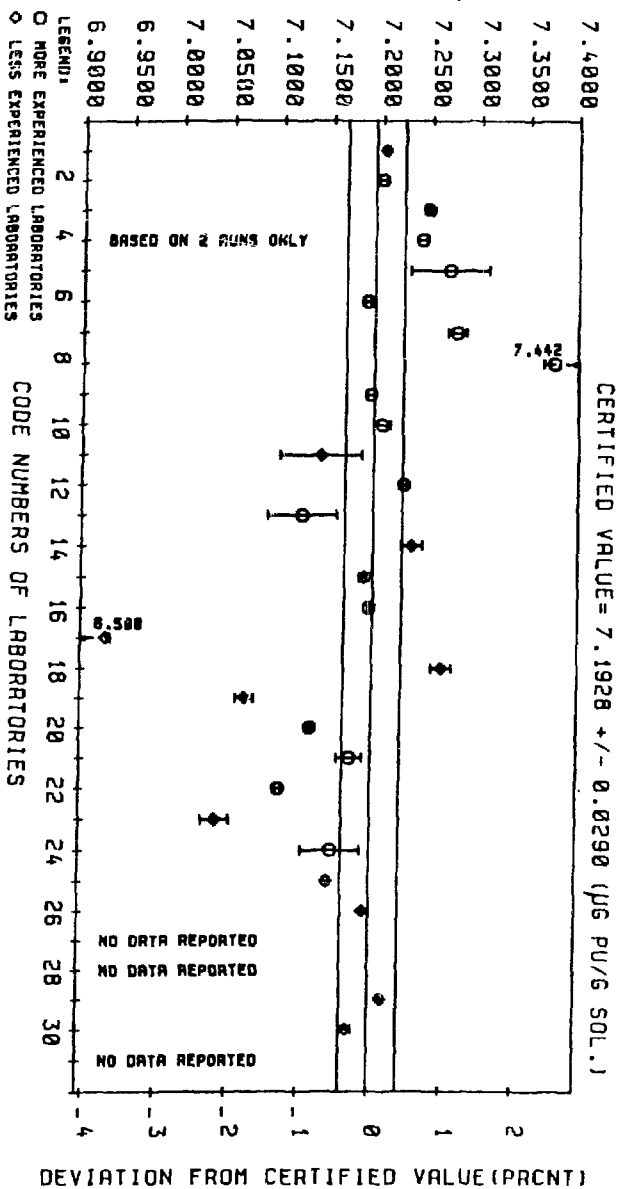


Figure 16: RESULTS OF IDA INTERLABORATORY EXERCISES ON ISOTOPE
DILUTION ANALYSIS OF SPENT FUEL SOLUTIONS

Table 3

Expected Magnitudes of the Major Error Factors in Accountability Analyses of Spent Nuclear Fuel Solutions by Isotope Dilution Mass Spectrometry (coefficient of variation, in %).

Error Factor	Component	
	Random (%)	Systematic (or calibration error) (%)
Isotope Ratio Measurement	0.3	0.1
Chemical Treatment and Isotopic Exchange	0.05	0.1
Sample and Spike Aliquoting	0.1	0.02 (a) or 0.1 (b)
Tracer Calibration	0.0	0.2
Dilution Factor	0.1	0.02 (a) or 0.1 (b)
Sampling	0.1	unknown
Bulk Volume Measurement	0.2	0.2
Calculation, Rounding and Transcription	0.05	0.1
Propagated Errors		
(i) Single Measurement	0.41	0.33 (a) 0.36 (b)
(ii) Average of 2 Measurements	0.29	0.33 (a) 0.36 (b)

(a) Sampling, spiking, dilution by volume

(b) Sampling, spiking, dilution by weight

Table 4

Results of the Rictex Experiment on Tracer Technique for Tank Calibration Verification

TRACER	$\bar{d}$ (%)	SD (%)
Lu	0.22	1.0
Pb	-0.69	0.61
Nd	0.43	0.91

$\bar{d}$ mean relative difference from reference value

SD standard deviation of relative differences

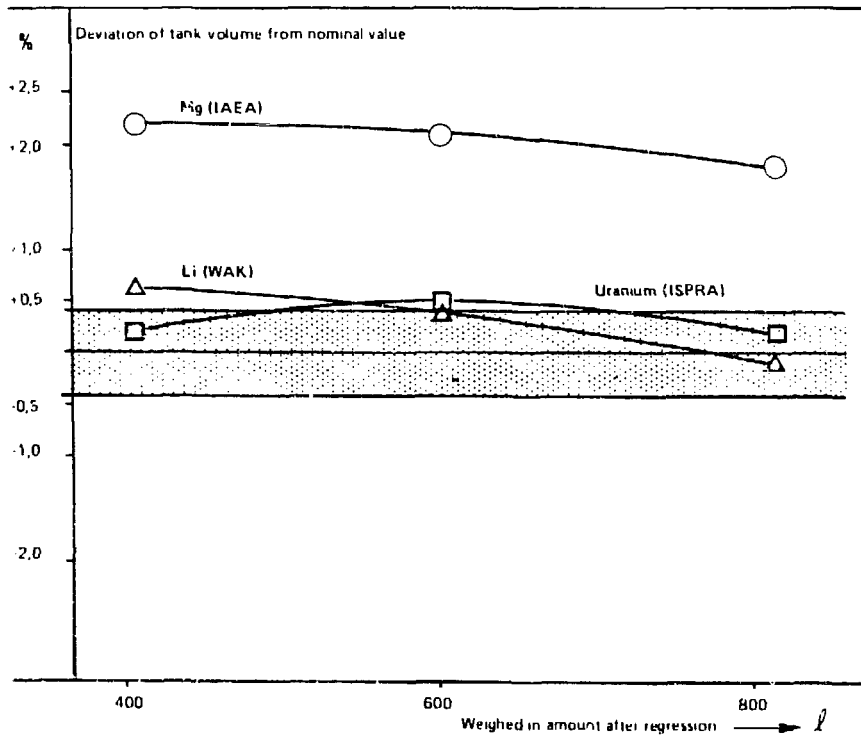


Figure 17. RESULTS OF A FIELD TEST OF TRACER TECHNIQUE FOR THE VERIFICATION OF TANK CALIBRATION AT WAK

Figure 17 and Table 4 give the results of two field experiments. The addition of the tracer in the tank and the isotope dilution measurement are the two major steps that determine the performance of the technique.

It is particularly difficult to assure that the tracer is quantitatively transferred into the tank and that it will mix uniformly in the tank. This is true with any tracer, so it is recommended that the active storage tanks of the future plants be designed to facilitate the application of tracer techniques.

Light elements like Li and Mg are difficult to analyze because their isotopes fractionate rapidly in the mass spectrometer. Samples spiked with magnesium are difficult to protect against contamination by chemicals like talc that are in common use in analytical laboratories.

On the basis of the results reported at this time, Pb, Nd and Lu appear to be the most promising tracer elements for such applications.

2.6. New Developments

Some progress can be expected in the near future as a result of recent developments in instrumentation and methods.

2.6.1. Automatic Mass Spectrometers

Fully automatic thermal ionization mass spectrometers became commercially available in 1979 and are spreading widely in the analytical laboratories. When in good operating condition, these instruments are very stable; they have an excellent sensitivity and are able to collect 1 ion for every 500 atoms loaded on the filament; and their resolution is very high, so that the fluctuations of the base line are smaller than 0.0002% of the major ion peak, one mass unit away from the major isotope position. With such characteristics and good computer software, precisions of 0.1 - 0.2% and less should be normal practice in isotope ratio measurements.

A further advance occurred in 1982-1983 when automatic instruments were equipped with a multidetector assembly, which allows simultaneous rather than sequential analysis of the isotopes of interest (Figure 18). With such a system, major isotope ratios can be measured readily with a precision of 0.05%. However, the fluctuations of the isotope fractionation effect from one filament loading to the next continue to limit the precision. It will remain difficult to calibrate such instruments to better than 0.1 - 0.2% unless appropriate certified isotope mixtures become available for all isotopes of interest. Mixtures of ^{242}Pu and ^{239}Pu , or mixtures of ^{244}Pu and ^{239}Pu are still not available as certified reference materials.

Precision thermal ionization quadrupole mass spectrometers constitute a third development that can be of interest in safeguards. These instruments are very compact and are potentially appropriate for in-field measurements.

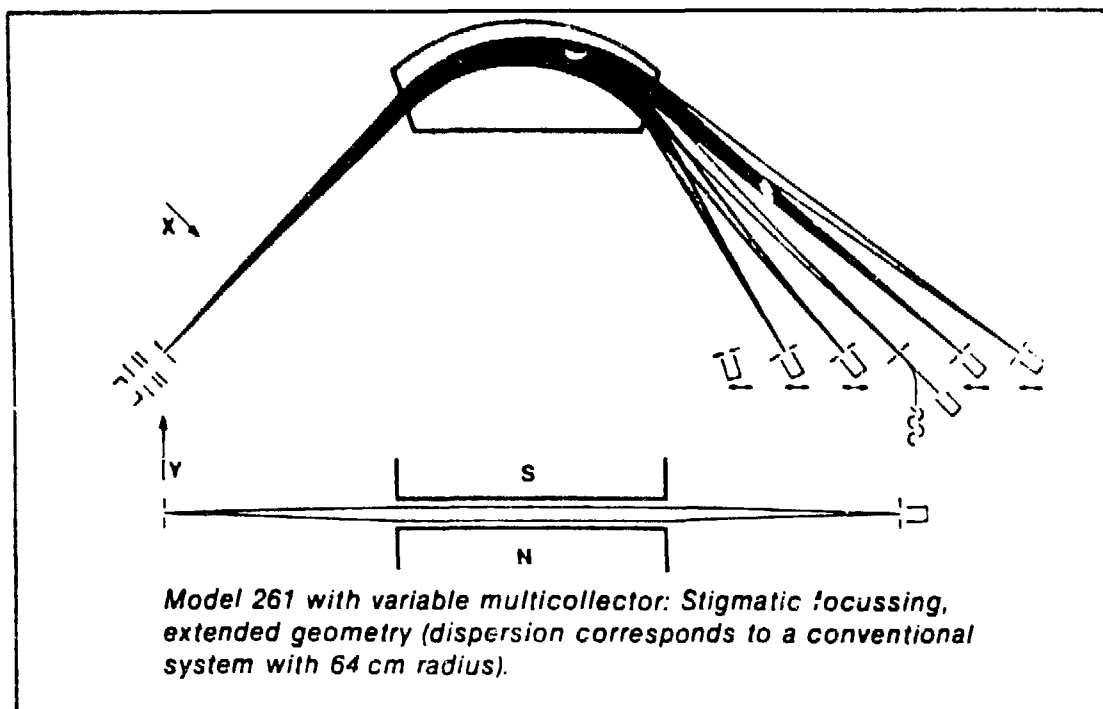


Figure 18

2.6.2. Automatic Laboratories

An even greater number of more accurate analyses will be required to ensure an adequate accountability of the fissile materials in future plants that will have annual throughputs of 800 to 2000 tons of spent nuclear fuels. The designers of such plants are developing fully automatic analytical laboratories to address this challenge.

2.6.3. Internal Calibration of Mass Spectrometers

We said above that the precision and the accuracy of isotope ratio measurements by mass spectrometry is ultimately limited by the fluctuations of isotope fractionation effects from one filament loading to the next. It has been proposed to spike the sample with a well characterized mixture of two tracer isotopes in order to overcome this limitation; the measured ratio of the tracer isotopes observed on an individual analysis of the spiked sample is compared to its reference value; and one calculates the actual calibration factor to be applied to the result of the analysis of an individual filament loading.

The approach has been tested with isotopic and isotope dilution analyses of uranium, using a mixture of ^{233}U and ^{236}U tracer isotopes. Filaments loaded with samples in solution from following conventional techniques can be analyzed with precisions and accuracies of 0.01% after internal calibration. Resin bead samples have been analyzed with precisions and accuracies of 0.1 - 0.2%.

The technique is being tested now to analyze both uranium and plutonium, using a spike containing a mixture of ^{233}U , ^{236}U , ^{242}Pu and ^{244}Pu tracer isotopes.

The feasibility of the technique will depend upon the long term availability of gram quantities of these enriched isotopes.

As the irradiation proceeds the mass of uranium in the core and the abundance of the ^{235}U isotope decrease, while the mass of plutonium and transplutonium elements and the abundances of the heavier isotopes of plutonium, ^{240}Pu , ^{241}Pu and ^{242}Pu , go up. The balance of nuclides following irradiation of a uranium fuel in a light water reactor (LWR) may be written as follows

$$M_i(\text{U}) = M_F(\text{U}) + F + M(\text{Pu}) + \text{TP} \quad (16)$$

where $M_i(\text{U})$, $M_F(\text{U})$ are the initial and final masses uranium in core

F represents the mass of fissioned uranium

$M(\text{Pu})$, TP are the masses of plutonium and transplutonium elements in core after irradiation.

If one defines the burn up, B , as the fraction of fissioned uranium

$$B = \frac{F}{M_i(\text{U})} \quad (17)$$

the final mass of uranium is equal to

$$M_F(\text{U}) = M_i(\text{U}) \frac{1 - B}{1 + \frac{M(\text{Pu})}{M_F(\text{U})} + \frac{\text{TP}}{M_F(\text{U})}} \quad (18)$$

Theoretically, given the proper description of the reactor, the initial composition of the nuclear fuel, its distribution in the reactor core, and various other parameters nuclear physicists can calculate the burn-up, the masses of uranium, plutonium and transplutonium elements to be expected after irradiation. They can also predict the isotopic composition expected for these elements. Conversely they can define the functions relating the burn-up B and the mass fractions, $\frac{M(\text{Pu})}{M_F(\text{U})}$, $\frac{\text{TP}}{M_F(\text{U})}$ to the final isotopic composition of

the fuel. Such functions are called Isotopic Correlations. Correlations can also be defined between the isotopic compositions of uranium and plutonium. Table 5 lists a number of correlations that were tested for practical use. D_{235} and D_{236} designate the changes in the isotopic abundances of ^{235}U and ^{236}U isotopes during irradiation.

Thus, if the initial enrichment of the fresh fuel U_{50} is known, and if the enrichment of the spent fuel U_5 is measured, one may calculate

$$D_{235} = U_5 - U_{50} \quad (19)$$

Table 5

Examples of Isotopic Correlations

Correlation	Function	Variable	Property
1	$\frac{M(Pu)}{M_F(U)}$	D 235	insensitive to changes in burn-up, but sensitive to variations in neutron spectrum
2		$\frac{^{239}Pu}{^{240}Pu} * \frac{^{242}Pu}{^{240}Pu}$	insensitive to variations in neutron spectrum, but sensitive to changes in burn-up
3		$\frac{^{240}Pu}{^{239}Pu}$	
4		$100 - ^{239}Pu$	
5	B	D 235	sensitive to variations in neutron spectrum, but based on precise analytical results
6		D 236	based on less precise results, but sensitive to variations in neutron spectrum
7		$\frac{^{242}Pu}{^{240}Pu}$	
8	D 236	D 235	
9	$(^{239}Pu)^2$	D 235	
10	$\frac{^{240}Pu}{^{239}Pu}$	D 235	

Correlations 1 and 5 provide respectively a prediction of the plutonium mass fraction and of the burn-up:

$$\frac{M(\text{Pu})}{M_F(\text{U})} = f_1(D 235) \quad (20)$$

$$B = f_5(D 235) \quad (21)$$

The resulting values are entered in equation (18) to yield a prediction of the final mass of uranium, as the mass fraction of transplutonium elements is usually smaller than 0.1% and can be neglected.

A predicted value can now be calculated for the mass of plutonium:

$$M(\text{Pu}) = M_F(\text{U}) f_1(D 235) \quad (22)$$

3.2. Information needed

An extensive set of information must be available to make full use of the potential of isotope correlations.

3.2.1. Reference Correlations

The isotope correlations functions should be known before the reprocessing starts. Ideally such reference correlations should be derived from nuclear physics calculations applied to appropriate reactor models, also called reactor codes. These theoretical correlations should themselves be corroborated by careful experimental verifications based on accurate analyses.

In the previous years, empirical correlations have also been established on the basis of historical data. Such correlations are difficult to qualify and should be used only with great caution as they may be biased by uncontrollable analytical errors.

The correlations are computed and valid for the date of discharge of the fuel from the reactor.

3.2.2. General Reactor and Fuel Data

General characteristics of reactor and fuel available from released publications can suffice to establish the model. They include the number of rods by fuel assembly, the nature of cladding and structural materials (Zr, stainless steel), burnable poison, if applicable, and geometric characteristics such as the fuel rod diameter, pellets diameter and water gap dimensions.

3.2.3. Fuel Fabricator's Data

The total initial mass of uranium and its isotopic composition must be known from the declarations of the fuel fabricator or safeguards verifications.

3.2.4. Reactor Operator's Data

In addition to the general information required to define the reactor code, specific information is also needed on the operational history of the core, especially charge and discharge times. Individual assembly history (time in and out of core, etc...) is of interest for the evaluation of anomalies that may be detected in the correlations.

The reactor operator's declaration must include the description of the content of each spent fuel assembly in total mass of uranium, plutonium, ^{235}U isotope and burn-up, ^{236}U , and the plutonium isotopic composition.

3.2.5. Reprocessor's Data

150 to 1200 kg of fuel will be dissolved per input batch, depending upon the volume of the dissolver and input accountability tanks. An input batch rarely corresponds to the dissolution of a single assembly. The correspondence between assembly dissolution and input batches must be known.

In order to compare correctly the results obtained by isotope correlations and the accountability data on the input tank, the decladding losses, the amount of recycled materials and the mass of fuel retained in the heel of the dissolver tank must be negligible or properly estimated (Figure 19). If this is not the case, the two values will match only after accumulating results over several dissolver batches or even over the complete campaign.

3.3. Applications by Plant Operators

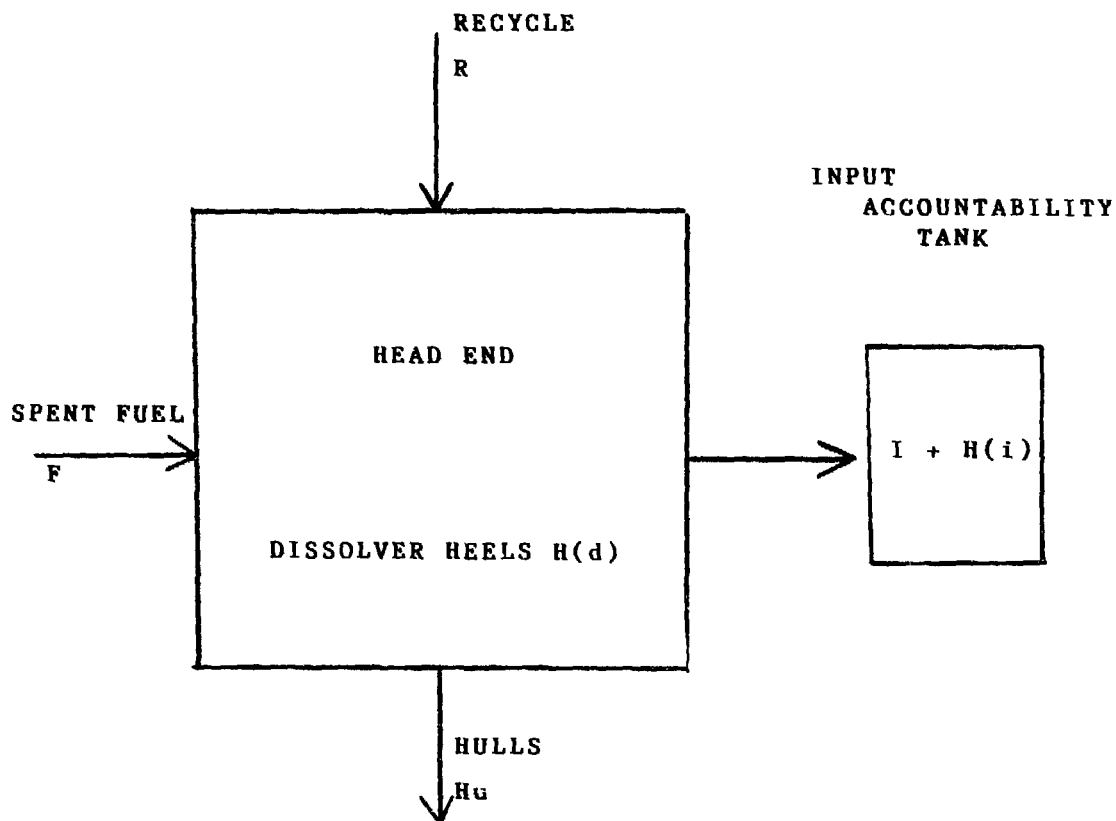
3.3.1. Internal Consistency Checks

Assume that the correlations of Table 5 are known, and that $\frac{M(\text{Pu})}{M_{\text{F}}(\text{U})}$, B, D 236, ^{239}Pu and $\frac{^{240}\text{Pu}}{^{239}\text{Pu}}$ are measured by analyzing samples taken

from the dissolver or from the input tank. An abnormally large difference between the analytical result and the value derived from the correlations detects an anomaly. Multiple cross checks may enable us to decide whether the anomaly is due to an error in the analyses or to a deviation from the model used to define the correlations.

If the analyses are incomplete, it is still possible and instructive to compare estimates of the same parameter obtained from different correlations. For example, the burn-up may be calculated by correlation 5 (vs D 235) or 6 (vs D 236). The two values will not agree if the measurement of the ^{236}U isotopic abundance is in error, if the neutron spectrum in the reactor differs significantly from the expected one or if the data used for the initial isotope abundances of ^{235}U or ^{236}U isotopes were wrong.

Even when no reference correlation is available, useful internal consistency checks can be made by regression analyses of the results of measurements accumulated as the campaign proceeds.



$$I = F + R - H_u - H(d)$$

$$\begin{array}{ccccccc}
 & & & \text{Recycle} & \text{Reel} & & \\
 & & & \downarrow & \text{(Disolver)} & & \\
 \text{Analysis} & \uparrow & A = I + H(i) = & F + R - H_u - H(d) + H(i) & & & \\
 \text{Input Tank} & & & \uparrow & \uparrow & \uparrow & \\
 & & & \text{Isotope} & \text{Hull} & \text{Heel} & \\
 & & & \text{Correlation} & \text{Monitor} & \text{(Input Tank)} &
 \end{array}$$

Figure 19. BALANCE OF FUEL MATERIAL AT THE HEAD END OF A REPROCESSING PLANT

In addition to analytical errors, these tests may detect changes in cladding, substitution of a spent fuel element with an unirradiated element, changes in initial isotopic composition, abnormal irradiation conditions, or other errors in the data bank. They are very important because confirmed anomalies may invalidate the application of isotope correlation techniques or they may detect situations of safeguards significance.

3.3.2. Estimate of Plant Input

After verification of the data through internal consistency checks, isotope correlations may be used to calculate the burn-up, the mass fraction of plutonium $M(\text{Pu})$, and the final mass of uranium and the mass of plutonium, as $M_F(\text{U})$ explained in section 3.1.

As shown in Table 6, the plant of La Hague had very good success in applying isotope correlations. The operator considers that his correlation techniques are well qualified. They are used routinely to confirm the validity of process control at the head end.

3.3.3. The Gravimetric Method

In the first part of this document we described the volumetric technique that is most commonly used to measure the flow of spent fuel entering the process area. The volumetric technique has three shortcomings:

- (a) it does not account for the losses that can occur in dismantling and dissolving the fuel and it includes the amount of fissile material that may be recycled (Figure 19);
- (b) the volume of the solution in the tank must be accurately measured; this is not easy and volume measurement errors can be a significant parameter of the measurement system (Table 3); and
- (c) the solution in the tank must be perfectly homogeneous so that the samples are well representative.

The plant of La Hague prefers using a gravimetric method that combines analysis with estimates by isotope correlations: the plutonium mass fraction $\left(\frac{\text{Pu}}{\text{U}}\right)$ is determined by IDMS analyses of samples of the spent fuel solution as described in (2.1). The burn-up is either measured by analyzing neodymium or estimated using heavy isotope correlations 5 and 6 of Table 5. Equation 18 is used to calculate the mass of uranium $M_F(\text{U})$ expected to have entered the dismantling cell. This is used to calculate the mass of plutonium expected to be carried by this fuel:

$$M(\text{Pu}) = M_F(\text{U}) \cdot \left(\frac{\text{Pu}}{\text{U}}\right) \quad (23)$$

The relative difference between the results obtained by this method and the data derived solely from isotope correlations has been, on the average, -0.05% per campaign, with a standard deviation of 0.33% for a single campaign. This observation was made on a series of campaigns involving the reprocessing of a total of about 510 metric tons of spent fuels.

Table 6

Results of Isotope Correlation Estimates at the Reprocessing Plant of La Hague

Reactor Type	PWR	BWR
Number campaigns	20	8
Number batches	505	224
Total uranium (in metric tons)	510	
<u>Burn up</u>		
$\bar{d}$ (%)*	- 3.2	- 1.7
SD between campaign (%)	$\pm$ 2.4	$\pm$ 1.5
SD between batches (%)	$\pm$ 2.6	$\pm$ 2.5
<u>Pu/U</u>		
$\bar{d}$ (%)**	- 0.35	- 0.45
SD between campaign (%)	$\pm$ 0.75	$\pm$ 0.85
SD between batches (%)	$\pm$ 1.3	+ 2.1

* Comparison between reactor data and value obtained by isotope correlations.

** Comparison between analysis and value obtained by isotope correlations.

3.4. Safeguards Verifications

Isotope correlations can be applied for safeguards verifications with various degrees of independence

3.4.1. Using Declared Data

Software packages have been developed to enable inspectors to perform isotope correlation calculations during inspection at reprocessing plants using the declarations of the fuel fabricator and reactor operator and the analytical results of the reprocessor.

The only independent factor that the inspector selects is the correlation and the procedure of evaluation. The advantage of the system is to signal anomalies on the spot and to provide timely estimates of shipper/receiver differences and differences between direct input analyses and isotope correlation estimates. The power of the approach is strengthened if:

- (a) the fabricator's data have been already verified;
- (b) assurance is obtained that the fuel is completely dissolved;
- (c) the operator's samples are proven to be representative and their analyses accurate; and
- (d) the reference correlation that is applied is valid for the fuel under examination.

3.4.2. Using Verified Isotopic Data

A higher degree of independence is achieved by performing isotope correlation calculations using the results of isotopic analyses done by a verification laboratory on independent samples taken during inspection. The preparation of such samples and their analyses are much simpler and quicker procedures than the ones that are required to perform complete analyses including element assay by IDMS.

3.4.3. Verification of Correlations

At least a fraction of the dissolver batches must be sampled and submitted to a complete and independent isotopic and isotope dilution analysis in order to verify the validity of the correlations used in inspection verifications. If the observations differ significantly from the reference correlations, safeguards verifications can rely only on inspection remeasurements.

3.4.4. Performance and Error Factors

Little or no data have yet been published on the actual performance of safeguards verifications by isotope correlation techniques. The Agency is not implementing the techniques but is proceeding with an evaluation of various correlations systems. The validation of the system is the first difficulty and the interpretation of observed anomalies is always be a difficult task. A number of national and international exercises have been carried out or are underway to address these problems.

The estimate of the mass of spent uranium by isotope correlations is about as precise as the fuel fabricator's declaration. The precision of plutonium estimates is expected to approach the performance of direct verification analyses and is limited both by deviations from the model and errors in the isotopic analyses of the spent fuel (Table 7).

When comparing the results of analyses and the correlation estimates, one must take into account the uncertainties in the measurements of the volumes of solutions, the analyses of heels in the dissolver, input accountability tanks and recycled solutions, the monitoring of hulls, and other potential losses between the storage pond and the input tank.

3.5. Other Isotope Correlation Methods

3.5.1. Radioactive Fission Products and Transplutonium Isotopes

The emission of gamma radiation by radioactive fission products like ^{137}Cs at 662 keV, ^{134}Cs at 796 keV and ^{144}Ce - ^{144}Pr at 2186 keV can be measured directly on the spent fuel assembly. The activity ratios $^{134}\text{Cs}/^{137}\text{Cs}$ and $^{144}\text{Ce}/^{137}\text{Cs}$ have been correlated with the burn-up and the cooling time of the fuel.

The spent fuel also emits neutrons originating from spontaneous fissions and (α, n) reactions. For spent fuels submitted to burn-up equivalent to more than 20 000 MWd/t and cooled for more than 2 years, ^{244}Cm is the main source of neutrons. This neutron emission E is correlated with the initial enrichment of the fuel, its burn-up, and cooling time to compute the mass of plutonium:

$$B = a E^t \quad (24)$$

$$M(\text{Pu}) = b \cdot f(B) \cdot g(^{235}\text{U}) \cdot h(R) \cdot E^{0.125} \quad (25)$$

Plutonium masses have been estimated in a non-destructive mode with precisions of 0.3 to 1.2% and accuracy of + 0.6 to 1%.

3.5.2. Stable Isotope Correlations

Among the numerous stable isotopes produced as a result of reactor fissions, only the isotopes of krypton, xenon and neodymium have retained attention for isotope correlation techniques. ^{148}Nd is analyzed routinely as a burn-up indicator. The ratios of ^{145}Nd , ^{146}Nd and ^{148}Nd may be used as correlation variables.

Gaseous xenon and krypton isotopes can be analyzed in the off gases of the dissolver. It has been proposed to install fixed measurement stations that would provide a continuous monitoring of the dissolver.

3.5.3. Across The Plant Element and Isotope Balance

If the plant has no loss and its hold-up remains constant, the material output should be equal to its input. It is of interest to calculate the element balance passing through the plant to provide estimates of hold-up changes and of the magnitude of the Material Unaccounted For (MUF).

A balance can also be computed for each isotope rather than for each element. If no recycling or substitution takes place without being noted, the mean isotopic composition of the input and output should be equal. Assume that a reprocessing campaign consists of n input batches and N output batches. Input batch i contains a mass m_i of fissile element of isotopic composition A_i . Output batch j has a mass M_j of fissile element of isotopic composition B_j .

Table 7

Sources of Errors in Isotope Correlation Estimates and Comparisons with Direct Analytical Measurements

METHOD	ELEMENT	URANIUM			PLUTONIUM		
	TYPE OF ERROR	SYSTEMATIC ERROR	CV OF SYSTEMATIC ERROR	CV OF RANDOM ERRORS	SYSTEMATIC ERROR	CV OF SYSTEMATIC ERROR	CV OF RANDOM ERROR
	SOURCE OF ERROR	(%)	(%)	(%)	(%)	(%)	(%)
ISOTOPE CALCULATION	FABRICATOR MASS		0.02	0.02		0.02	0.02
	FABRICATOR ISOTOPIC		0.02	0.02		0.02	0.02
	REACTOR MODEL	0.01	0.015	0.015	0.01	0.5	1.0
	NEGLECTING TRANSPLUTONIUM ELEMENTS	0.1	0.01	0.01	0.1	0.01	0.01
	SPENT FUEL ELEMENTS		0.02	0.02	- 0.5	0.3	1.0
	TOTAL	0.1	0.04	0.04	- 0.4	0.58	1.4
VOLUMETRIC METHOD	TANK VOLUME		0.1	0.3		0.1	0.3
	IDA		0.2	0.2		0.3	0.3
	TOTAL		0.22	0.36		0.32	0.42
DIFFERENCE	HULL LOSSES	0.1	0.02	0.02	0.1	0.02	0.02
	RECYCLING	VARIABLE	VARIABLE	VARIABLE	VARIABLE	VARIABLE	VARIABLE
	TANKS HEELS	± 0.1	0.05	0.05	± 0.1	0.05	0.05
	OVERALL	~ 0.1	0.23	0.37	~ -0.4	0.66	1.5

The mean isotopic composition of the input is

$$A = \frac{\sum_{i=1}^n m_i A_i}{\sum_{i=1}^n m_i} \quad (26)$$

The output mean is

$$B = \frac{\sum_{j=1}^N M_j B_j}{\sum_{j=1}^N M_j} \quad (27)$$

The equation

$$A = B \quad (28)$$

May also be considered as an isotope correlation.

4. CONCLUSION

The isotope correlation techniques allow one to calculate the expected input of the head end of the spent fuel reprocessing plant. They provide a verification of the shipper's and reprocessor's data with simpler and faster isotopic analyses than the isotope dilution analyses and volume measurements required in direct physical verifications.

However, isotope correlations methods alone cannot detect the absence of an element, incomplete dissolutions, or diversion of part of the dissolver solution. Such losses or diversions would become apparent only after the output products are analyzed and would show up as a MUF.

The direct physical and analytical verifications on the input accountability tank measures the real input to the process area. It can differ from the input of the head end if hull losses and recycled materials are not negligible. Such effects would show up as shipper/receiver differences unless they are accounted for by a direct measurement of the hulls and recycled reagents or by using the gravimetric method.

The direct volumetric method is also the only method to validate fully the isotope correlations.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

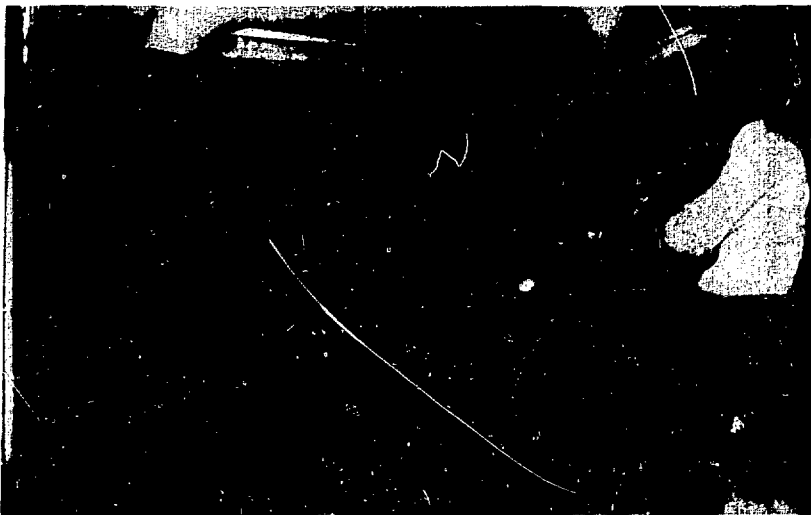
SESSION 16: TOUR LOS ALAMOS SAFEGUARDS RESEARCH AND DEVELOPMENT LABORATORY

Participants will tour the materials testing reactor (MTR) at Los Alamos National Laboratory and examine techniques for the measurement of irradiated nuclear fuel. Many of the instruments described in sessions 13-14 will be available for examination and some "hands on" use. These will include:

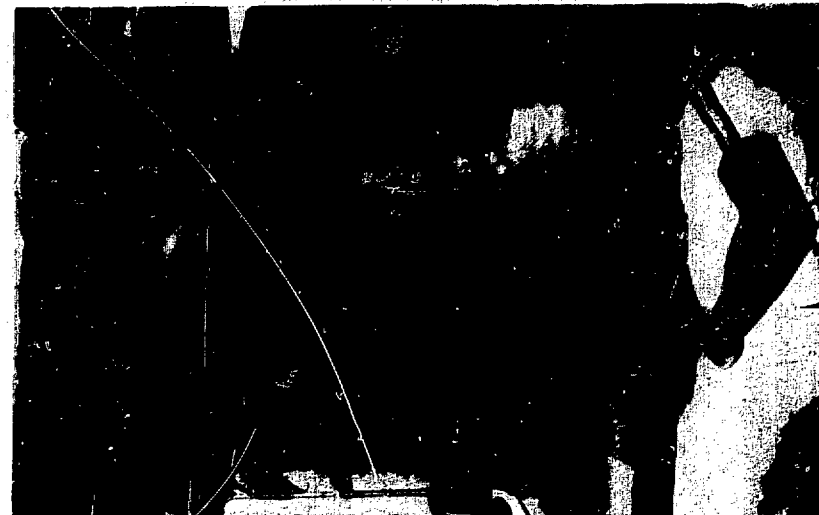
- Cerenkov radiation viewing instruments.
- Gross gamma-ray and neutron activity detectors, ION-1.
- High-resolution gamma-ray spectroscopy instrumentation for fission product assay.

After the session, participants will be able to:

- Describe the range of techniques and instruments available to the reactor operator or the safeguards inspector for the measurement or verification of irradiated fuel materials.
- Discuss the different levels of verification attainable using presently available instrumentation.
- Understand the limitations of fuel measurement techniques.



Terry Smith describes Omega West (Los Alamos materials testing reactor) to the tour group.



Participants use a night vision device to observe Cerenkov radiation from spent-fuel elements stored around edge of reactor core.



Gene Bosler discusses measurement data taken with ION-1 and ion chamber from irradiated MTR fuel element moved to cooling pond at Omega West Reactor.



Jim Halbig demonstrates use of portable MCA to collect spectra from GE detector viewing spent MTR assembly in cooling pond.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 16: TOUR OF THE OMEGA WEST REACTOR (OWR) AND
SPENT-FUEL MEASUREMENT DEMONSTRATION

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CONTENTS

- I. INTRODUCTION
- II. STATION 1: DESCRIPTION OF OMEGA WEST REACTOR
- III. STATION 2: CERENKOV VIEWING DEVICE
- IV. STATION 3: HIGH-RESOLUTION GAMMA SPECTROSCOPIC METHOD OF BURNUP DETERMINATION
- V. STATION 4: MEASUREMENT OF THE GROSS GAMMA ACTIVITY PROFILE OF AN MTR SPENT-FUEL ELEMENT

Tour participants will be divided into four groups, and the groups will circulate among the four demonstrations according to the following schedule:

TOUR SCHEDULE

	Station 1	Station 2	Station 3	Station 4
9:00				
	Introductory Lecture and Coffee Break			
10:00				
	Group 1	Group 2	Group 3	Group 4
10:30				
	Group 2	Group 3	Group 4	Group 1
11:00				
	Group 3	Group 4	Group 1	Group 2
11:30				
	Group 4	Group 1	Group 2	Group 3
12:00				
	LUNCH at LANL Cafeteria			

I. INTRODUCTION

Our safeguards groups are actively involved in the development of non-destructive techniques for the measurement of spent-fuel assemblies. This work is supported jointly by the Department of Energy under the Offices of Safeguards and Security and the Department of State under the program of technical assistance to the International Atomic Energy Agency. We have applied a variety of gamma-ray and neutron techniques to a wide range of fuel materials. Table I lists the types of fuel materials we have examined.

TABLE I
SPENT-FUEL MEASUREMENT EXERCISES

<u>Facility</u>	<u>Type of Fuel</u>	<u>Measurements</u>
Big Rock Point	BWR	Gamma ray, neutron
Zion Generating Station	PWR	Cerenkov, gamma ray, neutron
General Electric-Vallecitos	BWR	Gamma ray, neutron, destructive
Research Reactor	HWR	Gamma ray
Omega West Reactor	MTR	Cerenkov, gamma ray, neutron
General Electric-Morris	BWR, PWR	Cerenkov, gamma ray, neutron
Canadian Facility	CANDU	Cerenkov
Battelle Columbus Laboratory	PWR	Neutron, destructive
Loviisa, Finland	WWER	Neutron, gamma ray
Tihange, Belgium	PWR	Neutron, gamma ray
Obrigheim, Germany	PWR	Neutron, gamma ray
Gundremmigen, Germany	BWR	Neutron, gamma ray
Three Mile Island	PWR	Cerenkov, neutron, gamma ray

In addition to the emphasis placed on the experimental approach, we have developed calculational techniques that permit us to model accurately:

1. the buildup depletion and decay of fission products and transuranic isotopes in fuel materials under a variety of irradiation conditions, and
2. the transport of gamma ray and neutrons through the source material using Monte Carlo techniques.

The calculational modeling techniques permit us to simulate different sources and detector conditions and to evaluate the applicability of specific NDA techniques.

During the demonstrations today you will have the opportunity to use three instruments that were developed to measure the gross gamma ray, spectral gamma ray, and neutron signatures of spent-fuel assemblies. Table II lists the possible diversion scenarios and the detectability using each of the techniques. As mentioned in the formal presentations, a combination of gamma-ray and neutron techniques is the most effective approach to the verification of spent-fuel assemblies.

TABLE II

POSSIBLE DIVERSION PATHS FOR SPENT-FUEL ASSEMBLIES

<u>Diversiion Scenario</u>	<u>Change in Signatures</u>		
	<u>Gross Gamma</u>	<u>Spectral Gamma</u>	<u>Passive Neutron</u>
(a) Operator declaring incorrect values ^a	Detectable ^a	Detectable ^a	Detectable ^a
(b) Substitution of unirradiated counterfeit fuel assembly	Detectable	Detectable	Detectable
(c) Substitution of counterfeit assembly with gamma-ray source (i.e., ⁶⁰ Co) ^b	Not detectable ^b	Detectable ^b	Detectable ^b
(d) Substitution of counterfeit fuel assembly with only fission products replaced ^b	Not detectable ^b	Not detectable ^b	Detectable ^b
(e) Substitution of counterfeit fuel assembly with fission products and curium isotopes replaced ^b	Not detectable ^b	Not detectable ^b	Depends on the source distribution
(f) Partial removal of fuel pins within the assembly	Depends on amount removed	Depends on amount removed	Depends on amount removed

^aDepends on the precision of the measurement technique and the difference between the true and falsified values.

^bFor the scenarios (c), (d), and (e), two cases should be considered axial to reflect more precisely the effect of the source distribution. If the axial distributions of fission products and neutron sources are measured, it may be possible to detect changes in the profiles using the gross and spectral gamma or neutron signatures by comparing the measured results with theoretical calculations.

II. STATION 1: OMEGA WEST REACTOR

Description

The Omega West Reactor (OWR) uses an array of materials-testing-reactor-(MTR) type fuel elements supported inside a stainless steel tank. The core support structure and grid plate are aluminum. Light water circulated downward through the core at 3500 gal./min serves as moderator and coolant. The reactor is controlled by eight blade-type poison rods. The reactor tank is covered with a stainless steel lid that supports the control-rod drive mechanisms and also contains two large hatches to provide access to the core. A biological shield of heavy concrete in an irregular octagonal shape surrounds the tank and thermal column. Experimental ports open on all of the shield faces; a large thermal-column door opens on the east side. Instrumentation ports are in a recess at the bottom of the south face and extend underneath the core. Figure 1 is a cutaway illustration of the reactor.

Reactor Core

The reactor core consists of a 4 by 9 array of fuel elements and experiment or sample elements supported vertically in the aluminum grid plate. Eight boron-stainless steel control blades move in slots between fuel elements. A beryllium reflector is on the west face of the core, and a lead gamma shield is on the east face. The north and south ends of the core are water reflected (see Fig. 2).

Fuel Elements. The metal fuel elements are of the type developed originally for the MTR. Each element contains 18 curved fuel plates, 1.52 mm thick, mounted 2.97 mm apart in heavy aluminum side plates. Each fuel plate is made of a sheet of uranium-aluminum alloy sandwiched and hot-rolled between two 0.5-mm-thick sheets of pure aluminum. The ^{235}U bearing section of the elements is 60.96 cm long. A cylindrical end box welded to each end of the element positions it in the grid plate and permits water flow between fuel plates. Fuel element dimensions are shown in Fig. 3. Each fuel element contains 220 g of ^{235}U .

The original fuel elements contained 168 g of ^{235}U , but the addition of more in-core irradiation positions and the extension of the operating schedule to 120 h/wk in 1960 made it necessary to increase the element loading. Burnable poisons have not been used.

Core Base and Grid. The core is supported and positioned in the tank by an aluminum base casting so that its center is 1.07 m above the reactor room floor and 4.60 mm east of the tank center. A three-sided "enclosure" is formed on the south, west, and north by 2.59-cm-thick aluminum plates bolted to the base in a position providing 20.32 cm of space inside the enclosure walls for additional reflector material and a 15.24-cm-wide shelf outside these walls for more shielding, should either be needed. The enclosure side walls are penetrated by the beam and through ports.

The grid or matrix plate contains 54 holes in a 6 by 9 array, which fit the end boxes of the fuel elements; 54 position pins for precise location of each element; and 8 slots for the control rods. The type 356 aluminum alloy plate is 12.7 cm thick and contains additional holes for improved water flow between elements. The six rows of nine holes are occupied as shown in Fig. 2.

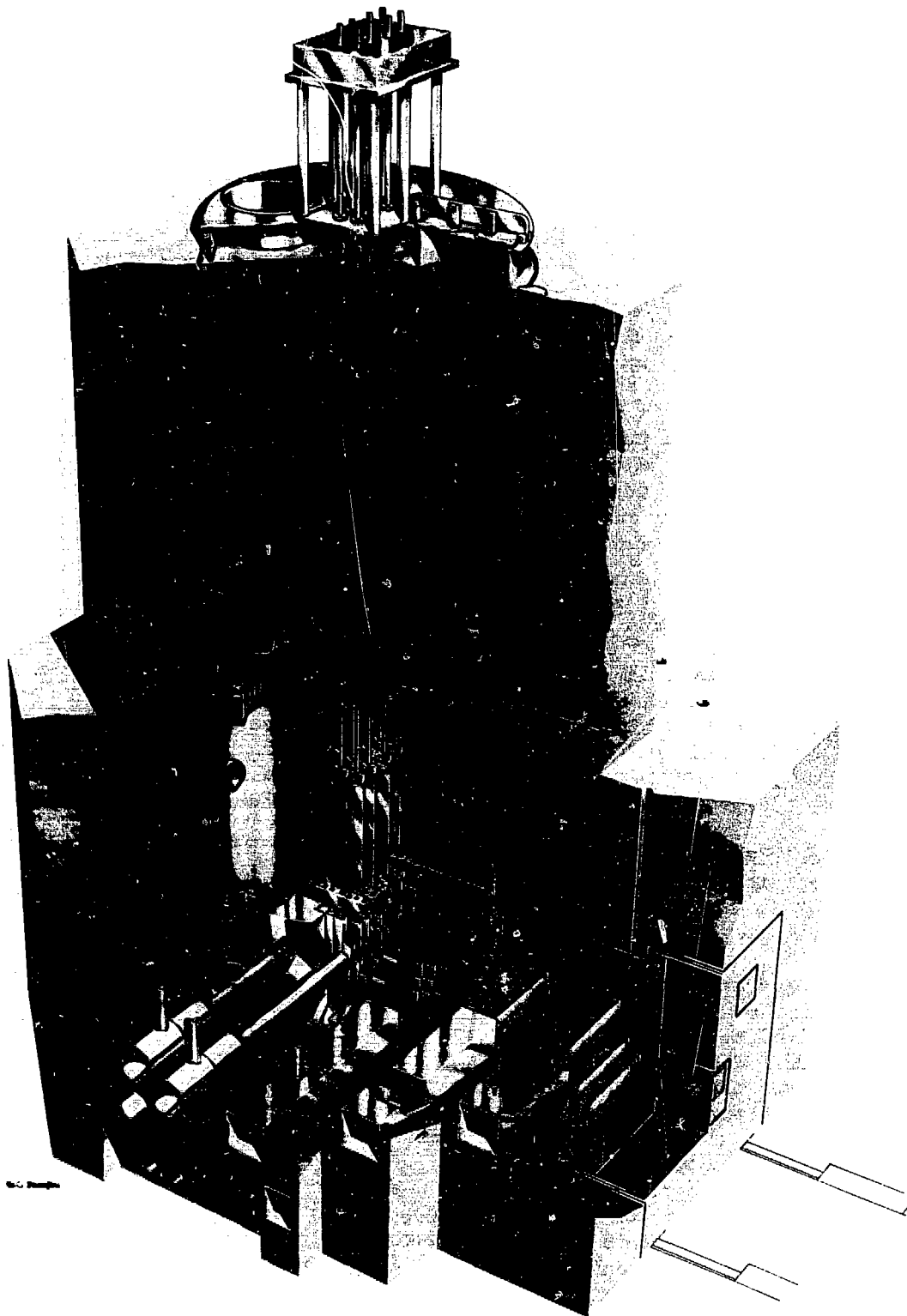


Fig. 1. Cutaway view of OWR.

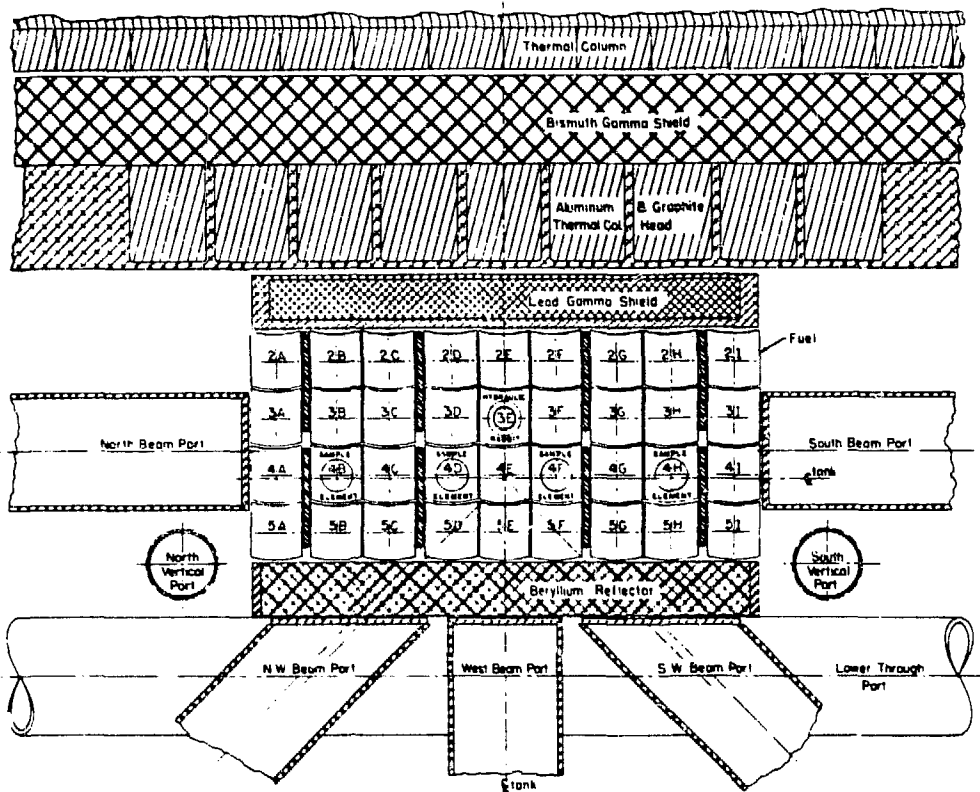


Fig. 2. OWR core layout.

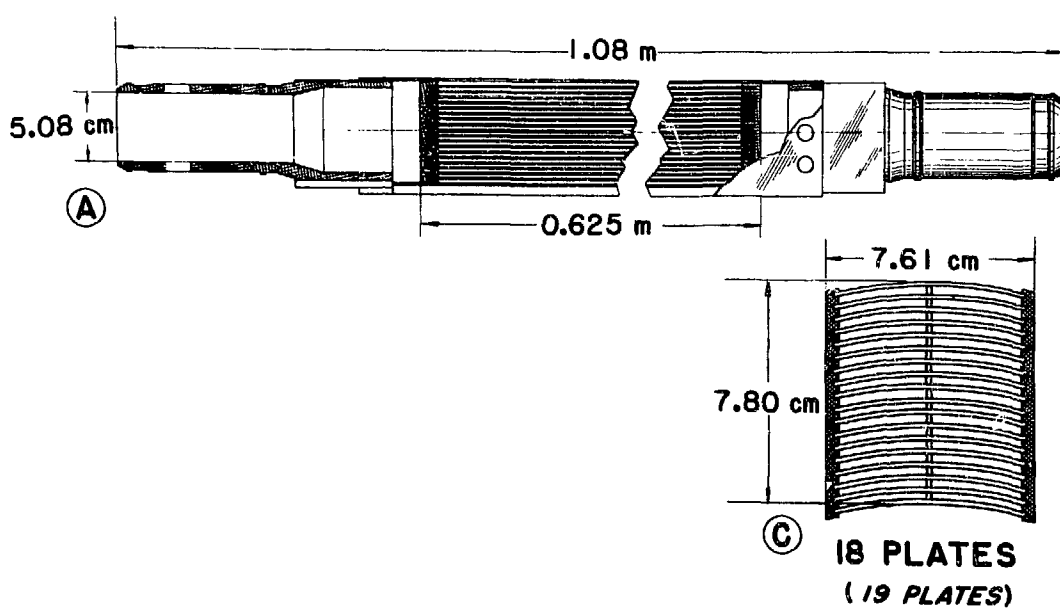


Fig. 3. OWR fuel element.

with rows 1 and 6 accommodating, respectively, the lead gamma shield and beryllium reflector.

A hold-down grid folds down over the top of the core and provides a positive stop to prevent inadvertent lifting of any fuel element from the matrix. The grid can be unlocked and lifted aside from either access hatch to permit core-loading operations. The open framework design allows access to the sample and experiment positions with the grid in the down position, as shown in Fig. 4.

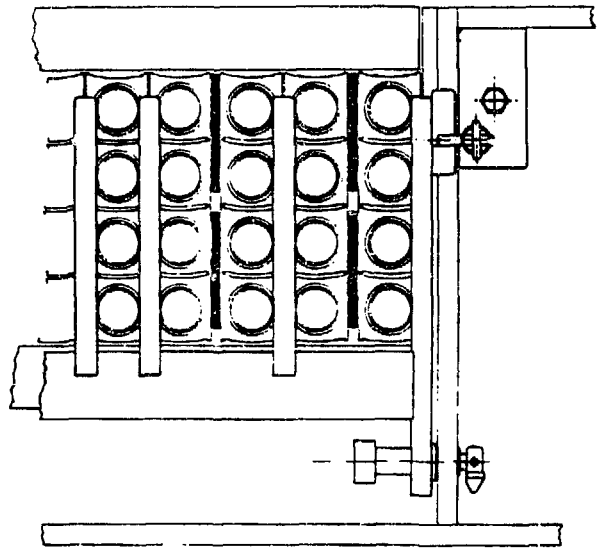


Fig. 4. Fuel element hold-down grid.

Control Rods. The reactor is controlled by eight independently movable, stainless steel control rod blades containing 1.2% natural boron by weight. The rods move in slots between fuel elements and are located in the core array, as shown in Fig. 2. The rods are in three sections, as shown in Fig. 5; the upper and lower sections are 6061-T6 aluminum. The center section, boron-stainless steel, 69.85 cm long, is fastened to the upper and lower sections with stainless steel screws through compound lap joints. The lower section necks down below the joint to allow clearance through the slots provided in the bottom grid. Inserts of stainless steel in the grid plate provide guide surfaces for the rods. Because of the step in the blade and because the blade length is such that the lower end clears the bottom of the reactor tank by only 15 to 18 cm in the down position, there is no possibility of a rod dropping out of the core.

Bolted to the top of each control rod is a latching device that can be operated from the top of the reactor with a long tool, facilitating replacement of control rods when necessary. The rods are attached by the latches to stainless steel extension rods, 2.54 cm in diameter and about 3.0 m long. The extension rods extend upward through bearing holes in the two support bridges across the reactor tank (see Fig. 1). The lower bridge is 3.35 m from the top of the tank and the upper is 1.5 m from the top. Just above the upper bridge the extension rods become discontinuous through an armature and electromagnet coupling assembly. In case of a scram or power failure, the magnetic coupling releases the rods, which then fall under gravity to their down positions. Dashpots under the armatures strike a plate on the upper bridge before the rods reach their full down position and brake their fall through the last 7.62 cm. Above the magnet assembly, the extension rods pass through chevron seals in the reactor tank lid. These provide a water-tight seal without hampering rod movement. Just above the lid, the extension rods are coupled through a rack and pinion to the control rod drive mechanisms. The drive mechanism for each rod consists of an ac motor and reduction gear box capable of lifting the rod through its 66-cm length of travel in 55 s. Motor controls are interlocked so that only one rod can be raised at a time.

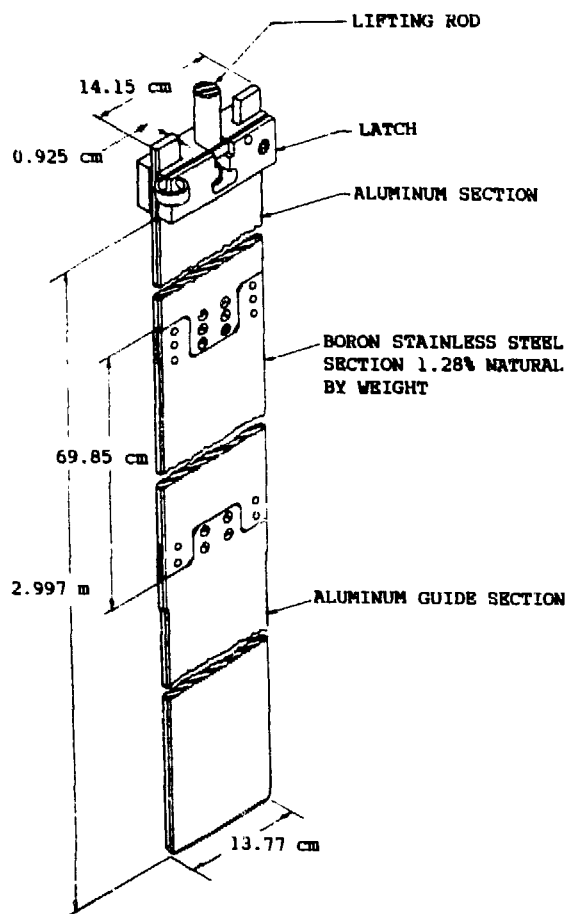


Fig. 5. OWR control rod.

Because stainless steel containing 1 to 2% boron suffers radiation damage after prolonged exposure, the control rods are routinely replaced after 90 000 MWh of operation. Tests on exposed rods have shown that this exposure level does not adversely affect their mechanical properties.

Beryllium Reflector. A reflector of twenty-one 10.16- by 10.16- by 10.16-cm beryllium bricks mounted in a nickel-plated mild steel and aluminum framework that rests over row 6 of the bottom grid plate enhances the thermal flux on the west face of the core and produces an abundance of photoneutrons for startup. The nickel-plated mild steel was chosen for the side plates because its thermal expansion coefficient closely matches that of beryllium. The average flux enhancement at the west face of the core is about 13%.

Core Loading. The fuel loading typical of present operation consists of a 4 by 9 array containing 31 fuel elements, with the remaining five positions being used as experimental and irradiation facilities. This loading was adopted in 1960 and has been standard since that time. Figure 2 shows this loading arrangement.

The fuel elements initially contain 220 g of ^{235}U each; however, the elements in a given core loading range from 0 to 33% in burnup, with the core

containing on the average about 5.9 kg of residual ^{235}U . In a normal cycle, four spent-fuel elements are replaced with new elements every 6 weeks to compensate for burnup. New elements are introduced at the ends of the core and shifted to progress toward the center during subsequent reloadings. This pattern tends to flatten the flux distribution.

Fuel element burnup is computed from wire flux maps and records of the position and duration of irradiation for each element.

REFERENCE

H. T. Williams, A. E. Lyle, O. W. Stopinski, C. L. Warner, J. L. Yarnell, and H. L. Maine, "1969 Status Report on the Omega West Reactor, with Revised Safety Analysis," Los Alamos Scientific Laboratory report LA-4192 (July 1969).

III. STATION 2: CERENKOV VIEWING DEVICE

The Cerenkov viewing device (CVD) is used to view the characteristic Cerenkov glow emitted from spent-fuel elements stored underwater. For irradiated fuel elements with cooling times greater than a few months, the Cerenkov glow cannot be seen in a normally lit room. This requires that the viewing of the glow be done in a darkened room or at least where the room light intensity is less than the Cerenkov glow and light reflected off the surface of the pool be minimized.

Two CVDs will be used to view the Cerenkov glow from the core and spent-fuel elements stored along the side of the reactor vessel. The Javelin Model 226 CVD contains a Generation I image amplifier, has a gain of 60 000 and image size of 40 mm and is used with a tripod because of its weight. The Varo Noctron V CVD is smaller and lighter than the 226 and can be used either with a tripod or hand-held. The Noctron V has a Generation II image amplifier with a gain of 50 000 and a 25-mm image. Both instruments can be fitted with a 35-mm camera to photograph the Cerenkov glow.

A great deal of qualitative information can be obtained by simply viewing the glow emitted from the spent fuel. Such information as the uniformity of the glow, the spatial distribution, the highly collimated nature of the glow emitted axially, and the general appearance of the element can be very useful to the inspector in establishing the authenticity of the assemblies inspected. This method of inspection and confirmation is both rapid and easy to perform.

REFERENCES

E. J. Dowdy, N. Nicholson, and J. T. Caldwell, "Irradiated Fuel Monitoring by Cerenkov Glow Intensity Measurements," Los Alamos National Laboratory report LA-7838-MS (ISPO-61) (1979).

N. Nicholson and E. J. Dowdy, "Irradiated Fuel Examination Using the Cerenkov Technique," Los Alamos National Laboratory report LA-8767-MS (March 1981).

IV. STATION 3: HIGH-RESOLUTION GAMMA SPECTROSCOPIC METHOD OF BURNUP

In high-resolution gamma spectroscopy measurements, it is possible to identify and measure gamma rays from individual isotopes. In this example, the gamma rays from ^{134}Cs and ^{137}Cs will be used to measure the burnup of the fuel at one location along the axis of a MTR fuel element. When used in conjunction with a profile measurement, as in today's exercise with the ionization chamber (Station 4), the burnup along the entire assembly can be determined.

The basis for this technique is the fact that the two isotopes are produced by different mechanisms. The ^{137}Cs is a fission product, so the amount of this isotope present is proportional to the burnup of the fuel. The ^{134}Cs , however, results from the fission product ^{133}Cs capturing a neutron. Two successive neutrons are needed to produce one atom of ^{134}Cs : the first one causes the fissioning and creates ^{133}Cs ; the second neutron changes ^{133}Cs in ^{134}Cs . From a simplified viewpoint, this implies that the amount of ^{134}Cs present is proportional to the square of the burnup. The ratio of the amounts of ^{134}Cs to ^{137}Cs is thus proportional to the burnup.

The advantage of this procedure over simply using ^{137}Cs by itself is that some complicating factors are eliminated. If ^{137}Cs were used alone, it would be necessary to take into account the geometry of the source-detector arrangement and deadtime correction, for example. Since these types of factors are the same for the two cesium isotopes, they cancel after taking a ratio. It is more customary to use ratios of corrected activity ratios instead of amounts. These two ratios differ only by a ratio of decay constants, so both are proportional to the burnup. [The corrected activity ratio is defined in Eq.(1).]

The spectrum of gamma rays coming from the fuel is displayed for us by a multichannel analyzer. This device sorts the gamma rays according to their energies and, hence, by their source isotopes. Figure 6 shows part of a typical spectrum. Some of the peaks are identified as to the source isotopes and energies. Today we shall use the ^{134}Cs peak with an energy of 796 keV and the ^{137}Cs peak with an energy of 662 keV. The areas under these peaks give the numbers of gamma rays captured during the counting time, and the analyzer is capable of finding these areas simply by summing the number of gamma-ray pulses detected in the peak region. There are always some background counts that are subtracted by the analyzer as well. The background adjustment technique is illustrated in Fig. 7. The regions on either side of the peak are joined by a straight line that runs under the peak. The area under this line that also falls under the peak is subtracted from the total area under the peak. This leaves what is called the net peak area.

The net areas must be modified further before a useful ratio is obtained. We will use three corrections here.

- Decay Time

We need to compare the amounts of the two cesiums present when the fuel was removed from the reactor because calibration curves are based on the time of discharge. To do this, the known decay constants are used to increase the present amounts to estimate initial amounts.

- Branching Ratio

A nucleus can decay by emitting more than one gamma ray. One peak, therefore, does not represent the total amount of the isotope's

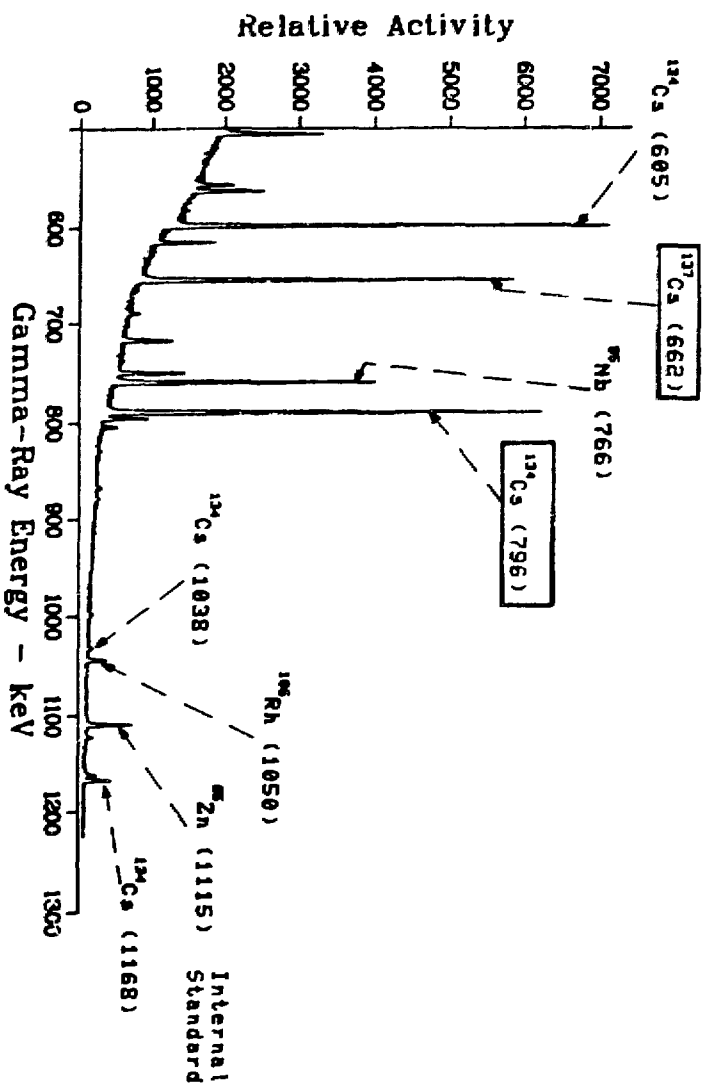


Fig. 6. Part of a typical spectrum.

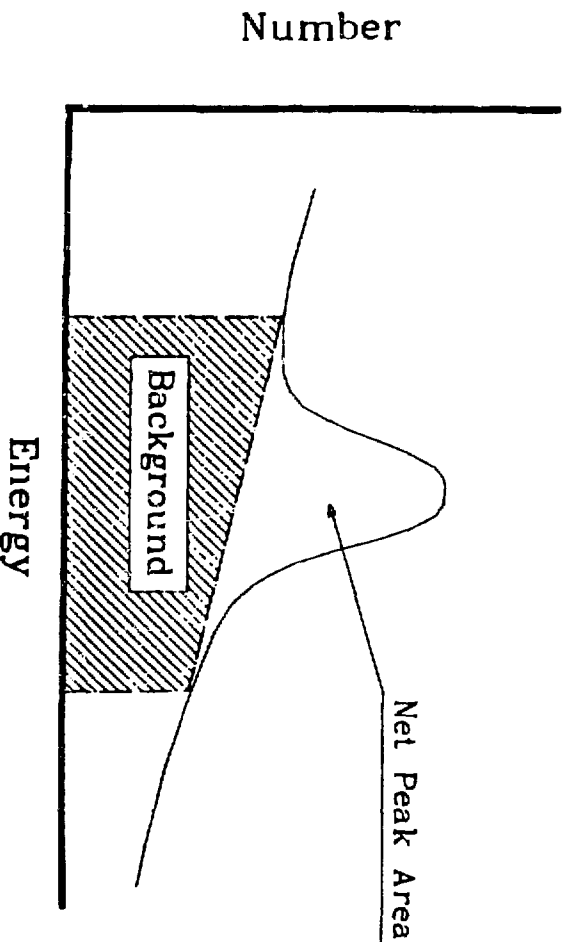


Fig. 7. Background adjustment technique.

activity. The branching ratio is the fraction of the total decay that goes into the decay of interest. By dividing by the branching ratio, the total activity is obtained.

- Relative Detection Efficiency

A detector does not have the same efficiency for gamma rays with different energies. By dividing the net peak area by the relative efficiency for that energy, the different peaks are normalized to the same effective efficiency.

Each isotope's peak is thus multiplied by a decay time factor, divided by a branching ratio, and divided by a relative efficiency.

$$\text{Activity} = \frac{(\text{Net peak area})(\text{Decay time correction})}{(\text{Branching ratio})(\text{Relative efficiency})} \quad (1)$$

For the demonstration, we will use fuel assembly 0-444, which was discharged from the reactor on May 11, 1984. As of today, June 11, 1985, this assembly has cooled 396 days. The decay corrections are:

$$^{137}\text{Cs} = e^{(6.3 \times 10^{-5} \text{ d}^{-1}) * (396 \text{ days})}$$

$$= 1.0253$$

$$^{134}\text{Cs} = e^{(9.2 \times 10^{-4} \text{ d}^{-1}) * (396 \text{ days})}$$

$$= 1.4395$$

The branching ratios for two peaks of interest, namely, 796 keV for ^{134}Cs and 662 keV for ^{137}Cs are:

$$^{137}\text{Cs} = 0.851$$

*Half-lives for ^{137}Cs and ^{134}Cs are 30.17 and 2.062 y, respectively. Decay constants are determined from $\lambda = \ln 2/T_{1/2} = 0.693/T_{1/2}$ and expressed in inverse days.

$$^{134}\text{Cs} = 0.941 \quad .$$

Relative efficiencies and net peak areas will be determined during the demonstration. The equation for the ratio of the two cesium isotope can thus be reduced to

$$\begin{aligned} \frac{\text{Cesium 134 activity}}{\text{Cesium 137 activity}} &= \frac{(\text{Net Peak area} - 134)(\text{Relative Eff-137})}{(\text{Net peak area} - 137)(\text{Relative Eff-134})} \cdot \frac{1.4395}{1.0253} \cdot \frac{0.851}{0.941} \\ &= 1.2697 \cdot \frac{(\text{Net peak area} - 134)}{(\text{Net peak area} - 137)} \cdot \frac{(\text{Relative Eff} - 137)}{(\text{Relative Eff} - 134)} \quad . \end{aligned} \quad (2)$$

When this ratio is used with a calibration curve, as shown in Fig. 3, it can quickly give the burnup of the fuel at the axial position where the spectrum was taken. The calibration curve must be made from either an absolute measurement of this type of fuel or from relative measurements of this type on assemblies that have verified burnup values.

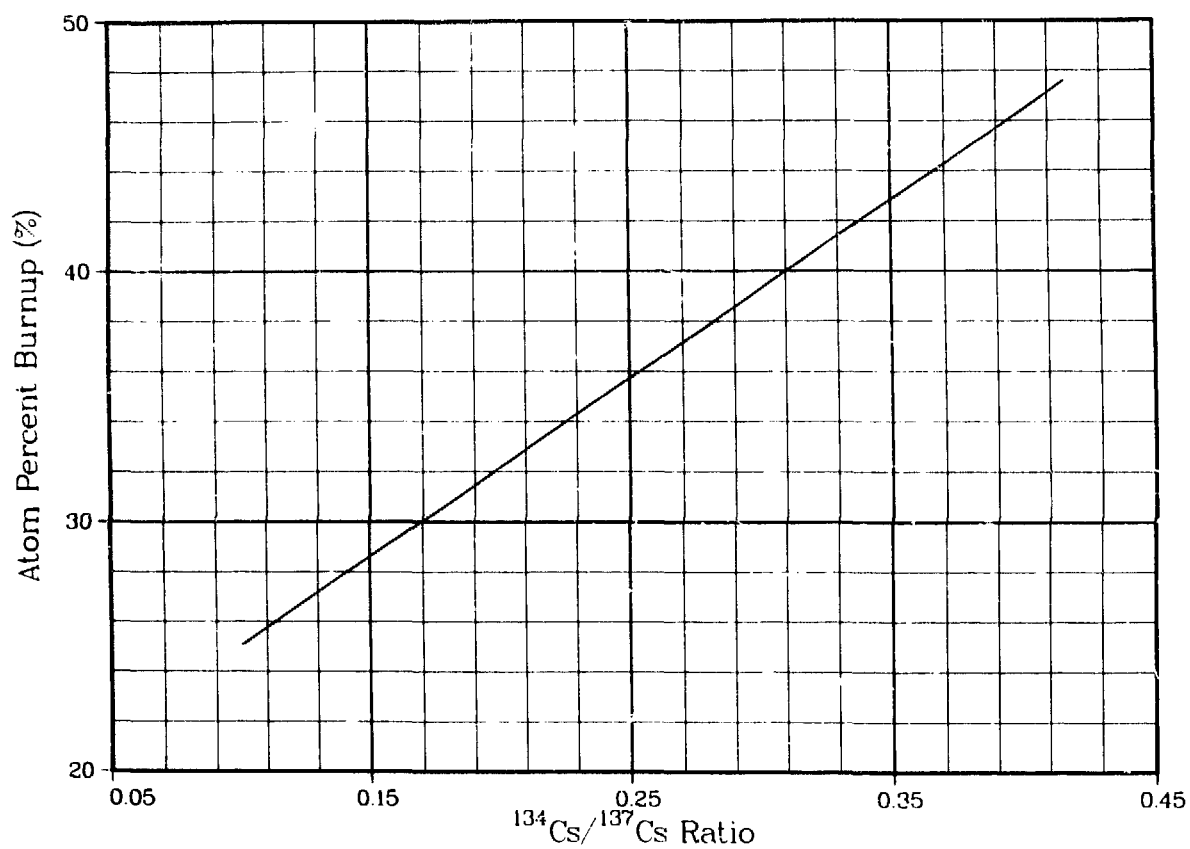
In summary, here are the steps we shall follow:

1. Obtain the spectrum on the multichannel analyzer;
2. Use the analyzer's built-in capabilities to find the net peak areas for the ^{134}Cs and ^{137}Cs peaks at 796 and 662 keV;
3. Substitute these areas into Eq. (2) along with relative efficiency values supplied by the instructor to find the activity ratio;
4. Use the calibration curve to find the atom per cent burnup from the activity ratio.

REFERENCES

- 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).
- J. R. Phillips, J. K. Halbig, D. M. Lee, S. E. Beach, J. R. Bement, E. Dermenjiev, K. Kaieda, and E. G. Medina, "Application of Nondestructive Gamma-Ray and Neutron Techniques for the Safeguarding of Irradiated Fuel Materials," Los Alamos Scientific Laboratory report LA-8212 (ISPO-77) (May 1980).
- J. R. Phillips, G. E. Bosler, J. K. Halbig, S. F. Klosterbuer, D. M. Lee, and H. O. Menlove, "Neutron Measurement Techniques for the Nondestructive Analysis of Irradiated Fuel Assemblies," Los Alamos National Laboratory report LA-9002-MS (ISPO-156) (November 1981).

SPENT-FUEL CALIBRATION CURVE



V. STATION 4: MEASUREMENT OF THE GROSS GAMMA ACTIVITY PROFILE OF AN MTR SPENT FUEL ASSEMBLY

We will give a brief overview of the problem and introduce you to an ion or neutron-detector electronics package (ION-1), an instrument that was designed to make rapid measurements of the gamma and neutron activity in the vicinity of a spent-fuel assembly. For those of you interested in more fully understanding the problem we invite you to ask for copies of papers that describe our measurements, calculations, and results in detail.

Safeguarding spent or irradiated reactor fuel using gross gamma-ray and neutron detection methods is a difficult problem. The quantities of specific isotopes in a spent-fuel assembly are a function of reactor type, initial enrichment, reactor operating history, and of course whether all the original elements are still in the assembly. The activities of these isotopes combine to give a gross gamma dose rate and gross neutron count rate. It is possible that the same gross gamma dose rate or neutron count rate can be obtained from two assemblies with different histories.

The measurement of gamma and neutron activities coming from an assembly provides a determination of gamma or neutron attributes of the assembly, that is, the assembly does or does not have gamma-ray or neutron activity. Unlike Cerenkov methods, which measure the light emission of the interaction of radiation in water, the ION-1 measures the primary radiation emitted from the assembly. Hence this primary attribute measurement is a higher level of verification than Cerenkov because any diversion scenario requires a dummy assembly to be composed of both gamma emitters and neutron emitters to confuse the interpretation of ION-1 data. To confuse the Cerenkov measurement, one need only make the dummy assembly from gamma emitters, for example, ^{60}Co rods.

Gross activity measurements have a more quantitative application at a spent-fuel storage pond where the fuel is typically of the same type and initial enrichment and has had essentially the same exposure history. In this case, the consistency of the operator declarations can be checked. The relative gross gamma and neutron activities can be measured and checked against those activities predicted by empirical or calculational studies of this type fuel. The use of the word consistency is prompted because an operator could declare shorter exposures and cooling times for all the assemblies in the pond to simulate a higher exposure and longer cooling time. While the quantity of individual isotopes would be significantly different in the two cases, the gross response change may not be detectable. In a diversion scenario that uses this type of false declaration, the declarations of the total pond inventory would have to be falsified in a consistent manner. This is a possible but demanding scenario.

An instrument has been developed at the Los Alamos National Laboratory that can make rapid relative gamma dose rate and neutron rate measurements. It is specifically designed to operate in the vicinity of a spent-fuel assembly in a fuel storage pond at a reactor. The gamma dose rate may range between 10 000 and 100 000 R/h. The instrument, ION-1, measures gamma dose by measuring the current in a gas-filled ion chamber. (Because of the high count rate, gamma-pulse counting is not practical). The dynamic range of this measurement spans 7 decades of detector response. When used with a ^{235}U -lined fission chamber (~135 mg) positioned 5-10 cm from a typical Westinghouse PWR assembly, neutrons are counted at rates of hundreds to thousands of counts/s. For the case of highly enriched fuel, very few neutrons are generated because of the

small amount of ^{238}U in the original fuel. Uranium-238 is the source material from which the neutron-producing isotopes are generated.

Unlike high-resolution gamma-ray spectroscopic (HRGS) measurements that can be used to determine how much of a specific isotope is in a sample, the gross gamma-ray measurement reveals only the dose rate resulting from all the isotopes. The advantage of the gross gamma measurement is the rapidity of the measurement (<10 s) compared to the time needed for HRGS measurements (~10-15 min). Because of their small size, the ION-1 detector and electronics are much less obtrusive and easier to use than the equipment required for HRGS measurements.

The two types of measurements complement each other. The ION-1 measurement can give a gross gamma profile measurement in a short time and hence a good statistical sample of fuel assemblies can be taken in only one day. Whereas the gross gamma-ray measurement alone can only give a consistency check among the assemblies, a single HRGS measurement on one assembly that was also measured with ION-1 will provide the calibration that will turn the relative ION-1 results into absolute measurements.

Today we will use ION-1 to obtain a gamma-dose rate profile of a MTR fuel assembly. (Because of the 93% original enrichment of the assembly we will not obtain very good statistics on the neutron count rates.) We will assign a burnup value to the assembly using the average activity obtained from the profile in conjunction with the burnup value obtained at a single point using the HRGS setup.

Equipment

1. Ion or neutron (ION-1) electronics package (see list of commands in Table III).
2. Ionization chamber.
3. Fission chamber (~140 mg ^{235}U) and preamplifiers.
4. Pipe for positioning ion and fission chambers.

Procedure

ALWAYS HAVE RADIATION PROTECTION PERSONNEL MONITOR GAMMA BACKGROUND DURING INITIAL SETUP.

1. Push INIT.
2. Because of the extreme sensitivity of ION-1 an electronics offset is required to establish the zero references of the different gain ranges. The first time ion chamber data are requested, you are prompted through the electronic offset sequence.
 - a. You are prompted to DISCONNECT IC CABLE. When the cable is disconnected, press NEXT to indicate you have completed the requested action and are ready to proceed to the next step.

- b. When NEXT is pushed the instrument starts to record the electronics offset on each of its eight ranges. The scale on which the offset is being taken is displayed in the liquid crystal displays (LCDs).
 - c. When all offsets are taken, you are prompted to RECONNECT IC CABLE and to push NEXT when that action is completed.)
 - d. Proceed through remaining initialization parameters, responding to questions asked by ION-1.
3. Push ACQ.

(If a background measurement has not been taken, a reminder NO BACKGROUND TAKEN YET appears. If you want to take a background now you must push END to exit the ACQ command, then BGND to initiate the background command. If you chose not to take a background now, simply push NEXT to proceed to the next step.)

4. Acquisition begins. During the acquisition the LCDs contain:

T = 308 N = 568
G = 2.979

TABLE III

ION-1 COMMANDS

When in the command response mode, all upper case keys are active.

ACQ	Acquire data for time specified.
PARM	Examine or change instrument parameters. The parameters included in the PARM command are count time in seconds, mode of accumulation, baud rate and software version.
BGND	Acquire background data for time specified.
LLD	Set the low-level discriminator of the neutron channel and display neutron count rate and total neutron counts with the present LLD setting.
GAIN	Select automatic or manual gain and display ion chamber reading with present gain.
VOLT	Display present values for high-voltage biases, battery voltage, and +5 and ±12 V supplies.
SCAN	Perform automated axial profile scan measurement.
INIT	Initialization of parameters.

where

T = count time remaining in seconds,

N = number of neutron counts from the fission chamber, and

G = present gross gamma signal from the ion chamber.

A question mark may appear after the gamma reading to signal that the instrument is searching for the optimum gain. When the best gain is found, the question mark is removed.

5. At the end of the acquisition the following information is available:

LCDs		Definition
ACQ STARTED AT 11:17 85-4-23	↓	(Time of start of acquisition)
GAIN = 163840	↑	(Gain for IC channel)
Automatic	↓	(Gain setting in automatic mode)
COUNTTIME = 30	↑	(Count time for this acquisition)
LLD = 30	↓	(Lower level discriminator setting)
GROSS NEUTRONS = 141.0 + -11.9 CTS	↑ ↓	(Total number of neutron measured during this acquisition)
GROSS GAMMAS = 0.0829 + -0.0001	↑ ↓	(Gross gamma reading)
NET NEUTRONS = 141.0 + -11.9 CTS	↑ ↓	(Net number of neutrons)
N = 0.4700 + -0.040 C/S	↑	Neutron count rate
G = 0.0829 + -0.0001		Gamma reading

The final two lines are displayed in the LCDs. You may move through the total page of results by pushing ↑ or ↓ as prompted. To do another acquisition push NEXT. To exit the acquisition command, push END.

6. Continue using ION-1 to measure the relative gamma dose rate at 10-cm intervals along the length of the MTR fuel element. A data sheet has been provided for your use. Plot the axial profile on the data sheet and determine the average burnup.

REFERENCE

J. K. Halbig and S. F. Klosterbuer, "The Inspector-Instrument Interface in Portable NDA Instrumentation," in Proceedings of 1981 IEEE Nuclear Science Symposium, San Francisco, California, October 21-23, 1981.

Axial Pos.	ION-1 Reading
0	
10	
20	
30	
40	
50	
60	
70	

② Sum _____

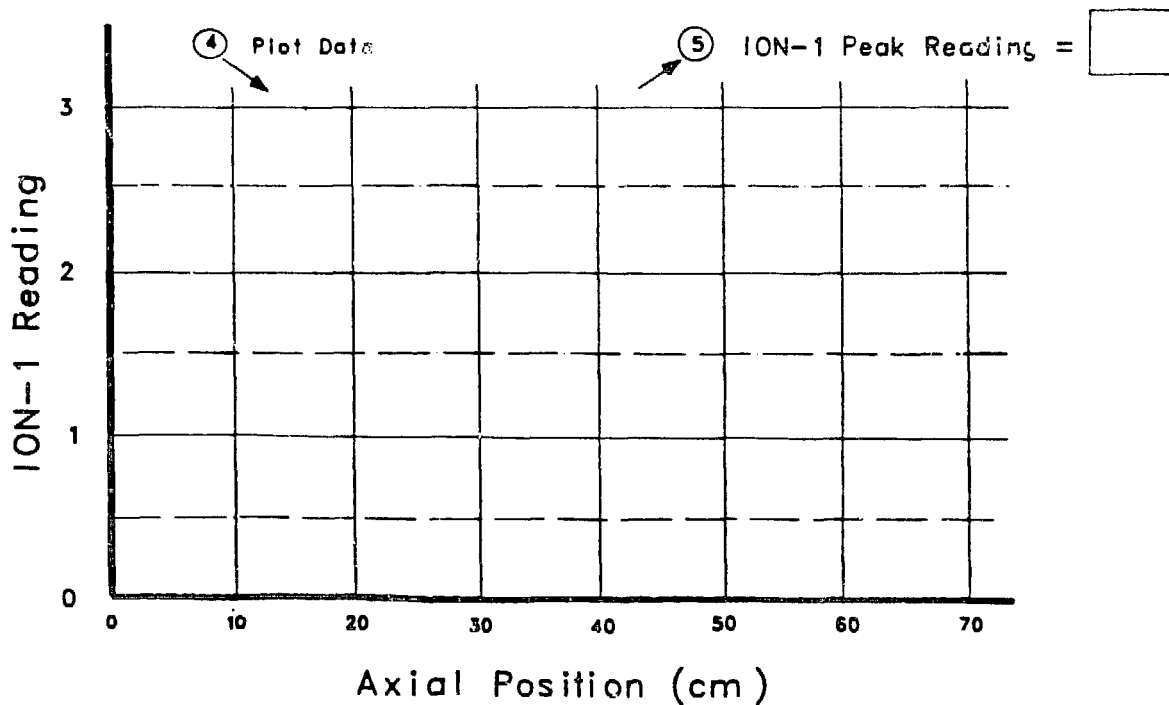
③ Ave. =

⑥ Ave. Burnup =
$$\frac{\text{Average Burnup}}{\text{Ave. ION-1 Reading}} = \frac{\text{Burnup at Peak}}{\text{ION-1 Peak Reading}}$$

⑦ Initial Loading = 220 gm

Fuel Remaining = $220(1 - \text{Ave. Burnup}/100)$

=





INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 17: PHYSICAL PROTECTION IN RELATION TO IAEA SAFEGUARDS

This session will review the general structure of the safeguards system, the SSAC interfaces, and physical protection principles, equipment, and techniques. In addition, the interactions between the state, the facility operator, and the IAEA will be described.

After the session, participants will be able to:

- Describe the role of physical protection in the state safeguards system.
- Understand the relationship of physical protection to the IAEA safeguards measures of material accounting and containment and surveillance.
- Understand the responsibilities of the state, the facility operator, and the IAEA in the application of these measures.
- Understand the basic physical protection principles, equipment, and techniques.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 17: PHYSICAL PROTECTION IN RELATION TO IAEA SAFEGUARDS

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I. INTRODUCTION

State systems of accounting for and control of nuclear material (SSAC) are a fundamental part of the state's and the IAEA's safeguards systems. This session describes how the part of the safeguards system that is operated directly by the IAEA in carrying out its verification activities, including field inspections, relates to the systems operated by the individual states. In particular, the physical protection function in the state system and the related equipment and techniques will be discussed. The differences between containment and surveillance (C/S), which is an integral part of the IAEA operational system, and physical protection, which is not, will be described.

II. GENERAL STRUCTURE OF THE SYSTEM

What is normally referred to as "IAEA safeguards" is based on the IAEA Statute and later documents that define the system in more explicit terms. The Statute provides that safeguards shall be established and administered so as to ensure, insofar as the Agency is able, that peaceful nuclear activities do not contribute to any military purpose. The states have direct responsibility for dealing with the possibility that para-military or guerrilla groups or terrorist organizations might divert safeguarded resources for weapon use.

A distinction must be made between the overall system of safeguards and the component parts that are operated respectively by the state and by the IAEA. Unauthorized actions by individuals and/or subnational groups are the concern of the police and internal-security authorities of the state; other concerns relate to the state itself. The system therefore operates on two levels:

1. The state ensures physical protection and regulates the handling, control, and accounting of nuclear materials.

2. The IAEA promotes high standards of physical protection and material accounting, and independently verifies the material accounting data provided under state direction.

Physical protection is a matter that involves the state's internal security, and hence an area where the state's sovereign authority is most carefully recognized and maintained. For that reason, the Agency's role in physical protection is limited to the promotion of guidelines and high standards, and the provision, upon request from the state, of training and advisory assistance. In spite of this limited role, physical protection as administered by the state must be considered a fundamental element of IAEA safeguards.

III. SSAC INTERFACES

The state system of accounting for and control of nuclear material (SSAC) consists of organizational arrangements on the national level to ensure the achievement of safeguards objectives, inter alia:

1. A national objective, to contribute to the deterrence, prevention, and detection of theft or unauthorized use of nuclear material by individuals or subnational groups.
2. An international objective, to provide the basis essential for the application of IAEA safeguards pursuant to the provisions of a safeguards agreement between the state and the IAEA.

The international component of an SSAC consists of three principal elements: (1) a legal framework within which the state exercises its control, (2) an organizational infrastructure at state and facility operator levels, and (3) a functional infrastructure at state and facility operator levels. The SSAC reports accounting data to the IAEA and its findings are subject to independent verification by the IAEA.

The two principal national functions of the SSAC are nuclear material accounting and control, and physical protection. Regulatory and licensing responsibilities are fundamental to each of these functions.

IV. NUCLEAR MATERIAL ACCOUNTING AND CONTROL

The objective of nuclear material accounting and control is to maintain current information on types, quantities, and locations of material. Essential

elements of nuclear material accounting are material measurements, record keeping, preparation and submission of accounting reports, and verification and analysis of these accounting data to determine correctness, accuracy of MUF, and evaluation of causes of MUF.

In the area of nuclear material accountancy, the state specifies the accountancy requirements, which are implemented by the facility operator. The accounting data are provided to the state for evaluation.

In nuclear material accounting there is a direct involvement of the IAEA. IAEA safeguards approaches are based on nuclear material accounting as a measure of fundamental importance, with containment and surveillance as important complementary measures.

Nuclear material accounting within the framework of IAEA safeguards begins with the material accounting activities which are undertaken by facility operators in response to requirements set by the SSAC, which may arise from agreements between the IAEA and the state. These activities and the corresponding accounting information generated are verified through independent IAEA verification and inspection. These efforts, after evaluation, provide one of the means of detecting diversion and of deterring diversion by the risk of early detection. They also provide the basis for statements regarding assurance that no diversion has occurred.

Nuclear material control within the framework of the state system is a set of measures for assigning responsibilities and maintaining awareness of the status and location of nuclear material.

V. CONTAINMENT AND SURVEILLANCE

Containment and surveillance (C/S), as used by the IAEA, provides information on movements of nuclear material and on the integrity of items (e.g., equipment, containers, etc.). C/S equipment includes seals, optical surveillance, and monitors. Those devices, supplied by the IAEA, are operated in an unattended mode for time intervals between IAEA inspections--often for periods of three months. Such unattended use places high importance on tamper resistance and reliability. In some cases C/S measures employed by the IAEA use the same

technology as that used in physical protection systems, although the function or purpose is different.

In the IAEA use of C/S equipment, there are interactions with the facility operator and possibly the state. In all cases, the facility operator must be assured that the C/S equipment will not affect or impair facility operations and safety. In some cases, the facility operator arranges for the installation of the C/S equipment.

C/S devices are utilized by the IAEA and the state and/or the facility operators. The distinction that must be recognized is the detection function of the IAEA and the detection and response function of the state/facility operators.

The use of optical surveillance provides a good example of this distinction. As a C/S measure, the IAEA uses unattended optical surveillance to monitor areas of safeguards interest during inspection intervals of 1-3 months. Review of the accumulated data during or subsequent to the inspection is accomplished to detect anomalies which are discussed and resolved with the state/facility operator. On the other hand, in the use of optical surveillance by facility operators/state authorities, the imagery is generally monitored in real or non-real time by security forces. Upon detection of an unauthorized event, these forces have the authority and capability to act upon the information in a manner intended to deter or prevent the event. Because of this distinction, in this paper C/S is covered only as an IAEA safeguards measure.

VI. PHYSICAL PROTECTION

Physical protection measures are employed by states and facility operators to prevent sabotage of nuclear facilities and unauthorized removal of nuclear materials. The functions of physical protection systems are to control authorized and unauthorized personnel, restrict access to nuclear material, and provide rapid and comprehensive response to locate and recover missing material. The physical protection systems are designed to provide an effective balance between the basic elements of detection, delay, and response.

In the United States, the Nuclear Regulatory Commission (NRC) specifies the required physical protection measures to be applied to commercial nuclear

facilities. The facility operator installs and operates the physical protection system. The NRC periodically inspects the facilities, insuring that specified physical protection measures are maintained on a continuing basis and that regulations and standards are complied with. The entire system is governed by a set of regulations to which all facilities must adhere in order to obtain and retain an operating license. In the case of government owned facilities, the Department of Energy (DOE) performs functions similar to the NRC.

In general, then, the flow of physical protection responsibilities is from the state to the facility, commercial or state owned; the state specifies the required physical protection systems, and the facility operator installs and operates the systems, including response actions as required. This general flow of responsibility is typical for most IAEA member states which engage in nuclear activities.

The IAEA has no responsibility for the provision of a state's physical protection system or for the supervision, control, or operation of such a system. However, in order to promote uniformly high standards for the protection of nuclear material, the IAEA provides recommendations in INFCIRC/225 on the requirements for physical protection of nuclear material in use, transit, and storage. Also, at the request of the state, the IAEA provides advice to state authorities with respect to their physical protection systems. The specific physical protection measures which are applied to a particular facility are determined by the state. That determination is based on factors specific to the state, including threat perception, economics, and political infrastructure.

To further the goal of achieving effective physical protection, the IAEA and the US DOE sponsor a comprehensive international training course on physical protection, which is conducted at Sandia National Laboratories on a regular basis. The attendees, which include personnel from the developing nations, are given comprehensive information on physical protection principles, techniques, and equipment. Workshops are held to provide participants with experience in design of comprehensive physical protection systems for various types of nuclear facilities.

VII. SUMMARY OF SYSTEM INTERACTIONS

The physical protection, material accounting and C/S measures, responsibilities, and interactions that have been discussed are summarized in Figure 1. The principal elements of the SSAC are material accounting and physical protection.

In material accounting, the IAEA provides recommendations and advice to the state on the structure and format of the accounting system. The state specifies, regulates, and inspects the accounting system used by the state or commercial facility operator. The operator implements the system, providing the accounting data to the state. Finally, the state supplies the accounting data to the IAEA for independent verification. This verification is achieved through review of the state supplied data, together with inspections to compare data, perform independent measurements, and to review the complementary information from the C/S equipment. From all of these operations, it can be seen that material accounting involves strong interaction between the IAEA, the state, and the facility operator.

Physical protection presents a different picture. The state and facility operator interactions are essentially the same as for material accounting. In physical protection, however, the state or facility operator have no responsibility to the IAEA. Related data is kept within the state. However, upon request the IAEA will assist the state with physical protection system recommendations and advice.

The main functions of the physical protection system are to detect, assess, delay, and respond. In some cases, data from the physical protection system is used to complement the material accounting system in a manner similar to the C/S measures used by the IAEA. Should the physical protection system fail, the material accounting system is designed to identify the type and quantity of material missing.

VIII. PHYSICAL PROTECTION EQUIPMENT AND TECHNIQUES

Physical protection is realized through the use of a variety of equipment and techniques. For approximately ten years, Sandia National Laboratories has had a comprehensive program to develop and test a wide variety of physical protection equipment. Principal equipment and techniques included in that program are:

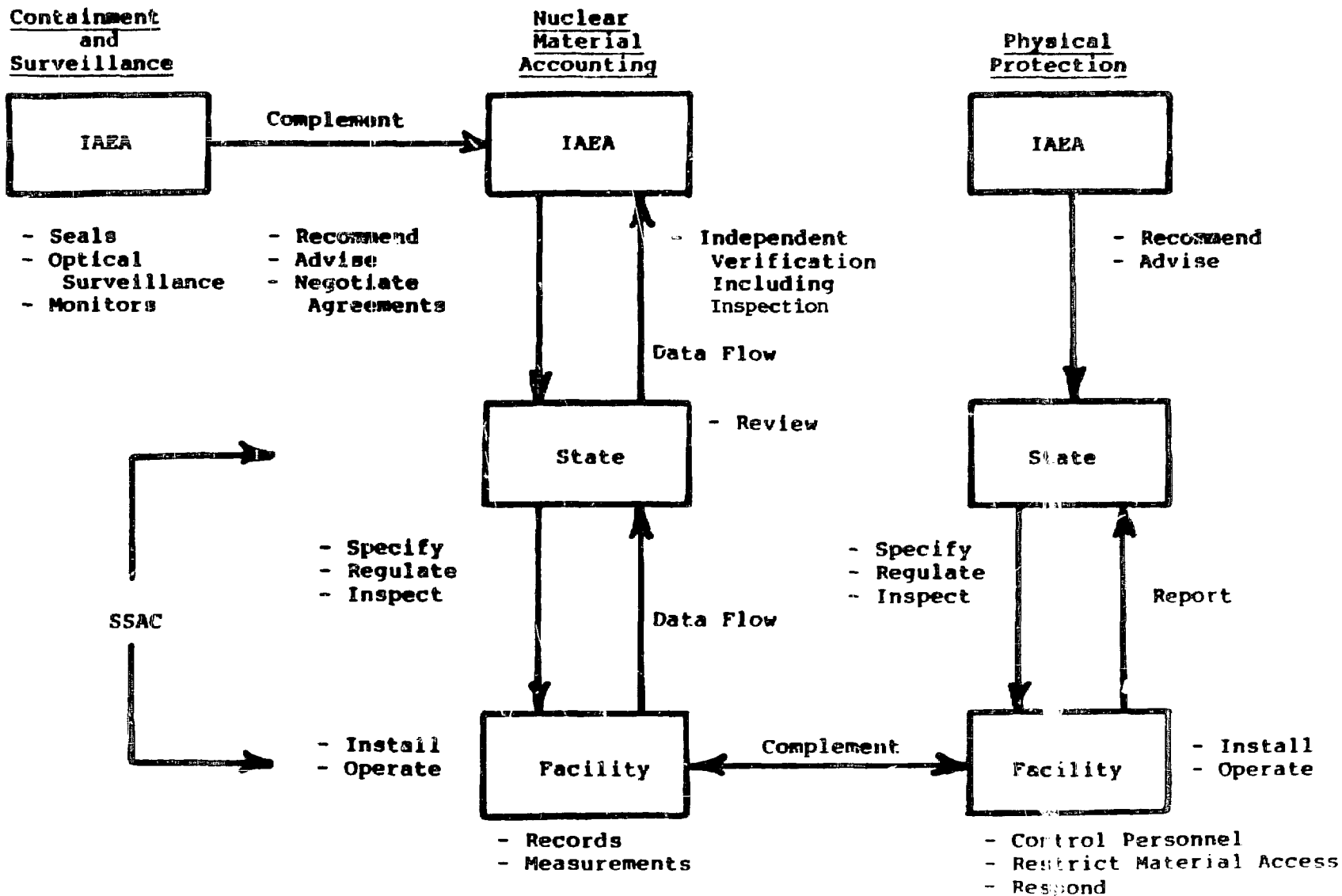


Figure 1.
The Safeguards System Structure.

1. Exterior intrusion detectors - Used to provide an indication of entry into restricted exterior areas. Technologies include microwave, electric field disturbance, active infrared (beam breakers), buried line (seismic, magnetic, balanced pressure, ported coaxial cable), and taut wire.
2. CCTV equipment - Used to provide continuous coverage of interior and exterior areas of safeguards interest and used to assess the causes of alarms generated from physical protection equipment.
3. Portal monitors - Used to identify personnel and to detect unauthorized movement of shielded and unshielded nuclear material through passageways of safeguards interest. Technologies include neutron detection, gamma detection, metal detection, and personnel identification (see item 6 below).
4. Interior intrusion detectors - Used to provide an indication of entry into restricted interior areas. Technologies include microwave, passive infrared, sonic, ultrasonic, balanced magnetic, continuity, capacitance, strain, and pressure.
5. Credential verification - Used to verify credentials. Technologies include photograph, tuned-loop, and magnetic code.
6. Personnel identification - Used to identify individuals. Technologies include fingerprint, voice, hand geometry, palm print, signature recognition, passwords, access codes, and photo matching.
7. Seals - Used to provide a positive indication of entry into a secured area or containers. Technologies include adhesive paper, metallic, fiber optic, and electronic.
8. Electromechanical devices - Used to prevent unauthorized use of facility equipment which is necessary to move or control nuclear material. Technologies include power interruption and valve actuation.
9. Passive and active barriers - Used to prevent or delay unauthorized entry into facility areas of safeguards interest. Passive barrier technologies include walls, fences, gratings, vaults, locking mechanisms, tiedowns, containers, and hardened doors. Aqueous foam and remotely controlled doors are the principle active barrier technologies. Active barriers are used to obstruct, on command, normal access areas.
10. Obscurants - Used to prevent or delay unauthorized entry into facility areas of safeguards interest. Smoke and aqueous foam are the principal obscurant technologies.

11. Guard force - Used to monitor personnel and equipment movement within the facility, and to assess and respond to safeguards alarms. Principal efforts involve equipment development and evaluation, and training.

A comprehensive physical protection system could use many of these types of equipment, with the specific technology being chosen to match the particular use-environment. Such a system must be carefully integrated to provide effective monitoring, alarm assessment, and response initiation. These functions are generally performed with display, control, and communications equipment located in a guard force control center.

IX. SUMMARY

In this session, physical protection, nuclear material accounting and control, and containment and surveillance have been discussed, with emphasis on the interactions of these measures within the context of IAEA safeguards. In addition, the current physical protection equipment and techniques have been reviewed.

The interactions can be summarized as follows. Although physical protection is a fundamental element of IAEA safeguards, it is solely a state/facility operator responsibility. While the IAEA has an interest in promoting the implementation of effective physical protection systems, it serves only in an advisory capacity.

Nuclear material accounting directly involves the state, facility operator, and the IAEA. Facility records and reports provided by the state are independently verified by the IAEA. The SSAC is of fundamental importance in this process.

Containment and surveillance measures are used by the IAEA. Installation and routine use of C/S equipment must be approved by the state and facility operator, and must not affect facility operations or safety.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 18: MATERIAL CONTROL AND ACCOUNTING AT A RESEARCH REACTOR

The session will describe the implementation of a state system of accounting and control (SSAC) at research reactors and critical facilities. Material accounting and control requirements, operating practice, staffing, and resource problems will be discussed. Emphasis will be placed on practical experience in actual operating facilities. The lecture will discuss current problems encountered at state and facility levels, and applications of advanced technology to help solve the problems.

After the session, participants will be able to:

- Describe typical research reactor and critical facility characteristics and the material control and accounting requirements of such facilities.
- Better appreciate operational practice, staffing, and resources needs for implementing an SSAC at research reactors and critical facilities.
- Compare actual facility SSAC operational performance with general SSAC requirements.
- Recognize current problems encountered at state and facility levels and have an appreciation of advanced technology that may help solve the problems.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 18: MATERIAL ACCOUNTANCY AND CONTROL FOR RESEARCH REACTORS AND CRITICAL ASSEMBLIES

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I. INTRODUCTION

The security of a community may be affected by the explosion of nuclear devices, dispersion of nuclear materials, and sabotage of nuclear facilities. Any of these events may have serious international consequences. In this context, each State party to the Non-Proliferation Treaty (NPT) is responsible for protecting nuclear material in peaceful use from theft or from diversion for production of nuclear explosive devices.

In order to fulfill this responsibility, three basic measures are employed:

- Nuclear material accountancy
- Nuclear material control
- Physical protection of nuclear material

While the first two components provide countermeasures to covert actions taken by unauthorized persons, the third provides countermeasures to overt actions. A combination of the first two measures is known as the State System of Accounting and Control (SSAC), and a combination of all of the three measures is called Domestic Safeguards.

This lecture is organized to discuss the SSAC for research reactors and critical assemblies based on SSAC operating experience.

II. FACTORS UNDERLYING SAFEGUARDS CONSIDERATIONS

In a general sense, attractiveness, accessibility, and concealability are major factors underlying safeguards considerations.

A. Attractiveness

As far as nuclear explosive devices are concerned, the following two considerations are essential in the examination of attractiveness. First is the total amount of nuclear material available at a facility, because a certain threshold amount* is necessary to manufacture a single nuclear explosive device. The second consideration is the technical complexity for upgrading the nuclear material available at the facility to nuclear weapons grade, which means highly enriched uranium metal or clean plutonium metal.

Usually the following three processes, single or combined, are employed for upgrading nuclear material to weapons grade.

*Not clearly defined yet, but tentatively 8 kg for plutonium and 25 kg for highly enriched uranium.

- Chemical processes, for separating individual elements and converting them into metallic form.
- Reprocessing, for chemically separating fissile materials contained in spent fuel.
- Enrichment, for physically separating fissile and nonfissile isotopes. In principle, gas diffusion, centrifuge, nozzle, or laser separation techniques are involved.

Concerning the relative complexity of these processes, it is commonly understood that chemical processes involve the simplest technologies, enrichment involves the most complex technologies, and reprocessing is in between. In this context, highly enriched uranium metal fuel and clean plutonium metal fuel are top ranking in attractiveness because no upgrading is necessary. Fresh fuel elements of either highly enriched uranium alloys or mixed oxide are also very high ranking in attractiveness because a simple chemical separation process can upgrade them into weapons grade material. Low-enriched uranium fuel (less than 20%) ranks lowest in attractiveness because a complex enrichment process is inevitable for upgrading it to weapons grade material.

For radiation hazards induced by dispersion of nuclear material, rather different features will be taken into consideration for evaluating attractiveness. For example, highly enriched uranium ranks low, while plutonium ranks very high due to its toxicity.

B. Accessibility

In the scenario of covert removal of nuclear material for unauthorized purposes, access to nuclear material is the first premise. Accessibility to nuclear fuel is influenced by various factors: radioactivity of the fuel (associated mainly with fission products), location of the fuel, and mode of operation of the facility. Access to irradiated fuel containing significant amounts of fission products, while possible, could be lethal. According to investigations done by Oak Ridge National Laboratory, radiation risks may be rated as follows:

- High rating: radiation level greater than 10,000 R/h; about 50% lethal for a few minutes exposure.
- Medium rating: radiation level lower than 10,000/h but greater than 10 R/h; significant physiological effects for a 1-day exposure.
- Low rating: radiation level lower than 10 R/h.

Accessibility of nuclear material depends on its location in use or in storage. Barriers that must be cleared for gaining access to the location of the material will affect the amount of time needed. For example, it is almost impossible to access fuel elements loaded in reactors under high power operation. However, it is easy to access fresh fuel stored in a storage vault.

The mode of operation of a facility is also an important factor. Nuclear material is more accessible in facilities that have frequent fuel movement than in facilities whose movement of nuclear fuel is almost frozen during the entire period of operation. Also, fuel material that is moved by machine is significantly less accessible than fuel moved manually.

C. Concealment

Concealment means any covert actions taken to escape from detection by authorities. These actions include tampering with surveillance devices

falsifying or manipulating records, and replacing nuclear material with material of lower strategic value (that is, dummy material). Not so difficult may be the concealment of a theft of nuclear fuel when the fuel consists of a large number of small pieces that have low radiation levels. However, the fuel subassemblies of power reactors and research reactors are large enough to be difficult to conceal. Furthermore, removal of an irradiated subassembly requires heavily shielded containers to protect the handler from radiation hazards. Such a removal requires the assistance of heavy duty equipment like cranes, tracks, and off-loading devices, and therefore concealment would be extremely difficult. Generally speaking, the smaller the unit, the easier its concealment.

III. REACTOR STRUCTURE

"Reactor" means any device in which a controlled self-sustaining fission chain reaction can be maintained. What the reactor looks like may be visible in the word "pile," named by E. Fermi for his first device that achieved a self-sustaining fission chain reaction. It means piling on small elements of nuclear material and moderator material in an appropriate ratio. Then a self-sustaining chain reaction can be achieved automatically when the pile reaches a certain volume, namely, a critical volume. If the volume of the pile exceeds the critical volume, the fission rate in the pile increases with a certain acceleration rate. This situation is known as supercritical and the opposite situation is called subcritical. The combination of fuel and moderator make up the most essential component of a reactor, known as the reactor core.

A research reactor is a reactor used for basic or applied research or for training. The fission heat is generally removed by coolant but is not used as a heat source. A critical assembly is known as a zero-power reactor. It is distinguishable from a research reactor in that it normally has no special provision for cooling, is not shielded for high-power operation, has a core designed for great flexibility of arrangement, and uses fuel in a readily accessible form that is frequently repositioned and varied to investigate various reactor concepts.

Research reactors as well as critical assemblies in use today are of great variety, in that most are custom made. It is outside of the scope of this lecture to scrutinize each type of research reactor, so I am going to illustrate only some representative types of research reactors and critical assemblies, focusing on safeguards interests.

A. Material (Engineering) Test Reactor

Material test reactors are used to provide very high neutron flux (about 5×10^{14} n/s/cm²) for irradiation test of reactor fuels and various materials. Their operating power is between 30 and 250 MW, depending on core design. Each fuel assembly contains about 20 uranium-aluminum fuel plates clad with aluminum. Usually, highly enriched uranium of 93% is used as fuel. A typical fuel subassembly contains about 0.5 kg of ²³⁵U.

The reactor core consists of several tens of subassemblies leading to a few tens of kilograms of ²³⁵U in the core. The core is usually equipped with an in-core test loop in which irradiation experiments are conducted with a variety of monitoring equipment. The core is contained in a large pressure vessel, but the pressure is not as high as that of power reactors. The pressure vessel is completely surrounded by a concrete shield or by water for radiation shielding.

Refueling is rather frequent, leading to the need for storage of about a hundred fresh fuel subassemblies in storage vaults (for assuring continuity of operation) and for storage of about 300 spent fuel subassemblies in a storage pond on the reactor site. This means that a considerable amount of highly enriched uranium exists outside the core.

Taking into account the mode of operation, the location of the fuel subassemblies, the total amount of fuel, the enrichment, size, and weight of the subassemblies, the total number of subassemblies and the radiation level, the ranking of attractiveness, accessibility, and concealability is considered to be:

	<u>Attractiveness</u>	<u>Accessibility</u>	<u>Concealability</u>
Fresh fuel	High	High	Medium
Fuel in core	Medium	Low	Low
Irradiated fuel	Medium	Medium	Low

The capability of clandestine plutonium production using the high neutron flux available in the reactor is also to be examined.

B. CP-5-Type Reactor

The CP-5-type reactors are heavy-water-moderated research reactors used to provide neutrons for a variety of basic research purposes. Their operating power is about 5-10 MWt to provide a thermal neutron flux of about 5×10^{13} and a fast flux of about 5×10^{12} n/s/cm². The fuel subassemblies contain three concentric cylinders of uranium-aluminum alloy with a uranium weight per cent of 15 to 22. Usually, highly enriched uranium of 93% is used and one subassembly contains 0.2 kg of ²³⁵U. The reactor core consists of about 20 fuel subassemblies containing about 4 kg of ²³⁵U. The core is located in a closed cylindrical tank filled with heavy water and surrounded by a biological shield of concrete.

Refueling is rather infrequent, for example, once every 3 years. With on-site storage capability of 10 fresh fuel subassemblies and 20 spent fuel subassemblies, one can assure continuity of reactor operation. The total fuel inventory, including fresh fuel, irradiated fuel, and fuel in the core, is about 10 kg of ²³⁵U. While the rankings of accessibility and concealability are nearly the same as those for the material test reactor, the ranking of attractiveness is low because of the small inventory of fuel.

C. Pool-Type Reactor

Pool-type reactors are simple low-power reactors designed for use as aids in reactor physics education courses and to provide a convenient research tool. The reactor core is compact and is suspended in an open pool of water that serves as a neutron moderator, coolant, and radiation shield. This arrangement provides considerable flexibility since the fuel and experimental apparatus can easily be positioned from the top of the water pool.

Usually the operating power is about 1 MW to provide a neutron flux of about 3×10^{13} n/s/cm² for thermal neutrons and 1×10^{12} for fast neutrons. The typical fuel subassembly consists of about 10 fuel plates made of uranium-aluminum alloy. One subassembly contains about 0.2 kg of ²³⁵U, and about 10 subassemblies make up a core. There is almost no fuel turnover so that there is no spent fuel storage pond. For some reactors, the reactor pool is used for

storing spent fuel, when it is needed. Only a few fresh fuel subassemblies are stored on-site. The fuel inventory is about 5 kg of ^{235}U , including fresh- and irradiated-fuel subassemblies.

D. Critical Assembly

A critical assembly, also known as a zero-power reactor, is a physical model of a specific type of power reactor expected to be built. It simulates the composition of the reactor constituent materials and geometry that will affect the neutron spectrum and spatial distribution. Critical assemblies are used for optimization studies of the core composition, geometrical arrangements, and core parameters.

Critical assemblies have the following specific features that distinguish them from research reactors:

- The operating power is very low, such that most operations are conducted at 10-100 W and, at most, 2 kW for short-term operation.
- No special cooling provision is provided.
- The core is designed to be accessible for arrangement by hand, providing great flexibility. The core is a simple frame structure of racks for temporary loading of fuel subassemblies that compose the reactor core. The induced activity is usually kept low to permit access.
- Fuel elements consist of a large number (a few thousands to a few tens of thousands) of small pieces of plates or pins of nuclear material in a form that is readily accessible by hand. For loading and unloading convenience, normally 10 to 50 fuel plates or pins are assembled together in an appropriate case to form a fuel subassembly. The fuel subassemblies are frequently varied or repositioned by hand for investigating a variety of reactor compositions and geometries.
- Loading and unloading fuel subassemblies into and out of the core are carried out by hand, with or without simple tools.
- Radiation shielding does not enclose the assembly, but the assembly is located in a sealed, shock-resistant, concrete building.

Accessibility and concealability are both high ranking factors for critical assemblies. In many cases, the fuel plates (or pins) consist of highly enriched uranium and plutonium and the inventory is large. Consequently, critical assemblies are among the most difficult facilities to safeguard.

IV. SPECIFIC FEATURES OF RESEARCH REACTORS AND CRITICAL ASSEMBLIES

The modes of operation and techniques employed for accounting and control of nuclear material in research reactors and critical assemblies differ in many respects from those used in bulk-handling facilities, such as reprocessing plants and fuel fabrication plants. Some specific features of these facilities that may influence the design of a system of accounting and control will be discussed in this section.

A. Itemized Facility

Generally, reactor fuel is solid, except for a fluid fuel reactor like a molten salt reactor. The fuel may be metallic uranium, an alloy of uranium or plutonium, or a ceramic such as an oxide. The solid fuel is shaped into plates, pellets, or pins and grouped into units called fuel elements. The

dimensions of these fuel elements depend on the purpose and design of the reactor. As a protective measure, all fuel elements are clad with stainless steel, zirconium alloys, or aluminum to prevent the release of fuel or fission products, as well as to prevent direct contact with the coolant. The cladding also serves as a part of the structure of the fuel element.

For handling convenience, 10 to 50 elements are normally assembled together in a square or a pseudo-circular cross section by means of a grid structure to form a fuel subassembly. For quality control and production control purposes, identification marks and numbers are usually engraved on the surfaces of the fuel elements and fuel subassemblies during their manufacture in the fuel fabrication plant. It is quite unusual to breach fuel cladding or to disassemble elements at the reactor site. In this context, a reactor or critical assembly can be dealt with as an itemized facility, in contrast to a bulk facility.

B. Laboratory-Use Material

A great variety of nuclear materials are used at research reactor and critical assembly facilities. In addition to reactor fuels, many other types of nuclear materials are used, such as fission detectors, experimental samples for reactivity measurements, and specimens for irradiations, etc. These nuclear materials used in the laboratory are usually very small in quantity, but have a wide variety of physical and chemical forms. These laboratory nuclear materials are frequently encapsulated, canned, or fabricated; are loaded into or out of research reactors; and are frequently transferred to and from various laboratories for investigation. Because of these features, the laboratory nuclear materials are usually controlled under a separate arrangement from reactor fuels.

C. High-Attractiveness Ranking of Fuel

A nuclear reactor must be fueled with fissionable materials, whether it is designed to be a power reactor, research reactor, or critical assembly. The distinguishing difference between nuclear fuels used in commercial power reactors and research reactors is the higher attractiveness ranking of fuel used in research reactors. The fresh fuel for today's commercial power reactors is low-enriched uranium. Spent fuel from power reactors contains plutonium produced from the ^{238}U during neutron irradiation in the reactor core. But plutonium is accompanied by highly radioactive fission products while the spent fuel is at the reactor site.

Unlike power reactors, research reactors and critical assemblies need to use the highly enriched uranium or plutonium contained in fresh fuel elements to provide:

- High neutron flux in a convenient and economic way
- Versatile use of the research reactor
- Mockup experiments of more advanced reactor concepts utilizing plutonium as a converter or breeder

Frequent verification of the fuel inventory is necessary for timely detection of missing fuel, namely, detection within the time required to convert different forms of nuclear material to weapons grade.

D. MBA and KMP Structures

Nuclear material accounting and control must be implemented, based on the structure of material balance areas (MBA) and key measurement points (KMP). Typical KMP and MBA structures employed in research reactors and critical assemblies are shown in Fig. 1. As mentioned in Sec. IV.B, two different groups of nuclear material are used in research reactors and critical assemblies: reactor fuel, which is a major part of the inventory, and laboratory material, which is a very minor part of the inventory. These two groups are assigned to different MBAs for convenience: a reactor-fuel MBA and a laboratory-use MBA.

As shown in Fig. 1, while each reactor facility usually consists of a single independent MBA, laboratory nuclear material used at different facilities can be combined into a single MBA because of its small quantity. In the generalized model of a reactor complex, shown in Fig. 1, the KMPs include: a reactor KMP, a fresh fuel storage KMP, a spent fuel storage KMP, and KMPs in other locations, including workrooms and service areas.

E. Nonroutine Mode of Operation

Research reactors and critical assemblies are not production plants, and usually they operate in a nonroutine mode that is dependent on experimental schedules. In this regard, an inspection that requires a facility shutdown is

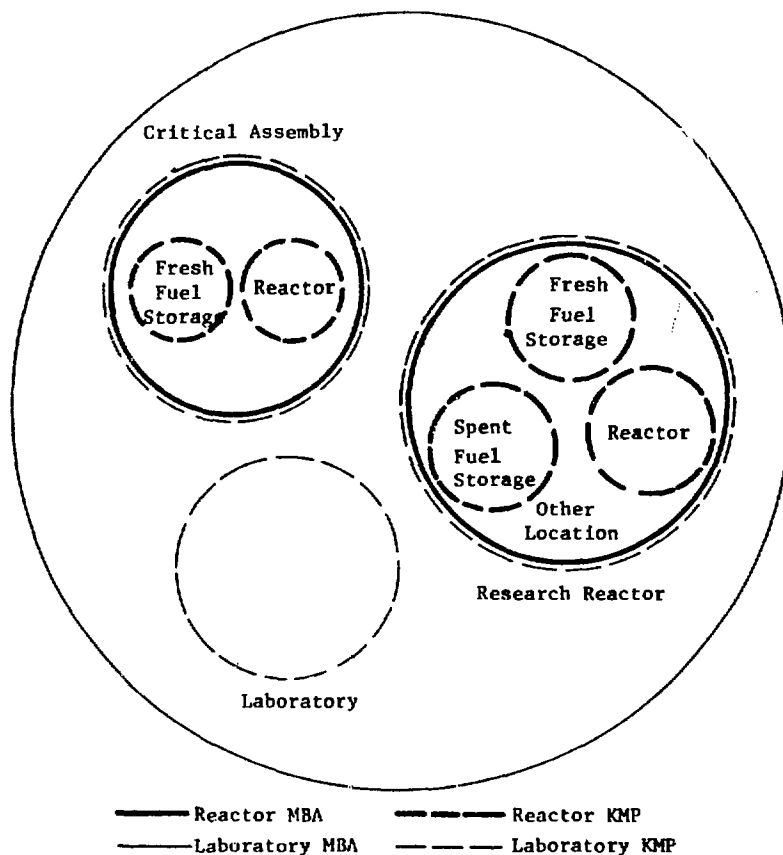


Fig. 1. MBA and KMP structures in research reactors and critical assemblies.

not as serious as it is in commercial power plants, if the timing of the inspection is appropriately coordinated with the experimental schedule. Otherwise, the inspection may lead to disruption of essential experimental conditions.

F. Low Frequency of Fuel Transfer

At reactors, fuel usually stays in the same location for a long time, whereas at reprocessing and fabrication plants, material usually flows through the plant within a few days. Therefore, at research reactors and critical assemblies, containment and surveillance can be used to alleviate the burden associated with material accounting and control.

V. REQUIREMENTS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIAL

The State is required to establish and maintain a State System of Accounting for and Control (SSAC) of nuclear materials conforming to the Safeguards Agreement pursuant to the NPT. "Accounting for nuclear material" is interpreted as the activities carried out to establish the quantities of nuclear material within defined environments and the changes in those quantities. The essential elements of nuclear material accounting are material measurement, record keeping, preparation and submission of accounting reports, and verification and analysis of these accounting data to determine correctness, accuracy of MUF, and evaluation of the cause of MUF.

Control of nuclear material is interpreted as administrative or management control over the possession, use and transfer of nuclear material and control over the organization, function and performance of a nuclear material accounting system within the State.

A. Underlying Fabric of the SSAC

The International Atomic Energy Agency (IAEA) must apply its safeguards, in accordance with the terms of the Safeguards Agreement, in such a manner as to enable it to verify the findings of the SSAC (ascertaining that there has been no diversion of nuclear material from peaceful uses to nuclear weapons or other nuclear explosive devices). The IAEA, in its verification, must take due account of the technical effectiveness of the SSAC and avoid unnecessary duplication of the State activities (Provisions 7 and 31, INFCIRC-153).

Considering the IAEA's limited financial resources and qualified personnel, the IAEA's objective of early detection of diversion is a significant challenge that would be extremely difficult to fulfill without high-level SSACs leading to the implementation and maintenance of efficient accounting and control procedures by each facility operator. Therefore, a structured, effective program of accounting and control at the facility level is essential.

In this context, the SSAC will consist of three main activities:

- Facility accounting and control activity in compliance with the SSAC
- State inspection activity
- Information audit activity

B. General Requirements and Their Simplification for Research Reactors and Critical Assemblies

The IAEA's Advisory Group meeting, held at Brno in July 1975, recommended general requirements for the SSACs. I think you have already learned through other sessions in this course about facility-independent SSAC features: authority and responsibility for the establishment, implementation, and

maintenance of the SSAC, and elements of the SSAC that consist of promulgation and review of regulations, licensing, information systems, categorization of nuclear materials, control of compliance, sanctions, and technical support.

To avoid duplication, this section focuses on discussion of the requirements of SSAC implementation in research reactors and critical assemblies, based on my experience in Japan.

As seen in the foregoing sections, research reactors and critical assemblies are characterized as itemized facilities. In such facilities, accountancy of nuclear material can be simplified by the use of item counting and containment and surveillance measures. Inspection can be based on attribute sampling and identification of items. In the following discussions, therefore, the general requirements on accounting and control at the facility level will be summarized, focusing on their simplification and taking into account specific features of research reactors and critical assemblies.

Requirement 1: The facility operator should set forth the organizational unit responsible for material control and accountancy (MCA) at the facility.

Comment: Responsibility for MCA must be assigned to a single designated individual. For item facilities, there are no bulk materials and few measurement control programs, so that one individual may be designated as responsible for MCA and for reactor operation and maintenance.

Requirement 2: The facility operator should provide information on facility design and operation in sufficient detail to permit evaluation of the adequacy of the MCA.

Comment: Information must be provided for detailed design of

- the structure of inventory and flow KMPs,
- item/batch identification and counting procedures,
- containment and surveillance measures, and
- procedures for estimating nuclear loss and production in normal reactor operation, and for estimating nuclear production capability.

Requirement 3: The facility operator should establish an MBA and KMP structure.

Comment: It is convenient to establish two MBAs for a research reactor or critical assembly: one for reactor fuel and the other for laboratory material such as samples, foils, and fission chambers. While one facility consists of one MBA for reactor fuel, the laboratory-use MBA can include laboratory nuclear material in other facilities as shown in Fig. 1. For designing inventory KMPs, the following four KMPs are usually considered:

- Fresh fuel storage
- Reactor core
- Spent fuel storage
- Other locations

The "other locations" KMP is for temporary inventory, such as irradiated fuel contained in a shielded transport cask awaiting shipment from the facility.

Requirement 4: The facility operator should provide a measurement system for determination of the quantities of nuclear material received, produced, shipped, lost, or otherwise removed from inventory.

Comment: Since research reactors and critical assembly inventories generally have a very slow turnover, shipping and receiving to and from the MBA is infrequent and its verification procedure is also simple. Nuclear fuel is usually received into the MBA on the basis of shipper's data measured by a fuel manufacturer. During transportation from the manufacturer to the facility, a seal system is applied for assuring the quality and quantity of the fuel. Usually, no direct measurements take place at the facility. Consequently, the shipper's data provide the facility with the basis of the MCA during the fuel's stay at the facility. Nuclear production and loss are estimated on a calculational basis using reactor operation parameters. For internal transfer of nuclear fuel inside the MBA, item identification and counting is used for MCA.

Requirement 5: The facility operator should establish procedures for evaluating the precision and accuracy of measurement and estimations of measurement uncertainty.

Comment: Not applied for research reactors and critical assemblies because they are item facilities.

Requirement 6: The facility operator should establish procedures for identifying, reviewing, and evaluating differences in shipper/receiver measurements.

Comment: Usually the shipper/receiver differences observed for research reactors and critical assemblies are differences in item counting.

Requirement 7: The facility operator should establish procedures for physical inventory taking (PIT).

Comment: PIT for research reactors and critical assemblies can be based on attribute sampling, and a statement of the MUF can be made on the basis of item counting. In this context, item identification procedures should be established. Duplication of identification numbers engraved on the surface of irradiated fuel elements, although possible, would be difficult. Counting of items in containers can be skipped if such containers can be uniquely identified and their integrity can be ensured. PIT usually occurs once or twice each year.

Requirement 8: The facility operator should establish procedures for the use of containment and surveillance measures.

Comment: Containment and surveillance measures can be applied more effectively in itemized facilities than in bulk facilities for alleviating the burden and intrusiveness caused by implementation of the MCA. In general, three different types of surveillance equipment are now available:

- Seals or intrusion detectors to ensure integrity of containment
- Doorway monitors to detect removal of nuclear material
- Film camera or CCTV with video recorder to store pictures for subsequent viewing

Requirement 9: The facility operator should establish procedures for detection of unmeasured inventory/loss.

Comment: Not needed for itemized facilities.

Requirement 10: The facility operator should establish a system of records and reports for each MBA, showing the inventory of nuclear material and the change in that inventory, including receipts into, transfers within, and transfers out of the MBA.

Comment: Reports can be compiled, based on shipper's data (measured by the manufacturer) for fresh fuel and data on corrections of nuclear production and loss, which can be estimated on a calculational basis using reactor operation parameters. Usually, no direct measurements for material accounting are conducted on the facility sites, so that reactor operation records and shipper's data are essential for the MCA.

C. Inspection

Compliance of the facility operator with the requirements of the SSAC by the State can be assured only by means of inspection. As mentioned earlier, nuclear material accounting depends to a great extent on the level of the facility activity. In this regard, SSAC activity is focused on inspection. The purpose, scope, and mode of inspection are set forth in the Safeguards Agreement, its Subsidiary Agreement, and relevant domestic laws and regulations.

1. Degree of Assurance. There are two alternative State inspection modes: Level I and Level II, depending on different degrees of assurance required. Briefly, Level I activity includes only examination and observation of the operator's measurements and records to assure facility compliance with the SSAC. Level II activity includes not only the Level I activity but also independent measurements conducted by the State inspector, for detecting errors, falsification of items, etc.

A group of States, the European Atomic Energy Community (EURATOM), has decided to conduct Level II activity by using EURATOM inspectors. Japan, as a State who intends to introduce large-scale use of nuclear energy supported by comprehensive fuel cycle facilities, has also decided to conduct Level II activity by using State inspectors and taking into consideration international contributions. It was a great challenge for a single State to establish and maintain an independent measurement capability using State inspectors; it is still a significant challenge.

2. Inspection Goals. Inspection goals consist of three parameters, detection quantity, detection time, and detection probability (the probability of detecting missing nuclear material from a specified MBA). As a guideline in establishing inspection activities, the goal values of these three parameters currently used by the IAEA are as follows. The detection quantity is interpreted as the significant quantity given in Table I. The detection time is interpreted as the maximum time that may elapse between diversion and its detection by inspection. According to the current guideline, the detection time corresponds in order of magnitude to the conversion time given in Table II. The detection probability is normally set at 95% for detecting diversion of a significant quantity of nuclear material.

TABLE I
SIGNIFICANT QUANTITY OF NUCLEAR MATERIAL

	<u>Material</u>	<u>Significant Quantity</u>	<u>Safeguards Apply To</u>
Direct-Use	Pu*	8 kg	Total element
Nuclear Material	U-233	8 kg	Total isotope
	U[U-235 > 20%]	25 kg	U-235 contained

- Plus rules for mixtures where appropriate -

Indirect-Use	U[U-235 < 20%]**	75 kg	U-235 contained
Nuclear Material	Th	20 t	Total element

- Plus rules for mixtures where appropriate -

*For Pu containing less than 80% Pu-238.

**Including natural and depleted uranium. (From IAEA glossary)

TABLE II
CONVERSION TIME OF NUCLEAR MATERIAL

<u>Beginning Material Form</u>	<u>Conversion Time</u>
Pu, HEU or U-233 metal	Order of days (7-10)
PuO ₂ , Pu(NO ₃) ₄ , or other pure Pu compounds; HEU or U-233 oxide or other pure compounds; MOX or other non-irradiated pure mixtures containing Pu, U[U-233 + U-235] > 20%; Pu, HEU and/or U-233 in scrap or other miscellaneous impure compounds	Order of weeks (1-3)*
Pu, HEU or U-233 in irradiated fuel**	Order of months (1-3)
U containing <20% U-235 and U-233; Th	Order of one year

*This range is not determined by any single factor but the pure Pu and U compounds will tend to be at the lower end of the range and the mixtures and scrap at the higher end.

**Criteria for establishing the irradiation to which this classification refers are under review. (From IAEA glossary)

There has been no agreed consensus on these for goal values for use in the SSAC yet. However, considering the IAEA's basic safeguards concepts and the objectives of the SSAC, these goals should, in principle, provide a high degree of assurance for the SSAC. Some investigations have been made for clarifying the relationship between these goals as used by the IAEA and the SSAC. General trends found in these investigations suggested that the detection quantity used in the SSAC will be smaller than that used in the IAEA by about a factor of 0.7.

3. Escort and Facility Operation for Inspection. The author is not familiar with the situation in other States, but in Japan the Safeguards Agreement was concluded between the Japanese Government and the IAEA. Strictly speaking, the facility operator is the third party to the Agreement. Facility operators are responsible for conforming not only to the domestic law and regulations of the SSAC but also to other laws and regulations concerning safety, radiation exposure control, etc. In this context, there are some constraints for conducting inspections:

- Neither the State inspector nor the IAEA inspector can operate any part of the facility. Facility operation can be done only by operator personnel upon request of the State Inspectorate.
- Agency inspectors may be accompanied by State inspectors, provided that the IAEA inspectors are not thereby delayed or otherwise impeded in the exercise of their functions.
- Both State inspectors and IAEA inspectors must be escorted by facility operator personnel during inspection of a facility.

4. Coordination of Inspection Plan. A facility must be inspected by two different authorities, namely, the IAEA and the State. Since both the State and IAEA inspections include independent measurements, an appropriate coordination of inspection plans between the IAEA and the State is very important to minimize the burden of inspection on the operator.

In Japan, the following procedures are taken to coordinate the inspection plan. First, the Japanese Inspectorate decides on the State inspection plan and informs the IAEA. If acceptable, the IAEA inspection takes place at the same time as the State inspection. When measurements are conducted by the IAEA, the State inspector accompanies the IAEA inspector for observation. When the State conducts measurements, the IAEA inspection might be appropriately simplified, depending on the decision of the IAEA inspector.

5. Record and Report System. A system of accounting and operating records and of reports of each MBA conforming to the Safeguards Agreement must be established. The State system must conform to any bilateral agreements that require preservation of the rights of the States supplying nuclear material.

VI. MATERIAL CONTROL OPERATING PRACTICE

This section details material control operating practice as applied to the Japan Material Test Reactor (JMTR), the training reactor JRR-4, and the Fast Critical Assembly (FCA) now under operation by the Japan Atomic Energy Research Institute (JAERI). These are selected as typical examples from a variety of research reactors.

In Japan, the facility rules for implementing MCA are established by Provision 61-8 set forth in the Law for Control of Nuclear Source and Fuel Materials and Nuclear Reactors: namely, a user of internationally controlled nuclear materials must formulate the facility rules for MCA and obtain the sanction of the Prime Minister before starting to use it. The facility rules should be formulated at least in compliance with any requirements conforming not only to the Safeguards Agreement but also to any bilateral agreements.

As mentioned in the previous section, the research reactor and critical assembly are item facilities, so measurements involve rather simple item counting and attribute measurement. Therefore, control procedures are important elements of facility-level MCA. Control is implemented based on the facility rules for MCA. Considering the high strategic value of nuclear material used for research reactors and critical assemblies, containment and surveillance measures are particularly important for facility-level MCA.

Although different in depth of achievement, depending on the quality and quantity of nuclear fuels used at different reactors, facility rules for MCA are very similar, regardless of reactor type.

A. Facility Rules for MCA

1. Responsibility. Two different structures must be organized for providing appropriate separation of functions and an independent check-up capability.

- (a) One structure is managed by the facility nuclear fuel manager (hereafter called the manager) who is designated to take comprehensive responsibility for implementing the MCA in each facility.
- (b) The other structure is managed by the nuclear fuel section chief (hereafter called the chief) who is responsible for providing an independent check-up capability and the central ledger for all reactors and laboratories operated by JAERI and for compiling official material accounting reports for the State.

2. Material Accounting. Material accounting must be conducted in batch/item units at each KMP.

3. Receiving.

- (a) Before receiving nuclear material, the manager must secure the consent of the chief in advance.
- (b) After receiving nuclear materials, the manager must, on a timely basis, provide the chief with inventory change information, including:
 - Date of inventory change
 - Reason for inventory change
 - Type and quantity of nuclear material
 - Assigned MBA and KMP
 - Batch identification
 - Number of items in the batch
 - Measurement basis
 - Origin of the material

The origin of the material is required for conforming to bilateral agreements, not the Safeguards Agreement.

4. Shipping.

- (a) Before shipping nuclear material, the manager must secure the consent of the chief in advance.
- (b) After shipping nuclear material, the manager must, on a timely basis, provide the chief with inventory change information.

5. Custody in Inventory KMP.

- (a) Upon taking custody of nuclear material in the KMP, advantage must be taken of any opportunity to use containment and surveillance measures to help ensure material accountability.
- (b) The manager must arrange all batches/items in the KMP so as to permit convenient clarification of batch data, including:
 - Categorization of nuclear material
 - Batch identification number
 - Weight of nuclear material in the batch
 - Number of items in the batch
 - Origin of nuclear material

6. Internal Transfer.

- (a) Before internal transfer of nuclear material between KMPs, the manager must secure the consent of the chief in advance.
- (b) After internal transfer of nuclear material, the manager must, on a timely basis, provide the chief with inventory change information.

7. Rebatching. If one of the following changes appears in the batch, the manager must provide the chief with the following information.

- Change in physical form
- Change in chemical form
- Change in enrichment
- Change in irradiation condition, as appropriate
- Change in batch container, as appropriate
- Change in sublocation, if provided
- Change in measurement basis
- Change in origin

8. Retained Waste. Not important in research reactors.

9. Measured Discard. Not important in research reactors.

10. Nuclear Loss and Production. After discharge of nuclear material from the reactor core, the manager must provide the chief with information

concerning production of special nuclear material and the loss that results from transmutation.

11. Continuity of Knowledge of Fuel Inventory. In implementing the MCA, the manager must take necessary steps to maintain continuity of knowledge of all of the fuel inventory in order to detect any theft or unknown loss of nuclear material without delay.

12. Physical Inventory Taking (PIT).

- (a) The manager must conduct the PIT at least once every 6 months.
- (b) The chief must conduct the PIT at least once every year.
- (c) The manager must provide the chief with operating procedures and forecast dates in advance. The procedures should take into consideration, inter alia, the following:
 - Containment and surveillance measures adopted
 - Stratification of nuclear material
 - Containment and surveillance measures adopted
- (d) The chief must also provide the manager with operating procedures and expected PIT dates in advance.
- (e) The chief or the manager must, without delay, inform the other of the results of the PIT.

13. Procedure for Unusual Situations. If an unusual incident or circumstance leads the manager or chief to believe that there is or may have been a loss of nuclear material that exceeds the specified limits, the manager or chief must immediately take action to determine the cause of such loss.

14. Accounting Procedure. The manager must provide detailed procedures for accounting for nuclear material that is received, shipped, produced, consumed, transferred internally, and inventoried.

15. Maintenance of Equipment. The manager must conduct maintenance and calibration of equipment used for MCA at appropriate frequency to assure that it is ready for use at any time.

16. Records. The manager must maintain the following records for a specified period.

- Facility operating data used to establish changes in the quantity and composition of nuclear materials
- Location of fuel subassemblies
- Relevant source data with respect to nuclear loss and production, including the integrated thermal power produced by the reactor and the burnup for each fuel material on discharge
- Information on any accident that results in a loss of nuclear material
- Maintenance and calibration records of equipment used for MCA

17. Reports. The chief must provide the government with all necessary information conforming to the domestic law and regulations.

B. Implementation

Taking into consideration the specific features of research reactors--namely, itemized fuel elements, infrequent internal and external movement of fuel elements, and high strategic value of the fuel--material control in combination with containment and surveillance can significantly alleviate the burden caused by safeguards implementation, especially at the facility level. Fundamentally, the first step in implementing control is based on a policy of written approval for transfer of fuel elements by a responsible person. The second step is to facilitate detection of noncompliance with material control procedures.

Inner JAERI fuel transfer must be conducted according to the Inner-JAERI Nuclear Material Transfer Transaction form shown in Fig. 2. Receiving and shipping transactions (Figs. 3 and 4) are used for nuclear material transfer to and from JAERI, respectively. These transactions should clearly define what the item is, who issued and received it, when it was issued and received, its previous location and its present location. Every transfer of custodianship should also be reworded in these transactions. These transactions should be accompanied by the timely initiation and completion of the forms.

1. Inner-JAERI Transfer Transaction. As shown in Fig. 2, this transaction consists of three sections: deliver, change for, and receiver. The right-hand side on the transaction form is for computer input cards; the computerized central ledger system compiles all official reports to the government, after accountancy consistency checks conforming to domestic law and regulations.

Since research reactors have infrequent inter-MBA transfers, inter-KMP transfer in an MBA becomes important for nuclear material control. In the following paragraphs, an example is given to show how to interpret the transaction for material control purposes.

Usually, the research reactor MBA is divided into four inventory KMPs: KMP-A for the fresh fuel storage vault, KMP-B for the reactor, KMP-C for the spent fuel storage canal, and KMP-D for other locations as mentioned before. While KMP-A, KMP-C, and KMP-D are usually under the custodial responsibility of the fuel coordinator, KMP-B is under the custodial responsibility of a reactor coordinator or an experiment coordinator at the critical assembly.

Transfer from KMP-A to KMP-B is conducted step by step in accordance with the following schedule.

- Upon request of the experiment/reactor coordinator, the fuel coordinator initiates the transaction by filling in relevant data in the deliver section and by signing in the space marked Credit in the transaction form.
- The transaction is submitted to the manager for approval of the transfer.
- The manager must secure the consent of the chief, who signs the transaction form at the left bottom corner.
- When the manager approves the transfer by placing his signature in the deliver section, the manager provides the fuel coordinator with keys for opening the fuel storage vault door and disconnects the surveillance alarm system for the fuel storage vault.

Date of issue 昭和 年 月 日		Date of deliver 昭和 年 月 日 時 分	
Deliver section		JOB No. Manager Sub-manager Credit	
(Signature) (Signature) (Signature)			
M B A KMP Room			
Batch name		Regulation Ownership	
		☐ Reactor use ☐ JAERI ☐ PNC ☐ Laboratory Use ☐ Gov. ☐	
Category of material			
☐ NU ☐ SU ☐ Source material ☐ DU ☐ Pu ☐ Heavy water ☐ Th ☐ U-233			
Total weight		Fissile weight	
t kg y		t kg g	
Pu production		No. of items	
t kg g		☐ M ☐ T ☐ N ☐ L	
Physical form		Chemical form	
(Code)		(Code)	
Manufacturer		Origin	
		Measures of transportation	
		(Code)	
☐ → MBA ☐ Retained waste ☐ → KMP ☐ Measured discard ☐ → Room ☐ Nuclear loss ☐ Nuclear production ☐ Change in status ☐ Enrichment ☐ Measurement basis ☐ Rebatching ☐ Category ☐ Rebatching			
Receiver section		JOB No. Manager Sub-manager Debit	
(Signature) (Signature) (Signature)			
M B A KMP Room			
Batch name		Regulation Ownership	
		☐ Reactor use ☐ JAERI ☐ PNC ☐ Laboratory Use ☐ Gov. ☐	
Category of material			
☐ NU ☐ SU ☐ Source material ☐ DU ☐ Pu ☐ Heavy water ☐ Th ☐ U-233			
Total weight		Fissile weight	
24 t kg g		34 t kg g	
Pu Production		No. of items	
49 t kg g		58-61	
Physical form		Chemical form	
(Code)		(Code)	
Container		Irrad./quality	
		(Code)	
Date of receipt 昭和 年 月 日 時 分			
Nuclear fuel Section; Chief		Computer input coordinator	
(Signature)		(Signature)	

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日本原子力研究所

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Fig. 2. Inner-JAERI nuclear material transfer transaction.

Date of issue 昭和 年 月 日		M B A		Consigner	
Shipper's organization and address		Name		(Signature)	
Batch name		Contract name			
Category of material					
☐ NE ☐ DE ☐ RE		☐ Pu ☐ U-235		☐ Source material ☐ Heavy water	
Total weight		Fissile weight		Fissile ratio	
24 t kg g		34 t kg g		44-47	
Pu Production		No. of items		Measurement basis	
48 t kg g		58-61		☐ M ☐ I ☐ N ☐ L	
Physical form		Chemical form			
(Code)		(Code)			
Containment		Irradiation status/quality			
(Code)		(Code)			
Manufacturer		Origin		Measures of transportation	
				☐ Trailer ☐ Truck	
Carrier					
Receiving section		JOB No.		Manager Sub-manager Consignee	
M B A KMP Room		(Signature)		(Signature)	
(Signature)		(Signature)		(Signature)	
Containment		Receipt for			
(Code)		☐ Purchase ☐ From JAERI's ☐ Borrowing ☐ Oarai site ☐ Trust ☐ ☐ Return ☐ ☐ Donation			
Regulation		Ownership			
☐ Reactor use ☐ Laboratory Use		☐ JAERI ☐			
Date of receipt		年 月 日 時 分			
(Note)					
昭和 年 月 日					
Nuclear Fuel Section; Chief			Computer input coordinator		
核燃料課長 係長 係 (Signature)			保安管理室 (Signature)		
イ 処 ン ア フ ト 課					

1-2 区分	3-6 登録番号	7-12 受入年月日
21		
14-15 核物質	16-23 受入物件バッチ名	
計量データ (24-61) 欄		

1-2 区分	3-6 登録番号	7-12 (修正年月日)
22		
13-15 受入JOB	16-19 受入MBA	20-24 KMP受入ROOM
25-28 受入事由	29-30 事由\$10	31-33 向位体
34-39 製造加工業者	40-41 供給国	
42-43 輸送	44-45 測定	46-47 受入後物
48-50 払出元MBA		

Computer input card arrangement conforming to code 10 of Subsidiary Arrangement

法定10年間保存

Fig. 3. Nuclear material receiving transaction.

- Witnessed by the experiment/reactor coordinator, the fuel coordinator conducts the fuel transfer from the storage vault to the reactor.
- Immediately after completion of the transfer, the fuel coordinator must return the key, and the manager must actuate the alarm system.
- After completing the transfer, the experiment/reactor coordinator must fill in, without delay, relevant data in the receiver section and place his signature in the space marked Debit in the transaction. Then custodianship is transferred from the fuel coordinator to the experiment/reactor coordinator.
- The transaction is submitted to the manager. After the manager approves completion of the transfer with his signature in the receiver section, he submits the complete transaction to the chief for recording and reporting purposes.

For transfer between other combinations of KMPs, like from KMP-B to KMP-C, the procedures are almost the same as those mentioned above except that the Credit and Debit can both be signed by the same person, that is, the fuel coordinator for transferring irradiated fuels. Transfer from KMP-B to KMP-A can be made only for critical assemblies. In this case, the Credit should be signed by the experiment coordinator and the Debit should be signed by the fuel coordinator.

2. Detection of Noncompliance with Control Procedures. While there is a general understanding of the functions, measures, and procedures associated with material accountancy and physical protection, the scope of nuclear material control seems to be not so clearly defined, especially at the facility level. There is considerable overlap in nuclear material control and physical protection at the facility level. Examples of such overlap can be seen in written approval procedures for access to fuel, designation of responsible personnel for fuel movement, and detection of intrusion by personnel without approval. However, the material control system neither delays attempts by intruders nor responds to intruders.

The manager is primarily responsible for detecting noncompliance with control at the facility level. Generally, two measures are employed for detecting noncompliance:

- Visual checks of control documents and consistency checks of accounting
- Surveillance appropriately incorporated with containment

Surveillance means the collection of information through devices and/or human observation in order to detect unauthorized movement of nuclear fuel. Containment means physical barriers that in some way physically control the movement of nuclear material.

The design philosophy of containment/surveillance devices for IAEA use is different from that for the material control at the facility level because of differences in implementation. At the facility level, surveillance is implemented by the manager who is always at the facility site (referred to as a continuous inspection mode in IAEA terminology). For IAEA safeguards, surveillance is implemented by IAEA inspectors who visit the facility site only intermittently. Therefore, the IAEA's surveillance devices are usually more elaborate; for example, they can operate unattended and have a high capability for indicating tampering. Shared use of IAEA surveillance devices with the

facility operator seems to be unlikely at present because common use would provide facility personnel with operational details of the IAEA devices.

Generally, surveillance devices and physical barriers used for physical protection can also be used for material control. For fresh fuel elements, JAERI's policy requires that all fresh fuel inventory KMPs, including critical assembly reactor rooms, must be protected by intrusion sensors and by double locks of different types. The surveillance system that contains these sensors is monitored by the manager on the facility site during working hours. After working hours, the system is switched to the central guard station and the manager is informed from the guard station when an alarm is actuated.

The on-site monitoring panel of the surveillance system is shown in Fig. 5. The alarm can be released only by the manager, who uses an appropriate key operation and special code number input.

Fresh fuel transfer within the facility must be conducted under constant surveillance by a responsible person. For the spent fuel canal area, intrusion detectors are equipped on a case-by-case basis. However, intrusion detectors do not seem to be necessary in that area for material control purposes. Due to very high radiation, movement of spent fuel requires the operation of specialized heavy-duty equipment; thus, such movement takes a long time, even for short distances so that, while possible, it would be very difficult to do without detection by the manager.

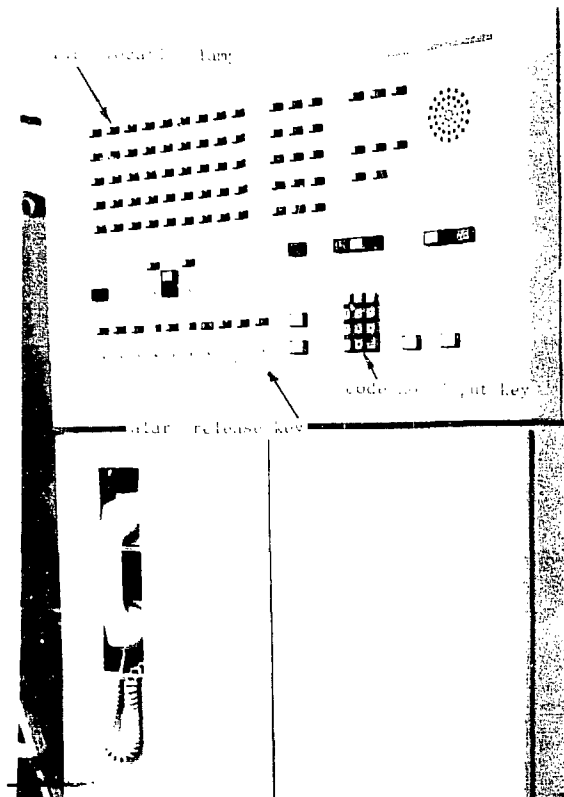


Fig. 5. On-site monitoring panel of surveillance system.

VII. OPERATING PRACTICES OF MATERIAL ACCOUNTING AND INSPECTION

This section describes the material accounting and inspection practices employed at the JMTR, JRR-4, and the FCA. Inspection activities conducted by the State and the IAEA may be divided into two different modes: routine inspection and verification of PIT.

PIT is conducted by the facility operator. The State and the IAEA's verification must include, inter alia, independent measurements or observation. Therefore, coordination of the PIT by the operator, the State, and the IAEA is particularly important to minimize intrusiveness. The PIT schedule is decided by the operator who informs the State and the IAEA at least 30 days in advance. The State and the IAEA can then coordinate their PIT activities to carry out verification of the PIT simultaneously with the operator's schedule.

Routine inspections are conducted by the State and the IAEA simultaneously. In concept, the routine inspection is conducted on a flow measurement basis, whereas item counting and identification of fuel in inventory KMPs are carried out by routine inspection at itemized facilities. The routine inspection effort is focused on inventory KMPs that might be the origin of significant material flow. In this regard, routine inspections usually start with the fuel in the core of research reactors.

Routine inspection activities consist of, inter alia,

- Audit activity: examination of records, verification of self-consistency, and consistency with reports.
- Observations: visual observation of movement of nuclear fuel and of measurements conducted by the operator.
- Containment and surveillance: installation and maintenance of surveillance equipment and verification of integrity of containment.
- Measurement: independent measurement of nuclear material by inspectors.

A. Operator Activity at JMTR, JRR-4, and FCA

For research reactors and critical assemblies, facility MCA activities are initiated when nuclear fuel is transferred from a fuel manufacturer to fresh fuel storage; the MCA activities are terminated when irradiated fuel is transferred to a reprocessing plant.

Nuclear fuel is received into the facility on the basis of the fuel manufacturer's measured values (as witnessed by facility operator personnel) and on the basis of identification numbers, which are protected by seal. The facility operators do not usually carry out direct measurements of nuclear fuel except for item counting and related attribute NDA for identification purpose, if relevant. Consequently, shipper's data provided by the fuel manufacturer are the basis of MCA for research reactors and critical assemblies.

Routine facility MCA activities are focused on maintaining continuity of knowledge of the inventory based on inner transfer transactions, compliance with the facility MCA rules mentioned in Sec. VI, and detection of noncompliance by consistency checks of relevant documents and by containment/surveillance.

Besides these activities, the facility operator must provide the IAEA and the State with the assistance needed for inspections. Such assistance involves, inter alia, record preparation, fuel handling, facility operation for conducting the inspection, and escort of inspectors during the inspection.

Subject to the current information system, the facility inventory change report (ICR) as of the end of the month is issued to the State by the fifteenth day of the following month. It takes the State reporting system about 2 weeks to prepare relevant State reports for the IAEA. Mailing takes another 2 weeks. As a result, the monthly inventory change report is made available to the State inspector 1 month after inventory and to the IAEA inspector 2 months after inventory. The delay may be even larger, depending on travel schedules.

In addition to the problem of time delays, official reports of ICR, MBR, and PIL do not provide inspectors with sufficient information for conducting inspections, because these reports do not show the location of batches in the KMPs, seal numbers, and updated inventory information. Consequently, the operator is asked to prepare an updated inventory summary for inspector use, including:

- Detailed locations of batches in the reactor core and storage area
- Identification numbers of seals and their location
- Updated itemized list of inventory

B. Operating Practice Employed at JMTR

1. Description of Facility. The JMTR is a tank-type light-water-moderated and cooled reactor with 93%-enriched uranium/aluminum alloy fuel. The maximum operating power is 50 MW thermal (0.5 MW/ℓ core) for providing neutron fluxes of 4×10^{14} n/s/cm² for thermal neutrons and 4×10^{14} n/s/cm² for fast neutrons. The reactor was designed for material irradiation tests, isotope production, and fundamental research.

Horizontal and vertical arrangements of the JMTR complex are shown in Figs. 6 and 7. As shown in these figures, a cylindrical reactor vessel is submerged in a reactor pool of 6 mφ by 11.8 m depth. There is a long water canal of 6.2 m depth connecting the reactor pool and the spent fuel subassembly trimming pool. The reactor pool and water canal, partitioned by a movable gate, are shown in Fig. 8. The movable platform car with jib crane shown in Fig. 8 is used for moving spent fuel subassemblies under water in the canal. A typical fuel subassembly is shown in Fig. 9. Nineteen 1.27-mm-thick fuel plates of 93%-enriched uranium/aluminum alloy with aluminum cladding are assembled in a 7.7- by 7.7-cm-square aluminum tube. The fuel subassembly has a 120 cm length, of which the core length is 77.8 cm. The total weight of one subassembly is about 7.9 kg, including 0.28 kg of ²³⁵U per subassembly.

The core is contained in a cylindrical stainless steel pressure vessel of 3 mφ by 9.5 m height. A top view of the pressure vessel is shown in Fig. 10. Refueling is carried out through the central hatch on the top cover of the pressure vessel. An inside view of the pressure vessel is shown in Fig. 11. There are 204 squares in the matrice and the central 7 by 5 positions are used for the core. Usually about 22 positions are used for fuel subassemblies. Five control-rod positions are also provided in this central region. Beryllium subassemblies surround the central region as reflectors. Many in-pile loops and pipes used for irradiation tests are located in the beryllium region close to the core.

The core cooling-water flow rate is designed to be 6000 m³/h, which corresponds to 140 m³/h per fuel subassembly. The inlet water pressure is 14 kg/cm² and the pressure drop is about 3.3 kg/cm². The inlet-outlet temperature difference is about 10°C.

The normal operation mode consists of three shifts per day and six cycles per year. Each cycle consists of a 24-day operation, including a 2-day shutdown for refueling in the middle of the cycle and a 3-week shutdown for reactor maintenance. An off-load refueling mode is employed twice during each cycle.

2. Location of Fuel and Mode of Fuel Transfer (KMP Structure). The fuel subassembly cannot be changed inside the facility except for minor trimming of the aluminum guide tubes on the upper and lower parts of the fuel plates before shipping. Normal inventory and locations at the facility are as follows:

- Fresh fuel storage storage vault: about 100 fresh fuel subassemblies
- Reactor core: about 22 subassemblies
- Water canal: about 200 subassemblies

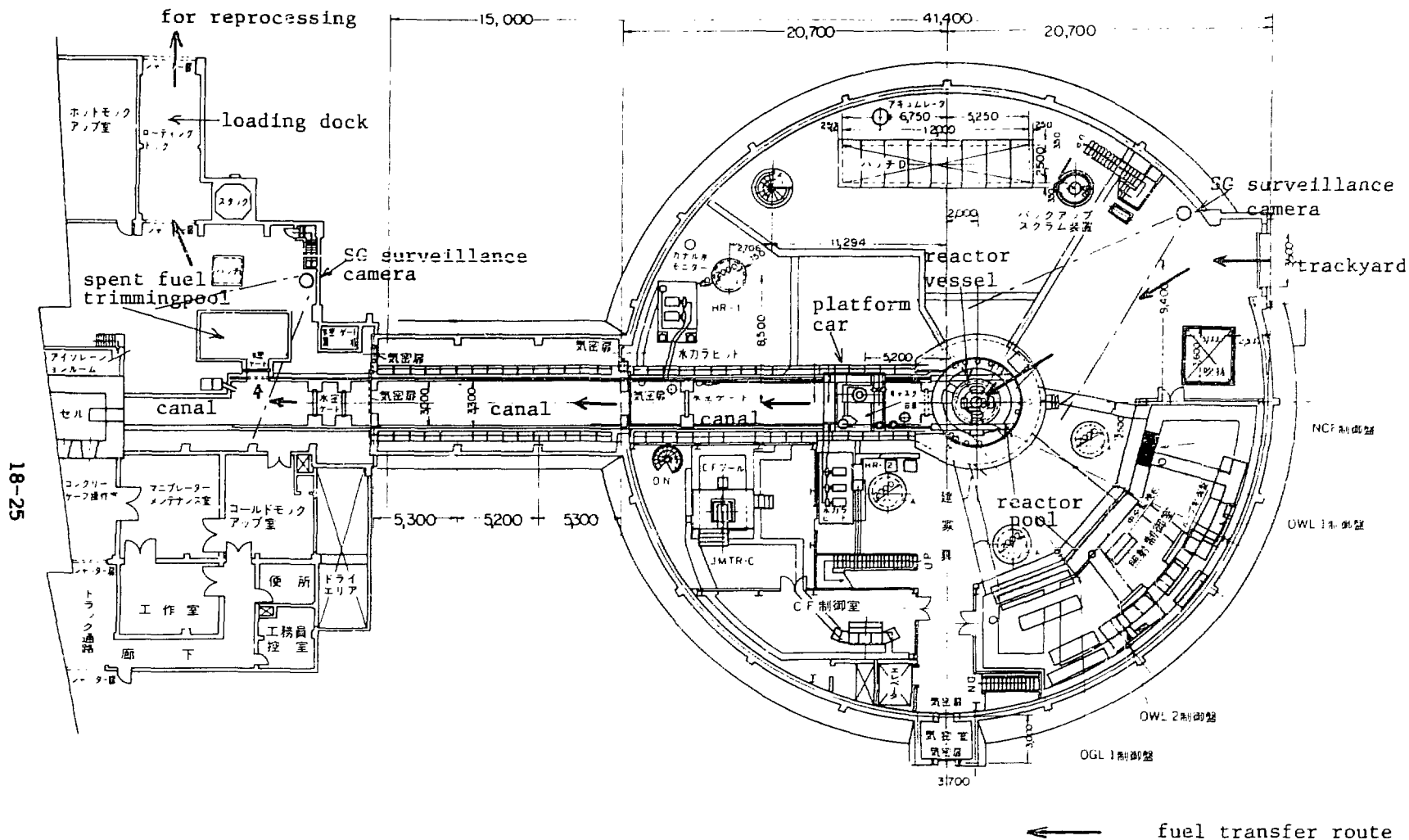


Fig. 6. Horizontal plan of JMTR complex. .

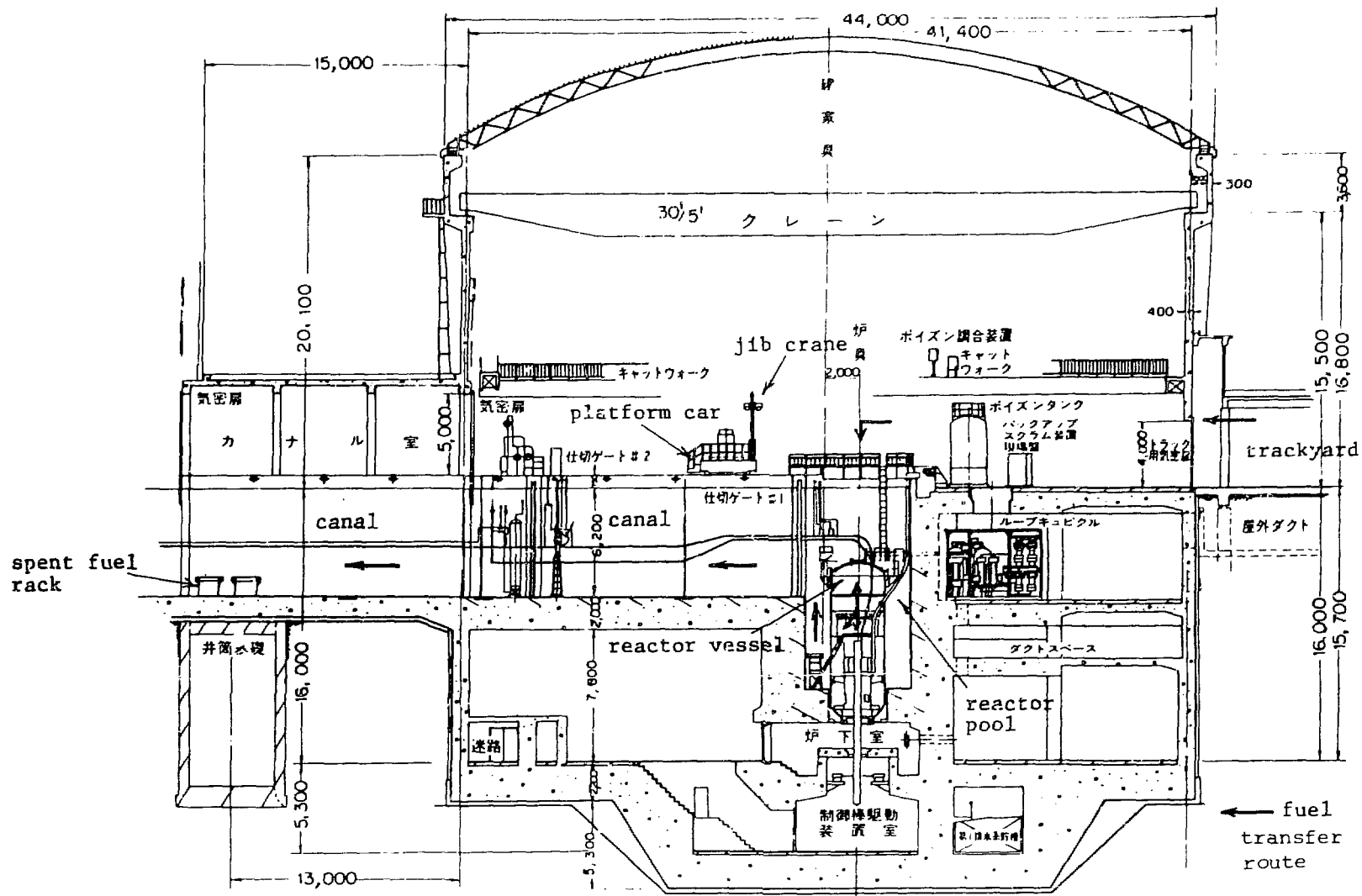


Fig. 7. Vertical plan of JMTR complex.

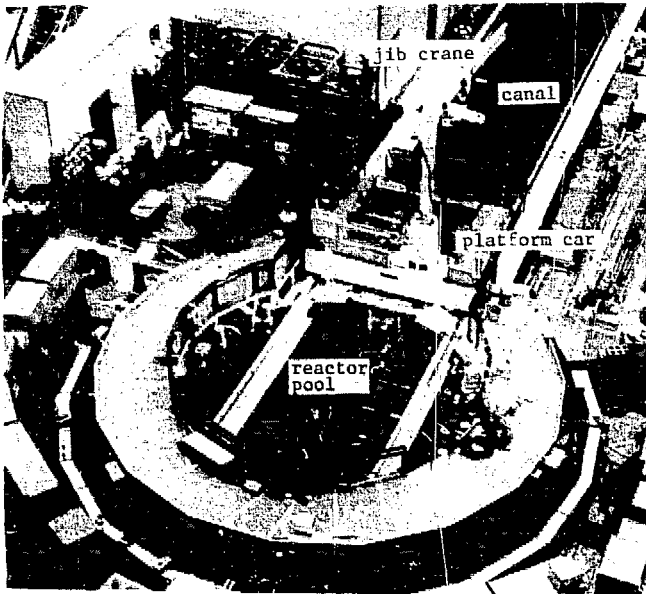


Fig. 8. JMTR reactor surroundings.

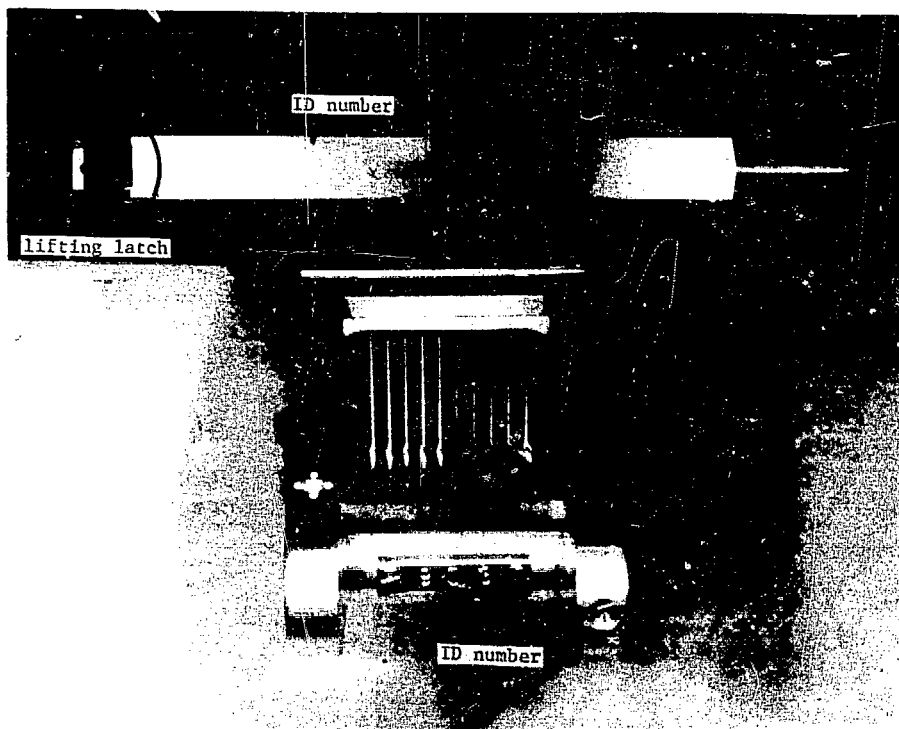


Fig. 9. JRR-4/JMTR fuel subassembly.

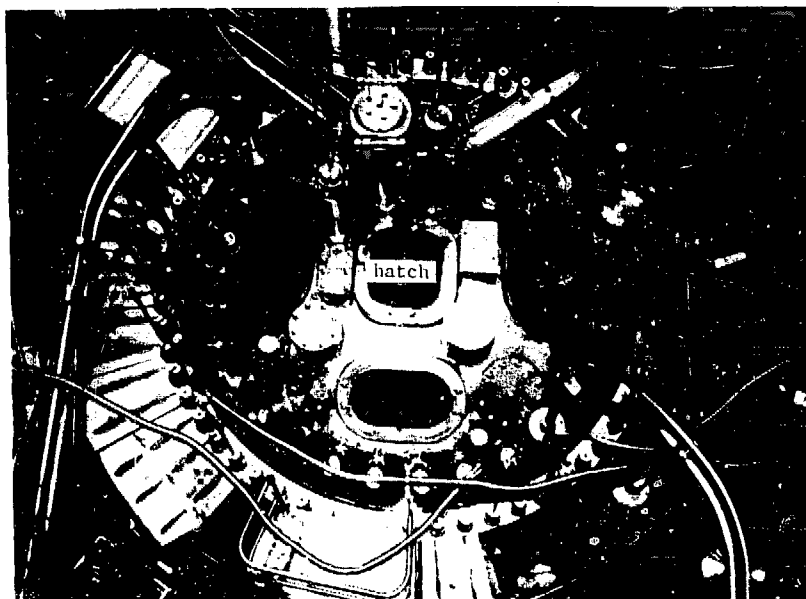


Fig. 10. Top view of JMTR pressure vessel.

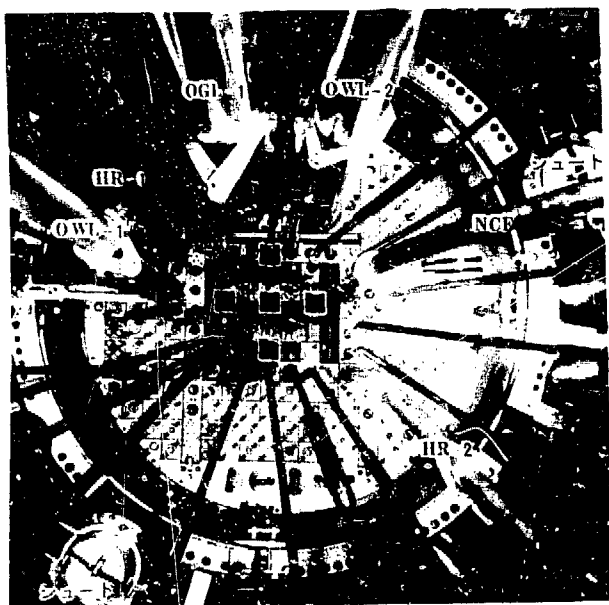


Fig. 11. Inside view of JMTR pressure vessel.

Major fuel subassembly transfers are conducted in the following manner:

- The fresh fuel subassemblies are received from the manufacturer and stored in the fresh fuel storage vault. Frequency of receipt: about five times per year.
- Upon request, the fresh fuel subassemblies are transferred by hand cart from the vault to the reactor room through the track yard shown in Figs. 6 and 7. Frequency: about twice per cycle.
- After reactor shutdown, the water level of the reactor pool is brought down to the level of the top cover of the vessel. The bolts of the central hatch are unfastened by hand tools, then the central hatch shown in Fig. 10 is removed by the jib crane shown in Fig. 8. The fuel subassemblies are moved into or out the reactor core through the central hatch located under water in the pressure vessel by using handling tools.
- Irradiated fuel subassemblies are transferred under water into the canal using the jib crane on the platform car. Irradiated subassemblies are stored in a rack under water as shown in Fig. 12.
- After the aluminum guide tube is trimmed at the spent subassembly trimming pool shown in Fig. 6, the spent fuel subassembly is placed in the transfer cask and shipped. Frequency: about twice per year.

The inventory and flow KMPs are structured as shown in Fig 13.

3. Accounting Procedure. Each fuel subassembly consists of one batch and one item. The identification number is engraved on the surface on the subassembly as shown in Fig. 9. Relevant data for each batch/item consist of:

(a) For fresh fuel subassembly:

- Identification number
- Weight of total and fissile uranium and the chemical composition based on the shipper's data.

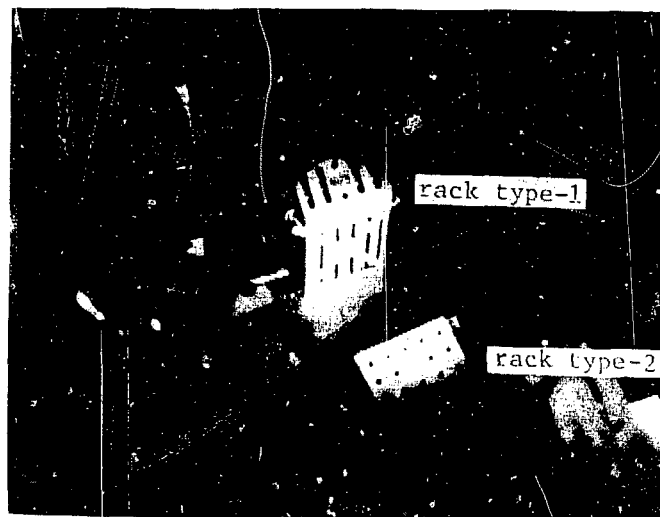
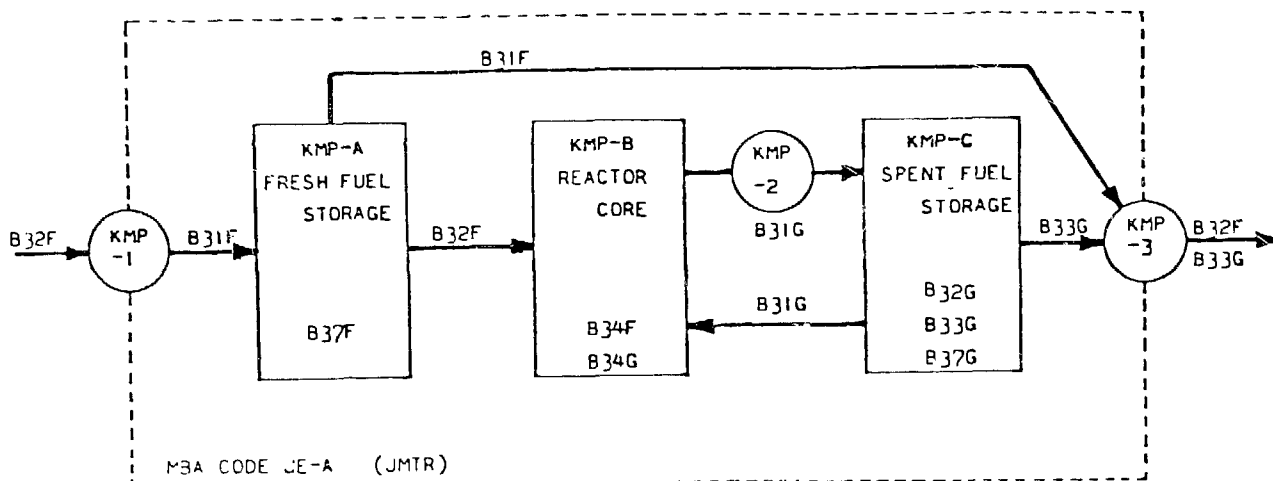


Fig. 12. JMTR spent fuel storage rack under water canal.



KMP-1	RECEIPT OF FRESH FUEL FROM OTHER MBA ACCIDENTAL GAIN, DE-EXEMPTION	(B32F)	SPENT FUEL STORAGE
KMP-2	NUCLEAR LOSS/ PRODUCTION IN FUEL DISCHARGE FROM THE REACTOR CORE	(B31G)	1) TEMPORARY STORAGE 2) REACTOR POOL 3) CANAL # 1 4) CANAL # 2 5) CANAL # 3 6) SFC POOL 7) SHIPPING CASK
KMP-3	SHIPMENT OF FRESH FUEL TO OTHER MBA SHIPMENT OF SPENT FUEL TO OTHER MBA ACCIDENTAL LOSS, EXEMPTION	(B32F) (B33G)	

Fig. 13. KMP structure of JMTR.

(b) For an irradiated fuel subassembly:

- Identification number
- Weight of total uranium, fissile uranium, and total plutonium as calculated to allow for nuclear loss and production.

Nuclear loss and production are based on neutronic calculations normalized to total reactor power (obtained from the product of the temperature difference of inlet and outlet cooling water and its flow rate). The average burnup is about 40%, and the calculated nuclear loss and production agree with the measured loss and production obtained by the Savannah River Reprocessing Plant in the United States, to within about 5% error. Since shipper/receiver differences are corrected by the receiver, in principle, there is no shipper/receiver difference for research reactors and critical assemblies.

According to the inner-JAERI transfer transactions (shipper transactions and receiver transactions shown in Figs. 2, 3, and 4), fuel subassembly transfers are conducted based on item counting, item identification, shipper's data, and calculated nuclear loss and production for each item. These transactions are issued for each batch/item. The number of transactions issued in 1981 was:

(a) Inventory change	519	
• Receiving		162
• Shipping		119
• Nuclear loss		119
• Nuclear production		329

4. Containment and Surveillance. Items are not allowed to be stored, even temporarily at locations other than fresh fuel storage, the spent fuel storage canal, and the reactor because of potential safeguards risks. According to JAERI's policy, all fresh fuel storage areas are under continuous surveillance by intrusion sensors. Intermittent surveillance by responsible facility personnel is used for the spent fuel canal and reactor area.

Both the State and the IAEA apply independent seals for each fuel subassembly in the fresh fuel storage. They also apply independent camera-surveillance systems at the two different locations shown in Fig. 6. Two cameras, one for the State and the other for the IAEA, cover the reactor pool area; two others cover the spent fuel subassembly trimming pool area.

5. Inspection.

a. Routine Inspection. Routine inspection takes place simultaneously by one or two State inspectors and one or two IAEA inspectors. The average frequency is once a month, and the inspection takes 1 day. The operator is notified 1 week in advance. Major routine inspection activities consist of:

- Explanation of updated inventory summary paper
- Record audit inspection
 - (1) Examination of records
 - (2) Consistency check
- Fresh fuel storage
 - (1) Subassembly item counting
 - (2) Sealing of subassembly after NDA by SAM-II
 - (3) Visual check of seal integrity and identification number
 - (4) Seal replacement on a random sampling basis of 10% of all seals (as a check for tampering)
- Spent fuel canal inspection
 - (1) Item counting of spent fuel subassemblies in underwater storage rack (Fig. 12)
 - (2) Visual check of identification number engraved on the surface of the subassembly (Fig. 9) by lifting the subassembly from the underwater storage rack to about 3 m from the water surface (random sampling of 10% out of all subassemblies in the canal)
 - (3) Maintenance of surveillance camera, tampering check, and film replacement

Routine inspection does not include observation of fuel subassemblies in the reactor core so that it does not impact on reactor operation. The facility operator contributes about a 5 man-day effort to each routine inspection.

b. Physical Inventory Taking (PIT). PIT is conducted on three different levels simultaneously--the IAEA, the State, and the facility operator--using the sampling plan calculated by the State. It takes about 3 days and its frequency is twice a year. Verification activities conducted for PIT are the same as for routine inspections with the following exceptions:

- Fuel subassemblies in the reactor core are verified by item counting and by identification number checks on a random sampling basis of 5 out of 27 fuel subassemblies in the core, including 5 control rods. The reactor is shut down for this purpose, and the central hatch (Fig. 10) is opened. A handling tool is used to lift up fuel subassemblies from the reactor core to an appropriate level in the reactor pool for observing identification numbers.
- The sampling rate used for checking identification numbers for the PIT is greater than that used for the routine inspection; for example, of about 200 subassemblies stored in the underwater storage rack in the canal, about 50 are sampled for the PIT, while about 10 are sampled for the routine inspection.
- The facility operator must submit the material balance report and physical inventory list to the State within 15 days after completion of the PIT.

Usually, the operator man-power necessary for conducting the PIT is 8 x 3 man-days, and for verification of the PIT the State needs about 2 x 3 man-days. The IAEA also needs 2 x 3 man-days. There is no MUF except for the missing fuel subassembly as a whole.

C. Operational Practice Employed at JRR-4

1. Description of Facility. The Japan Research Reactor No. 4 (JRR-4) is a swimming-pool-type light-water-moderated and -cooled reactor with 93%-enriched uranium/aluminum alloy fuel. The maximum operating power is 3.5 MW. The reactor is operated for training purposes and for basic research to some extent. Since the total inventory of JRR-4 is approximately 10 effective kilograms, JRR-4 is selected in this lecture as an example of an insensitive facility, in contrast to the sensitive facility mentioned in the next section.

Horizontal and vertical drawings of the JRR-4 facility are shown in Figs. 14 and 15, respectively. As shown in these figures, a movable bridge suspends the reactor core in the water pool, 9 m below the water surface. The fuel subassembly is similar to that used for the JMTR (shown in Fig. 9), except that the smaller number of fuel plates in one fuel subassembly leads to 157 g of ^{235}U per fuel subassembly. The core is composed of an 8 by 8 square matrix frame structure (suspended by the bridge) in which 20 matrix positions are filled with fuel subassemblies. The spent fuel subassemblies are stored in racks located on the bottom of the water pool as shown in Figs. 16 and 17. The normal operation is one shift per day and 4 days of operation per week.

There are four different inventory KMPs at the facility: the fresh fuel storage vault assigned as KMP-A; the reactor core assigned as KMP-B; the spent fuel storage area located in the water pool, assigned as KMP-C; and the "other locations" assigned KMP-D. The normal fuel inventory in each KMP is:

- KMP-A 5 subassemblies
- KMP-B 20 subassemblies
- KMP-C 20 subassemblies
- KMP-D normally zero

Fig. 14. JRR-4 pool horizontal section.

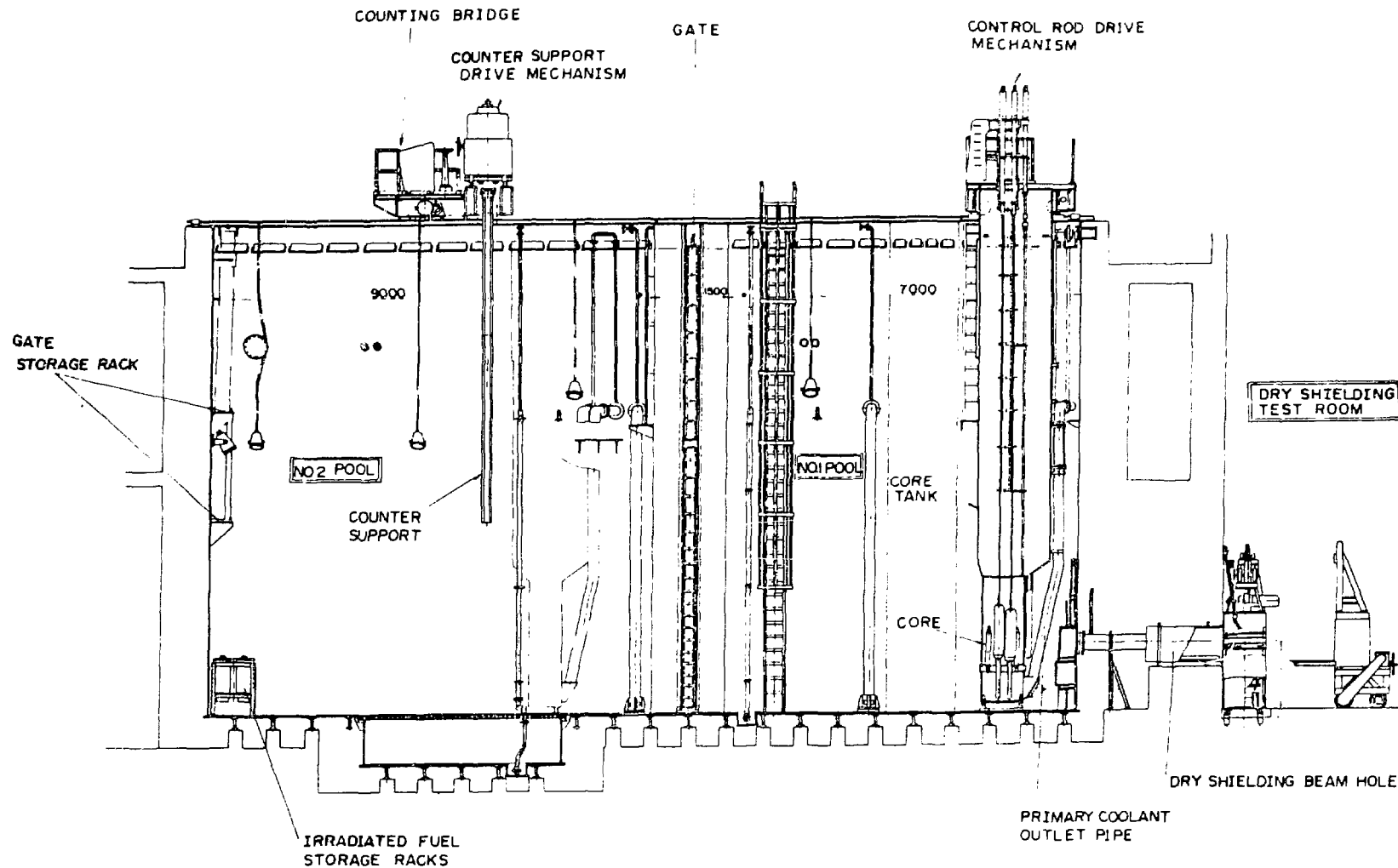


Fig. 15. JRR-4 pool vertical section.

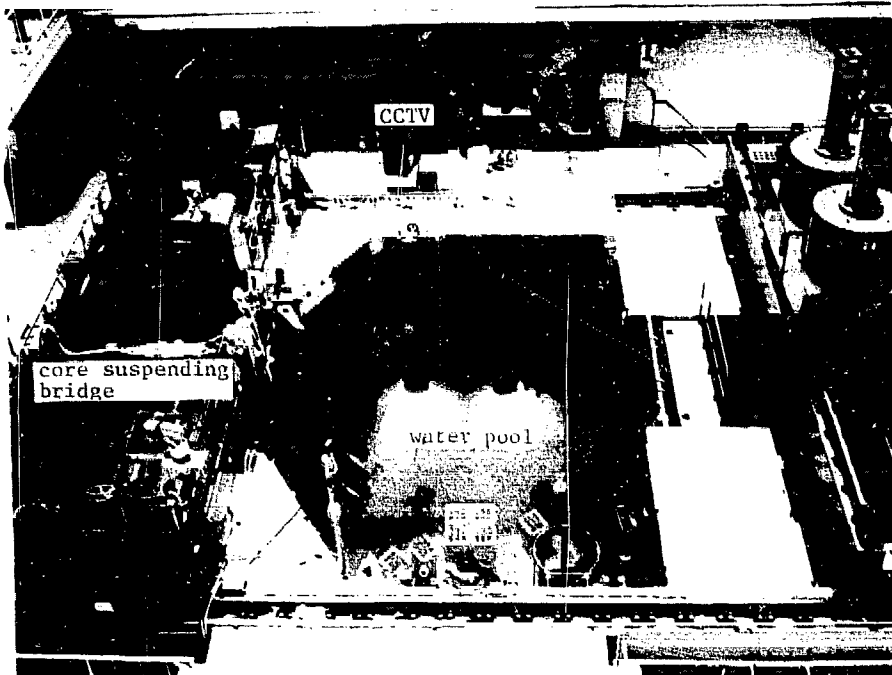


Fig. 16. JRR-4 reactor surroundings.

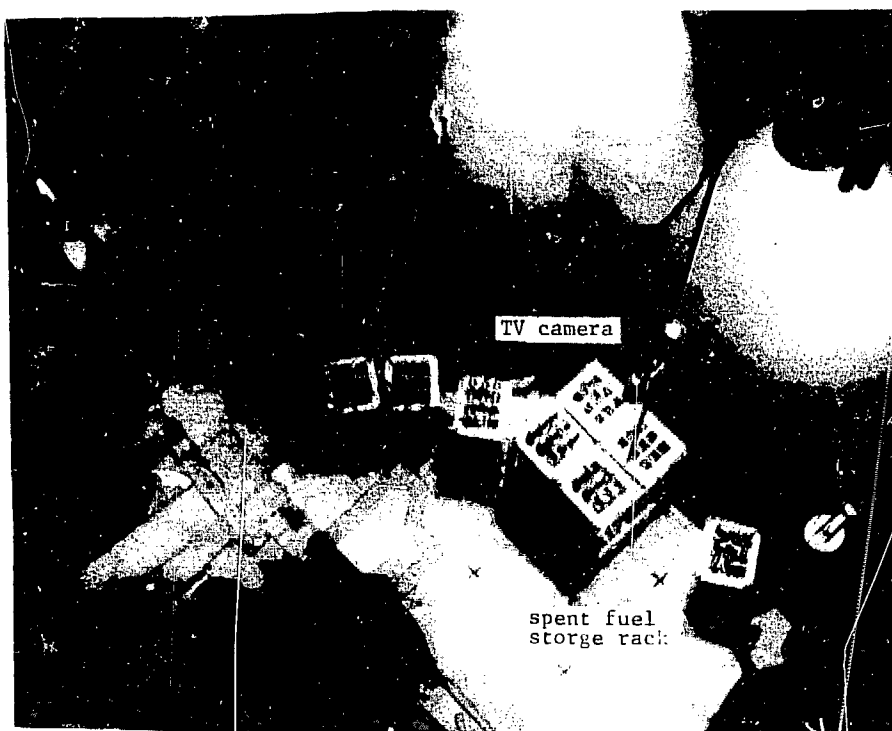


Fig. 17. Underwater TV camera looking at spent fuel subassembly.

The accounting procedures applied for JRR-4 are exactly the same as those for JMTR, but the frequency of fuel transfer is very low. The number of transactions issued in 1981 was:

(a) Inventory change	4	
• Receiving		0
• Shipping		0
• Nuclear loss		2
• Nuclear production		2
(b) Inter-KMP transfer	6	

2. Inspection. As mentioned previously, JRR-4 is a typical example of an insensitive facility owing to the small inventory of about 10 effective kilograms, so that the inspection effort allocated for JRR-4 is small. In this context, no routine inspections take place for JRR-4. The PIT is done once a year during 1 day.

The main activities carried out for the PIT are item counting and checking of identification numbers engraved on the lifting latches of subassemblies as shown at the bottom of Fig. 9. Item counting and identification number checks are conducted using an underwater CCTV system. Figure 17 shows an underwater camera looking at a spent fuel rack on the bottom of the water pool, and Fig. 18 shows the TV display of the identification number engraved on the lifting latch. For conducting the PIT, the State and the operator need only 2 man-days and 4 man-days, respectively.

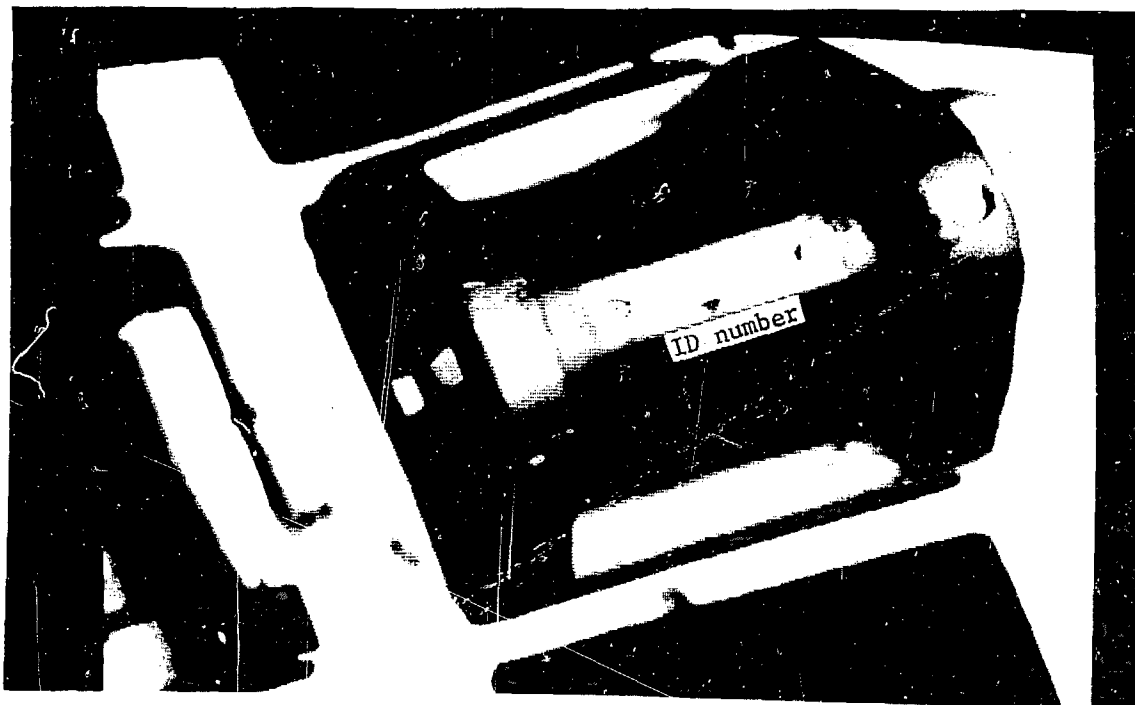


Fig. 18. TV display for spent fuel subassembly ID number on lifting latch.

D. Operating Practice Employed at FCA

The FCA is of the horizontal split type used for fast-reactor physics research. Each half of the split assembly consists of a matrix of horizontal stainless steel tubes. Dimensions of the individual tubes are 55.2 mm square by 1324 mm long, with walls 1 mm thick. As shown in Fig. 19, the two halves are positioned face to face. One of the halves is stationary and the other moves horizontally on steel rails away from or toward the fixed half.

When these two halves stand open, the two faces of the assembly are accessible for hand loading and unloading of fuel drawers. As shown in Fig. 20, fuel plates, blocks, and structural material are stacked in the fuel drawers to simulate the desired composition of the reactor core, blanket, and reflector. When the core, blanket, and reflector configurations are completed by loading drawers into the stainless steel matrices, the two halves are closed to make the reactor critical.

The reactor operating power is usually very low, a few tens of watts to a few hundred watts. The operation mode is one shift per day and 4 days operation per week. The total inventory is about 300 kg of plutonium, 200 kg of 93% enriched uranium, and about 350 kg of ^{235}U in 20%-enriched uranium, so that the FCA is selected in this lecture as a typical example of sensitive item facilities.

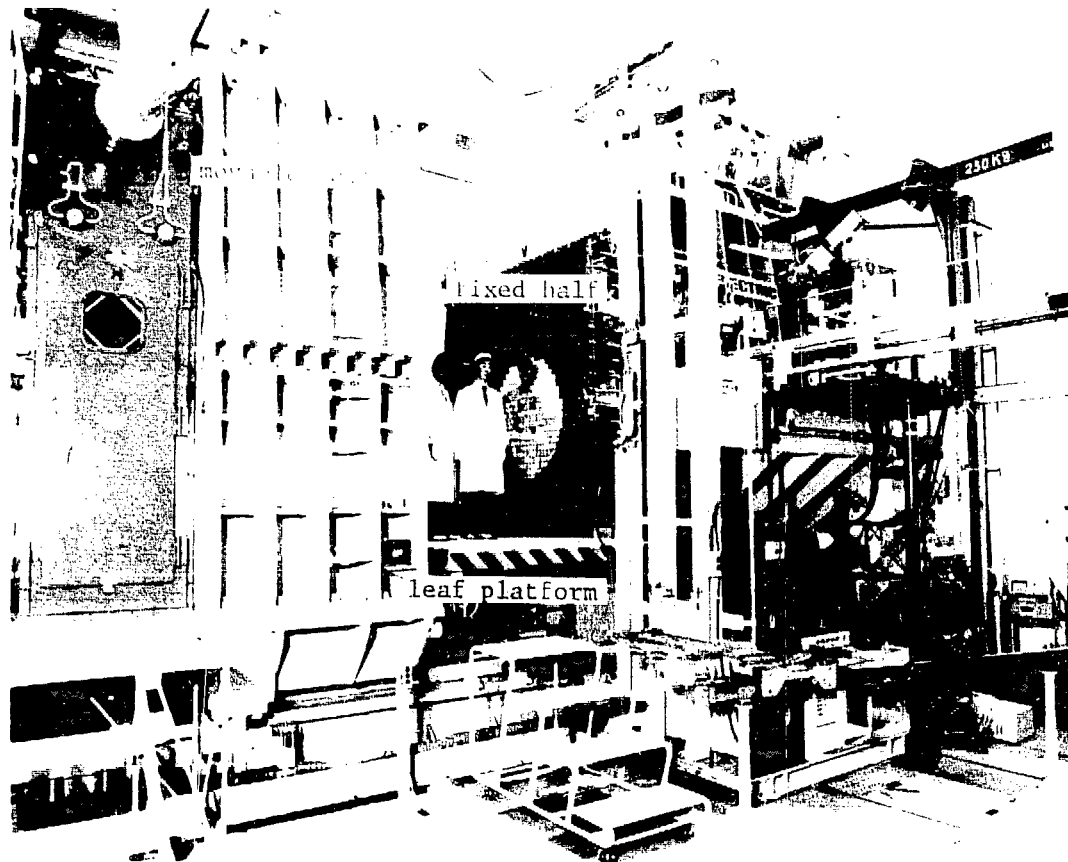


Fig. 19. FCA: Split-table-type fast critical assembly.

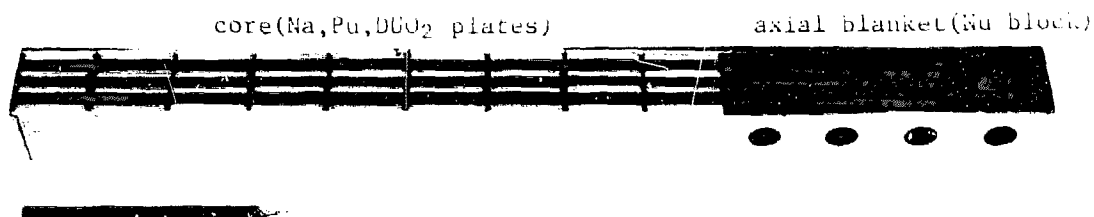


Fig. 20. Typical fuel drawer of FCA.

Fuel elements for experiments in the FCA are metallic plates and blocks. These items are carefully machined to stack well in the fuel drawers and are controlled in quality and quantity. Typical examples of fuel plates are shown in Fig. 21. Of course, the plutonium plates are hermetically sealed in stainless steel cladding. Enriched uranium plates are coated with a KEL-F type of nonhydrogeneous resin. As shown in Fig. 21, an identification number is engraved on each plate.

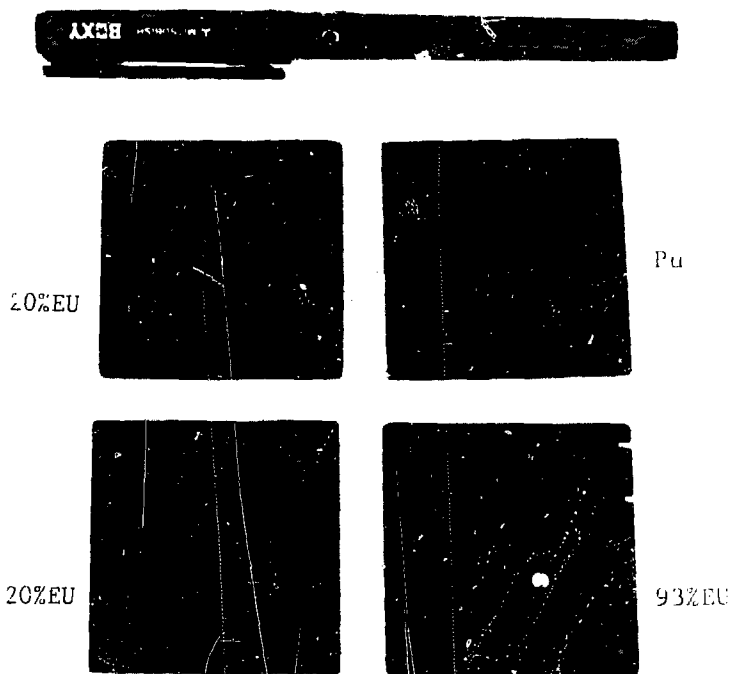


Fig. 21. Typical fuel plates of FCA.

2. Location of Fuel and Mode of Fuel Transfer. The layout of the FCA is shown in Fig. 22. The fuel is located in two KMPs: KMP-A, assigned for the fuel storage vault, and KMP-B, assigned for the reactor containment building. The fuel-handling area is a transit area where fuel is stacked in fuel drawers or tested for safety regulations. It is against regulations to leave fuel elements in this handling area.

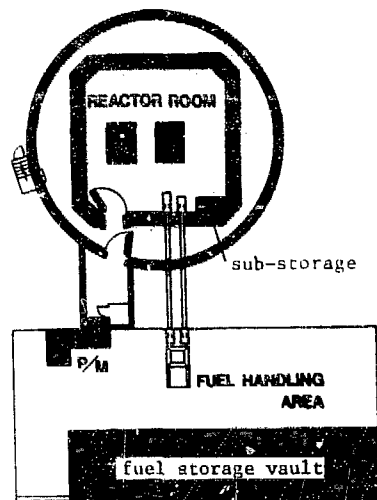
Due to nonroutine use of the facility, it is difficult to give the ratio of fuel inventories in KMP-A and KMP-B. It depends on what experimental program is going on. Usually, about 10% of the core inventory is stored in the substorage in KMP-B.

Two modes of core changes can be achieved by changing the arrangement of the fuel plates stacked in the fuel drawers (Fig. 20). One is a minor core change that can be achieved using only the fuel plates stored in the substorage in KMP-B; rearrangement of fuel plates in the fuel drawers is conducted in the reactor room. The other mode is a major core change that covers almost the whole core region, leading to fuel transfer from or into the fuel storage vaults. In this case, rearrangement of fuel plates in the fuel drawers is conducted in the fuel-handling area shown in Fig. 22. Minor core changes take place frequently (a few times a day at most); major core changes are infrequent (once or twice a year). Loading or unloading fuel drawers into or from each half of the machine assembly and rearrangement of fuel plates in the drawers are carried out manually.

Generally, critical assemblies have minimal inventory turnover, leading to no inventory changes for long periods. For the FCA, however, a trivial registrational transfer from the reactor MBA to the laboratory MBA is made just before every PIT of the reactor MBA, for dealing with broken pieces of depleted uranium oxide ceramics used for reactor mockup experiments. During loading and unloading, about a hundred ceramic plates are broken. The broken pieces are transferred into the laboratory MBA so that item accounting can be retained for the reactor MBA.

In the storage vault, plutonium fuel plates and enriched uranium plates are placed in bird-cage-type storage containers (Figs. 23, 24, and 25). Then each storage container is deposited at a location assigned for it in the storage vault. Each container contains fuel plates having the same specifications in size and composition.

3. Accounting Procedures. The batch and item structure used for fuel plates at the FCA depends on the location of the fuel plates.



CONTAINMENT BY REACTOR BUILDING

Fig. 22. Illustration layout of FCA facility.

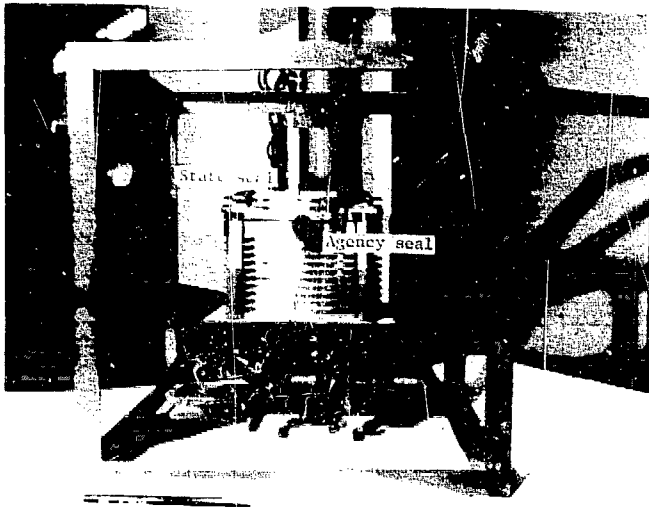


Fig. 23. Plutonium fuel plate storage.

Fig. 24. Inside view of plutonium plate storage container (40 plates inside).

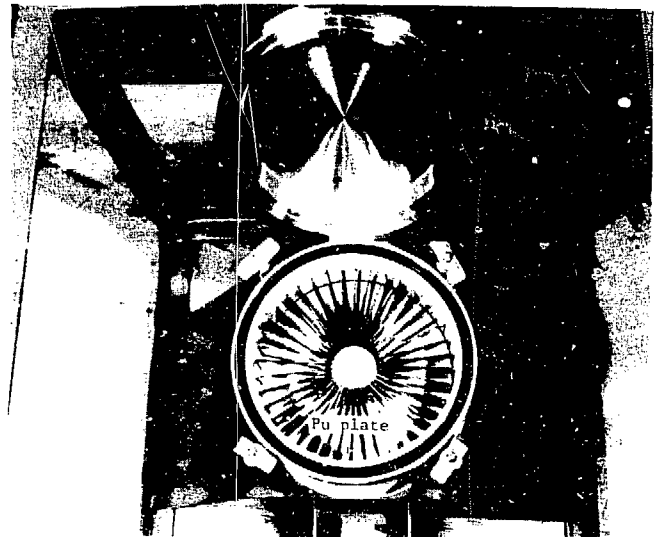
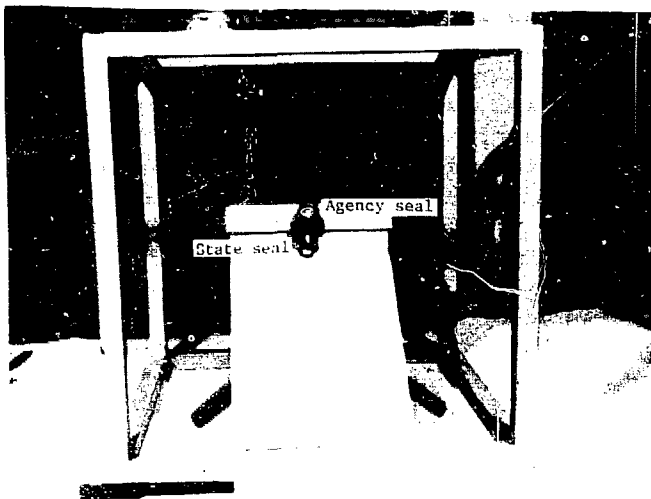


Fig. 25. Enriched uranium plate storage container.



(a) In the storage vault, an item is defined as one fuel plate, and a batch is defined as one storage container holding fuel plates having the same specifications. The corresponding batch data consist of:

- Item identification by color code
- Number of items
- Weights of total and fissile materials and chemical compositions based on shipper's data
- Assigned location in the storage vaults

(b) In the reactor core, an item is defined as one fuel drawer and a batch is defined as a number of drawers containing the same nominal weights of each type of nuclear material. The corresponding batch data consist of:

- Number of items
- For fuel drawers, the number of fuel plates and other nuclear material counted by standard size
- For fuel drawers, the weights of total and fissile materials and chemical compositions and physical form based on shipper's data
- Location in the core

(c) In the substorage area, an item is defined as one equivalent fuel plate of standard size, and a batch is defined as fuel plates of the same chemical and isotopic composition and physical form. The responding batch data consist of:

- Number of items, namely, number of fuel plates counted by standard size
- Weights of total and fissile materials and chemical compositions based on shipper's data
- Location in the substorage area

Except for decay correction of ^{241}Pu , there is no significant nuclear loss or production, owing to low-power operation of the assembly. The decay of ^{241}Pu must be adjusted only when shipment of plutonium fuel plates takes place.

Material accounting is conducted based on item counting, item identification, and shipper's data. Due to the high attractiveness, high accessibility and high concealability of the fuel used at the FCA, material accounting relies on containment and surveillance measures to a great extent, especially at the facility level. In principle, identification of items is achieved by attribute NDA measurements on a random sampling basis.

One of the specific features of fuel transfer in critical assemblies is frequent changes of batch composition. While a fuel subassembly defined as one batch may not be manipulated on the facility site, a fuel drawer defined as one batch for the FCA is changed frequently in composition for conducting experiments. Such batch composition changes that result in no inventory change at each KMP must be reported and recorded as "rebatching." The number of fuel transfer transactions issued in 1981 was:

(a)	Inventory change	1	
	• Receiving		0
	• Shipping		1 (from reactor MBA to laboratory MBA)
	• Nuclear loss		0
	• Nuclear production		0
(b)	Inter-KMP transfer	205	
(c)	Rebatching	524	

4. Containment and Surveillance. Storage of nuclear material, even temporarily, is not allowed at locations other than the fuel storage vault and the reactor room. The fuel drawer make-up for major core changes is conducted by the responsible facility personnel in the fuel-handling area shown in Fig. 22. All fuel storage locations are under continuous surveillance by intrusion detectors. In addition, a highly sensitive SNM detector is at the exit door of the fuel-handling area to detect unauthorized removal of fuel. The combination of these measures can detect noncompliance with nuclear material control and assure continuity of the inventory knowledge. In this context, at the facility level, measures used for physical protection overlap considerably.

The State and the IAEA apply seals for assuring containment of the nuclear fuel in the storage vault. As shown in Figs. 23 and 25, the State and the IAEA apply independent seals for each container in the storage vault. No containment and surveillance measures have been implemented yet for fuel in the reactor room because the technology is still premature. Research and development efforts are currently concentrated on implementing a personnel portal monitor to assure containment of nuclear fuel in the reactor room and thereby alleviate the burden associated with frequent inspections conducted by the IAEA and the State.

The personnel portal monitor is a multisensor system accommodating metal detectors, intrusion sensors, and a CCTV surveillance-recording system. It can detect removal of a single metallic fuel plate and can also detect any tampering actions without hindering normal passage of personnel. The system is designed for unattended use, with a remote interrogation capability for the State or IAEA. As illustrated in Fig. 26, there are two metal detection units: MD1 and MD2. The MD2 is used by the facility operator to check for careless removal of metal objects, such as tools. MD1 is used by the State or the IAEA to detect unauthorized removal of fuel plates from the reactor room. The combination of a CCTV system and intrusion sensors is used for detecting tampering. The portal monitor is located at the passage between the reactor room and the fuel-handling area as shown in Fig. 22. Relevant field tests are being conducted by the IAEA and the operator, under the coordination of the Japanese support program to the IAEA, before the portal monitor is implemented.

5. Inspection.

a. Routine Inspection. Two routine inspection modes are applied at the FCA for the time being. One mode is a routine inspection that consists of seal integrity examinations and book audit examinations focusing on fuel in the storage vaults. The other mode is a routine inspection that consists of, inter alia, item counting and attribute NDA measurements based on random sampling; it focuses on fuel in the reactor room. The NDA measurements now being applied are being used temporarily until such time as satisfactory containment and

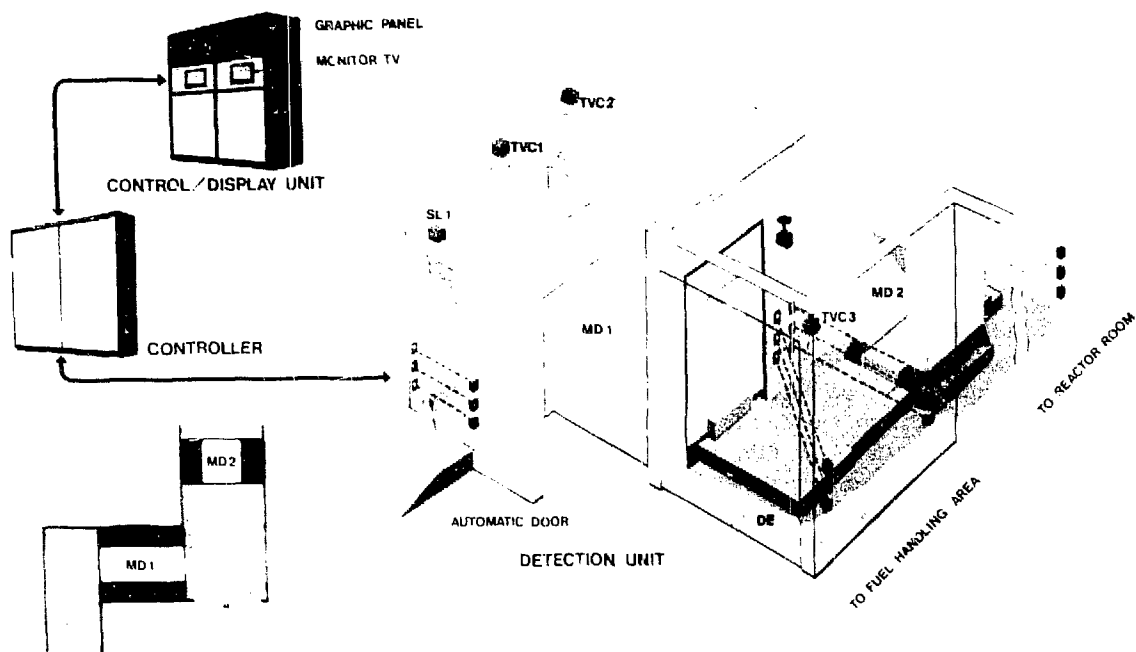


Fig. 26. FCA portal monitor system.

surveillance techniques are available for the reactor room. Such NDA measurements are conducted by the IAEA, with State inspectors acting as observers.

Routine inspections are conducted every 2 weeks, one mode after the other. Normally, two State inspectors and two IAEA inspectors take 1 day for inspection of the storage area. Inspection of the reactor area takes 2 days, including setting up and checking NDA equipment; it is usually conducted by two state inspectors and three IAEA inspectors. For supporting each routine inspection, about a 10 man-day effort of the facility operator is needed.

In order to minimize the impact on reactor operation, NDA equipment is set up on a Friday and measurements made of fuel in the core on a Monday. Major routine inspection activities consist of:

- Explanation of updated inventory summary paper by operator
- Book audit
- Fuel storage vaults
 - (1) Visual check of seal integrity and identification numbers
 - (2) Seal replacement on a random basis of about 5% of all seals (as a check for tampering)
- Reactor room
 - (1) Item counting of fuel in the reactor core and substorage
 - (2) Random sampling of items from the core and substorage

- (3) Attribute NDA measurements
- (4) Return of sampled items to the reactor or substorage

b. Physical Inventory Taking (PIT). PIT must be conducted whenever a major core change takes place, but at least once a year. PIT is conducted simultaneously by the IAEA, the State, and the operator with independent measurements using the sampling plan calculated by the State. PIT includes the following activities:

- Preparation of updated inventory summary paper
- Record audit
- Visual check of seal integrity and identification number
- Replacement of seals on a random sampling basis
- Item counting of all items without seals
- Stratification of fuel plates, and samples taken from each stratum on a random basis for NDA measurement
- Attribute NDA measurement of each sample
- Weighing of each sample
- Preparation of the PIT report

The sample size N_s for a stratum is calculated by the following equation,

$$N_s = N \cdot (1 - \beta\gamma) ,$$

where β is the nondetection probability of 0.05, γ is the ratio of the fissile weight contained in each item to the significant quantity given in Table I, and N is the total number of items in the stratum.

Provided that no seals are applied for fuel containers, typical sample sizes used at the FCA are:

<u>Stratification</u>	<u>No. of Items</u>	<u>Sample Size</u>
Plutonium	9,000	105
93% enriched uranium	3,000	25
20% enriched uranium	12,000	14

The NDA measurements are conducted by gamma-ray spectrometry using a germanium detector and an 4,000-channel analyzer. In principle, the attribute measurements are derived by comparing the intensity of the 414-keV characteristic gamma ray from ^{239}Pu or the 185-keV gamma ray from ^{235}U emitted from each sample with that emitted from the standard sample.

The measurement time for each sample is about 200 both for plutonium and enriched-uranium fuel plates, to obtain results within a 3% error. If a measured result differs more than 30% from the expected value, the item is considered to be defective.

The necessary man-power and days for conducting the PIT are subject to change, depending on the amount of fuel (with seals) that can be excluded from item counting and sampling. The following table shows the man-power and days spent for the PIT in the past several years.

<u>Year</u>	<u>Working Days</u>	<u>State (Man-days)</u>	<u>Operator (Man-days)</u>
1979	11	30	140
1980	16	54	205
1981	9	39	123

VIII. STAFFING AND RESOURCE NEEDS

At JAERI, facility rules of MCA (mentioned in Sec. VI) were first implemented in May 1978, conforming to the amendment of the Domestic Law for Control of Nuclear Reactors, Nuclear Fuel Material and Nuclear Source Material. Before implementing the facility rules of MCA, some facility-level MCA activities were conducted for reactor operational purposes. These included item counting, as a measure for assuring safety of operation; shipper's data examination; calculation of ^{235}U depletion and plutonium build-up; and reactor power calibration for assuring reactor performance. The major concern of these activities is to prevent mistakes in operation.

Conversely, the major concern of the MCA conforming to safeguards requirements is to detect intentional unauthorized movement of nuclear fuel. In this context, the major activities newly introduced for implementing the MCA are mostly associated with frequent and elaborate PIT, routine inspection, and compilation of official reports of material accounting in a timely manner. This section discusses the staffing and resources needed to fulfill these duties.

A. Staffing

Two sections accomplish the MCA for research reactors and critical assemblies at JAERI: the Nuclear Fuel Section, for accomplishing facility-independent common duties, and the Reactor Operation Section, for accomplishing facility-specific duties.

1. Nuclear Fuel Section. At JAERI, there are six research reactor MBAs, four critical assembly MBAs, and one laboratory MBA. The major facilities operated by JAERI are listed in Table III. Reactor operations managers take primary responsibility for the MCA at their respective facility. As mentioned in Sec. VI, the Nuclear Fuel Section has been established to maintain the central ledger, to compile official reports of material accounting to the government, and to provide clerical services for inspection of all research reactors and critical assemblies operated by JAERI. Four persons are now working full time in the Nuclear Fuel Section.

2. Facility Level. As mentioned in the foregoing sections, there are no bulk materials or analytical measurements in the itemized research reactor facilities. Therefore, the MCA responsibility and the reactor operation responsibility at the facility level can be overlapped, which is better than establishing a separate MCA responsibility because the MCA activities can be accommodated with minimal intrusiveness and expenditure.

Considering the type of work and technical capability necessary to conduct a PIT and to assist routine inspection, it may be appropriate to use the reactor operation staff, not full-time assignees, for the MCA at the facility level.

TABLE III

SUMMARY OF MCA ACTIVITIES AT MAJOR FACILITIES OPERATED BY JAERI

Facility Name	Operating Power	Nuclear Fuel	No. of Transfer Transactions Issued/Year			Inspection Days and Man-Power/Year		
			Inventory	Inter-KMP	Rebatch	Days	ARIE-1	ARIE-2
JRR-2	10 MW	93% EU 20 kg	140	130	0	2	2	1
JRR-3	10 MW	1.5% EU 11. Ekg, plus 8 kg of Pu	50	50	0	12	25	18
JRR-4	3.5 MW	93% EU 10 kg	4	6	0	1	2	1
NSRR	0.3 MW pulse 2×10^4 MW	20% EU	0	0	0	1	1	1
		2.2 Ekg	0	0	0	1	1	1
JMTR	50 MW	93% EU 80 kg	500	400	0	15	40	30
FCA	10-100 W	Pu 300 kg 93% EU 200 kg, 20% EU 70 Ekg	1	550	200	50	200	200
Laboratory use			1000	500	500	8	20	10

A summary of the MCA activities performed at the major research reactors and critical assemblies operated by JAERI is given in Table III. This table shows the total number of reactor fuel-transfer transactions issued for a year and the annual routine inspection effort in man-days for the State inspector (ARIE-1) and for the IAEA inspector (ARIE-2). However, these figures should be interpreted as rough estimates because they change, depending on various circumstances.

From the allocations shown for ARIE-1 and ARIE-2, it is clear that inspection activity is focused on facilities that contain more than one significant quantity of plutonium and highly enriched uranium. In such facilities, it is obvious that reactor operation personnel should be used at the facility level for accommodating inspection and PIT. According to the author's observation, the increment of operation personnel needed at the facility level (ARIE-3) is roughly estimated by:

$$\text{ARIE-3} = F \times \text{ARIE-1} \quad \text{for Level II (State)}$$

$$\text{ARIE-3} = F \times \text{ARIE-2} \quad \text{for Level I (State)}$$

where F is a multiplier of 2.5 to 3.5

In summary, a 1 man-year increment for JRR-3 and JMTR and a 3 man-year increment for the FCA (other than the Nuclear Fuel Section) may be reasonable.

B. Resource Needs

The cost of the resources needed at the facility level for implementing the MCA is categorized into three parts: operating cost, equipment cost, and relevant research and development cost.

1. Operating Cost. Operating costs are given in terms of man-years of effort since the cost of salaries, fringe benefits, and other relevant administrative cost can vary over a wide range. To obtain the total cost for 1 man-year, a factor of 2.5 is common as a multiplier of an individual's salary. As described in the foregoing sections, the MCA activities at the facility level are conducted by personnel who perform reactor operation in research reactors and critical assemblies. The average salary of this staff is assumed to be \$25,000, which is also the salary assumed for personnel in the Nuclear Fuel Section.

According to the empirical relation mentioned in the previous section, the accumulated increment of man-power at the facility level for implementing the MCA is estimated to be 1 man-year for JMTR and 3 man-years for FCA (negligible for JRR-4). Rough estimates of the cost are:

JMTR	$\$25,000 \times 2.5$	$= \$62,000$
FCA	$\$25,000 \times 3 \times 2.5$	$= \$187,500$
JRR-4	Negligible	
Nuclear Fuel Section	$\$25,000 \times 4 \times 2.5$	$= \$250,000$

2. Equipment Cost. Since there is no bulk material in research reactors, neither direct measurements nor NDA measurements are carried out. Item identification is performed by visual checks of identification numbers engraved on the surfaces of fuel subassemblies, with the assistance of a TV camera, if appropriate. For critical assemblies, attribute NDA measurements are conducted for item identification of small pieces of fuel elements on a random sampling basis. The only expenses needed for the Nuclear Fuel Section are for computer software and running time. Thus, the equipment cost for implementing the MCA, excluding the cost of equipment for physical protection, is estimated as follows:

JMTR	\$15,000 for underwater TV camera
JRR-4	\$15,000 for underwater TV camera
FCA	\$40,000 for gamma-ray spectrometer \$40,000 for HLNCC
Nuclear Fuel Section	\$30,000 for computer software \$20,000 for computer running time

The estimated cost of annual maintenance and replacement parts account for 10 to 20% of the total. Although these costs should be interpreted as approximate, it is apparent that equipment costs are a minor part of resource needs for implementing MCA in research reactors and critical assemblies.

3. Research and Development Cost. As described in parts 1 and 2 of this section, most of the cost of implementing MCA is due to personnel. When the IAEA and STATE inspector cost is taken into account, personnel cost may become an unreasonably large portion of the total safeguards cost. Evidently, a great amount of personnel cost is necessary at facilities where a large number of ARIE is allocated. For such facilities, the investment for developing appropriate containment and surveillance techniques might reduce the number of ARIE. One example may be seen in the development of a personnel portal monitor at the FCA. About \$500,000 has been allocated for this purpose.

Since most of the existing reactors were built at a time when safeguards was not a major concern, safeguards measures generally were not considered in the design of facilities. Consequently, some modifications of existing facilities and their operation modes may be necessary for adopting containment and surveillance techniques. There is little incentive for spending research and development funds for facilities that require a medium amount of inspectors, like JMTR. No incentive is found for using the funds for facilities that use few inspectors, like JRR-4.

IX. PROBLEMS ENCOUNTERED AND LESSONS LEARNED

Specific problems encountered and lessons learned in implementing MCA activities for research reactors and critical assemblies have been mentioned for each reactor in the foregoing sections. In this section, the author will add a few touches on general considerations.

A. Application of Containment and Surveillance (C/S)

MCA for research reactors and critical assemblies can rely, to a great extent, on item identification and C/S measures. However, safeguards measures were not usually considered in the design of these facilities. Consequently, application of C/S at present is suffering from the inconvenience of setting up and checking C/S equipment. This type of inconvenience could be alleviated at a small cost, if appropriate consideration is given to installing C/S equipment at an early stage of facility design.

Since no direct measurements are conducted, inspectors of research reactors and critical assemblies having large inventories spend most of their time wiring containers, racks, or subassemblies of fuel, using flashlights to read seal identification numbers, or counting fuel items. While the mission of inspection is very important, the type of work currently needed for conducting inspections is like a typical odd job. Considering that IAEA and State inspectors must be highly qualified people because of certain features of the mission, introduction of updated technologies might be a high priority for improving the cost effectiveness of inspections, especially for reactors with large inventories.

B. On Perception Gap, Communication, and Mutual Understanding

Whereas technology is a very important element of safeguards, a more important element that should be touched upon is the objective validity of safeguards. This involves political problems.

Although the safeguards issue is related to the safety issues, there is a substantial difference between the underlying fabric of these two issues. Safety hazards are due to unknown phenomena, machine trouble, or human error, but safeguards hazards are always accompanied by an intentional human action.

Human action takes place as a result of various human interactions, either at the national or subnational level.

In this context, safeguards issues are sometimes more political than technical in nature. When it comes to political issues, mutual understanding of the thinking of different parties either at the international or domestic level is a key to peacefully overcome difficulties encountered. The fundamental way people think in a given community is usually related to the cultural background particular to the community.

For many reasons, such as the isolated island geography of Japan, the homogeneous race and cultural background, and the generally enjoyable climate, Japanese people used to be unfamiliar with the meaning of security and safeguards. Therefore, after the new concept of international safeguards was introduced, many perception gaps in looking at safeguards have been observed. These gaps are apt to be found between different parties; for example, the inspector and those inspected. A different perspective is found in the sceptic who suspects that the safeguards specialist seems too willing to emphasize unrealistic scenarios. Therefore, one of the most difficult tasks for us has been to improve the mutual understanding of different parties by extensive communication.

As mentioned before, the fundamental way people think depends on their cultural background. Therefore, communication should be developed case by case, considering the specific features of each community, and preferably, the communication should make use of examples.

Recently, Japanese utility people had an opportunity to exchange safeguards views with a guest from the United States. He is an influential person in international safeguards. It was very impressive for me that in this discussion he referred to a famous Japanese garden to help communication with Japanese people. Let me tell you a story about it with certain modifications.

Figure 27 shows a rock garden at the RYOAJI Temple in Kyoto, Japan. The designer is unknown, but he must have lived at least 500 years ago. As seen in the picture, the garden is seemingly very simple. All you can see is a few rocks on a carpet of clear white sand. The garden used to provide ZEN philosophers with a space for meditation for a long time. For meditation, what does the white sand mean--Ocean, Space, or Universe? And Continent or Earth for rocks? It depends on your imagination.

Now let me interpret as follows. Place a few people on different sides of this garden and then ask them to count the number of stones in the garden. Now ask them for their results. To your amazement, all the observers will come up with different answers. Some of them will have counted 14, some of them 13, and some of them even as few as 12. Now what is right? In fact, none of them, since there are 15 stones in the rock garden. It is the result of the clever and ingenious arrangement by the designer that, depending on your relative position around the garden, you will observe a different number of stones, and no one can observe 15 stones from any one position.

What I am trying to illustrate is this: Several persons can take a look at the same object at the same time and yet come up with different points of view, all depending on their relative position and their perception of the object. If communication is poor among them in such a situation, the situation may be serious enough to introduce suspicious consequences among them. Only elaborate communication can develop mutual understanding of their positions and develop further mutual cooperation to find out the right number of stones. Different points of view and different perceptions are even necessary in the beginning. Otherwise, they will miss a chance to know what is the truth.

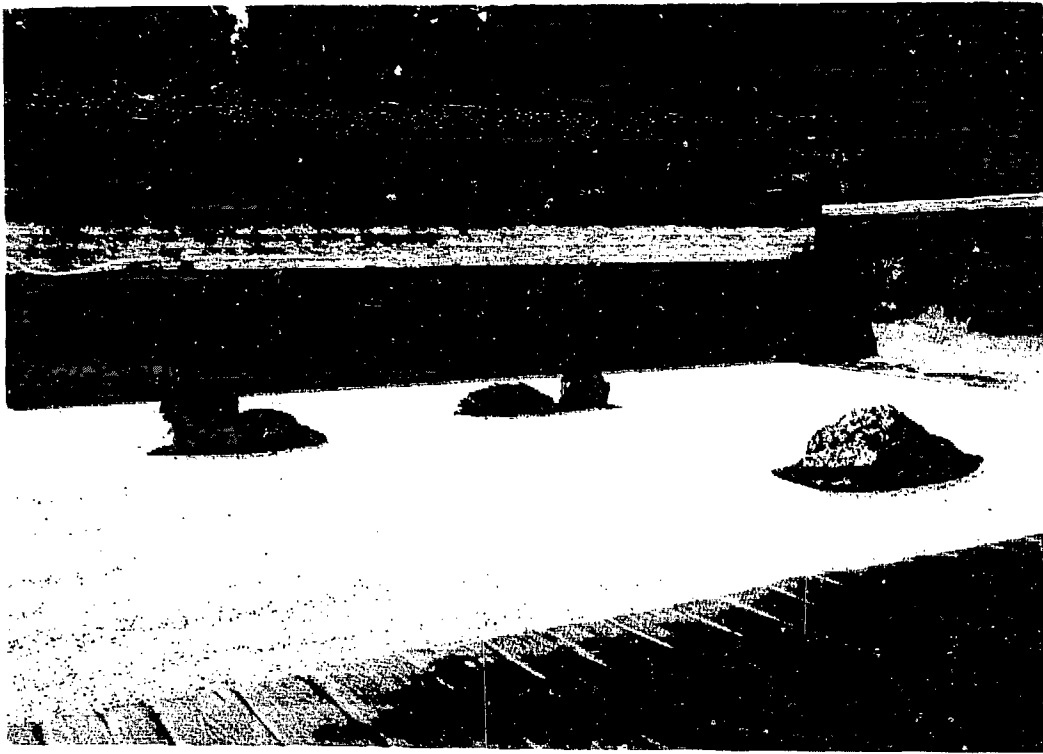


Fig. 27. Rock garden at the RYOANJI Temple in Kyoto, Japan.

This philosophy could also well be applied for effective implementation of safeguards. Since safeguards do not produce any direct commercial profit, mutual understanding and consensus are absolutely necessary to obtain the staff and resources needed.

I believe that one of the major purposes of this course is to provide us with an excellent chance for good communication and to develop mutual understanding and cooperation among all of us from different countries.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

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REACTORS AND CRITICAL FACILITIES

This session is intended to provide the technical details of an effective SSAC which member states may establish and maintain for research reactors and critical facilities. Facility-level systems designed along these lines would meet national and international requirements for nuclear material accounting and control.

After the session, participants will be able to:

- Describe the material control and accounting requirements and establish materials balance areas for a typical research reactor facility.
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- Discuss the interaction of the facility system with the IAEA system.

ISPO-237

FINAL REPORT

DETAILED DESCRIPTION OF AN SSAC AT THE FACILITY LEVEL

FOR CRITICAL FACILITIES

March, 1985

USA

PROGRAM FOR
TECHNICAL ASSISTANCE
TO IAEA SAFEGUARDS

IAEA

Department of Energy

Office of Safeguards & Security

PREPARED BY

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Foreword

During the course of the preparation of this report many have questioned the purpose of the report and its role in international safeguards in relation to existing documents such as INFCIRC/153 (corrected), INFCIRC/66/Rev. 2 and IAEA/SG/INF/2. The first two documents provide requirements for IAEA safeguards systems while the latter provides the basic structure for State Systems of Accounting for and Control of Nuclear Material (SSAC). This report is intended to provide the technical details of an effective SSAC which Member States may use, if they wish, to establish and maintain their SSACs. It is expected that systems designed along the lines described would be effective in meeting the objectives of both national and international systems for nuclear material accounting and control. There is no intention that this report would add to, subtract from, or amend in any way the rights and obligations defined for the IAEA and States in documents INFCIRC/66/Rev. 2, INFCIRC/153 (corrected) or any specific safeguards agreement concluded with the IAEA pursuant to these documents.

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Part 1. INTRODUCTION

1.1 General

There are two primary objectives toward which the development and maintenance of a state system of accounting for and control of nuclear material (SSAC) can be directed. There is a national objective to provide for detection of nuclear material losses or unauthorized use or removal of nuclear material in the interest of public health and safety as well as for economic interest. There is also an international objective to provide the basis for the application of IAEA safeguards pursuant to the provisions of agreements between the State and the IAEA. While systems designed specifically for one or the other of these objectives may differ in some details there are many technical elements which are the same. To assist Member States with regard to the system directed toward the international objective the IAEA issued IAEA/SG/INF/2, "Guidelines for States' Systems of Accounting For and Control of Nuclear Materials". The guidelines describe the legal structure and organizational and functional elements of an SSAC at both the State and the facility level "for the organization and functions of the SSAC with respect to obligations arising from Safeguards Agreements concluded by a State with the IAEA". Although the guidelines provide a basic structure for an SSAC, some States have expressed a need for more detailed guidance with regard to the technical elements in the design and operation of SSACs for both the national and the international objectives. To meet this need the present document has been prepared, describing the technical elements of an SSAC in considerable detail.

It is intended that this document will not only assist States in establishing their national systems, but contribute to their understanding of the IAEA's needs in carrying out its obligations, and thereby help make international safeguards more effective, efficient, and less burdensome to the State and operator. It is also hoped that the document will be a useful training tool for those responsible for the operation of an SSAC at either the state or the facility level.

It is emphasized that the descriptions in this document are of possible SSAC's, and are not meant to be prescriptive in nature. There are many ways to design an effective SSAC which will meet national objectives for nuclear material accounting and control and at the same time will enhance the IAEA's ability to achieve its objectives. In fact, the SSACs of a number of states show considerable variation, while still being satisfactory from the IAEA's point of view. Consequently, the material presented here should be regarded as illustrative and representative of good practice. That is to say, on the basis of its experience the IAEA is able to say that SSACs designed and operated at the facility level along the lines described here are capable of meeting national system objectives as well as fulfilling the undertakings of the State with respect to international safeguards, and of enhancing the ability of the IAEA to carry out its responsibilities.

1.2 Principal Elements of an SSAC

The guidelines presented in IAEA/SG/INF/2 provide a basic structure for an SSAC but do not address the details of the specific elements of SSACs. The technical and functional elements presented in the structure are interrelated, in that the same subjects are addressed in different ways, depending on the application. For example, material measurements are considered with respect to establishing MBAs, with respect to material flows, and with respect to physical inventory taking. The guidelines in IAEA/SG/INF/2 include seven categories of information which can be defined and described so that they include all of the elements of an SSAC. By considering these seven categories of information as principal elements of an SSAC, a detailed description for each of the seven can be prepared to cover all the subject matter and to show the relationships of the SSAC functional elements and requirements identified in IAEA/SG/INF/2. The seven principal elements are:

- (1) Nuclear Material Measurements
- (2) Measurement Quality
- (3) Records and Reports
- (4) Physical Inventory Taking
- (5) Material Balance Closing
- (6) Containment and Surveillance
- (7) Organization and Management

Two levels of an SSAC may be defined: the State Authority level and the facility level. While the functional separation between these two levels may vary from State to State, in general the Authority level establishes the requirements and criteria for the SSAC and the facility operator implements the system in compliance with Authority requirements and criteria. The guidelines presented in IAEA/SG/INF/2 addressed both levels of an SSAC. The detailed descriptions of the principal elements presented in this document address only the facility level of an SSAC with the Authority level being addressed only to the extent necessary to show relationships between the two levels.

1.3 Purpose and Scope of this Document

The purpose of this document is to provide a detailed description of a system for the accounting for and control of nuclear material in a critical facility which can be used by a facility operator to establish his own system in a way which will provide the necessary information for compliance with a national system for nuclear material accounting and control and for the IAEA to carry out its safeguards responsibilities. A discussion of the IAEA to carry out its safeguards responsibilities. A discussion of the IAEA safeguards approach for critical facilities is presented in IAEA STR-87.

The scope of this document is limited to descriptions of elements 3, 4 and 5 of the first five of the seven principal SSAC elements identified in 1.2 above. Elements 1 and 2 are not included because there are essentially no measurements made for nuclear material control and accounting purposes at a critical assembly facility. The last two elements, Containment and Surveillance and Organization and Management have not been included because it was thought more useful to concentrate on the nuclear material accounting aspects of the SSACs. The other two elements may be treated at a later time. The descriptions of the three elements will be at the facility level, with specific examples for each of the three for facilities like the reference facility. By the conclusion of the document a detailed description of an SSAC for the reference facility will have been provided. Some of the examples will be based on discussions in the published safeguards literature cited in the list of references at the end of each Part. These discussions will not be

repeated in detail here. To understand the examples more fully, the reader is urged to consult the references.

Some discussion is also devoted to the IAEA's philosophy and practices in the belief that a clearer understanding of its needs will foster that spirit of cooperation so essential to the application of effective and efficient safeguards with a minimum of intrusion.

The descriptions in this document are not intended to add to, subtract from, or amend in any way the rights and obligations defined for the IAEA and the States in documents INFCIRC/66/Rev.2, INFCIRC/153 (Corrected), or any specific Safeguards Agreements concluded with the IAEA, but only to provide information which States may find useful in the exercise of their rights and the discharge of their obligations under their national requirements and under their agreements with the IAEA.

1.4 The Reference Facility

It is useful to base the examples in a guide such as this on a reference facility. There are two basic types of critical assemblies, the vertical integral and the horizontal split-table. These are described briefly in Appendix A and at greater length in two references (IAEA-STR-87, LA-7315). The nuclear material accounting and control procedures are essentially the same for the two types, i.e. accounting for and control of many small pieces containing nuclear material. For this reason only one type, the horizontal split-table type has been described in Appendix A as the reference facility. The example given will be based on this reference facility.

References for Part 1:

IAEA/SG/INF/1: IAEA Safeguards Glossary, IAEA, Vienna, 1980.

IAEA/SG/INF/2: Guidelines for States' Systems of Accounting for and Control of Nuclear Material, IAEA, Vienna, 1980

INFCIRC/66/Rev. 2: The Agency's Safeguards System, IAEA, Vienna, 1968.

INFCIRC/153 (Corrected): The Structure and Content of Agreements Between The Agency And States Required In Connection With The Treaty On The Non-Proliferation Of Nuclear Weapons, IAEA, Vienna, 1971.

IAEA STR-87: S. Ermakov, Safeguards Approach for Critical Facilities with Large Inventory of Nuclear Material, System Studies Section, Division of Development and Technical Support, IAEA Department of Safeguards, STR-87, April 1979.

LA-7315: D. D. Cobb, J. L. Sapir, E. A. Kern, R. J. Dietz, Concepts for Inventory Verification in Critical Facilities, Los Alamos Scientific Laboratory, Report LA-7315, December 1978.

Part 2. RECORDS AND REPORTS

2.1 Introduction

Reports based on the records are the results of the nuclear material accounting system. The verification and evaluation by a national or international safeguards authority of these reports and their supporting records and accounting system make possible statements of the effectiveness of safeguards. It is the consistency of the data that appear in the various related levels of the records and reports that provide the basis for identification of anomalies which may indicate a loss or diversion of nuclear material. Depending upon the national regulations and the local situation, the preparation of data to be transmitted to the IAEA may occur at various places. In some cases it will be the facility itself, where the accounting reports would be compiled and completed in the very form in which they ultimately reach the IAEA. In other cases, the facility personnel will only prepare the data and transmit them to some other body, e.g. a national authority, which will then complete the reports to the IAEA. In any case it is the facility data that are basic to the final reports to the IAEA. The following description provides information regarding a system of records and reports which includes the necessary data elements for a nuclear material accounting system that will permit verification and evaluation. The general discussion in Secs. 2.2 - 2.5 is followed by detailed examples in Sec. 2.6 showing how, in typical cases, the various record and report forms would be filled out. The examples are given for a system of records kept manually. Many facilities would have computer systems but the data elements would be the same.

2.2 Accounting Records

Accounting records are defined in the IAEA Safeguards Glossary (IAEA/SG/INF/1) as a set of documents kept at a nuclear facility showing the quantity of each type of nuclear material present, its distribution within the facility, and any changes affecting it. To provide an effective nuclear material accountancy system a primary requirement is that the accounting records accurately show the nuclear material inventory and all inventory changes that affect it. Another requirement is that the accuracy and

consistency of such records is capable of verification. Accounting records consisting of the following three components can meet these requirements at a critical assembly facility:

- a) Ledgers. These are the records in which the inventory changes are summarized and totaled for specific time periods.
- b) KMP perpetual inventories. These are the records which reflect the internal disposition of the inventory times and changes in batch composition.
- c) Supporting documents. These provide the primary data to be entered into the records for each inventory change including internal batch and item changes.

In facilities where there are a large number of inventory change transactions, Inventory Change Journals (sometimes called transaction journals) are used as records of original entry which are periodically totaled and entered into the ledgers. Such journals are not needed at a critical facility because of the limited number of inventory change transactions.

2.2.1 Ledgers

A ledger is the record of final entry which provides a summary of all inventory changes to determine the book inventory, i.e. the quantity of nuclear material that should be present at the facility. Because no journals are maintained, entries to the ledgers are made on the basis of the supporting documents for the respective accounts. At a critical facility these entries normally would be receipts and shipments or perhaps de-exemptions and exemptions. Separate ledgers should be kept for different material types. In the reference facility these would need to be four ledgers, i.e. plutonium, enriched, natural and depleted uranium. Figure 2.1 is an example of a general ledger format that could be used. It also may be useful to keep subsidiary ledgers for the inventory KMPs but the totals for the respective KMP perpetual inventory lists could serve the same purpose, i.e. provide a book inventory total for each KMP.

Figure 2.1

General Ledger

Facility				MHA:		Element Code:		Isotope Code:		Unit: Kg/g		
Material Description:												
Line No.	Date	ICD No.	Receipts		Other Additions		Shipments		Other Removals		Inventory	
			Element	Isotope	Element	Isotope	Element	Isotope	Element	Isotope	Element	Isotope
1												
2												
3												
4												
5												
6												
7												
8												
9												
10												
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2.2.2 KMP Perpetual Inventory Records

At a critical facility, inventory changes normally are limited to infrequent receipts and shipments. Once material has been received at the facility there normally is no change that affects the quantity of the facility inventory unless more material is received or some is shipped away, e.g. broken fuel pieces. However, at the critical facility internal transfers are of more importance than the infrequent changes due to receipts or shipments. Even though internal transfers do not change the inventory size they do change the location of the material and usually change the batch composition. It is important for the physical inventory taking by the facility operator and physical inventory verification by the IAEA that the results of these internal transfers be recorded. Detailed knowledge of the disposition of items and batches within the facility can be maintained by recording the results of the internal transfers in some manner that is capable of being changed each time an internal transfer occurs, i.e. perpetual inventory up-dates. Any format or procedure can be used that will show which items and batches are on inventory in each KMP including which are in the reactor core and which are in the intermediate storage area which exists at most facilities. One means of doing this is the use of perpetual inventory batch cards such as shown in Figure 2.2; and Figures 2.19 and 2.20 on pages 47 and 48. Note that space is provided for an inventory tag number so that these cards can be used in physical inventory taking. A new card or a revised card is prepared for each new batch that is formed when material is transferred within the facility. The old card is marked superseded so that one can tell whether one has the most current card or not. These cards in the examples are shown as being prepared manually. In most facilities computers would be available to maintain these data. While computer systems might not use separate batch cards but produce computer printed lists for each change, the principle is the same.

2.2.3 Supporting Document

Although, as noted above, inventory changes which affect the size of the facility inventory are infrequent at critical facilities, each such change is entered in the general ledger and each such entry should be supported by data

Figure 2.2

KMP-B Perpetual Inventory Card

Batch Identity: Batch Location:
Number of Items: Core Sign: ☐
Item Description:
Date: Date Superseded:
Inventory Tag Number: Card No.: PIC-

Item Composition

<u>Fuel Piece</u> <u>Description</u>	<u>Number</u> <u>of Pieces</u>	<u>Total Quantity (kg/g)</u>			
		<u>Per Item</u>		<u>Per Batch</u>	
		<u>Element</u>	<u>Isotope</u>	<u>Element</u>	<u>Isotope</u>

Continued to: PIC-

in supporting documents. Even though infrequent each such change should be fully described in supporting documents.

Even though internal transfers do not change the total facility inventory and they are not entered in the general ledger it is important that such transfer be fully described in supporting documents, i.e. internal transfer forms, to assure an accurate and complete statement of the composition of the inventory in each KMP. In addition, if the facility attachment called for batch follow-up, internal transfers involving changes in batch composition would need to be reported to the IAEA. Rebatching procedures are described in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements.

Supporting documents for either external or internal movement of materials provide the link between the data in the ledgers or perpetual inventory records and the original source data, which in the case of critical facilities would be primarily fuel fabricator's data. In particular, the supporting documents for internal transfers should provide complete item and batch data to facilitate physical inventory taking and verification. Supporting document data should include:

- a) Inventory Change Identity. Some means should be provided to differentiate a specific inventory change from all others so that it can be referenced in the records and traced during audit. This normally is done by consecutive inventory change numbers with perhaps a different series for each type, i.e. receipts, shipments or internal transfers.
- b) Date of Change. This should be the date that the inventory change actually occurred, not the date the supporting document was prepared, although the supporting document should be prepared on the date of the change if possible.
- c) Type of Change. For a critical facility this usually would be an internal transfer with an occasional receipt or shipment.

- d) **Material Description.** This should include identification of material type and isotopic enrichment, as appropriate.
- e) **Material Movement.** This would be the information which identifies a change in location of the material. i.e. receipt, shipment or internal transfer.
- f) **Batch Identity.** A batch is defined in the Safeguards Glossary as a portion of nuclear material handled as a unit for accounting purposes at a key measurement point and for which the composition and quantity are defined by a single set of specifications or measurements. Batch structure is discussed in section 2.5.2. One way of establishing and treating batches at a critical facility is described in Appendix A Section A.4. For internal transfers, in particular, the identity of all batches involved as well as identity of all items in the batches, should be included so that increases and decreases can be assigned to the appropriate batches. A rebatching procedure such as described in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements could be used, especially if such rebatching is to be reported to the IAEA.
- g) **Number of items.** The number of fuel pieces received as a fabrication batch, the number of fuel pieces or fuel drawers assigned to a facility batch or the number of items shipped as a batch.
- h) **Batch Data.** These are defined in the Safeguards Glossary as the total weight of each element of nuclear material and in the case of plutonium and uranium the isotopic composition when appropriate.
- i) **Batch Data Basis.** This should provide the basis from which the batch data entries were derived. For a critical facility this would be primarily fuel fabricator's data.

2.3 Operating Records

From a nuclear material accountancy standpoint operating records consist of the documentation of the results of facility systems and procedures which provide data relevant to accounting for nuclear material. At critical facilities such systems and procedures usually are limited to the Physical Inventory Taking procedures. Operating records are defined specifically in the Safeguards Glossary as:

- a) those operating data which are used to establish changes in the quantities and composition of nuclear material;
- b) the data obtained from the calibration of tanks and instruments and from sampling analyses, the procedures employed to control the quality of measurements, and the derived estimates of random and systematic error;
- c) a description of the sequence of the actions taken in preparing for, and in taking, a physical inventory, to ensure that it is correct and complete; and
- d) a description of the actions taken to ascertain the magnitude and cause of any accidental or unmeasured loss that might occur.

2.3.1 Nuclear Material Measurements

Items a) and b) in the Glossary definition of operating records would not be the actual result of the facility systems because the quantities of nuclear material at the facility are based on the fuel fabrication data. These data, however, would be included in the facility records in support of receipt or shipment entries in the ledgers. Any nuclear material measurements made at the facility normally would be NDA verification of receipts or during physical inventory taking. As discussed in section 2.2.2 the supporting documents which record the internal transfer of the fuel pieces are of importance to material accounting. These documents can be considered as operating records

resulting from the operation of the facility's system of material transfers. The only measurements made, however, normally would be piece count and item identity.

2.3.2 Physical Inventory Taking

Item c) in the Glossary definition of operating records deals not only with results of the physical inventory taking procedures but includes the procedures themselves. Part 3 addresses Physical Inventory Taking, including the use of written physical inventory procedures. Part 3 also addresses the documentation of physical inventory results in terms of the physical inventory item lists which provide source data for the physical inventory summary and reconciliation, which can be considered the supporting document for entering a MUF adjustment, if any, into the records.

2.3.3 Losses

Item d) in the Glossary definition of operating records refers to actions that would be non-routine, occurring only when an accidental or unmeasured loss occurs. The action to be taken following an accidental or unmeasured loss of nuclear material will vary depending on the degree of seriousness of the incident with regard to its effect on such things as facility operations, personnel health and safety and nuclear material accountancy. From a nuclear material accountancy standpoint the primary concern is to determine the quantity of material lost. Documentation of the actions taken and procedures followed in determining the quantity lost will permit evaluation of the incident in terms of reliability of the quantity statement and of the cause of the loss, i.e. was it an actual accident that would account for the loss or could some material have been diverted? The following information would be useful to include in documentation of the results of a loss investigation:

- a) to the extent possible the identity, material type and quantity of material missing;
- b) basis for establishing the quantity missing;

- c) the time and circumstances of the discovery of the loss; and
- d) the most probable loss mechanism

The information recorded for items a) and b) would provide source data to adjust the accounting records for the loss and to provide some estimate of the accuracy of the quantity missing. At a critical facility any loss would be expected to be one or more fuel pieces of one type or another for which the quantity of nuclear material should be known.

2.4 Accounting Reports

There are three primary types of reports associated with a nuclear material accountancy system: Inventory Change Reports, Material Balance Reports, and Inventory Reports (Physical Inventory Lists). Other reports may be called for by facility management or a State Authority but only the three primary types are addressed here. While safeguards agreements call for reports of these types to be submitted by or through a State Authority to the IAEA, it is the facility nuclear material accountancy system that provides the data for the reports whether they are prepared by the facility operator or by a State Authority. For purposes of discussion it will be assumed that the facility operator prepares these reports essentially as called for in applicable safeguards agreements.

2.4.1 Inventory Change Reports

Inventory change reports as being considered here are the periodic summary reports of all of the inventory changes that occurred during a given period. At a critical facility inventory changes that affect the size of the facility inventory occur infrequently. Often inventory change reports to the IAEA are called for only when an inventory change occurs. The data for these reports can be taken directly from the individual inventory change documents. The data requirements for inventory change reports that may be specified in safeguards agreements by State Authority or by facility management should be taken into consideration when the format and content of inventory change documents are established. Report preparation can be much easier if all of

the required data elements are in the inventory change documents and search of supporting documents is not needed to complete a report. It would further facilitate report preparation if the inventory change documents had essentially the same format as that used in the inventory change reports. Subsequent examination and verification also would be easier if there were consistency in format and content of the accounts and the reports. For NPT Safeguards Agreements reporting, the IAEA has established the Inventory Change Report format in Form R.01.1/c shown in Figure 2.3. Instructions for completion of this form are given in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements and will not be repeated here. Different formats have been established for inventory change reporting under other safeguards agreements, such as INFCIRC/66, but essentially the same data elements would be needed regardless of format.

2.4.2 Material Balance Reports

A material balance report is a statement of the balance of material on hand taking into account all inventory changes, adjustments and corrections, and a physical inventory. One of the adjustments would, of course, be the adjustment for the difference between the book inventory and the physical inventory, although at a critical facility this normally would be expected to be zero. The data for the material balance report should be available from the ledger accounts, including the physical inventory data, which are obtained from the physical inventory taking and become the ending inventory in the ledger. The data requirements for material balance reports that may be specified in safeguards agreements, by State Authority or by facility management, should be taken into consideration when the format and content of the ledger accounts are established. Report preparation can be made easier if all of the required data elements are in the ledger accounts and search of other accounting records is not needed to complete a report. Subsequent examination and verification also will be easier if there is consistency between the report and the ledgers. For NPT Safeguards Agreements reporting, the IAEA has established the Material Balance Report format in Form R.03 shown in Figure 2.4. Instructions for completion of this form are given in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements

Figure 2.3

INVENTORY CHANGE REPORT (ICR) FORM R.01.1/c

COUNTRY FACILITY MATERIAL BALANCE AREA										PERIOD COVERED BY REPORT: FROM TO REPORT No.										PAGE No. OF PAGES		SIGNATURE	
ENTRY No.	CONTINUATION	DATE OF INVENTORY CHANGE	MBA/COUNTRY		TYPE OF INVENTORY CHANGE	EMP CODE	NAME OR NUMBER OF BATCH	NUMBER OF ITEMS IN BATCH	MATERIAL DESCRIPTION	ELEMENT	ACCOUNTANCY DATA			UNIT Lb/g	WEIGHT OF FISSILE ISOTOPES (URANIUM ONLY) (g)	INVENTORY CODE	MEASUR. BASIS	CONCISE NOTE	CORRECTION TO				
			FROM	TO							WEIGHT OF ELEMENT	REPORT No.	ENTRY No.										
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2																							
3																							
4																							
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MATERIAL BALANCE REPORT (MBR) FORM R.03[illegible]

and will not be repeated here. Other safeguards agreements, such as INFCIRC/66, may not specifically call for material balance reports based on physical inventories, but many of the same data elements would be required for the periodic reports called for by such agreements.

2.4.3 Inventory Report

An inventory report may be a statement of a book or a physical inventory at the end of a period. A book inventory report would be based on the data from the ledger accounts. A physical inventory report would be based on the data generated during a physical inventory taking. Various levels of detail may be called for in an inventory report by safeguards agreements, State Authority, or facility management. For example, facility management may require only a summary report of physical inventory, while an NPT Safeguards Agreement calls for a physical inventory listing which lists separate batches including batch identity and batch data for each. The data elements needed to provide the required levels of detail in reports should be taken into consideration in establishing physical inventory taking procedures. For NPT Safeguards Agreements reporting, the IAEA has established the Physical Inventory Listing (PIL) format in Form R.02/c shown in Figure 2.5. Instructions for completion of this form are given in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements and will not be repeated here. Procedures for physical inventory taking which will generate the necessary data for NPT reporting are discussed in detail in Part 3. Other safeguards agreements differ in that some call for book inventory reports and others for physical inventory reports. The procedures discussed here should assist in either situation.

In addition to inventory changes involving actual changes or movements or material there are adjustments and corrections which result in changes in the recorded inventory quantities. Shipper-receiver differences are one type of adjustment. These result from measurements performed at the facility on nuclear material previously recorded at the shipper's value, although at a critical facility, receiver's measurements are usually only MDA verification measurements, if any, and not sufficiently accurate to be used as a basis for

PHYSICAL INVENTORY LISTING (PIL) FORM R.02/c

19-20

adjusting the records. If, however, a significant shipper-receiver difference is found it should, of course, be investigated. This could result in a corrected document being issued by the shipper which would serve as the supporting document for a correction of the shipper's data that had been recorded in the facility records.

Another type of adjustment is material unaccounted for (MUF), which results from a physical inventory taking, although it is unlikely that there would be a MUF at an item facility such as a critical assembly. The supporting document for such an adjustment would be the physical inventory summary and reconciliation supported by the final physical inventory listing, which would record the results of the physical inventory taking. See Part 3, Physical Inventory Taking.

Corrections of mistakes in the data should be supported by appropriate documentation. This could be done by issuing a corrected support document such as a receiving report, shipping report, or internal transfer form to record the data supporting the correction.

Regardless of how all of these various data are recorded or how they are generated, it is most important that the data are identified with the items and batches and that proper data flow causes these data to be recorded and referenced in the accounting records, i.e. general ledgers and KMP perpetual inventory records, so that accurate accounting reports can be issued and the data can be traced to their origins to substantiate such reports.

2.5 Data Handling

2.5.1 Data Flow

Regardless of where the final nuclear material accountability reports are prepared, the source of the data is at the operating level of the facility. Quantity data are based on fabricator's data which are recorded in the accounting and operating records of the facility. When material is received or shipped, i.e. an actual change in the facility inventory quantity,

inventory change records are made to support appropriate entries in the general ledgers. Figure 2.6 is an example of a form that could be used for such records. As has already been discussed, internal transfer accounting at a critical facility is an important part of the facility material accounting system. Figure 2.7 is an example Internal Transfer and Rebatching Form that could be used to record internal transfers between KMPs. Other operating records are generated containing physical inventory data as discussed in Part 3. Figure 2.8 shows the data flow and relationships among source data from the fabricator, operating records and supporting documents, accounting records and accounting reports.

When an inventory change occurs at the facility an inventory change document (ICD) such as Figure 2.6 is prepared. For receipts, these documents may be prepared by the fabricator. For shipments, they would be prepared by the facility operator. In either case the quantity data would be based on fabricator's data. Other information would be determined on the basis of each occurrence, e.g. change code, various dates, shipper and receiver identity, etc. Batch identity is included in the data on the forms to provide the link between the source data and the other records and reports.

Distribution of completed ICD's may vary but as a minimum the MBA records office would receive a copy. In case of material receipts or shipments at least one copy would be sent to the shipper or receiver. The State Authority also may require that they receive copies. The ICD data is entered in the proper general ledger accounts upon receipt by the MBA records office. Cross-references to the ICDs are included in the ledger account entries to permit tracing data back through supporting documents to the source data. As discussed in section 2.2.2, KMP perpetual inventory records are an important means of material accounting in a critical facility. Internal transfer documents such as Figure 2.7 provide the data to maintain current KMP perpetual inventory records and to provide batch follow-up data as storage batches become reactor core batches and vice versa. These, in turn, can be used in physical inventory taking and verification and in preparing Physical Inventory Listing to be reported to the IAEA.

Figure 2.6

Inventory Change Document - Material Transfer

Material Description:										Mtl Code:		Unit: kg/g			
Shipper:						MDA:		Receiver:				UHA:			
						Change Code:						Change Code:			
						EMP Code:						KMP Code:			
Line No.	Batch Identity	No. of Items	Element Code	Isotope Code	Net Weight	Percent Element	Element Weight	Percent Isotope	Isotope Weight	Net Weight	Percent Element	Element Weight	Percent Isotope	Isotope Weight	Notes
1															
2															
3															
4															
5															
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24															

Doc. No. HT...		Notes				Shipper-Receiver Difference:				Date Measured	
Shipping Date		Shipper Signature				Receiving Date		Receiver Signature:			

Internal Transfer and Rebatching Form

KMP-A Batch Data		Rebatching Data			KMP-B Batch Data		
		Quantity Rebatched (kg/g)		Fuel Piece Identity	Batch Identity:		Core Sign:
					No. of Items (Drawers):		
Batch Identity	No. of Items	Element	Isotope		Location:		Pieces per Item
					Quantity Per Item (kg/g)		
					Element	Isotope	
This form can be used to record transfers between KMP-A and KMP-B in either direction. The form is designed to show one batch of material in KMP-B which would consist of a number of drawers containing a variety of fuel pieces. The form shows the several batches in KMP-A from which material was taken to form the KMP-B batch if the transfer was from A to B or the several batches in KMP-A into which the KMP-B batch was divided if the transfer was from B to A. The data in the center of the form shows the quantity of material involved in the transfer. The KMP-A batch identity shows the container numbers involved in the transfer. The KMP-B batch data shows the location of the batch and shows the core sign which is the symbol used on the core map to show the location of the specific drawers which make up the batch. The quantities per item in the KMP-B batches are shown to facilitate physical inventory taking and verification.							
Transfer Direction: From KMP- to KMP-						Document No. IT-	
KMP-A Signature:			KMP-B Signature:			Date:	

If a material balance interval is closed with a physical inventory taking then the facility Physical Inventory Item List and the Physical Inventory Summary are prepared as described in Part 3. The MUF, if any, which results from the physical inventory reconciliation and the final ending physical inventory values are entered in the ledger accounts. The ending physical inventory value entered in the ledger becomes the beginning inventory for the next material balance interval. If the material balance interval is not closed with a physical inventory then no MUF or physical inventory values are generated to be entered in the ledger and a book inventory can be obtained from the ledger accounts.

2.5.2 Batch Considerations

The IAEA Safeguards Glossary defines a batch as "a portion of nuclear material handled as a unit for accounting purposes at a key measurement point and for which the composition and quantity are defined by a single set of specifications of measurements". Using this definition one could consider all of the fuel pieces of each material types and quantity to be a batch. This usually would result in a few batches with a very large number of items per batch. For example, for the reference facility described in Appendix A this would result in two plutonium batches of 5000 pieces each. While it would be possible to use this batch structure it may be more convenient and provide better control and accounting to use smaller batches related to some characteristic other than material type and quantity. One way is to identify batches with location. In the main storage area a batch could consist of all fuel pieces of like shape and composition stored in an individual storage container, i.e. birdcage, rack,....., etc. In the reactor core area a batch could consist of all fuel drawers having the same nuclear material composition. In intermediate storage a batch could be all fuel pieces or drawers of like kind and composition. This is the approach used in the example for the reference facility. There may be other batch structure considerations that a facility operator may choose to use or be required to use by a State Authority. Some facilities, for example, account for fuel with regard to country of origin with batches so identified, i.e. batch identity by country code and number. While there are several ways that batches may be structured, the structure will almost certainly be different in the reactor

core from that in main storage. The internal transfer document should provide the link between the batch structure and content in these two areas. The difference in batches in an intermediate storage area and the reactor core may also need to be recorded. An internal transfer document may not be called for here, however.

2.5.3 Shipper-Receiver Differences

It is unlikely that shipper-receiver differences will occur at research reactor facilities because the facility operator normally will accept the shipper's data. If any measurement is made it will be a non-destructive measurement for verification purposes and not to generate data which would be entered into the records. Because of this situation the critical facility records system need not include procedures for handling shipper-receiver differences. If an NDA verification measurement shows a large difference from shipper's values then investigation should be undertaken to determine the cause of the difference and to determine the correct quantity. In such cases a revised Inventory Change Document would be issued containing the correct quantity if it were different from the original shipper's values.

2.6 Example Records and Reports

2.6.1 Introduction

The facility used in this example is the horizontal split-table facility described in Appendix A. Figure A.2 is a material flow diagram. Table A.1 describes the fuel at the facility and Table A.2 shows the batch structure assumed. The facility is a single MBA with two flow and two inventory KMPs. The records and reports examples are given for a single eight month material balance interval. Complete sets of records for the material balance interval are not shown in the example because this would add unnecessarily to the size of the document.

2.6.2 Records and Reports System

Examples of various records are shown sufficient to describe the system and information flow without showing all data for the entire interval or for other intervals. The records and reports system for the example facility consists of the following:

1. Inventory Change Document - Material Transfer such as Figure 2.6. This document is used to record transfer into or out of the facility, i.e. the MBA. It probably would be a form specified by the State Authority for transfers between facilities.
2. Internal Material Transfer and Rebatching Form such as Figure 2.7. This is an internal document used to record transfers between KMPs within the MBA. These are pre-numbered forms with the prefix IT.
3. KMP Perpetual Inventory Record such as Figure 2.2. These are internal records that show the batches and items on inventory in each KMP, i.e. an itemized book inventory for each KMP. The records are kept current with data from internal transfer forms when material is transferred between the KMPs.
4. Facility Batch Data File, example not shown. This is a complete record of every fuel piece in the facility inventory including identity and nuclear material quantity for the pieces. The data entered is based on fabricator's data received through the inventory change documents for fuel material received. This record may be the fabricator's documents themselves.
5. Physical Inventory Summary and Reconciliation such as Figure 3.2 in Part 3. This provides the PIT summary and final data for closing the material balance and making the final ledger entries.

6. General Ledgers such as Figure 2.1. These documents are used to record and summarize all nuclear material transactions for each material type for the MBA as a whole.
7. IAEA Inventory Change Reports, Material Balance Reports and Physical Inventory Listing such as Figures 2.13, 2.14, and 2.15.

The following sections describe the use of the above documents at the reference facility. For convenience, all of the figures and tables for the examples are together at the end of the example section. The examples are shown as being prepared manually. In practice maintaining records of such a large number of items and changes would best be done by computer.

2.6.3 Inventory and Material Flow Assumptions

The example given in Part 3 describes the physical inventory taking as of 31 March 1984 at the end of an eight month material balance interval. It is assumed that a physical inventory also was taken as of 31 July 1983, eight months earlier. The 31 July 1983 physical inventory values are shown on line 1 of the General Ledgers for plutonium and enriched uranium, Figures 2.9 and 2.10 in the Inventory accounts. These are assumed values with no support being shown in the example. They are the ending inventory values for the interval ended 31 July 1983 and the beginning inventory values for the interval 1 August 1983 through 31 March 1984. Similar records would be kept for Natural and Depleted uranium but are not shown in the example.

In relation to the example eight months material balance interval the following sequence of events is assumed.

28 July 1983 - In preparation for a major core configuration change and physical inventory taking all material is removed from KMP-B, except core matrix blanket material, and returned to KMP-A main storage. Figure 2.11 shows the core map, and Figure 2.12 shows the perpetual inventory card for KMP-B. Figure 2.13 shows the first part of a perpetual inventory card for KMP-A.

- 31 July 1983 - Physical inventory is taken. Results are entered in the general ledgers for plutonium (line 1, Figure 2.9) and enriched uranium (line 1, Figure 2.10).
- 6 August 1983 - Drawers for the new core configuration are set up in KMP-A and transferred to KMP-B. Table 2.1 shows the drawer compositions to be used. Figure 2.14 shows the core map for the new configuration. Figures 2.15, 2.16, 2.17, and 2.18 show the Internal Transfer and Rebatching Forms for the transfer of the new fuel drawers from KMP-A to KMP-B. Figure 2.19 is an example of one of the perpetual inventory cards showing the KMP-B inventory. Figure 2.20 is an example of one of the perpetual inventory cards showing the inventory remaining in KMP-A.
- 8 August 1983 - Broken fuel pieces are shipped to fabricator for recovery. Figure 2.21 shows the material transfer form for the shipment. Entry is made in general ledger, Figure 2.10, line 2.
- 5 Dec. 1983 - Changes are made to the core configuration by replacing 56 drawers from intermediate storage. Figure 2.22 shows the core configuration after this change. The footnote on Figure 2.19 dated 5-12-83 shows the change in the inventory status on the cards.
- 8 March 1984 - New fuel plates are received from the fabricator. Entries are made in the general ledgers, Figures 2.9 and 2.10 lines 2 and 3 respectively.
- 28 March 1984 - In preparation for the major core change planned for April and the physical inventory scheduled for 31 March 1984 all material in the Driver and Test zones and in intermediate storage is returned to KMP-A main storage. A core map such as Figure 2.11 and perpetual inventory cards such as Figures 2.12 and 2.13 would show this situation.

31 March 1984 -- Physical inventory is taken. Results are entered in the general ledgers, line 3, Figure 2.9 and line 4, Figure 2.10. The ending inventory is shown in Table 3.1 with the example Physical Inventory Summary shown in Table 3.2

2.6.4 Example Transaction Recording

Using the sequence of events assumed in section 2.6.3, example transaction documents are described in the following paragraphs to show how each transaction would be entered in the facility records. The quantity and identity data needed to follow the discussions in the following paragraphs can be found in Table 3.1, page 58, which provides inventory data for the fuel, including average nuclear material content per piece and total quantities and number of items for each type. Table A.1 in Appendix A provides descriptions of the fuel including identity and nominal values. Table A.3 provides main storage batch identity information.

2.6.4.1 Internal Transfers

At the beginning of the material balance period, immediately after completion of the physical inventory taking as of 31 July 1983, all plutonium, enriched and natural uranium fuel pieces are in main storage in the birdcages and bins numbered as indicated in Table A.3 in Appendix A. (Note: Not all containers or bins would contain material.) A part of the depleted uranium is in the main storage and a part is in the reactor matrix. Perpetual inventory cards were prepared when the material was transferred back to storage in preparation for the physical inventory taking at the end of the prior material balance period, i.e. before the major core change of 6 August 1983. Figure 2.13 is an example of a main storage perpetual inventory card for the large size plutonium plates. Note that space is provided for inventory tag numbers so that these cards can be used for physical inventory taking. Similar cards would have been prepared for the other materials, i.e. small plutonium plates, large and small 93% enriched uranium plates, 20% enriched uranium plates, natural uranium blocks, and those depleted uranium blocks still in the bins. There would be eight different sets of cards for the eight different kinds of fuel pieces. More than one card for each kind might be needed if there is not

enough space to list all the batches as in the case in Figure 2.13. Only one perpetual inventory card would be needed for KMP-B because the only material remaining there was the blanket drawers. Figure 2.12 shows this card. The core map for the situation at this time, shown as Figure 2.11 would show only the blanket with the Test and Driver zones empty.

The drawers are prepared in KMP-A for the new core configuration shown in Table 2.1. An Internal Transfer and Rebatching Form is prepared for each KMP-B batch, i.e. one for the Test zone drawer, two for the two different drawer compositions in the Driver zone and one for the drawers in intermediate storage. The blanket zone is already filled. Figures 2.15, 2.16, 2.17 and 2.18 show these four transfer forms. The drawers are transferred from KMP-A to KMP-B along with the appropriate transfer forms and are placed in their designated places in the reactor matrix or in intermediate storage. A core map is prepared to reflect the addition of these drawers to the reactor matrix. It is shown in Figure 2.14.

Perpetual inventory cards are prepared for each batch similar to that for the blanket material. Figure 2.19 shows such a card for the Test zone batch of drawers. Two such cards would be prepared for the driver zone drawers and one for the drawers in intermediate storage. These are not shown in the example but would be similar to Figure 2.19.

Revised perpetual inventory cards for the material remaining in KMP-A would be prepared similar to Figure 2.13 but listing only those batches, i.e. containers, in which there is material. Figure 2.20 shows the KMP-A inventory card for the large plutonium pieces after the transfer for the new core configuration. The perpetual inventory cards for KMP-A for the inventory before the change such as Figure 2.13, would be marked as being superseded on 6 August 1983, the date of the transfer of the material to KMP-B.

During the course of the eight month period no major core configuration change is made but the 56 drawers in intermediate storage are put in the Test zone to replace 56 drawers removed and placed in intermediate storage. No internal transfer between KMPs is involved so no internal transfer form is prepared. The perpetual inventory card for the Test zone batch, Figure 2.19,

is annotated to show that 56 of 176 drawers are now in intermediate storage and the perpetual inventory card for the 56 drawers from intermediate storage is changed to show that the drawers are in the Test zone of the reactor. A new reactor core map is prepared to show the location of these 56 drawers. The new core map is shown in Figure 2.22. The annotation is shown on Figure 2.19 but the card for the intermediate storage drawers is not shown.

It is assumed that a major core configuration change is to be made in April and a physical inventory taking is scheduled as of 31 March 1984. The plan calls for all of the Test and Driver drawers to be changed so all the drawers are transferred from KMP-B to KMP-A and unloaded into the storage containers. Internal Transfer and Rebatching Forms similar to Figures 2.15, 2.16, 2.17 and 2.18 are used to transfer the material back to KMP-A. If there have been no changes made to the drawers in KMP-B, as is the case in the example, the same transfer documents could be used to return the drawers to KMP-A. A note on the forms showing when the drawers were returned should be sufficient and preclude the need for making out new transfer forms. In KMP-A perpetual inventory cards are prepared to show that all of the fuel pieces have been returned to the storage containers. It may be that the original cards can be used by noting the data that all of the pieces were returned. Figure 2.13 was an example of the original KMP-A inventory and also represents the card that would be prepared when the material has been returned to storage. If not all material is returned then appropriate transfer forms and inventory cards would need to be prepared to reflect the inventory status. The cards for the KMP-B inventory, such as Figure 2.19 would be marked superseded on 28 March 1984 as would the KMP-A cards, such as Figure 2.20.

2.6.4.2 Shipments Off-Site

During the preparation of the new drawers twelve of the large 93% enriched uranium fuel pieces and eleven of the 20% enriched uranium fuel pieces were broken and are to be shipped back to the fabricator for recovery. Figure 2.21 is an example of the material transfer inventory change document recording this shipment. The shipment is assumed to have been made to the fabricator in a different country, thus the shipping change code SF (shipment

foreign). The various codes on the form are those as given in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements. The quantity data used is that provided originally by the fabricator. This transfer represents a change in the facility inventory so the data from the material transfer is posted to the enriched uranium general ledger shipment account. These data are shown on line 2 of Figure 2.10. In addition to posting to the ledger the fuel pieces removal would need to be reflected on the perpetual inventory cards in KMP-A.

2.6.4.3 Off-Site Receipts

It is assumed that the current inventory was received at some earlier time and was inventoried as of 31 July 1983 as shown on line 1 of Figures 2.9 and 2.10. It is assumed that more fuel plates are needed for the core change planned for April 1984 therefore 1000 additional plutonium and 1000 additional 93% enriched uranium fuel plates are received a short time before the end of the example material balance interval, i.e. 8 March 1984. Figure 2.23 shows the document which records this material transfer. The shipment is assumed to have been made from a fabrication plant in another country, thus the shipping change code SF (shipment foreign) and receiving code RF (receipt foreign). The various codes on the form are those as given in Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements. The shipper's data has been entered on the form but no receiver's data would be entered because the shipper's (fabricator) data are accepted for the fuel pieces. The shipper's data are posted to the plutonium and enriched uranium receipts accounts. These data are shown on line 2 of Figure 2.9 for the plutonium and line 3 of Figure 2.10 for the uranium. In addition to posting to the ledger accounts perpetual inventory cards for the new material would need to be prepared. These are not shown but would be similar to Figure 2.13.

2.6.5 Example Accounting Reports

Facility attachments for critical facilities usually call for Inventory Change Reports (ICR) to be submitted within 30 days of an inventory change or a change in batches if batch follow-up is called for. Some facility attachments do call for batch follow-up so the example shows the rebatching

reporting on the ICR. In the example facility, inventory changes occurred when the broken fuel pieces were shipped off-site and when the additional fuel pieces were received. Rebatching occurred when the new core configuration was set up early in the material balance period and when the fuel pieces were returned to storage late in the period. There would therefore be two ICRs sent to the IAEA during the example period. One would be required for the broken fuel shipment and the early rebatching and another for the additional fuel receipt and the later rebatching. Figure 2.24 is an example of the ICR reporting the earlier transactions.

Lines 1 and 2 of Figure 2.24 show the shipment of the broken fuel pieces. Lines 3 through 8 show the decrease in the batches identified. Lines 9 through 19 show the increases in the batches identified. The inventory change code RM indicates the decrease and RP the increase in accordance with Code 10 of the General Part of Subsidiary Arrangements for NPT Safeguards Agreements. The data for lines 3 through 19 are taken from the rebatching records, Figures 2.15, 2.16, 2.17 and 2.18. For example, the quantity of 295700 grams for decrease in PUBLK on line 3 is the sum of the PUBLK quantities given in Figures 2.15 and 2.16.

The material Balance Report for the example material balance period is shown in Part 3, Figure 3.2. A comparison of the data in Figure 3.2 shows that the general ledger Figures 2.9 and 2.10 provide the data needed for the report.

Part 3 describes the physical inventory taking using the KMP perpetual inventory cards as the facility inventory item lists. The verified facility perpetual inventory cards and the facility physical inventory summary provide the data needed to prepare the Physical Inventory Listing, Figure 3.3, sent to the IAEA and to make the final inventory entries in the ledgers, line 3 and 4 of Figures 2.9 and 2.10 respectively.

References

IAEA/SG/INF/1: IAEA Safeguards Glossary, IAEA/SG/INF/1, International Atomic Energy Agency, Vienna, 1980.

Figure 2.9

General Ledger

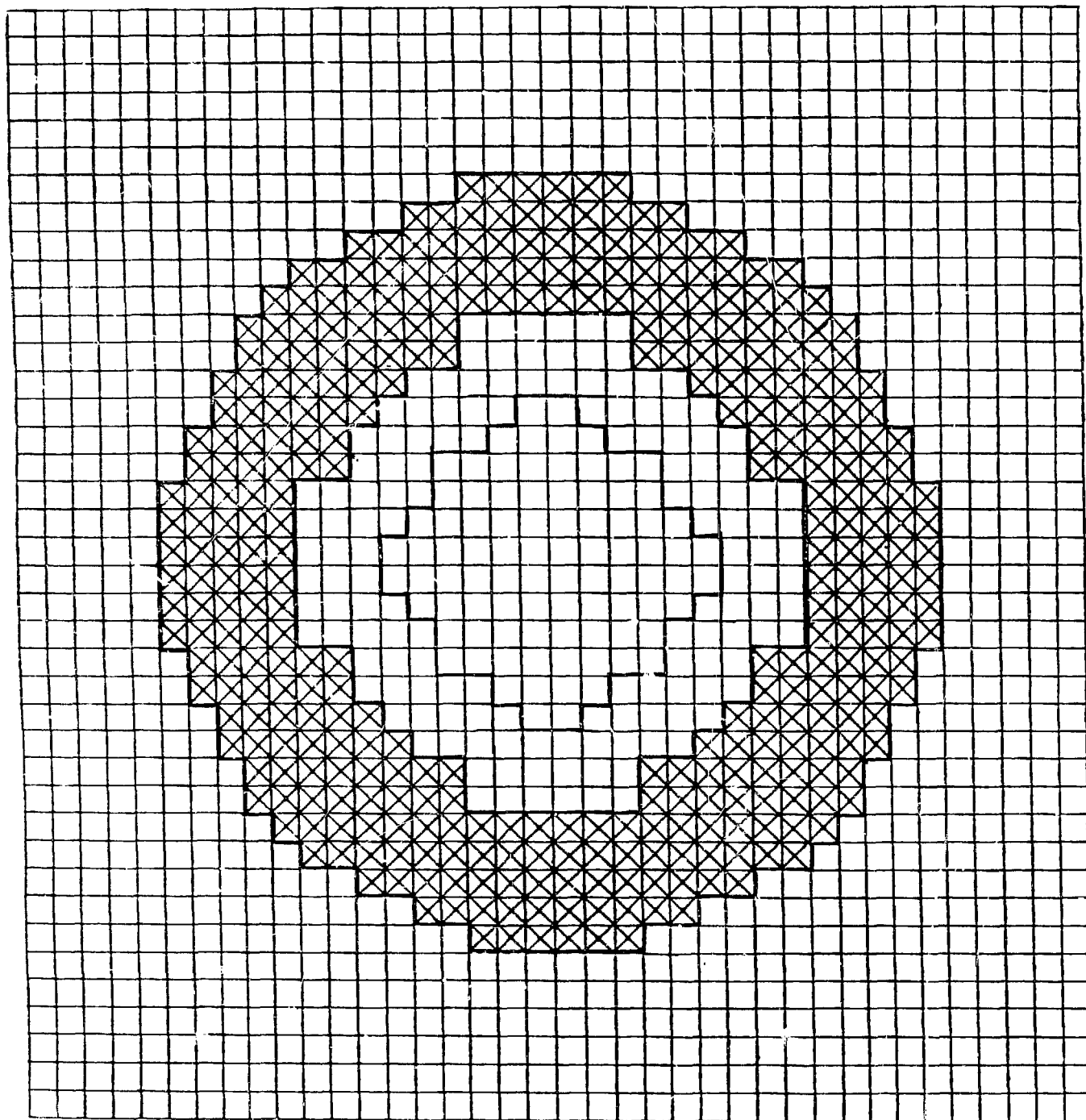
Facility: CAF			MBA: CAF				Element Code: P					
Material Description: Plutonium in critical assembly fuel plates				Isotope Code: -				Unit: grams				
Line No.	Date	ICD No.	Receipts		Other Additions		Shipments		Other Removals		Inventory ¹	
			Element	Isotope	Element	Isotope	Element	Isotope	Element	Isotope	Element	Isotope
1	310783	PIL									488.920	-
2	080384	MT-0131	75330 ²	-							564,250	-
3	310384	PIL									564,250	-
4												
5												
6	Explanatory Notes:											
7												
8	1. Running book inventory is calculated for each entry. Line 1 is the ending physical inventory											
9	for the prior material balance interval. Line 3 is the ending physical inventory for the current											
10	material balance interval.											
11												
12	2. Receipt entered from MT-0131, Figure 2.23, line 2.											
13												
14												
15												
16												
17												
18												
19												
20												
21												
22												
23												
24												
25												
26												
27												
28												
29												
30												

Figure 2.10

General Ledger

Facility: CAF				MBA: CAF				Element Code: E				
Material Description: 93% and 20% enriched uranium in critical assembly fuel plates				Isotope Code: G				Unit: grams				
Line No.	Date	ICD No.	Receipts		Other Additions		Shipments		Other Removals		Inventory ¹	
			Element	Isotope	Element	Isotope	Element	Isotope	Element	Isotope	Element	Isotope
1	310783	PIL									760,610	415,031
2	060883	MT-0029					1840 ³	1068 ³			718,770	413,963
3	080384	MT-0131	80160 ²	74533 ²							838,930	488,496
4	310384	PIL									838,930	488,496
5												
6												
7	Explanatory Notes:											
8												
9	1. Running book inventory is calculated for each entry. Line 1 is the ending physical inventory											
10	for the prior material balance interval. Line 4 is the ending physical inventory for the current											
11	material balance interval.											
12												
13	2. Receipts entered from MT-0131, Figure 2.23, line 1.											
14												
15	3. Total shipments entered from MT-0029, Figure 2.21, line 3.											
16												
17												
18												
19												
20												
21												
22												
23												
24												
25												
26												
27												
28												
29												
30												

Figure 2.11
Core Map With Only Blanket Zone Filled



DB -  - Blanket Zone

Effective Date: 280783



Table 2.1

New Core Configuration Fuel Drawer Compositions
(set up as of 6 August 1983)

Zone	Composition of Each Drawer	Numbers of Drawers [*]	Batch Identity	Core Sign
Blanket:	2 blocks DU 50x50x300 1 block DU 50x50x100	720	DB	☒
Test:	18 plates Pu 50x50x2 9 plates Pu 50x25x2 9 plates 20% 50x50x2 5 blocks NU 50x50x50	176	P2NT	☒
Driver:	5 plates Pu 50x50x2 8 plates Pu 50x25x2 9 plates 93% 50x50x2 5 blocks NU 50x50x50	152	P9ND	☐
	18 plates 20% 50x50x2 18 plates 93% 50x25x2 5 blocks NU 50x50x50	136	92ND	☒
Intermediate Storage:	36 plates Pu 50x25x2 18 plates 93% 50x50x2 5 blocks NU 50x50x50	56	P9NI	☐

* Total for both halves

Figure 2.12

KMP-B Perpetual Inventory Card

Batch Identity: DB Batch Location: Blanket
 Number of Items: 720 Core Sign: ☒
 Item Description: Depleted Uranium Blanket Drawers
 Date: 16-11-82 Date Superseded:
 Inventory Tag Number: Card No.: PIC-5011

Item Composition

Fuel Piece Description	Number of Pieces per Item	Total Quantity (kg/M)			
		Per Item		Per Batch	
		Element	Isotope	Element	Isotope
DUL	2	28.02	-	20,176	-
DUS	1	4.66	-	3,354	-
Total		32.68	-	23,530	-

Continued to: PIC-None

Figure 2.13

KMP-A Perpetual Inventory Card

Material Description: Plutonium 50 x 50 x 2 mm

Average Quantity per Item:

Element: 75.28 gms

Isotope: --

Item Description: Fuel Pieces - PUBLK

Date: 280793

Card No. PIC-0046

Date Superseded: 060883

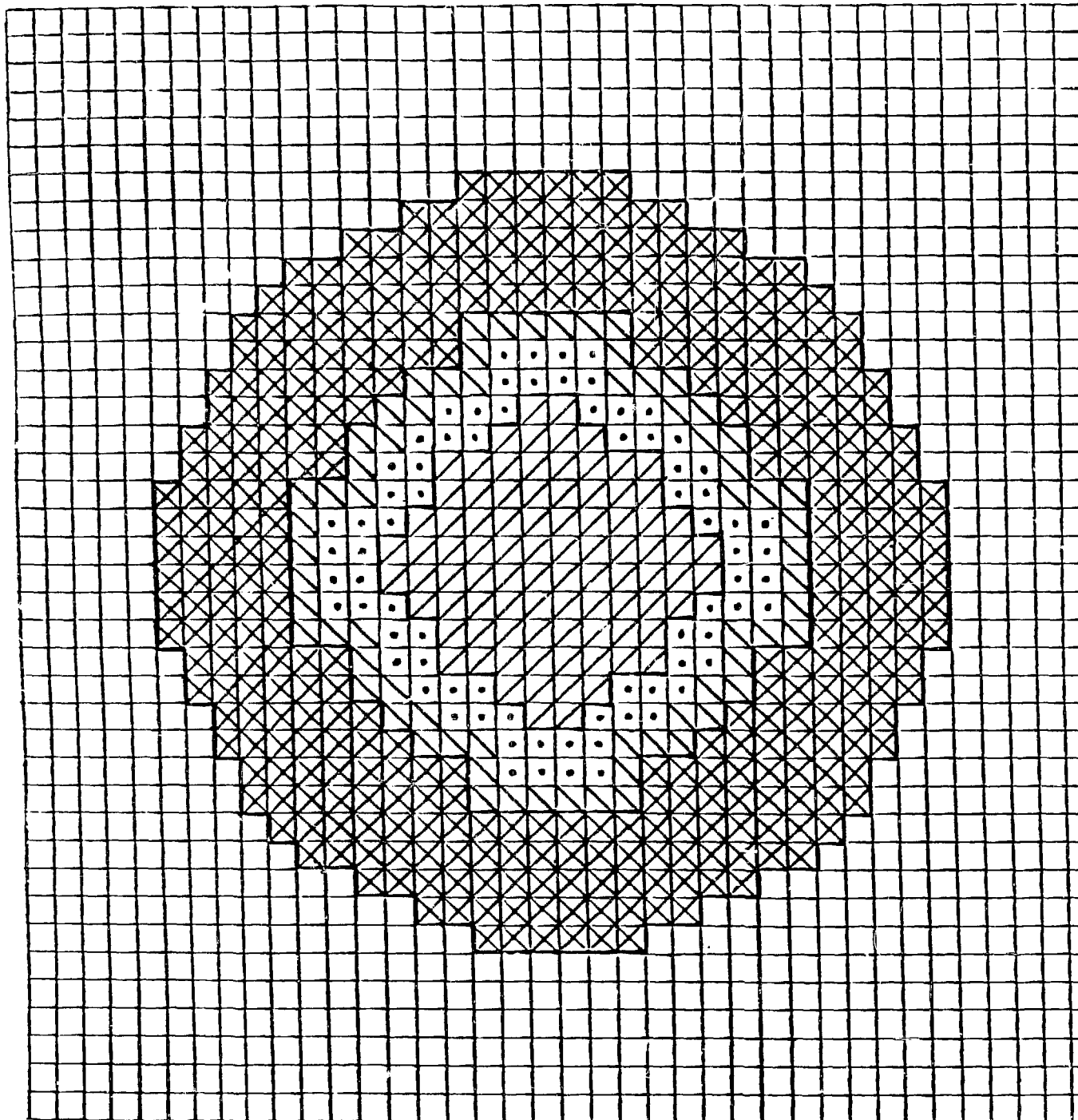
<u>Batch Identity</u>	<u>Number of Pieces</u>	<u>Inventory Tag No.</u>	<u>Quantity per Batch (~~kg~~/g)</u>	
			<u>Element</u>	<u>Isotope</u>
P-100	40		3011	-
P-101	40		3011	-
P-102	40		3011	-
P-103	40		3011	-
P-104	40		3011	-
P-105	40		3011	-
P-106	40		3011	-
P-107	40		3011	-
P-108	40		3011	-
P-109	40		3011	-
P-110	40		3011	-
P-111	40		3011	-
P-112	40		3011	-
P-113	40		3011	-
P-114	40		3011	-
P-115	40		3011	-
P-116	40		3011	-
P-117	40		3011	-
P-118	40		3011	-
P-119	40		3011	-
P-120	40		3011	-
P-121	40		3011	-
P-122	40		3011	-
P-123	40		3011	-
P-124	40		3011	-
P-125	40		3011	-

See PIC-0051 for Total
(not shown in example)

Continued to: PIC-0049
(not shown in example)

Figure 2.14

Example Core Configuration



DB -  - Blanket Zone

Effective Date: 060883

P2NT  - Test Zone

P9ND -  - Driver Zone

92ND - 

Figure 2.15

Internal Transfer and Rebatching Form

KMP-A Batch Data		Rebatching Data			KMP-B Batch Data		
Batch Identity	No. of Pieces	Quantity Rebatched (kg/g)		Fuel Piece Description	Quantity Per Item (kg/g)		Pieces per Item ²
		Element ¹	Isotope ¹		Element ¹	Isotope ¹	
P-100 + P-178 P-179	3160 8	238,487	-	PUBLK	1,355	-	18
P-500 + P-538 P-539	1560 24	59,495	-	PULMT	338	-	9
2-100 + 2-130 2-131	1550 34	126,847	25,230	20YEL	721	143	9
BIN 100 + Bin 107 Bin 108	800 80	2,115,520	-	MUGRN	12,020	-	5

Explanatory Notes:
¹ See Figure 2.19 for corresponding data entered on the perpetual inventory card.
² See Table 2.1

Transfer Direction: From KMP-A to KMP-B

Document No. IT0024

KMP-A Signature:

KMP-B Signature:

Date: 06-08-83

Figure 2.16

Internal Transfer and Rebatching Form

KMP-A Batch Data		Rebatching Data			KMP-B Batch Data		
Batch Identity	No. of Pieces	Quantity Rebatched (kg/g)		Fuel Piece Description	Quantity Per Item (kg/g)		Pieces per Item ¹
		Element	Isotope		Element	Isotope	
P-180 → P-198	760	57,213	-	PUBLK	376	-	5
P-541 → P-570 P-540	1200 16	45,673	-	PUWHT	300	-	8
9-100 → 9-133 9-134	1360 8	109,344	101,810	93RED	719	670	9
BIN 109 → Bin 115 Bin 108 Bin 116	700 20 40	1,827,040	-	NUGRN	12,020	-	5
Explanatory Note: ¹ See Table 2.1							

Transfer Direction: From KMP-A to KMP-B

Document No. IT0025

KMP-A Signature:

KMP-B Signature:

Date: 06-08-83

Figure 2.17

Internal Transfer and Rebatching Form

KMP-A Batch Data		Rebatching Data			KMP-B Batch Data		
Batch Identity	No. of Pieces	Quantity Rebatched (kg/g)		Fuel Piece Description	Quantity Per Item (kg/g)		Pieces per Item ¹
		Element	Isotope		Element	Isotope	
2-132 → 2-179	2400	196,036	38,992	20YEL	1,441	287	18
2-131	16						
2-180	32						
9-500 → 9-560	2440	98,263	91,492	93BLU	723	673	18
9-561	8						
BIN 117 → Bin 122	600	1,634,720	-	NUGRN	12,020	-	5
Bin 116	60						
Bin 123	20						
Explanatory Note: 1 See Table 2.1							

Transfer Direction: From KMP-A to KMP-B
 Document No. ITCJ26

KMP-A Signature:
 KMP-B Signature:
Date: 06-08-83

Figure 2.18

Internal Transfer and Rebatching Form

KMP-A Batch Data		Rebatching Data			KMP-B Batch Data		
Batch Identity	No. of Pieces	Quantity Rebatched (kg/g)		Fuel Piece Description	Quantity Per Item (kg/g)		Pieces per Item ¹
		Element	Isotope		Element	Isotope	
P-571 → P-620 P-621	2000 16	75,721	-	PUWHT	1,352	-	36
9-135 → 9-159 9-134	1000 8	80,569	75,018	93RED	1,439	1,340	18
BIN 124 → Bin 125 Bin 123	200 80	673,120	-	NUGRN	12,020	-	5
Explanatory Note: ¹ See Table 2.1							

Transfer Direction: From KMP-A to KMP-B

Document No. IT0027

KMP-A Signature:

KMP-B Signature:

Date: 06-08-83

Figure 2.19

KMP-B Perpetual Inventory Card

Batch Identity: P2NT	Batch Location: Test
Number of Items: 176	Core Sign: ☒
Item Description: Fuel Drawer with Pu and 20% Enr U	
Date: 06-08-83	Date Superseded:
Inventory Tag Number:	Card No.: PIC-5018

Item Composition

<u>Fuel Piece Description</u>	<u>Number of Pieces</u>	<u>Total Quantity (kg/g)</u>			
		<u>Per Item</u>		<u>Per Batch</u>	
		<u>Element¹</u>	<u>Isotope¹</u>	<u>Element¹</u>	<u>Isotope¹</u>
PUBLK	18	1,355	-	238,487	-
PUWHT	9	338	-	59,495	-
Total Pu		1,693	-	297,982	-
20 YEL	9	721	143	126,847	25,230
Total Enr U		721	143	126,847	25,230
NUGRN	5	12,020	-	2,115,520	-
Total Natural U		12,020	-	2,115,520	-

Explanatory Note:

1. See Figure 2.15 for source of data entered on card.

05-12-83: 56 drawers removed to intermediate storage in exchange for the
56 drawers identified as P9NI on PIC-5021. (Not shown in example).

Continued to: PIC-none

Figure 2.20

KMP-A Perpetual Inventory Card

Material Description: Plutonium 50 x 50 x 2 mm

Average Quantity per Piece:

Element: 75.28 grams

Isotope: --

Item Description: Fuel Pieces - PUBLK

Date: 06-08-83

Card No. PIC-0070

Date Superseded:

<u>Batch</u> <u>Identity</u>	<u>Number</u> <u>of Pieces</u>	<u>Inventory</u> <u>Tag No.</u>	<u>Quantity per Batch (g/g)</u>	
			<u>Element</u> ¹	<u>Isotope</u>
P-179	32		2,409	-
P-199	<u>40</u>		<u>3,011</u>	-
Total	72		5,420	-

Explanatory Note:

1. Source of data would be the difference between the totals on the perpetual inventory card Figure 2.13 and the totals transferred to KMP-B on internal transfer documents, Figures 2.15, 2.16, 2.17 and 2.18.

Continued to: PICnone

Figure 2.21

Inventory Change Document - Material Transfer

Material Description: Enriched Uranium 93% and 20% Broken Fuel Pieces										Mtl Code: DD8F		Unit: grams	
Shipper: CAF Country ABC					MBA: CAF Change Code: SF KMP Code: 2			Receiver: FAB Country DEF (REP)			MBA: Change Code: KMP Code:		

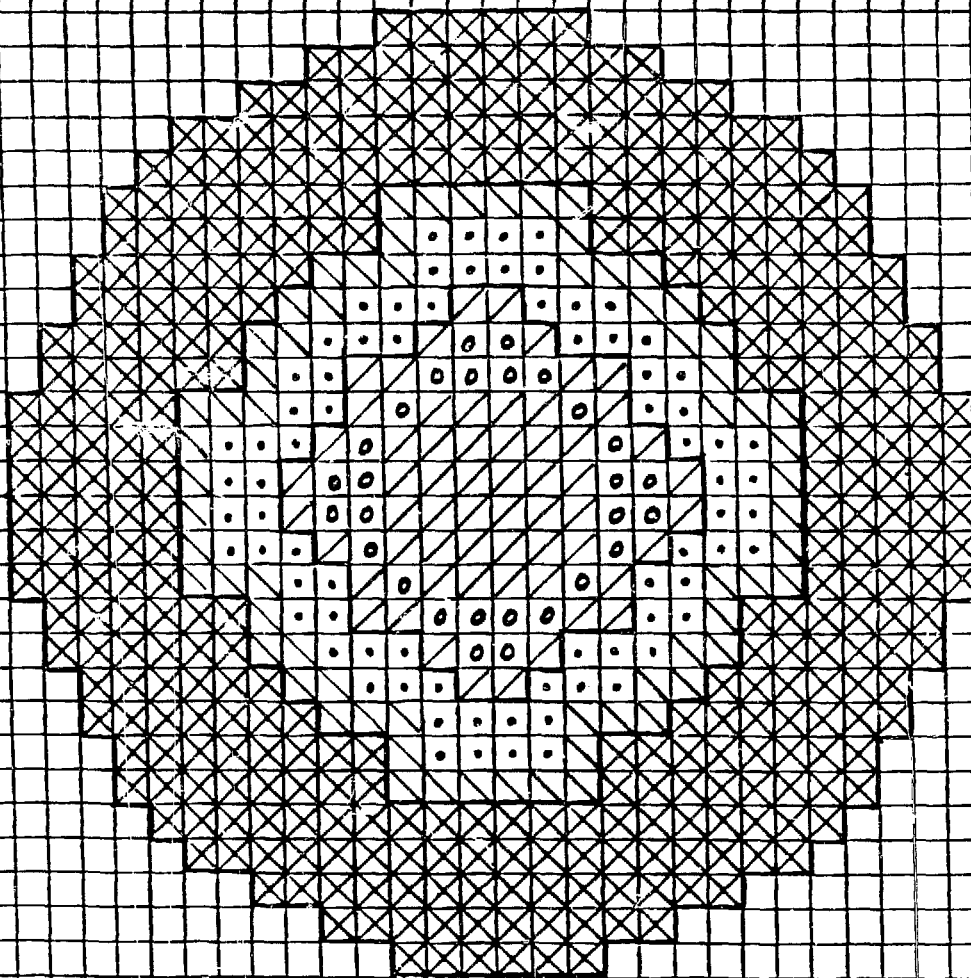
Batch Identity	No. of Items	Ele. Code	Iso. Code	Net Weight	% Ele.	Element Weight	% Iso.	Iso. Weight	Net Weight	% Ele.	Elem. Weight	% Iso.	Iso. Weight	Notes
1 9-0015 ²	12	E	G	NA	NA	959	93.1	893						
2 2-0015 ²	11	E	G	NA	NA	861	19.89	175						
3 Total	23			-	-	1840 ¹	-	1068 ¹						
4														
5														
6	Explanatory Note:													
7														
8	1. Entered to uranium ledger, Figure 2.10, line 2.													
9														
10	2. Batch identity in this case is the shipping container number where 9-0015 contains													
11	12 fuel pieces 93RED and 2-0015 contains 11 fuel pieces 20YEL - see note below.													
12														
13														
14														
15														
16														
17														
18														
19														
20														
21														
22														
23														
24														

Doc. No.	Notes	9-0015 contains 12 - 93RED fuel pieces	Shipper-Receiver Difference:	Date Measured:
MT - 0029		2-0015 contains 11 - 20YEL fuel pieces		

Shipping Date	Shipper Signature	Receiving Date	Receivers Signature:
06-08-83			

Figure 2.22

Core Map Showing 56 New Test Zone Drawers



DB -  - Blanket Zone

Effective Date: 051283

P2NT -  - Test Zone

P9NT - 

P9ND -  - Driver Zone

92ND - 

Figure 2.23

Inventory Change Document - Material Transfer

Material Description: Enriched Uranium and Plutonium Fuel Plates				Mtl Code: DDBF				Unit: grams			
Shipper: FAB		MBA: 3		Receiver: CAP		MBA: CAP					
Country DEF		Change Code: SF		Country ABC		Change Code: RF					
		KMP Code: 9				KMP Code: 1					

Batch Identity	No. of Items	Ele. Code	Iso. Code	Net Weight	% Ele.	Element Weight	% Iso.	Iso. Weight	Net Weight	% Ele.	Elem. Weight	% Iso.	Iso. Weight	Notes
1 93-020 ⁹	1000	E	G	-	-	80160 ¹	92.98	74533 ¹						
2 PU-040 ⁹	1000	P	-	-	-	75330 ²	-	-						
3														
4														
5														
6														
7														
8														
9														
10														
11														
12														
13														
14														
15														
16														
17														
18														
19														
20														
21														
22														
23														
24														

Explanatory Notes:

1. Entered to uranium ledger, Figure 2.10, line 3.
2. Entered to plutonium ledger, Figure 2.9, line 2.
3. Fabrication batch identity where each batch contains 1000 pieces, one of 93% U-235 and the other of plutonium pieces. Piece descriptions 930RG and PUBRN would be assigned by the critical facility and the pieces placed in bird cages to become batches of 40 pieces each.

Doc. No.	Notes	Shipper-Receiver Difference:	Date Measured:
MT - 0131		Shippers data accepted	
Shipping Date	Shipper Signature	Receiving Date	Receivers Signature:
01-03-84		08-03-84	

Figure 2.24.

INVENTORY CHANGE REPORT (ICR) FORM R.01.1/c

COUNTRY FACILITY MATERIAL BALANCE AREA										PERIOD COVERED BY REPORT: FROM 31 07 83 TO 06 08 83																																							
ABC CAF CAF										REPORT No. 28																																							
PAGE No. 1 OF 1 PAGES										SIGNATURE																																							
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	42	43	44	45	46	47	48	49	50
CONTINUATION		DATE OF INVENTORY CHANGE		MBA/COUNTRY		TYPE OF INVENTORY CHANGE		KMP CODE		NAME OR NUMBER OF BATCH		NUMBER OF ITEMS IN BATCH		MATERIAL DESCRIPTION		ACCOUNTANCY DATA		CORRECTION TO																															
				FROM	TO											WEIGHT OF ELEMENT	UNIT NO.	WEIGHT OF FISSILE ISOTOPES (URANIUM ONLY) (g)	CONCISE NOTE	REPORT No.	ENTRY No.																												
1		060883	CAF	FAB	SF	2	9-0015	12	DD8F	F	959	g	893	G	N																																		
2		060883	CAF	FAB	SF	2	2-0015	11	DD8F	F	881	g	175	G	N																																		
3		060883	CAF		RM	A	P100-198	3960	DD8F	P	295700	g		N																																			
4		060883	CAF		RM	A	P300-621	4880	DD8F	P	180889	g		N																																			
5		060883	CAF		RM	A	2100-180	4050	DD8F	E	322883	g	64222	G	N																																		
6		060883	CAF		RM	A	9100-159	2400	DD8F	F	189913	g	176828	G	N																																		
7		060883	CAF		RM	A	9500-561	2480	DD8F	E	98263	g	91492	G	N																																		
8		060883	CAF		RM	A	B100-125	2600	DD8F	N	6250400	g		N																																			
9		060883	CAF		RP	A	P2NT	176	DD4F	P	297982	g		N																																			
10		060883	CAF		RP	A	P2NT	176	DD4F	E	126847	g	25230	G	N																																		
11		060883	CAF		RP	A	P2NT	176	DD4F	N	2115520	g		N																																			
12		060883	CAF		RP	A	P9ND	152	DD4F	P	102886	g		N																																			
13		060883	CAF		RP	A	P9ND	152	DD4F	F	109344	g	101810	G	N																																		
14		060883	CAF		RP	A	P9ND	152	DD4F	N	1827040	g		N																																			
15		060883	CAF		RP	A	92ND	136	DD4F	F	294299	g	130484	G	N																																		
16		060883	CAF		RP	A	92ND	136	DD4F	N	1634720	g		N																																			
17		060883	CAF		RP	A	P9NI	56	DD2F	P	75721	g		N																																			
18		060883	CAF		RP	A	P9NI	56	DD2F	E	80569	g	75018	G	N																																		
19		060883	CAF		RP	A	P9NI	56	DD2F	N	673120	g		N																																			

Part 3. PHYSICAL INVENTORY TAKING AND MATERIAL BALANCE CLOSING

3.1 Introduction

The physical inventory is the bench mark of nuclear material accounting and control. Various means of control, accountancy and containment and surveillance can be employed to account for and control nuclear materials. However, a material balance based on a physical inventory that provides conclusive evidence of the physical presence of the material is the only means for the facility operator to assure that no significant losses or diversions have gone undetected. The frequency of physical inventory taking will depend on both national and international requirements and agreements. Of course, the other components of the material balance, receipts, shipments, discards, etc., also must be accounted for for the balance to be meaningful.

The following description provides information regarding inventory procedures, including pre-inventory procedures, to assure a complete and accurate physical inventory including assurance of data handling accuracy and integrity. This description also addresses the relationship of this element of an SSAC to other SSAC elements, descriptions of which are provided in other parts.

3.2 Inventory Organization and Planning

Physical inventories planned, organized and carried out according to written inventory procedures and instructions can be conducted with better assurance of being complete and accurate as well as being accomplished with a minimum of interference with facility operations. Although physical inventory taking at an item facility, such as a critical assembly, does not usually involve material measurements or the complications of material in process as is the case for bulk handling facilities, some organization and planning is needed to assure an accurate and complete physical inventory taking.

3.2.1 Assignment of Responsibilities

Clearly defined responsibilities, duties and authorizations together with proper orientation of personnel and assurance that each person understands his place in the inventory programme will materially assist in more rapid, trouble free conduct of physical inventories. Responsibilities and authorizations at the different organizational levels involved in the inventory should be clearly set forth in writing.

3.2.1.1 Facility Inventory Supervisor

One person who is familiar with the areas to be inventoried and with the principles and procedures of conducting physical inventories of nuclear materials should be assigned primary responsibility for planning, organizing and conducting the physical inventory. Specific statements should be included for the inventory supervisor's authority with regard to reactor shutdown or start-up, during and following the inventory.

3.2.1.2 Inventory Teams

Teams for conducting the inventory, i.e., the actual listing, tagging, measuring, etc., should consist of at least two persons. At a critical assembly there probably will need to be only one inventory team although there could be two, one for each inventory KMP. An inventory team should consist of one person who is familiar with the areas assigned for inventory and one person from another area or organizational unit to serve as a controller to assure accuracy and compliance with instructions. An inventory team should have no authority to deviate from specific instructions without approval of the Facility Inventory Supervisor.

3.2.2 Cutoff Procedures

One consideration in planning a physical inventory is establishing the timing of and controlling the various cutoffs necessary for accurate inventories. Cutoffs or cutoff procedures are so called because, at the

specified time, activities such as movement of material or posting to the records are stopped or cut off. The time of the inventory should be specified so that a well defined material balance interval can be established. Cutoff procedures should be established to assure that the quantity of material which results from the physical inventory taking accurately represents the material physically present at the specified inventory time and that this quantity is accurately reflected in the records at the same specified time. Material cutoff procedures should be coordinated with records cutoff procedures to assure that the same material balance interval is used for both.

3.2.2.1 Receipts and Shipments

Inventory instructions should specify a time at which material receipts and shipments are cut off. No more material should be received into or shipped from the facility after that time. This should present no problem at a critical facility because such movements of material are infrequent.

3.2.2.2 Internal Transfers

Material movements between the storage areas and the reactor should be stopped. This should present no difficulty at a critical facility because such movements are infrequent. If the physical inventory is to coincide with a fuel configuration change the inventory instructions should clearly specify whether transfers are to be made before or after the inventory. Control should be exercised to assure that the transfers have not been made or have been completed, whichever is specified.

3.2.2.3 Critical Assembly Operations

Taking a physical inventory does not necessarily require disassembly of the core material. If a physical inventory is taken without core disassembly, then statistical sampling based on the core map could be used to verify the presence of the core material. Better assurance of the presence of all of the material can be obtained, however, if a physical inventory is taken when the fuel in the assembly can be inventoried, e.g. during a fuel configuration change. Inventory procedures should include instructions for inventory of fuel materials in both storage and in the assembly matrix.

3.2.2.4 Records

The records for receipts and shipments should be cut off after the last receipt or shipment prior to the specified cutoff time and the books closed for the material balance interval. Records for internal transfers should be cut off after the last transfers prior to the specified cutoff time. The material cutoffs should be controlling, with the records brought up-to-date to the final material transactions. Thus, if the physical inventory is to be taken immediately after the fuel change has taken place, the records for the two KMPs should show the transfers made during the fuel change.

3.2.3 Facility Inventory Records and Stratification

Maintenance of perpetual inventory records for each KMP as discussed in Part 2 section 2.2.2 can assist in inventory taking, inventory verification and subsequent data processing by both the facility operator and the IAEA.

Some facility attachments call for the facility operator to provide the IAEA inspectors with a stratified inventory list for use in IAEA inventory verification. Facility attachments also call for the PIL which accompanies each material balance report to list all batches separately and to include identification and batch data for each. These two requirements can be met readily if the facility operator uses facility inventory records in the physical inventory taking which include batch and stratum information. Batch considerations at a critical facility are discussed in Part 2 section 2.5.2. Stratum is defined in the Safeguards Glossary (IAEA/SG/INF/2) as a grouping of items/batches having similar physical and chemical characteristics (e.g. volume, weight, isotopic composition, location) for the purpose of facilitating statistical sampling for measurements needed to establish and verify the material balance and its uncertainty (σ_{MUF}). For a critical facility strata should be easily defined. In the storage area the fuel pieces of like kind and quantity normally are stored together and could be considered a stratum. In the critical assembly area all the fuel drawers of the same composition, i.e. each batch, could be considered a stratum. In the example facility there could be eight strata in storage and one in the

critical assembly area as shown in Table 3.1. The depleted uranium in KMP-B is a single stratum of 720 items, i.e. drawers in the blanket matrix. If a physical inventory were taken while there were drawers in the other zones of the matrix such drawers could be considered four additional strata having compositions as shown in Table 2.1. Perpetual inventory records such as shown in Figures 2.12, 2.13, 2.19, and 2.20 would provide the necessary batch and strata information for physical inventory taking and inventory verification.

3.2.4 General Inventory Instructions

Inventory instructions should include a detailed description of the procedures to be followed in taking the inventory to assure that all items and materials are inventoried once and not more than once. Such description should include instructions for listing the inventory and tagging inventoried items and for control of inventory lists and tags. Inventory teams should be assigned specific blocks of numbered tags, and all such tags should be accounted for by the team to the Facility Inventory Supervisor. At a critical facility the perpetual inventory records could be used as the inventory item lists or such lists could be prepared from the perpetual inventory records. Alternatively, lists could be prepared by the inventory team directly from the material as it is inventoried. Inventory item lists should have a column for the tag number along with other columns for the inventory data which would identify each item or batch, show the associated quantity of nuclear material element and isotope and any other data pertinent for the inventory such as batch and strata identification. The perpetual inventory cards or the inventory item lists would be used to locate and identify each item or batch. Tag numbers could then be listed by each item or batch as it is tagged. The team should assure that all items physically present are tagged and checked against the list and that all items on the list are located, identified and tagged.

Every tag number assigned to a team should either appear on one of the lists assigned to that team or be returned to the Facility Inventory Supervisor. When an inventory item list or perpetual inventory card has been completed, i.e. all items tagged, it should be initialed by both team members

Table 3.1

Reference Facility Ending Inventory

March 31, 1984

<u>Strata Number</u>	<u>Material Type Code</u>	<u>Number of Items</u>	<u>Average Element Weight</u>	<u>Average Isotope Percent</u>	<u>Total Quantity (grams)</u>	
					<u>Element</u>	<u>Isotope</u>
KMP-A						
1	PUBLK	4000	75.28	-	301,120	-
	PUBRN	1000	75.33	-	75,330	-
2	PUWHT	5000	37.56	-	187,800	-
3	93RED	2988	79.93	93.11	238,831	222,376
	93ORG	1000	80.16	92.98	80,160	74,533
4	93BLU	3000	40.14	93.11	120,420	112,123
5	20YEL	4989	80.08	19.89	399,519	79,464
6	NUGRN	3000	2404	-	7,212,000	-
7	DUL	560	14011	-	7,846,160	-
8	DUS	1280	4658	-	5,962,240	-
KMP-B						
9	DUL	1440	14011	-	20,175,840	-
	DUS	720	4658	-	3,353,760	-

and any auditors or other observer, as appropriate. Completed lists should be collected by the Facility Inventory Supervisor.

The inventory instructions should make provision for anomalies and discrepancies. The inventory teams should have no discretion to deviate from the written instructions; therefore, the instructions should provide that the Facility Inventory Supervisor be called to resolve any problems.

3.2.5 Preliminary Inspection and Review

Prior to the conduct of a physical inventory the Facility Inventory Supervisor should conduct a preliminary inspection of the facility areas to be inventoried and review inventory instructions and procedures with the responsible personnel. Such inspections and reviews should be made sufficiently in advance to allow time for corrective action, if needed. Records should be reviewed to assure that they are current and that record clerks and accountants understand the records cut off procedures and times applicable to each set of records.

Inventory instructions should be reviewed in detail with each inventory team for the area it is to inventory to assure that each team member understands his duties and responsibilities.

3.3 Conduct of Inventory

The inventory should proceed according to instructions and plans. If proper planning, inspection and personnel orientation have been carried out, there should be a minimum of problems. Nevertheless, the Facility Inventory Supervisor should not be assigned to any specific activities, such as being a team member, so that he can be available at any time to take care of anomalies and to approve deviations from planned procedures.

3.4 Post-Inventory Activities

3.4.1 List and Tag Accuracy Check

Upon completion of the inventory in each area, before the area is released from cutoff the Facility Inventory Supervisor should inspect the area to assure that all material in the area has been tagged with current inventory tags. A random sample of the items in each should be checked against the inventory lists and a random sample of the times on the lists should be checked against the items in the areas to assure that items have been tagged and have been recorded accurately on the inventory sheets. It is neither practical nor necessary that these verification checks be 100%. A valid statistical sampling plan could be used for the two populations, i.e., the population of tagged items and the population of listed items. If these tests of the inventory accuracy do not show a high level of confidence, on the order of 95% or better, additional checks should be made or the area reinventoried. In addition to the list and tag accuracy checks for an area, the tags and lists assigned to the area team also should be accounted for. Upon completion of the entire inventory all tags and inventory item lists should be accounted for by the Facility Inventory Supervisor.

3.4.2 Cutoff Verification

Upon completion of the facility inventory, the cutoff procedures should be verified to assure that all internal transfers were recorded in more than one area; that material was inventoried in the proper area; and that all receipts and shipments were recorded properly with the respect to the cutoff times for the respective material movements.

3.4.3 Inventory Summary and Reconciliation

The raw data from the verified perpetual inventory cards should be summarized as soon as possible after the physical inventory taking is complete. The original cards and their summaries should be maintained under the control of the Facility Inventory Supervisor.

Inventory reconciliation involves comparing the results of the physical inventory to the inventory as stated in the records and resolving any differences to the extent possible by correction of errors in either set of data. A record should be made of all adjustments made during reconciliation to either the physical inventory data or to the records. Such adjustments to the inventory data should be approved by the Facility Inventory Supervisor. Adjustments to the records should be approved according to established facility control procedures for such adjustments. The final adjustment would be the material unaccounted for (MUF) which will bring the records into agreement with the physical inventory. At a critical facility it is unlikely that there will be any MUF.

3.5 Material Balance Closing

While it is the physical inventory that is the bench mark of nuclear material accounting and control, in that it provides evidence of the physical presence of the material, it is the material balance based on the physical inventory that permits the determination of whether significant losses or diversions have occurred undetected. Data, not only from the physical inventory, but from all components of the material balance are necessary so that the material balance and MUF resulting from the data can be evaluated and meaningful conclusions reached. For an item facility such as a critical assembly there should be no MUF in the same sense as in a bulk handling facility where measurement uncertainties occur. At a critical facility either all of the items are accounted for or they are not or perhaps there even may be more items than there should be. If not, then MUF is not evaluated in terms of σ MUF but in terms of missing or additional items. Discussion of MUF evaluation in terms of σ MUF is included in IAEA-STR-150 but has not been included in this document because it is not appropriate for a critical facility.

Even though MUF is not expected at a critical facility, closing a material balance for a time period spanned by two successive physical inventories involves the use of the material balance equation to calculate the balance or imbalance of material in terms of material unaccounted for (MUF). This basic equation is as follows:

$$\text{MUF} = \text{B} + \text{I} - \text{O} - \text{E}$$

where: MUF = Material Unaccounted For

B = Beginning Inventory

I = Input

O = Output

E = Ending Inventory

At a critical facility the material quantities associated with the four components of the equation are derived directly from the quantity data for each item associated with each of the equation components. Thus if all items are accounted for the quantities associated with them should result in a material balance with a zero MUF.

Upon completion of the physical inventory taking, the physical inventory material quantity data are totaled to establish the value for the ending inventory component of the equation. The data for the various items are obtained from the records to establish the other components of the equation, i.e. beginning inventory, inputs and outputs. From these three components the book inventory (B + I - O in the above equation) is established. Adjustments may be necessary for conditions, such as mistakes, that may be discovered during the inventory reconciliation. These adjustments are made, as appropriate, and the ending physical inventory is compared with this adjusted book inventory to determine MUF. Detailed information for data handling procedures to provide the appropriate data for use in the material balance calculation are given in Part 2 addressing the Records and Reports Element of an SSAC.

A material balance report for each MBA, in this case the entire facility, is called for in most facility attachments. The physical inventory results are combined with data from other facility records to produce such a report. This is addressed in the detailed descriptions of the Records and Reports Element of an SSAC. Each material balance report is required to have attached a physical inventory listing (PIL) which lists all batches, specifying material identification and batch data for each batch. This listing

should be easily prepared from the lists that result from the physical inventory taking if they are prepared and coded as described herein.

3.6 Example Physical Inventory Taking Procedures, Instructions and Data Handling

3.6.1 Introduction

The facility used in this example is the horizontal split-table facility described in Appendix A. Figure A.2 is a material flow diagram for the facility. The facility is a single MBA for IAEA safeguards purposes. The ending inventory for the example eight month material balance period ended 31 March 1984 is given in Table 3.1. The procedures and instructions given in the following sections are expressed in the imperative mode because they are examples of what would be issued at a facility. They are not proposed as requirements. Variations are, of course, quite possible. The example assumes the PIT is conducted during a fuel configuration change. This may not always be appropriate. The example assumes one inventory team. The operator may choose to use two or three teams. Different instructions would then be needed to take account of the different critical assembly status and the different team instructions.

3.6.2 General Inventory Procedures

Physical inventory of the facility shall be taken during each major fuel configuration change or each year whichever is most often. Scheduling shall be the responsibility of the Head of the Nuclear Material Control Group in consultation with the Facility Manager. When the inventory is scheduled to coincide with a fuel configuration change the actual physical inventory taking shall occur after fuel pieces from the prior configuration have been returned to storage and before the new fuel configuration is done. When no fuel configuration change is scheduled the critical assembly physical inventory will depend on the core matrix map and perpetual inventory records.

3.6.3 Responsibility and Authority

3.6.3.1 Facility Inventory Supervisor

The Head of the Nuclear Material Control Group is designated as the Facility Inventory Supervisor. As such he is responsible for planning, organizing and conducting the physical inventory. Specifically his responsibility and authority shall be as follows:

- (1) He shall be responsible for scheduling the inventory and the critical assembly shutdown if any.
- (2) Within the established inventory schedule he shall be responsible for and have sole authority to establish and enforce all cutoff times and procedures.
- (3) He shall be responsible for assuring that the physical inventory taking is conducted in accordance with established written procedures and instructions and shall have the authority to cause such actions as are necessary to attain such assurance. He also shall have the authority to authorize deviations from established procedures and instructions, if, in his opinion, such deviations are necessary to the satisfactory conduct of the inventory taking.
- (4) He shall not be a member of the inventory team but will be available at all times during the inventory to handle problems to assure satisfactory conduct of the inventory taking.
- (5) He shall be responsible for assigning personnel to the inventory team in consultation with the appropriate facility management persons.

3.6.3.2 Inventory Team

The inventory team shall be designated by the Facility Inventory Supervisor in consultation with other appropriated managerial staff. The team shall include one person from the Operations Group and one person from the

Nuclear Material Control Group. The inventory team shall be responsible for carrying out the physical inventory taking instructions applicable to each area and shall have no authority to deviate from such instructions without the approval of the Facility Inventory Supervisor.

3.6.4 Inventory Instructions

3.6.4.1 General Schedule and Cutoffs

A physical inventory of the entire facility will be taken for the material balance interval ending 2400, 31 March 1984*. Critical assembly shut-down and fuel change should be scheduled so that the inventory team can begin tagging no later than 0800, 1 April 1984*. New fuel configuration installation is scheduled 0800, 2 April 1984, provided that inventory taking, summarizing and reconciliation are complete and that IAEA verification has been completed. The following cutoff times and procedures shall be observed.

3.6.4.1.1 Receipts and Shipments

There shall be no receipts into or shipments from the facility after 2400, 31 March 1984 and before 0800, 2 April 1984. If material arrives at the facility for receipt it shall be held in quarantine until completion of the inventory and the Facility Inventory Supervisor notified of its presence.

3.6.4.1.2 Internal Transfers

There shall be no transfers between KMPs after 2400, 31 March 1984 and before 0800, 2 April 1984.

* These times and dates and all others that follow are meant to be illustrative only.

3.6.4.1.3 Critical Assembly

Because a fuel configuration change is scheduled, all fuel pieces not to be in the new configuration shall be returned to main storage prior to 2400, 31 March 1984. Fuel pieces shall be returned to the birdcage or bin appropriate for the kind of fuel piece and perpetual inventory cards prepared. Under normal conditions preparation for the new fuel configuration should be ready for start up by 0800, 2 April 1984.

3.6.4.1.4 Records

All transaction documents for transactions prior to 2400, 31 March 1984 shall be made available to KMP and Facility Records no later than 0800, 1 April 1984. All transactions up to 2400, 31 March 1984 shall be posted to the facility records and to the KMP records and the books closed as of the 2400 cutoff time. The KMP and facility book inventories as of the 2400 cutoff time shall be calculated.

3.6.4.2 Physical Inventory Item List and Stratification

Upon completion of the inventory preparation as directed in the preceding sections the Facility Inventory Supervisor shall prepare copies of the perpetual inventory cards for all materials in both KMPs. Each such card shall be considered a stratum for verification purposes. Figures 2.12 and 2.13 show examples of such cards as they would appear for the 31 March 1984 ending inventory. Note that the Inventory Tag number spaces have not yet been filled in. This would be done during the physical inventory taking.

3.6.4.3 Inventory Team Instructions

The inventory team will be provided with copies of the perpetual inventory for each KMP when they are ready to conduct the inventory in that KMP. They also will be provided with pre-numbered blank perpetual inventory cards, pre-numbered inventory tags and an Inventory Tag Reconciliation form (Figure 3.1). The inventory team shall proceed as follows:

Figure 3.1

Inventory Tag Reconciliation

Number of Items

1. Listed on perpetual inventory cards _____
2. Listed on blank perpetual inventory
cards _____
Total Items Tagged _____

Inventory Tags

1. Number assigned (from thru) _____
2. Deduct numbers not used
(from thru) _____
3. Deduct voided tags (Nos) _____

Tags used _____

Tags used should equal items tagged

Explain variances below:

Prepared by: _____

Date: _____

(a) KMP- A

- (1) One member of the team identifies each batch by container, i.e. birdcage or bin, number and counts the number of items in each unless the container has a facility or an IAEA seal and a label indicating the content. He calls out the batch identity, i.e. container number, the number of items per batch and the inventory tag number which he affixes to each container. Empty containers shall also be inventoried and tagged to assure that they are, in fact, empty.
- (2) The second team member records the inventory tag number on the copy of the perpetual inventory card in the proper column on the line corresponding to the identified batch. If there is no batch number on the list as called out by the first team member, as would be the case for empty containers, the number of both the container and the tag are entered on the blank perpetual inventory cards along with the number of items in the container, if any.

(b) KMP- B

- (1) One member of the team identifies from the core configuration map those positions in the matrix that should be filled and the batch identities for each batch. He identifies and counts the drawers in the respective positions in the matrix to determine that they agree with the core configuration map. For each batch he affixes an inventory tag to the core configuration map in line with the batch identity and calls out the batch identity and tag number. If there are any drawers in intermediate storage he also identifies, tags and calls and the pertinent information for them. He also observes that there are no drawers in the matrix or in the intermediate storage area other than those identified on the core configuration map or perpetual inventory cards. If there are these must be identified tagged and recorded. Empty spaces are not inventoried.
- (2) The second team member records the inventory tag number in the appropriate space on the copy of the perpetual inventory card corresponding to the identified batch. If there is a batch or

drawer found to be present but not listed on the cards, the identity and number of items are recorded on a blank card along with the inventory tag number for each such item or batch.

- (c) Both team members shall sign each of the perpetual inventory cards.
- (d) Both team member shall complete and sign the tag reconciliation form.
- (e) All perpetual inventory cards (used, voided or unused), unused inventory tags and the inventory tag reconciliation sheet are given to the Facility Inventory Supervisor.

3.6.5 Post Inventory Activities

3.6.5.1 List and Tag Check

Upon completion of the tagging by the inventory team the Facility Inventory Supervisor shall:

- (a) Account for all pre-numbered forms and tags assigned to the team.
- (b) Using an attributes sampling plan, randomly select a sufficient number of listed items from the inventory item lists to verify with 95% certainty that items listed were tagged and listed correctly, i.e. label data correctly listed on inventory sheets.
- (c) Inspect each area and randomly select a sample of the items in the area to assure that the items had been tagged and listed correctly, i.e. label data correctly listed on inventory sheets.
- (d) Cause such actions as he may deem appropriate to correct any discrepancies found.

3.6.5.2 Cutoff Verification

Upon completion of the inventory taking and list and tag check, the Facility Inventory Supervisor shall review the records to assure that all transfers were recorded properly in the records and that no transfers occurred or were recorded after the cutoff time. Any discrepancies found shall be corrected in the records so that the material involved in any transfers, receipts, or shipments is correctly documented in relation to the material balance interval ended 2400, 31 March 1984. Physical inventory listings also may need to be corrected for improper material movement after cutoff.

3.6.5.3 Inventory Summary and Reconciliation

Upon completion of the preceding checks, the Facility Inventory Supervisor shall prepare from the inventory item lists, physical inventory summaries for the facility. Table 3.2 is an example of such a summary for the example facility. This summary is compared to the book inventory for the facility and for the respective KMPs. Any differences are investigated and resolved to the extent possible by correcting mistakes in each set of data. Any corrections or other adjustments shall be clearly noted in the records that are changed. The final adjustment will be material unaccounted for (MUF), if any, made to the records to bring the MBA and facility records into agreement with the physical inventory.

A material balance report shall be prepared for submission to the IAEA along with a PIL using the appropriate IAEA forms. Figure 3.2 is an example Material Balance Report for the example facility. Figure 3.3 is an example of a Physical Inventory Listing on IAEA Form R.02/c for the material on inventory at the example facility at the end of the example material balance period.

References

IAEA/SG/INF/1: IAEA/SG/INF/1, IAEA Safeguards Glossary, IAEA, Vienna, 1980.

IAEA-STR-150: R.J. Jones, E.V. Weinstock, W.R. Kane, "Detailed Description of an SSAC at the Facility Level for a Low Enriched Uranium Conversion and Fuel Fabrication Facility", IAEA, STR-150 (Draft), May 1984.

Table 3.2

Physical Inventory Summary

Material and Batch Identity	Item per Batch	grams U	Total Quantity grams Pu	grams U-235
<u>KMP-A</u>				
PUBLK: P-100 thru				
P-199	40		301,120	
PUBRN: P-200 thru				
P-224	40		75,330	
PUWHT: P-500 thru				
P-624	40		187,800	
93RED: 9-100 thru				
9-173	40	236 593		220,292
9-174	28	2,238		2,084
93ORG: 9-175 thru				
9-199	40	80,160		74,533
93BLU: 9-500 thru				
9-574	40	120,420		112,123
20YEL: 2-100 thru				
2-198	50	396,396		78,843
2-199	39	3,123		621
NUGRN: Bin 100 thru				
Bin 129	100	7,212,000		-
DUL: Bin 300 thru				
Bin 313	40	7,846,160		-
DUS: Bin 500 thru				
Bin 511	100	5,589,600		-
Bin 512	80	372,640		-

Table 3.2 Physical Inventory Summary (cont'd)

Material and Batch Identity	Item per Batch	Total Quantity		
		grams U	grams Pu	grams U-235

<u>KMP-B</u>				
DB	720	23,529,600		-

Totals

Enriched U: Total FI	838,930	488,496
Book Inventory	838,930 [*]	488,496 [*]
MUF	0	0
Plutonium: Total PI	564,250	
Book Inventory	564,250 [*]	
MUF	0	
Natural U: Total PI	7,212,000	-
Book Inventory	7,212,000 ^{**}	-
MUF	0	
Depleted U: Total PI	37,338,000	-
Book Inventory	37,338,000 ^{**}	-
MUF	0	-

* See Figures 2.9 and 2.10 and Table 3.1

Figure 3.2.
MATERIAL BALANCE REPORT (MBR) FORM R.03

COUNTRY ABC										REPORTING PERIOD, FROM 01 08 83 TO 31 03 82									
FACILITY CAF										REPORT No. 30									
MATERIAL BALANCE AREA CAF										PAGE No. 1 OF 2 PAGES					SIGNATURE				
A, B, C, A, F, C, A, F, 0, 1, 0, 8, 8, 3, 3, 1, 0, 3, 8, 4, 2, 0, 3, 0, 1																			
ENTRY No.	CONTINUATION	ENTRY NAME	ACCOUNTANCY DATA				CONCISE NOTE	CORRECTION TO											
			ELEMENT	WEIGHT OF ELEMENT	UNIT kg/g	WEIGHT OF FISSILE ISOTOPES (URANIUM ONLY) (g)		REPORT No.	ENTRY No.										
1		Beginning Inventory																	
0.1		P.E. ☐	E	7,606,10	g	4,150,31	G		7										
		Receipts																	
0.2		R.F. ☐	E	8,016,0	g	7,453,3	G		7										
		Shipments																	
0.3		S.F. ☐	E	1,840	g	1,068	G		7										
		Ending Book Inventory																	
0.4		B.E. ☐	E	8,389,30	g	4,884,96	G		7										
		Adjusted Book Inventory																	
0.5		B.A. ☐	E	8,389,30	g	4,884,96	G		7										
		Ending Physical Inventory																	
0.6		P.E. ☐	E	8,389,30	g	4,884,96	G		7										
		Beginning Inventory																	
0.7		P.E. ☐	N	7,212,000	g				7										
		Ending Book Inventory																	
0.8		B.E. ☐	N	7,212,000	g				7										
		Adjusted Book Inventory																	
0.9		B.A. ☐	N	7,212,000	g				7										
		Ending Physical Inventory																	
1.0		P.E. ☐	N	7,212,000	g				7										
		Beginning Inventory																	
1.1		P.E. ☐	D	3,733,8000	g				7										
		Ending Book Inventory																	
1.2		B.E. ☐	D	3,733,8000	g				7										
		Adjusted Book Inventory																	
1.3		B.A. ☐	D	3,733,8000	g				7										
		Ending Physical Inventory																	
1.4		P.E. ☐	D	3,733,8000	g				7										
		Beginning Inventory																	
1.5		P.E. ☐	F	4,889,20	g														
		Receipts																	
1.6		R.F. ☐	F	7,533,0	g				7										
		Ending Book Inventory																	
1.7		B.E. ☐	F	5,642,50	g				7										

* In this case the adjusted book inventory is the same as the ending book inventory because there were no adjustments in the example. In such case the ending book inventory line could be omitted.

Figure 3.2. (cont'd)
MATERIAL BALANCE REPORT (MBR) FORM R.03

[illegible]

* In this case the adjusted book inventory is the same as the ending book inventory because there were not adjustments in the example.
7) In such case the ending book inventory could be omitted.

figure 3.3

19-75

Appendix A

Example Reference Facility

A.1 Introduction

The two basic types of critical assembly core arrangements are the horizontal split-table and the vertical integral. The horizontal split-table assembly consists of two approximately equal portions that move apart when the reactor is shut down. Fuel elements in the form of open "drawers" are loaded, usually manually, from between the halves into a horizontal matrix structure.

In the vertical integral assembly, full-length, self-contained fuel elements are suspended from a grid plate. There is no provision to separate large portions of the assembly during shut down. The necessary shutdown margin is obtained by withdrawing control elements. The fuel elements in many cases are assembled from component pieces such as plates, pin or rods for loading into the reactor. Even if such assembly and disassembly is not possible, there often are several different types of assemblies that are used in different configurations in experiments.

From the standpoint of nuclear material accounting and control both types of facilities present similar situations. Nuclear material inventories consist of hundreds and sometimes thousands of kilograms of uranium of various enrichments and plutonium in the form of many fuel assemblies, hundreds of fuel assembly components or thousands of small fuel pieces all in relatively pure form. These facilities are research tools designed for experimental flexibility. The core in either type is accessible and can be modified and re-assembled in a relatively short period. This complicates nuclear material accounting and control because many pieces are moved from place to place within a facility when modifications take place.

One aspect of these facilities simplifies nuclear material accounting and control. The reactor operates at very low power so that fuel burn-up is negligible. This also results in a relatively stable total facility inventory. Nuclear material receipts and shipments are infrequent and usually of small quantity, e.g. shipment of small quantities of broken fuel plates.

The procedures demonstrated in the examples in this document are based on a horizontal split-table type facility with the general characteristics described in the following sections. If one considers the fuel element of the horizontal split-table type to be a drawer filled with nuclear material fuel pieces to be accounted for and the fuel element of the vertical integral type to be an assembly of nuclear material fuel components to be accounted for one can see that the nuclear material accounting and control procedures would be essentially the same for the two types. Because of this it was thought to be unnecessary to provide examples based on separate types of reference facilities.

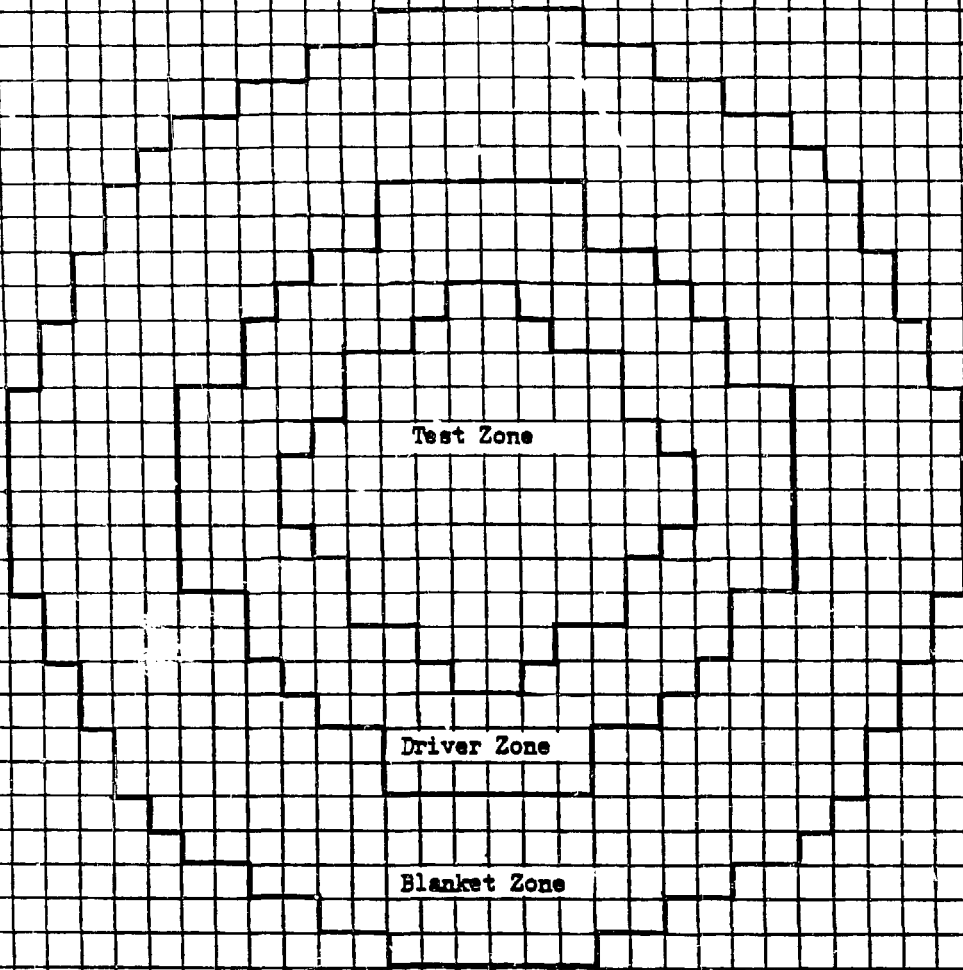
In the horizontal split-table reference facility the fuel pieces are assumed not to be individually identified, such as by serial number, but to be color coded by material type. It also is assumed that the sizes of the pieces are directly related to the quantity of contained nuclear material. At some facilities each fuel piece may be individually identified but color coding may still be useful to avoid having to check the serial number of each small piece. At some facilities the sizes of the pieces may not be related to the material content, i.e. all the pieces are the same size. In such cases the pieces would need to be coded as to material content with, perhaps, color codes representing average nuclear material content for batches of fuel pieces having the same material content within some tolerance limits. Such information would be needed for both material accounting and experimental purposes. At a vertical integrated facility the various components to be assembled into fuel elements, i.e. pins, plates, pellets, etc. also would need to be identified either individually or in batches for both material accounting and experimental purposes.

A.2 General Characteristics

For the purposes of preparing examples of nuclear material accounting and control procedures for critical assembly facilities the following assumptions are made:

- (a) The facility is a horizontal split-table type critical assembly having a 40 x 40 array of matrix tubes into which are inserted drawers approximately 700 mm long containing fuel pieces of various compositions along with various spacers and moderator materials. These drawers form the reactor core configuration which has three zones, i.e. test zone, driver zone, and blanket zone. The configuration of these three zones for the example material balance period is shown in Figure A.1, Example Core Configuration Map.
- (b) The fuel inventory contained in 29,000 fuel pieces consists nominally of 565 kg of plutonium, 440 kg of 93% enriched uranium, 400 kg of 20% enriched uranium, 7,200 kg of natural uranium and 37,000 kg of depleted uranium. Each fuel piece is color coded for material type, and quantity. Table A.1 shows the characteristics of the various pieces in the reference facility inventory at the end of the example material balance period.
- (c) The reactor is operated at a power level of two kilowatts or less so that nuclear production and nuclear loss are negligible except for Pu-241 decay which is calculated only when plutonium fuel pieces are shipped from the facility. Major core configuration changes occur infrequently, i.e. six to eight months, but minor changes occur often within the core inventory.
- (d) The facility inventory is maintained at the level indicated in b) above with major inventory changes occurring only every few years. Because the facility inventory contains significant quantities of nuclear material in IAEA inspection is conducted every two weeks. A facility nuclear material physical inventory is not conducted during each such inspection but is conducted in coordination with each major core configuration change or once each year whichever is most frequent.

Figure A.1
Example Core Configuration Map



Test Zone:	88 drawers, 2 halves	=	176	drawers
Driver Zone:	144 drawers, 2 halves	=	288	drawers
Blanket Zone:	360 drawers, 2 halves	=	<u>720</u>	drawers
	Total		1184	drawers

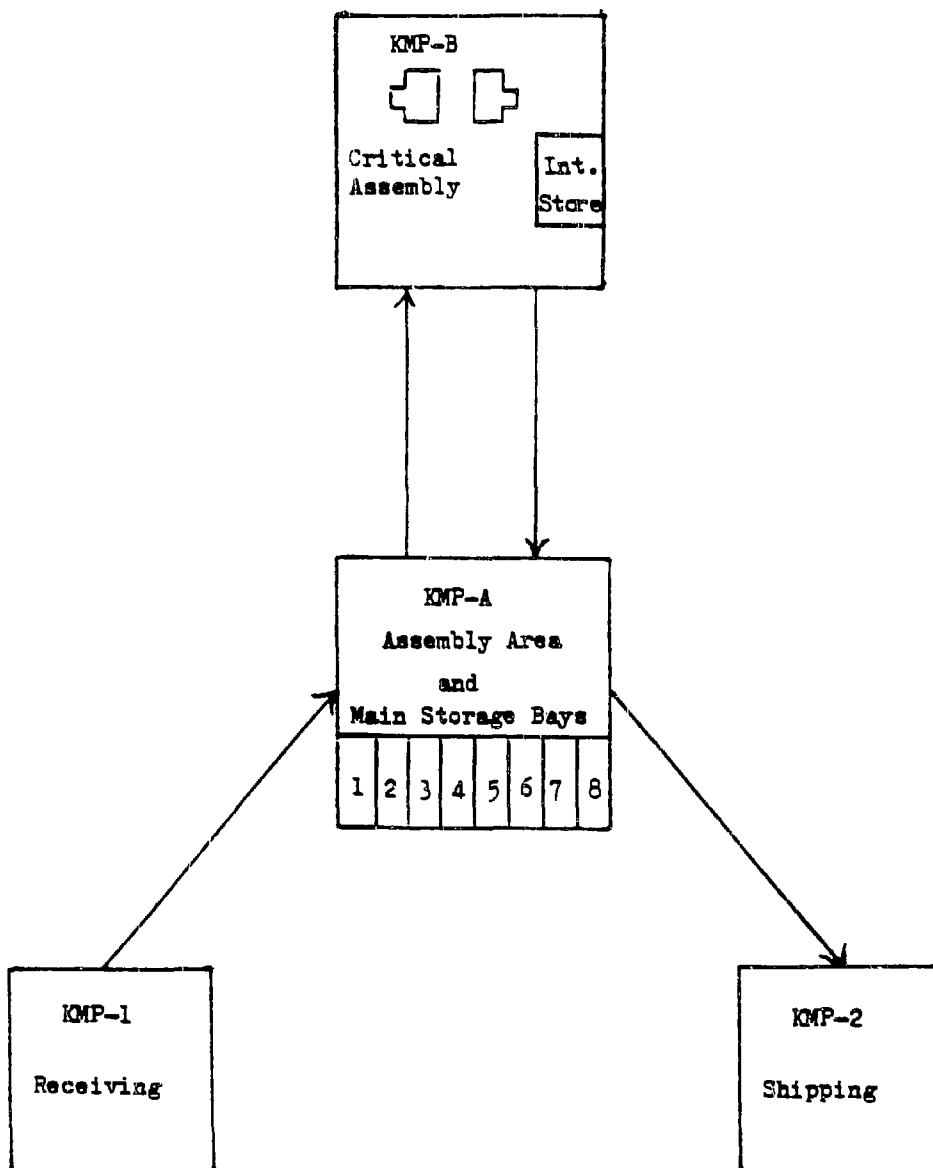
A.3 Nuclear Material Flow

The facility is a single MBA. Figure A.2 is a diagram of the flow of nuclear material through and within the MBA. There are only two flow KMPs, KMP-1 for receiving and KMP-2 for shipping. There are two inventory KMPs, KMP-A fuel assembly and storage and KMP-B the critical assembly core area. There is an intermediate storage area in the KMP-B area where a small inventory of fuel pieces is maintained for minor core configuration changes. The major flow of material occurs at the time of major core configuration change. At such time all fuel drawers to be changed are returned to KMP-A and unloaded to storage. New fuel drawers are assembled in KMP-A, transferred to KMP-B, and placed in the core. Some small quantity of material may be transferred to KMP-B intermediate storage from time to time to accommodate minor core changes. It is assumed that the main storage area includes birdcages and bins that can be sealed but that neither the intermediate storage area in KMP-B nor the reactor core can be sealed. To simplify the examples it is assumed that during the example time period none of the depleted uranium fuel pieces in the blanket are moved.

The facility is considered to be an item facility which is defined in the IAEA Safeguards Glossary as a "nuclear facility where all nuclear material is contained in identifiable items (e.g. fuel assemblies)". The determination of the quantity and composition of the nuclear material in the fuel pieces is based on the fabricator's data. At KMP-A the fabricator's data may be verified by the facility operator using NDA methods but the fabricator's data usually is accepted. Because individual fuel pieces are not individually identified an average quantity of nuclear material per piece is used, based on the fabricator's data for batches of fuel pieces.

Figure A.2

Critical Facility Nuclear Material Flow Diagram



A.4 Batch Composition

Batching considerations are discussed in section 2.5.2. In the reference facility each fabricator's shipment of the same type fuel pieces is considered a batch but is not retained for internal material control. The fuel pieces are stored in the main storage area bays (KMP-A) in special containers for criticality control, i.e. birdcages for the plutonium and enriched uranium pieces and in bins for the natural and depleted uranium blocks. Each container contains only one kind of fuel and is considered a batch. The container number is the batch identity. Table A.3 shows the container/batch identity for the reference facility. In the critical assembly area (KMP-B), fuel pieces of different material types may be mixed in various configurations in the fuel drawers. Spacing materials and moderators also are placed in the drawers but are not addressed in this example. In KMP-B a fuel drawer is considered an item and all drawers containing the same composition of nuclear material are considered a batch. A composition code is the batch identity along with a core sign to identify and locate individual drawers in the batch. The composition code uses letters and numbers to identify the fuel materials in the drawers in the critical assembly area (KMP-B) with P, 9, 2, N and D representing plutonium, 93% enriched uranium, 20% enriched uranium, natural uranium and depleted uranium respectively. The letters T, D, and B in the last position of the composition code identify the zone location of the drawer, i.e. Test, Driver and Blanket respectively. The letter I in this position identifies intermediate storage. Table A.4 shows the drawer composition, composition code and core sign for four different drawer compositions assumed for the example. In there intermediate storage area in KMP-B a batch could be a drawer, a group of drawers, or a group of like kind fuel pieces stored together depending on the material in the area. It is further assumed that one step in preparation for the physical inventory taking is to return all material in the intermediate storage area and all material in core drawers that will be changed to main storage in the respective fuel piece containers. The item-batch structure for the reference facility is shown in Table A.3.

References:

IAEA/SG/INF/1: IAEA Safeguards Glossary, IAEA, Vienna, 1980

IAEA-STR-150: R. Jones, E. Weinstock, W. Kane, Detailed Description of an SSAC at the Facility Level for a Low Enriched Uranium Conversion and Fuel Fabrication Facilities, System Studies Section, Division of Development and Technical Support, IAEA Department of Safeguards, STR-150 (Draft), May 1984.

Table A.1
Reference Facility Fuel Material

Item Description	Nominal Quantity per piece (grams)	Nominal Inventory		Material Description (dimensions in mm)
		Pieces	Material (kg)	
PUBLK	75	4000	300	50x50x2-Pu plates-black
PUWHT	37.5	5000	188	50x25x2-Pu plates-white
PUBRN	75	1000	75	50x50x2-Additional Pu plates-brown
93RED	80	3000	240	50x50x2-93% Enr U plates-red
93BLU	40	3000	120	50x25x2-93% Enr U plates-blue
93ORG	80	1000	80	50x50x2-Additional 93% Enr U plates-orange
20YEL	80	5000	400	50x50x2-20% Enr U plates-yellow
NUGRN	2400	3000	7,200	50x50x50-Natural U blocks-green
DUL	14000	2000	28,000	50x50x300
DUS	4660	2000	9,320	50x50x100 Depleted U blocks - not color coded

At an actual facility it would be expected that there would be a greater variety of fuel pieces. For example the plutonium fuel pieces might have different percentages of fissile content and there might be more different sizes and shapes. The varieties in the example have been limited to those described above to keep the example simple and more easily understood.

Table A.2

Item - Batch Structures

<u>Location</u>	<u>Item</u>	<u>Batch</u>	<u>Batch Identity</u>	<u>Batch Data</u>
Receipts	One fuel piece	One shipment of like pieces	Fabricator Batch Number	1. Item color code 2. Number of items 3. Average nuclear material content per item
Main Storage	One fuel piece	One Storage Container	Storage Container Number, See Table A.3	1. Item color code 2. Number of items 3. Fabricator's average nuclear material content for each item 4. Storage Container numbers
Critical Assembly	One fuel drawer	All drawers containing the same nominal weights of nuclear material	Composition Code-See Table A.4	1. Drawer Numbers 2. Number of drawers 3. Number of fuel pieces of each kind 4. Quantity of nuclear material of each type 5. Core location code
Intermediate	Same as for main storage if fuel pieces are not in drawers.			
	Same as for critical assembly if fuel pieces are in drawers.			
Shipments	One fuel piece	One shipment of like pieces	Shipment container numbers	1. Item identity 2. Number of items 3. Nuclear material content per item.

Table A.3

Main Storage Batch Identity

<u>Material</u>	<u>Item Description Code</u>	<u>Container Numbers¹ (Batch Identity)</u>	<u>Fuel Pieces per Birdcage/Bin</u>	<u>Containers In Use 31 March 1984</u>
Pu-50x50x2	PUBLK + PUBRN	P-100 → P-499	40	P-100 → P-224
Pu-50x25x2	PWWHT	P-500 → P-999	40	P-500 → P-624
93% Enr U-50x50x2	93RED + 93ORG	9-100 → 9-499	40	9-100 → 9-199
93% Enr U-50x25x2	93 BLU	9-500 → 9-999	40	9-500 → 9-574
20% Enr U-50x50x2	20YEL	2-100 → 2-499	50	2-100 → 2-199
Natural U-50x50x50	WUGRN	Bin 100 → Bin 299	100	Bin 100 → Bin 129
Depleted U-50x50x300	DUL	Bin 300 → Bin 499	40	Bin 300 → Bin 313
Depleted U-50x50x100	DUS	Bin 500 → Bin 799	100	Bin 500 → Bin 512

Explanatory Note:

- The container numbers listed are those assigned to the specific kind of fuel piece. In the actual storage area not all containers would contain fuel pieces depending on how many pieces were on inventory as indicated in the last column for the 31 March 1984 ending inventory.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 20: IAEA SAFEGUARDS EXPERIENCE AT RESEARCH REACTORS

The research reactors and critical facilities under IAEA safeguards will be described briefly with emphasis on the types and quantities of nuclear material found at such facilities. The nature and requirements of a typical facility attachment will be presented. Records and reporting requirements will be discussed. IAEA inspection procedures and measurement instrumentation will be described in detail. The analysis of inspection data to verify the nuclear material inventory will be considered. The session will emphasize actual experience at research reactor facilities under IAEA safeguards.

After the sessions, participants will be able to:

- Describe the major activities during IAEA inspection of research reactor facilities.
- Describe the records and reports prepared by the facility operator for the IAEA.
- Discuss the inspection procedures and measurement instrumentation used during IAEA inspections at research reactors.
- Understand how the IAEA activities help to meet the safeguards goals of detection and deterrence.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 20: IAEA SAFEGUARDS EXPERIENCE AT
RESEARCH REACTORS

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International Atomic Energy Agency

I. The title of this session "Safeguards Experience at Research Reactors (RR) and Critical Facilities" (or Assemblies) (CA) reads like a rather ambitious statement in view of the large variety of facility types and wide spectrum of features. In fact, this title stems from the "Standardized Installation Categorization" as in para. 53 of the IAEA's Glossary and it is used for SG planning purposes; with regard to the implementation of IAEA-SG, the facility categories are further subdivided, such as "research reactors," "material testing reactors," "TRIGA reactors," and "critical/subcritical assemblies."

Regarding the spectrum of features:

- a) the nuclear fuel used by these facility types varies from high-enriched uranium (HEU, up to 93% U-235) and plutonium to natural uranium;
- b) several distinctly different fuel designs can be identified;
- c) the thermal power output ranges from some 10 kilowatts to over 100 megawatts;
- d) they may be used for materials testing, radioisotope production, training, or nuclear physics studies.

Glossary, para. 67 reads:

"Research reactor - a reactor used as research tool for basic or applied research or for training. The fission heat is generally removed by the coolant at low temperature and is usually not used."

Glossary, para. 70 reads:

"Critical assembly (facility) - a research tool consisting of a configuration of nuclear material which, by means of appropriate controls, can sustain a chain reaction. It is distinguishable from research or power reactors in that it normally has no special provisions for cooling, is not shielded for high power operation, has a core designed for great flexibility of arrangement, and uses fuel in a readily accessible form which is frequently repositioned and varied to investigate various reactor concepts..."

These are in a condensed form the major features of RRs and CAs which we will have to remember. For the purpose of this presentation we will define the following groups:

- Group A: RRs with less than 25-MW(Th) power output.
 CAs with less than 1 SQ of nuclear material (core).
- Group B: RRs with more than 25-MW(Th).
- Group C: large CAs, more than 50 SQs.

This presentation is organized in 4 subsections, and each of them will be evaluated, as applicable, against the 3 facility groups. Section II will describe the SG-relevant features, but not the technical or operational design details; Section III will deal with IAEA-SG aspects, such as agreement types, inspection goals, Facility Attachment provisions, facility specific SG-approaches; Section IV will describe IAEA inspection procedures, including measurement techniques and available instruments.

A few statistical data (Fig. 1) should help to see the SG-relevance of RRCA facilities within the right perspective (all data are rounded and reflect the status as of end 1984):

a) RRCA	NPT = 150 non-NPT = 30	Total = 180	
b) RRCA less than 25 MW(Th) (less than 1 SQ)	= 165	Total = 180	[Group "A"]
" more than 25 MW(Th)	= 12		[Group "B"]
Large scale CA	= 3		[Group "C"]
c) RRCA less than 1 SQ	= 128	Total = 180	
" 1 to 5 SQs	= 37		
" more than 5 SQs	= 15		

Please note that the vast majority of such facilities are safeguarded under an NPT-type agreement, and also that an even larger number belong to Group "A" as defined above.

II. S G - r e l e v a n t F e a t u r e s

R e a c t o r p o w e r is one of the most significant parameters; it is an indicator for the achievable neutron flux. Studies have shown that research reactors operating at more than 25 MW(Th) are at least theoretically good instruments for producing significant quantities of plutonium or uranium-233 per year. Of course a few additional conditions have to be met, such as incore available space for the introduction of fertile material and heat-transfer/removal possibilities; the expected neutron flux (n/cm^2 sec) is above 1×10^{13} .

B u r n u p as high as 50% and more is reported for some of the larger reactors but nominal values are more like 20-30% for reactors of about 1 MW(Th) and above. For reactors at power levels less than 1 MW(Th), burnup becomes negligible and therefore the average lifetime of fuel elements is measured in years [at 1 MW(Th) about 4-5 years]; fuel lifetime in a 250-kW(Th) TRIGA reactor is estimated at 10 years.

The a c c e s s i b i l i t y of the fuel highly depends on the reactor type and design. The RRs of greater than 25 MW(Th) (Group "B") currently under SG are all tank-type reactors which means that the core cannot be observed during operation. In some cases all fuel-handling operations associated with the core are carried out remotely, and the same applies to the introduction or removal of experimental (or other) material. Most of them have test loops or test channels for locating material to be irradiated, including, of course, also undeclared production of fissile material.

The fuel of research reactors has been designed in a variety of shapes and configurations, such as: plates, assemblies, rods, discs, pellets, and fuel solutions. Fuel type or design or number of fuel items in the core is a function of the uranium enrichment, e.g., HEU is mostly fabricated as plates in an alloy with aluminum; intermediate enrichments are either fabricated as plates or homogeneously dispersed in some other material such as polyethylene or with zirconium as the hydride (TRIGA fuel); LEU or NU is being fabricated similarly to LWR and HWR type fuel (as oxide); and NU-metal fuel is also in use.

Most of the smaller RRs (Group A) currently under SG are of the swimming pool type. The fuel assemblies usually contain HEU. Due to the design the core is normally visible and accessible for measurements. Also the number of fuel assemblies is relatively small, and the total amount of nuclear material on inventory (fresh-fuel, core) at such a facility is below 1 SQ and more often much smaller than 1 SQ.

The identifiable locations at RRs and CAs would be (as applicable) the fresh-fuel storage, the core, the spent-fuel storage, and eventually a laboratory or hot cell facility; in particular the latter one could have an impact on MBA structure and inspection procedures. RRs and CAs are usually considered to be and treated as item facilities. In the majority of cases the nuclear material is contained in discrete items or sealed sources; however there is not always an unalterable identification number available. Nevertheless the item-accountancy principle is a justifiable choice.

There are a very limited number of large scale critical assemblies. Fig. 3 lists some of them, distinguished according to the two major types, namely vertical integral (VI) and horizontal split bed (HSB). In a few words only this means the VI type uses fuel assemblies in a matrix (D_2O or other moderator) in a vertical arrangement housed in a tank or calandria; the more important HSB type basically consists of two halves movable horizontally and thus forming the core.

A CA can just sustain a controlled nuclear chain reaction without any additional neutron source. It does not produce power in any significant quantities. In contrast to research reactors they have no provision for heat removal and only limited radiation shielding; the absence of a cooling system alone makes them less eligible for potential undeclared irradiation of fertile material (Pu production). This is also supported by the fact that the neutron flux is only in the order of 10^{10} (neutrons/cm² sec) as compared to more than 10^{14} in a research reactor.

The large inventory of HEU as well as plutonium at these large critical assemblies puts them amongst the most sensitive facilities from a SG point of view. They may contain hundreds of SQs of HEU or Pu in the form of oxide or even metal. The SG risk becomes comparable to that of a reprocessing plant or mixed-oxide fuel plant. The types of fuel vary. Sometimes platelets or pins are used which are assembled in drawers; in other cases rods are used which consist of metallic cans filled with fuel pellets, and the rods are then packed into fuel assemblies. The material is readily accessible since the sub-items of drawers or assemblies are small, such as coupons 5 x 5 cm x 1.5 millimeters. A CA is a search tool designed for fuel arrangement flexibility

and this is achieved by using fuel in an accessible form to study different core configurations. Further, the fuel is in an easily "convertible" form. Plutonium is usually in form of an alloy, but may even be in metal form; HEU coupons in metal form just plated with a layer of color are quite common, i.e., one really operates here at the borderline of the minimum conversion/detection time.

IAEA-SG treat critical assemblies as item-accountancy facilities, even though items as small as a coupon or a pellet do not carry identity numbers.

The content and isotopic composition of the items in a critical assembly are well known (at least to the operator) and it remains basically unchanged because the burnup is negligible. The radiation level is low and items can be handled (and measured) safely.

The inventory at a CA is generally static, with a limited number of inventory changes to and from other facilities; however, due to the required experimental core-configuration flexibility many internal movements of fuel items from and to a nearby storage place are to be expected. This has at least two consequences:

1. A book audit becomes a relatively simple activity. Mainly a comparison of ending and beginning book inventory is needed. No source documents for receipts and shipments need to be checked, and no state reports have to be reviewed, and
2. The MBA structure should be defined such that these internal movements are covered and that they do not trigger the issuance of state reports.

III. IAEA-SG ASPECTS

As shown above, the majority of RRs and CAs are safeguarded under an NPT-type agreement according to document INFCIRC/153, titled "The Structure and Content of Agreements between the Agency and States required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons," better known as "full-scope" safeguards or "blue-book safeguards."

This means that all provisions apply to RRs and CAs as they do for all other categories of nuclear facilities; the only important variable parameter is the inspection effort. Facilities handling less than 5 effective kilograms (i.e., less than 1 SQ) of nuclear material (NM) per year (the larger of inventory or throughput) should only be inspected once per year (reference para. 79 of INFCIRC/153). It follows that only a small number of facilities--mainly the 3 large-scale CAs--will absorb most of the IAEA's inspection effort.

Equally important in the context of these considerations is para. 81 of INFCIRC/153, which says that the determination of the actual routine inspection effort (ARIE) should take into account:

- form of NM (bulk or discrete items);
- chemical composition (oxide, alloy, metal);
- LEU or HEU;
- accessibility (CA fuel in small items, plus accessible);

- effectiveness of SSAC system
- state's nuclear fuel cycle (characteristics of facilities)
- international interdependence
- technical SG developments.

According to Fig. 1 there are approximately 30 RRs and CAs under a non-NPT-type agreement, better known as INFCIRC/66/Rev.2 or "Safeguards Document." If the RR is designated a "principal nuclear facility" then the same procedures become applicable as for reactors. We will talk about this elsewhere. If the RR or CA is designated as a "research and development facility" then no accounting or operating records have to be kept; only accounting reports need to be submitted to the IAEA. Further, such an R&D facility containing less than 1 ekg is not inspected at all, and up to 5 ekg only once per year; otherwise para. 57 of INFCIRC/66/Rev. 2 applies regarding inspection frequency.

The inspection goals for the different types and sizes of facilities under discussion here are principally the same, namely:

"The capability of detecting the diversion of one SQ of nuclear material; one SQ may range from a few items (HEU or Pu) to several thousand items (natural uranium). The timeliness detection goals are 4 weeks for HEU and Pu, and 12 months for natural U and LEU."

A diversion scenario for the large number of Group A facilities (less than 25 MW, less than 1 SQ) would have to focus on two aspects: firstly on simple removal of the available fuel, which however would only yield less than 1 SQ of material therefore one inspection per year is deemed sufficient; secondly on the possibility of undeclared production of Pu, or simply the accumulation of Pu in spent-fuel. For details I must refer to the document IAEA-TECDOC-233 (1980), a guidebook entitled "Research Reactor Core Conversion from the use of HEU to the use of LEU Fuels." Principally the Pu production is directly proportional to the thermal power and inversely proportional to the U-enrichment of the fuel, and another general conclusion is:

- large RRs [more than 25 MW(Th)] are theoretically capable of producing one or more SQs of fissile material (Pu, U-233);
- some smaller RRs [less than 25 MW(Th)] could be modified to enable them to be operated at more than 25 MW(Th).

This means that SG inspection activities must first be able to detect such undeclared production, and secondly to detect relevant design changes. Fig. 4 lists some examples for such design changes.

In view of the variety of the subject facilities it is hardly possible to describe facility specific SG approaches and individual inspection activities as well as the frequency of their implementation. In order to avoid "confusion by means of details" a few but rather generalized statements are offered:

- nuclear material accountancy measures are the fundamental SG measure, whereas C/S measures will usually play a secondary role;
- examination (audit) of records, in particular operating records, and the verification of inventory will be prevailing activities; inventory changes occur rather seldom such that flow verification as well as evaluation of State's reports become less important;
- most RRs and CAs (because of less than 1 SQ) require only one inspection per year which would be a PIV; this long interval (apart from unfavourable design characteristics) is prohibitive regarding appli-

cation of seals and surveillance.

- the potential of undeclared Pu production requires to emphasize repetitive design information (DI) verification activities, for which there should be provision in the Facility Attachment; a complete and up-to-date DI is a prerequisite;
- large RRs, LEU fueled, may require an increase of inspection frequency due to Pu accumulation in spent-fuel;
- large scale critical assemblies require almost monthly PIVs, or at least interim PIVs of unsealed HEU and Pu; the highly repetitive character of these activities is demanding on both sides, the facility operator and the inspectorate; careful inspection planning and standardization of inspection activities are possible means to minimize ARIE.

As for any other category (power reactors, FF plants, etc) of facilities the IAEA has also model Facility Attachments for RRCAs developed. They are the legal instrument and at the same time a guide-book for the actual implementation of safeguards.

The major subjects are:

- the whole facility usually is one MBA;
- flow KMPs for receipt and shipment of fuel;
- inventory KMPs are, for example, storage rooms, reactor room or core, assembly area;
- strategic points for C/S measures (eventually at large CAs), such as: core, or parts of core, storage cells (seals), storage cassettes (seals), bird cages (seals), drawers (seals), assemblies (seals), assembly area (surveillance), access routes to storages or core (surveillance);
- provisions for accounting and operating records;
- provisions for state's reports;
- inspections, inspection effort, and scope of inspections;
- IAEA's statements on results of and conclusions drawn from inspection activities.

It should be noted that prior to the negotiations of a Facility Attachment a facility operator (through the state authority) has to submit the Design Information, including detailed power operating characteristics, according to a prescribed format laid down in the "Design Information Questionnaire" (DIQ).

IV. INSPECTION PROCEDURES/MEASUREMENT TECHNIQUES

It is obvious that the design of the fuel largely determines the technique for its nondestructive assay; destructive analysis is nowhere applicable at RRCA facilities. Fuel plates, assemblies, separate fuel rods, fuel discs, aqueous homogeneous fuel solutions, coupons, fuel drawers, fuel birdcages, etc. are some possible variations and almost each of them requires specific measurement setups.

F r e s h - f u e l verification generally is a combination of item counting, item identification (where applicable), attributes test of the fuel, and variables test for total nuclear material content. The U-235 enrichment

of an item can be measured with the (semi-) portable Mini-MCA (PMCA) which is a Ge detector plus microprocessor assisted multichannel analyzer. An infinitely thick layer of material is required. If it is possible to verify the weight (net) it can be combined with the enrichment measurement to calculate the total U-235 content of a given item.

Alternatively, with appropriate gamma-spectrometry measurements (combined with transmission measurements) the total U and U-235 in MTR-type fuel can be determined, e.g., using the PMCA with an interactive software programme written specifically for NDA of MTR-type fuel.

Neutron coincidence counting techniques (in the passive or active mode) provide an advantage over gamma-ray techniques because of the high penetrability of the neutrons. The assay result reflects the total U-235 content of an item and not only of the surface layer. The instrument used is the Active Well Coincidence Counter (AWCC). Depending on the availability of good standards an accuracy of 1-2% is achievable. If only a section of an item (e.g., large assembly) can be measured, then the result is U-235 content per unit length and has to be combined with a separate determination of the active length.

HEU metal fuel coupons, thousands of which are found at large CAS, can be assayed with a PMCA or similar instrument for enrichment as a good attributes measurement; in combination with weighing this becomes an accurate variables test and permits calculation of total U and U-235.

Our Far-East Section colleagues have also successfully used the AWCC to verify stacks of several HEU fuel coupons. The individual coupons may be randomly selected from a number of other items, such as drawers or boxes or birdcages, as representative samples, and are then assayed together in one measurement.

For large fuel assemblies (e.g., of the NRX or NRU type) the so-called Uranium Neutron Coincidence Collar (UNCL) using Am Li sources has been suggested. Again the assay result is U-235 content per unit length and needs to be supplemented by a separate verification of the active length of the complete fuel assembly.

The High Level Neutron Coincidence Counter (HLNCC), which gives a measure of the non-fissile content of plutonium-bearing items (Pu-240, 242, 238), and a high-resolution gamma-spectroscopy system, such as the PMCA which provides information on the isotopic composition, are used for the measurement of Plutonium. Los Alamos Laboratory has developed a variety of configurations of neutron coincidence counters such as the standard (hexagonal) type, the drawer counter, and the birdcage counter. For the measurement of whole trays (or drawers) the HLNCC and a gamma detector are arranged in such a way that the gamma detector looks at the tray at the same time and by one measurement the whole Pu content as well as the isotopic composition is available. This procedure is used both as an attributes test and, with a greater accuracy via larger counting time, as a variables measurement. The system is interfaced with a portable computer to allow on-the-spot evaluation of the measurement data.

Often plutonium coupons are stored in a container type called birdcage (container with metal frame as spacer around); a birdcage counter HLNCC,

which can take the whole birdcage without opening it, is used to measure in an attributes or variable mode the plutonium content.

Usually all these NDA instruments are calibrated independently (usually in another lab) using standards of a similar composition and geometrical configuration, either as primary or secondary or just working standards. In some cases it has been possible to prepare working standards (e.g., stack of coupons or birdcage) at the facility where they are kept under IAEA seals. The material-ownership remains with the facility.

Further improvements are, however, desirable regarding the calibration and normalization procedures both for gamma as well as neutron based techniques. The IAEA is in the process or has even started to set up a computer file for instruments and available standards and calibration data.

It should be mentioned that some facilities, particularly those subject to a high inspection-frequency such as large critical assemblies, permit the on-site storage of IAEA equipment under IAEA seals.

Another procedural aspect of IAEA safeguards particularly at the larger and more frequently inspected facilities is the application of statistical concepts. The starting point of any physical verification of the material is usually the stratification according to the "measurability" and a basic requirement is the verification of the presence of items, i.e., the verification of the population of items in a given material stratum. A number of several thousand items (not unusual at a large CA) prohibits even an item count, not to say a qualitative NDA. Therefore according to well-known procedures [$n = N(1 - \beta)^{1/X/M}$] two sample sizes for randomly selected items are calculated:

1. the attributes sample size and related measurements are to detect completely missing items,
2. the variables sample is to detect a fractional (or even bias) falsification of items. In both cases a 95% detection probability is assumed.

Spent fuel from high-power research reactors can be verified by carrying out semi-quantitative or attributes measurements. For example, observation of the Cerenkov radiation from spent-fuel (provided sufficiently high burnup and low cooling time) offers a useful attributes test for the presence of spent fuel. A problem with this technique is that it often requires temporary reduction of light levels of the environment. The Cerenkov viewing device is also an often-used tool to verify/confirm presence of core fuel at small RRs particularly of the pool type.

At present there is no practical technique available for the quantitative measurement of plutonium content in spent fuel. Only the lately developed ION-1 detector is available, an instrument originally foreseen for LWR spent fuel. It uses fission chambers to measure gross neutron emission and ion chambers to measure gross gamma. These measurements can be correlated with the operator's declared burnup and time-since-discharge of the fuel, and at least allow consistency checks within a given population of items. However, considering the ION-1 application to spent-fuel of RRs which use HEU, it must be kept in mind that the burnup of HEU fuel involves different rates of buildup of transuranic isotopes than in LEU fuel, i.e., the correlation of neutron emission data to burnup is different!

Just for the sake of completeness the C/S measures should be mentioned briefly. Seals are seldom applied at RRs and CAs, eventually at CAs if sealing of parts of inventory in between PIVs is feasible, or as a temporary measure applied to certain rooms where for example a PIV lasts several days. The type of seal applied is either the well-known metal cap seal, or paper seals. Surveillance can hardly be used, since at large CAs as well as many small RRs too many movements occur of personnel and of equipment where no nuclear material is involved. Potential application at larger RRs (e.g., spent-fuel blocks) may turn out to be useful, and has been reconsidered in recent times.

Finally, since I will refer to IAEA's SIR (Safeguards Implementation Report) criteria elsewhere, here I only would like to mention that facilities handling (inventory or throughput) less than 1 SQ of nuclear material are not evaluated for the SIR regarding the attainment of inspection goals.

REFERENCE MATERIAL

IAEA Safeguards Equipment Handbook.

STR-118, "Safeguarding Research Reactors."

STR-147, Draft, "Safeguards System Analysis for NRX-type Research Reactors."

Draft SIR for 1984.

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Fig. 1. Statistical Data

2. Research Reactors with more than 25 MW(Th)
3. Large Scale Critical Assemblies
4. Undeclared Irradiation vs. Design Change

FIG. 1.

S T A T I S T I C A L D A T A

GROUP "A"	- -	Research Reactors with less than 25 MW(Th) Critical Assemblies with less than 1 SQ
GROUP "B"	.. -	Research Reactors with more than 25 MW(Th)
GROUP "C"	- -	Large Scale Critical Assemblies, 50 SQs

(a)	RRS & CAs	NPT	=	150
		non-NPT	=	30
(b)	"A" RRs & CAs	<25 MWth/<1 SQ	=	165
	"B" RRs	>25 MWth	=	12
	"C" Large Scale CAs		=	3
(c)	RRs & CAs	<1 SQ	=	128
	RRs & CAs	1 to 5 SQs	=	37
	RRs & CAs	>5 SQs	=	15

RESEARCH REACTORS with >25 MW(Th)

REACTOR	REACTOR POWER [MW(Th)]	NOMINAL CORE LOADING	FUEL ELEMENT	BURNUP	AVERAGE THERMAL NEUTRON FLUX (n/cm ² sec)	FUEL CONSUMPTION
FR-2	44*	2100 kg LEU	7 rods per element		1×10^{14}	6 elements per 10 ³ Mwd
BR-2	100	12 kg HEU	6 concentric tubes		9×10^{14}	
ESSOR	37	6 kg HEU/ 700 kg Nat. U	6 rods per element	45% in HEU 5-12000Mwd/t of Nat. U	7×10^{13}	
TRR	40	8 Mt Nat. U/ 4.5 kg HEU	cylindrical rods	800 Mwd/t	1.2×10^{14}	
WR-1	60	2800 kg HEU	18 rods per element	5000 Mwd/t	1.5×10^{14}	
NRU	125	21 kg HEU	12 rods per element		1×10^{14}	40-50 kg/y
NRX	30	21 kg HEU	7 rods per element		1×10^{14}	
JMTR	50	7 kg HEU	19 MTR plates per element	15-30%	3×10^{14}	
JOYO	50		MOX fuel pins			
HFR	45	11 kg HEU	19 MTR curved plates	20%	1×10^{14}	
HBWR	25	750 kg LEU	7 rods per element		3×10^{13}	
MARIA	30	7 kg HEU	6 circular tubes per element		3×10^{14}	
R-2	50	9 kg HEU	18 MTR curved plates per element	20-30%	2×10^{14}	

* shut down 1981.

| F I G. 3. |

L A R G E S C A L E C R I T I C A L A S S E M B L I E S

HORIZONTAL SPLIT BED	•FCA	JAPAN
	*ZPPR	USA
	*ZPR-6	USA
	*YPR-9	USA
VERTICAL INTEGRAL	*MASURCA	FRANCE
	•SNEAK	FRG
	•DCA	JAPAN
	•PROTEUS	SWITZERLAND
	*ZEBRA	UK
	*BFS	USSR

| F I G. 4. |

U N D E C L A R E D I R R A D. / D E S I G N C H A N G E

<u>DIQ NO.</u>	<u>TITLE</u>	<u>DESIGN CHANGE</u>
9.	Normal operating mode	May have to be changed to a greater number of operating days per year.
10.	Coolant	May require greater cooling capability
21.	Blanket, reflector	Insertion of target material in these areas
34.	Inventory	Increase in fresh-fuel consumption (also related to #37, refueling requirements)
39.	Irradiated fuel disposition	If this is diversion path for the irradiated targets, there would be increased quantities of spent-fuel to manage



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 21: MATERIAL CONTROL AND ACCOUNTING AT A CANDU REACTOR

A brief description of a CANDU reactor will begin the session. The material control and accounting system at CANDU facilities will be discussed. Equipment for fuel accounting, video and film surveillance, diversion monitoring, and containment and surveillance of irradiated fuel in storage will be described. Emphasis will be on actual experience in operating facilities. The impact of reactor safeguards on the plant operator will be discussed so that participants will be able to design a reactor safeguards system that meets national and international nonproliferation objectives with a minimum of interference to plant operators.

After the session, participants will be able to:

- Describe the major characteristics of a CANDU reactor and the special requirements of a material control and accounting system for a CANDU power station.
- Describe the special instrumentation available for use at CANDU facilities.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 21: MATERIAL CONTROL AND ACCOUNTING AT A CANDU REACTOR: THE INSTRUMENTED SAFEGUARDS SCHEME

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Atomic Energy of Canada Limited

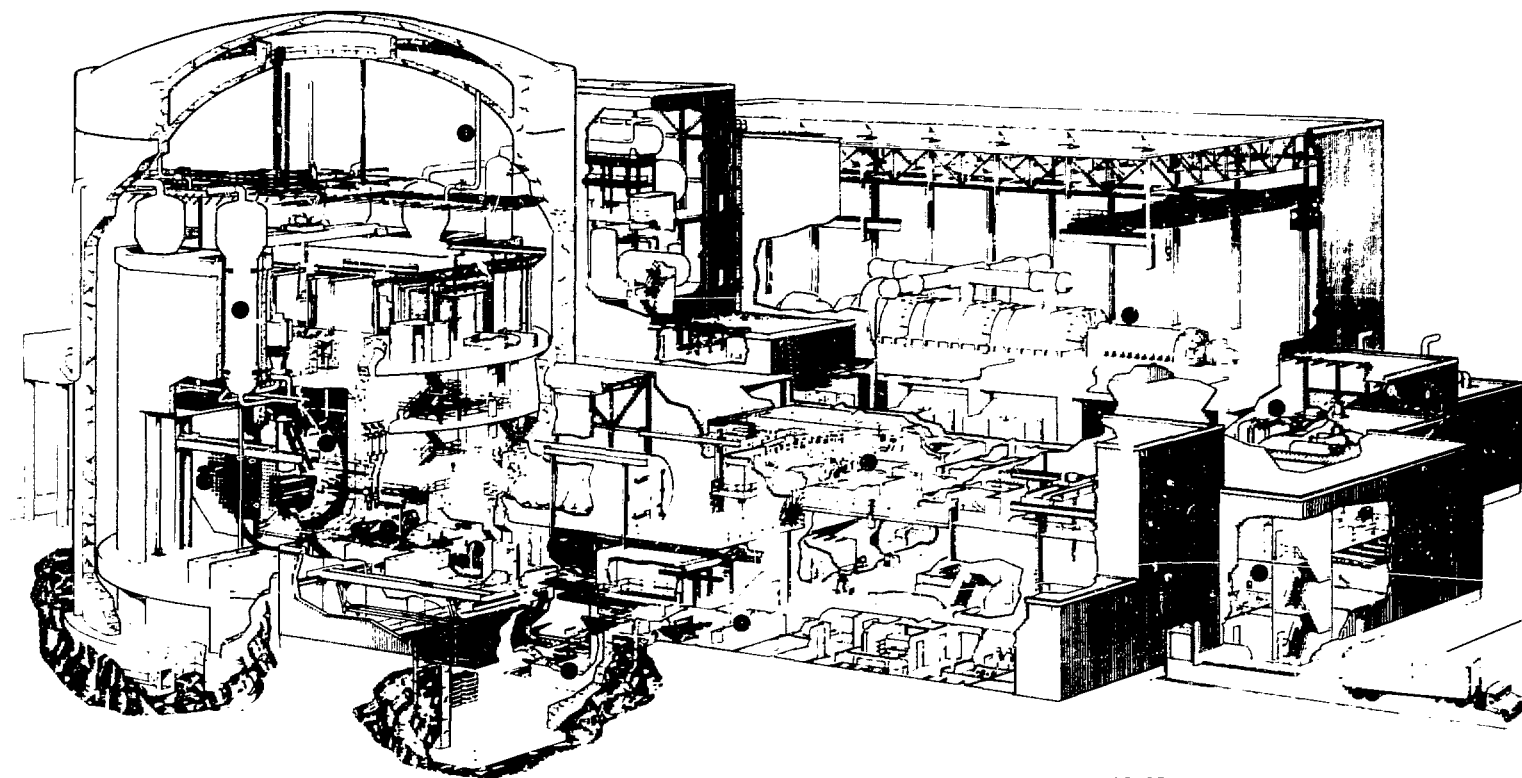
1. THE CANDU REACTOR

The CANDU is a natural uranium, heavy-water-cooled reactor with on-power refuelling (Fig. 1). It is in use in Canada, Argentina, India, Korea, and Pakistan. The next CANDUs to come on-line will be in Romania. In capacity factor, CANDUs frequently lead other reactor types by a considerable margin. This is partly attributed to on-power refuelling, a feature which makes the practical aspects of safeguards and accounting somewhat different. Canada and the IAEA have thus worked together to develop the equipment and methods for safeguarding on-load refuelled reactors using the internationally accepted fundamentals of accounting and containment/surveillance.

The CANDU reactor (Ref. 1) has no pressure vessel but is an array of horizontal pressure tubes, each 10 cm in diameter (Fig. 2). These contain the cylindrical fuel bundles (Fig. 3). The number and length of tubes determines the total power output, which is between 500 and 850 MW for current commercial models. The tubes are supported in a precise array by the vertical ends of a large horizontal cylindrical tank called the calandria. Each pressure tube is surrounded by a larger tube (calandria tube) which provides a thin gas annulus between hot heavy water in the pressure tube, and heavy water in the calandria which acts as an additional moderator and is at a temperature less than 100°C (Fig. 4). All control and safety devices are located in the unpressurized calandria and so there is no concern about having to drive these into a vessel during an accident.

The pressurized heavy water is circulated in two independent circuits (with one half the tubes in each) to carry heat to steam generators. Normal (light) water on the secondary side is converted to steam to drive the turbines (Fig. 5).

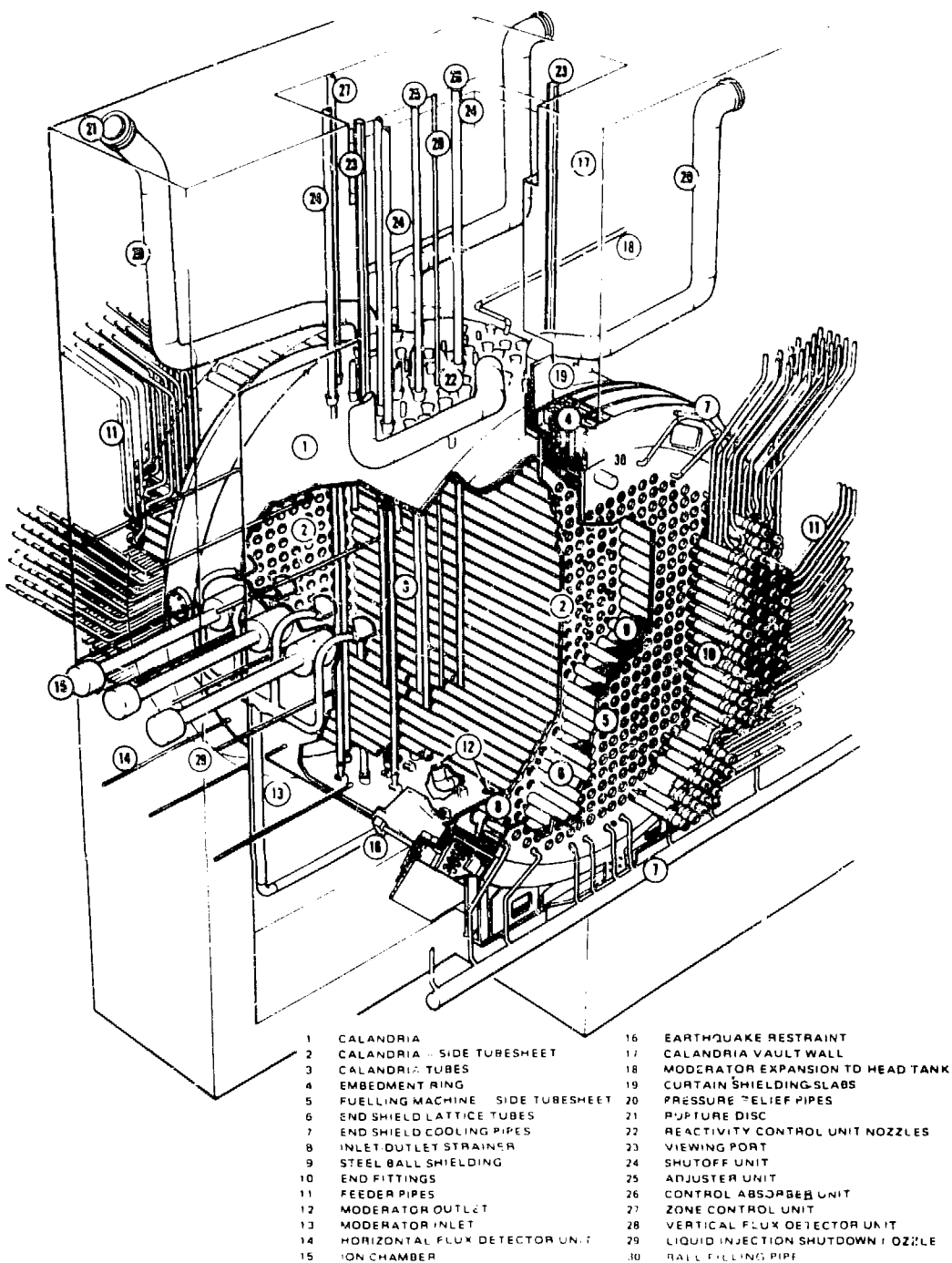
CANDU reactors are either built within single containment structures (like the 600-MW units in Quebec, New Brunswick, Argentina, and Korea) or with shared containments like those in Ontario. While relatively unimportant to the operation or to the concepts of safeguards, this has a significant impact on the detailed safeguards approach, and these differences will be mentioned in the next paper. The information in this paper, in general, refers to 600-MW reactors.



1 REACTOR	7 SPENT FUEL INSPECTION AND STORAGE
2 FUELLING MACHINE	8 MAINTENANCE AND INSPECTION
3 STEAM GENERATOR	9 CONTROL CENTRE
4 DOUSING SYSTEM	10 MECHANICAL WORKSHOP
5 NEW FUEL HANDLING	11 WATER TREATMENT
6 SPENT FUEL HANDLING	12 TURBINE AND GENERATOR

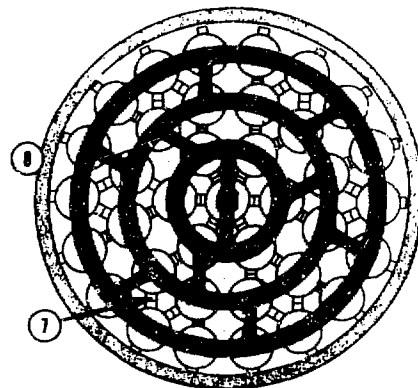
600 MW(e) NUCLEAR GENERATING STATION

FIG. 1.

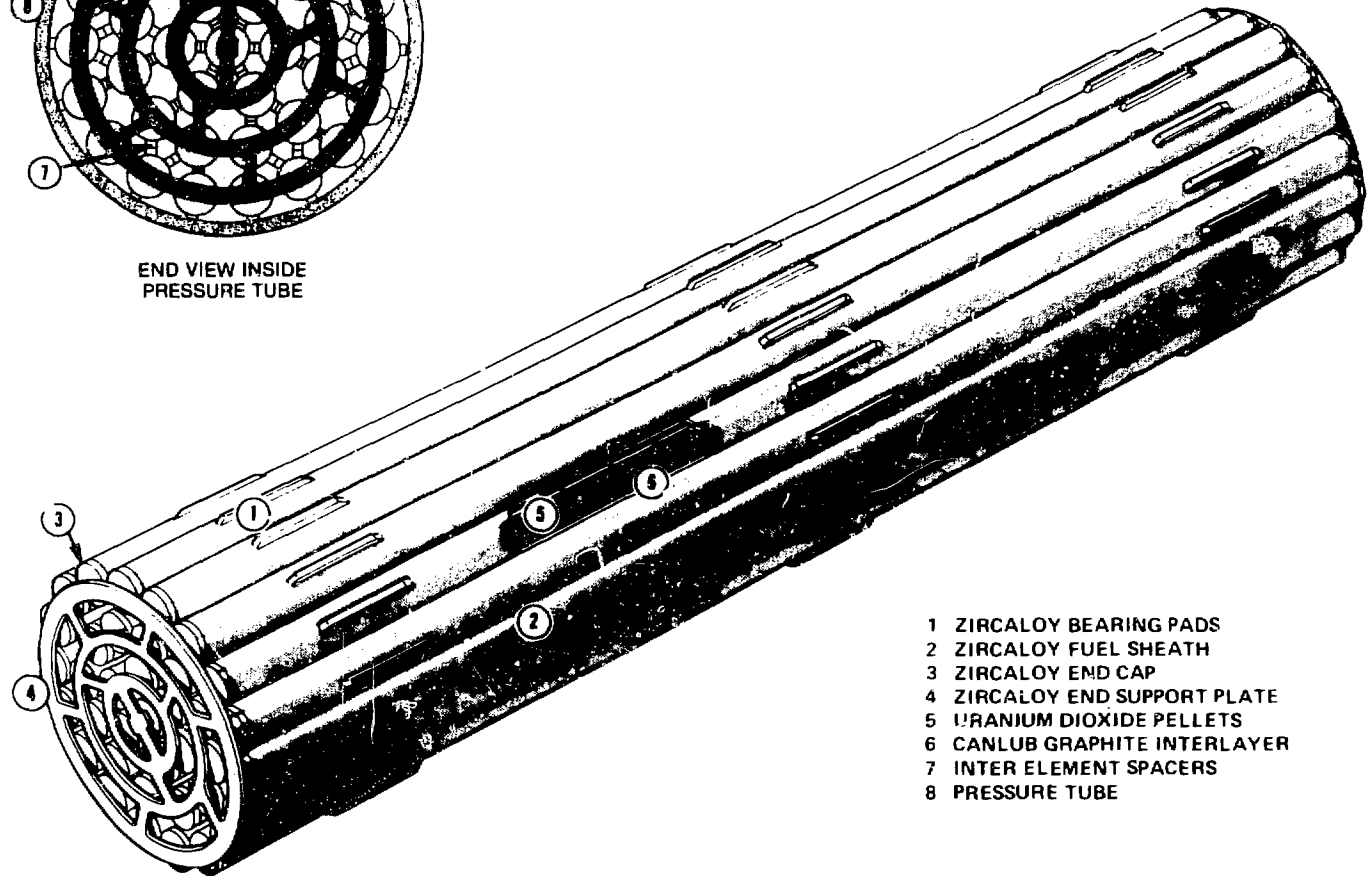


REACTOR ASSEMBLY

FIG. 2.



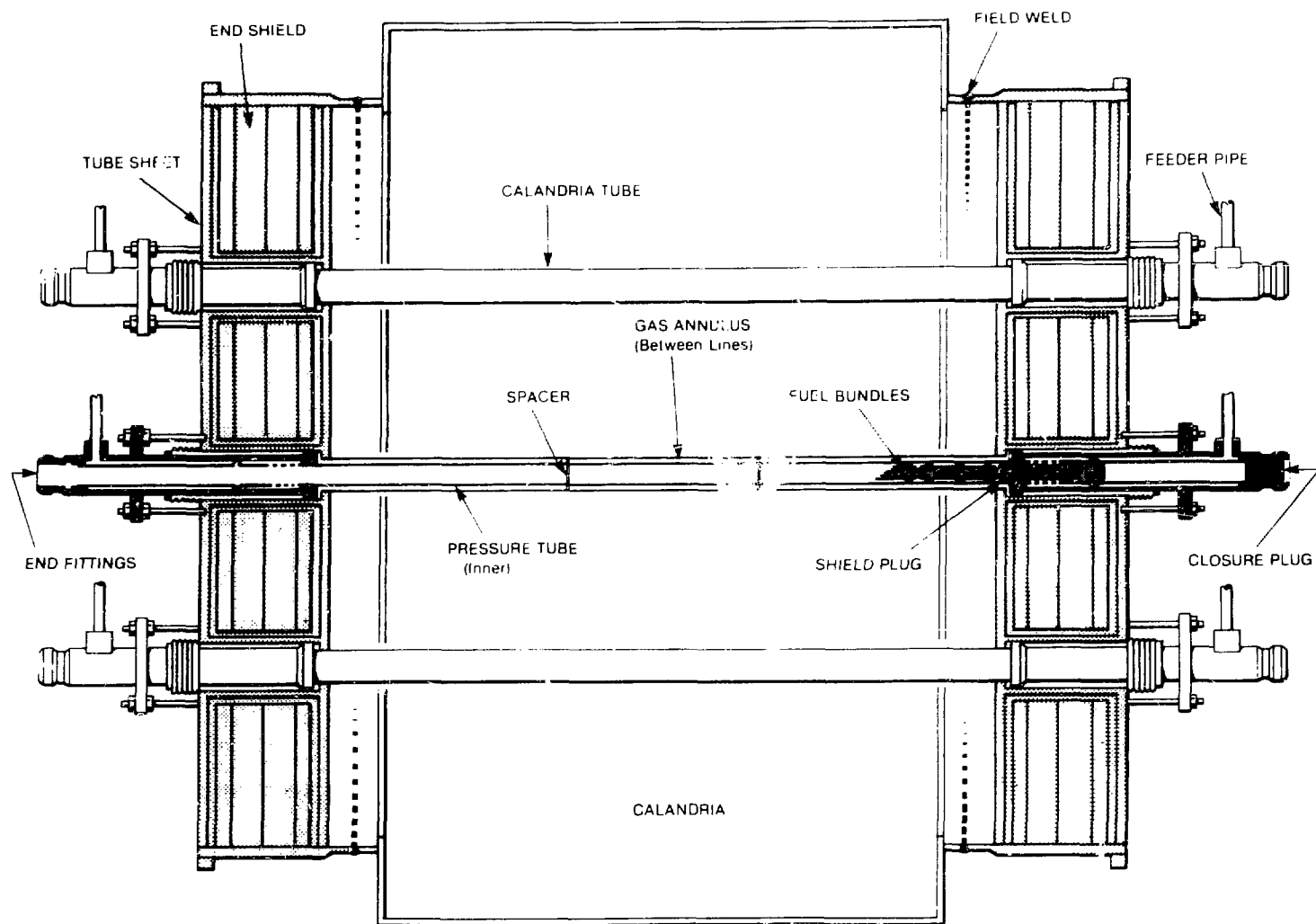
END VIEW INSIDE
PRESSURE TUBE



- 1 ZIRCALOY BEARING PADS
- 2 ZIRCALOY FUEL SHEATH
- 3 ZIRCALOY END CAP
- 4 ZIRCALOY END SUPPORT PLATE
- 5 URANIUM DIOXIDE PELLETS
- 6 CANLUB GRAPHITE INTERLAYER
- 7 INTER ELEMENT SPACERS
- 8 PRESSURE TUBE

37-ELEMENT FUEL BUNDLE

FIG. 3,



REACTOR CORE SCHEMATIC

FIG. 4.

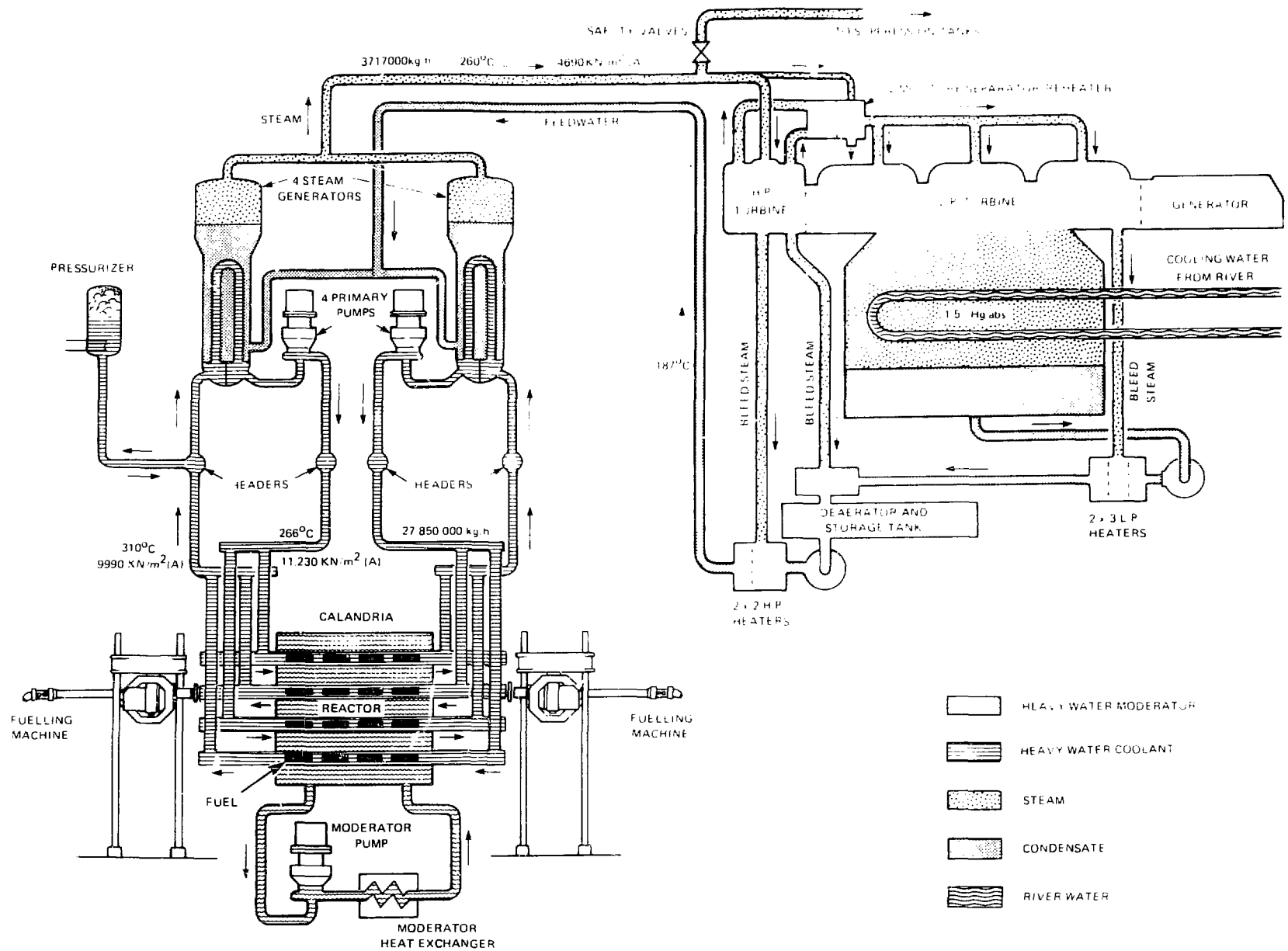


FIGURE 2.2-4 CANDU NUCLEAR POWER SYSTEM

FIG. 5.

2. FUEL

CANDU fuel is a ceramic of natural uranium oxide encapsulated in Zircaloy tubes. Thirty-seven such tubes form a single fuel bundle, 50 cm long and slightly less than 10 cm in diameter. The U^{235} in the fresh fuel is that of the natural ore (0.71%). Parenthetically, it can be noted that fresh CANDU fuel actually contains less U^{235} than the spent fuel from LWRs.

Normally, fuel resides in the core for approximately one year and emerges with a burnup of about 8000 Mwd/t. This corresponds to a plutonium concentration of approximately 4.0 g/kg of original uranium in the fuel, likewise somewhat lower than for LWRs. However, not all fuel is discharged with the maximum burnup. In order to maintain reactivity as poisons accumulate, and to keep the neutron flux distribution uniform, fuel is added, removed, and shuffled regularly. Thus, the discharged fuel has a wide range of plutonium concentrations.

3. FUEL HANDLING SYSTEM

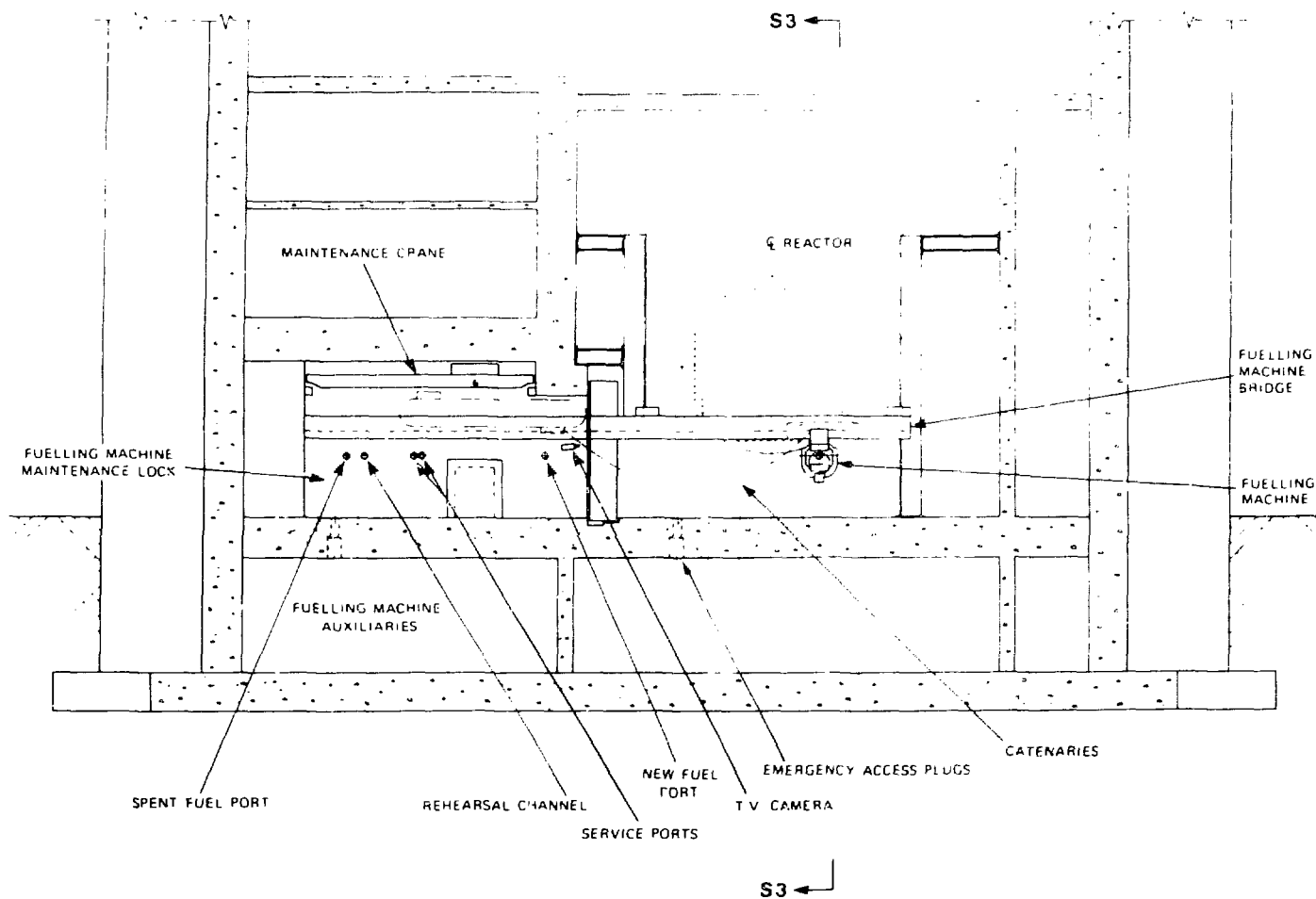
The fuel handling system is designed to:

- provide facilities for the storage and handling of new fuel;
- refuel the reactor remotely while it is operating at any level of power, and
- transfer the irradiated fuel remotely from the reactor to the storage bay.

The fuel changing operation is based on the combined use of two remotely controlled fuelling machines operating on each end of a fuel channel. Either machine can load or receive fuel. New fuel bundles, from one fuelling machine, are inserted into a fuel channel and the displaced irradiated fuel bundles are received into the second fuelling machine at the other end of the fuel channel. The direction of loading is in the direction of coolant flow in the fuel channel being fuelled and alternates from channel to channel.

The fuelling machines receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port (Fig. 6).

The entire operation is directed from the control room through a preprogrammed computerized system. The control system provides a printed log of all operations and permits manual intervention by the operator.



FUELLING MACHINE VAULT AND MAINTENANCE ROOM

FIG. 6.

New fuel is received in the new fuel storage room in the service building. This room accommodates six months' fuel inventory and can store temporarily all the fuel required for the initial fuel loading.

When required, the fuel bundles are transferred to the new fuel transfer room in the reactor building (Fig. 7). The fuel bundles are identified and loaded manually into the magazines of the two new fuel ports. Transfer of the new fuel bundles into the fuelling machines is remotely controlled.

Irradiated fuel received in the discharge port from the fuelling machine is transferred into an elevator which lowers it into a water-filled discharge bay. The irradiated fuel is then conveyed under water through a transfer canal into a reception bay, where it is loaded onto storage trays and passed into the storage bay (Fig. 8).

The discharge and transfer operations are remotely controlled by station staff. Operations in the storage bays are carried out under water using special tools aided by cranes and hoists. To limit the spread of contamination, defective fuel is inserted into cans under water before transfer to the defective fuel bay.

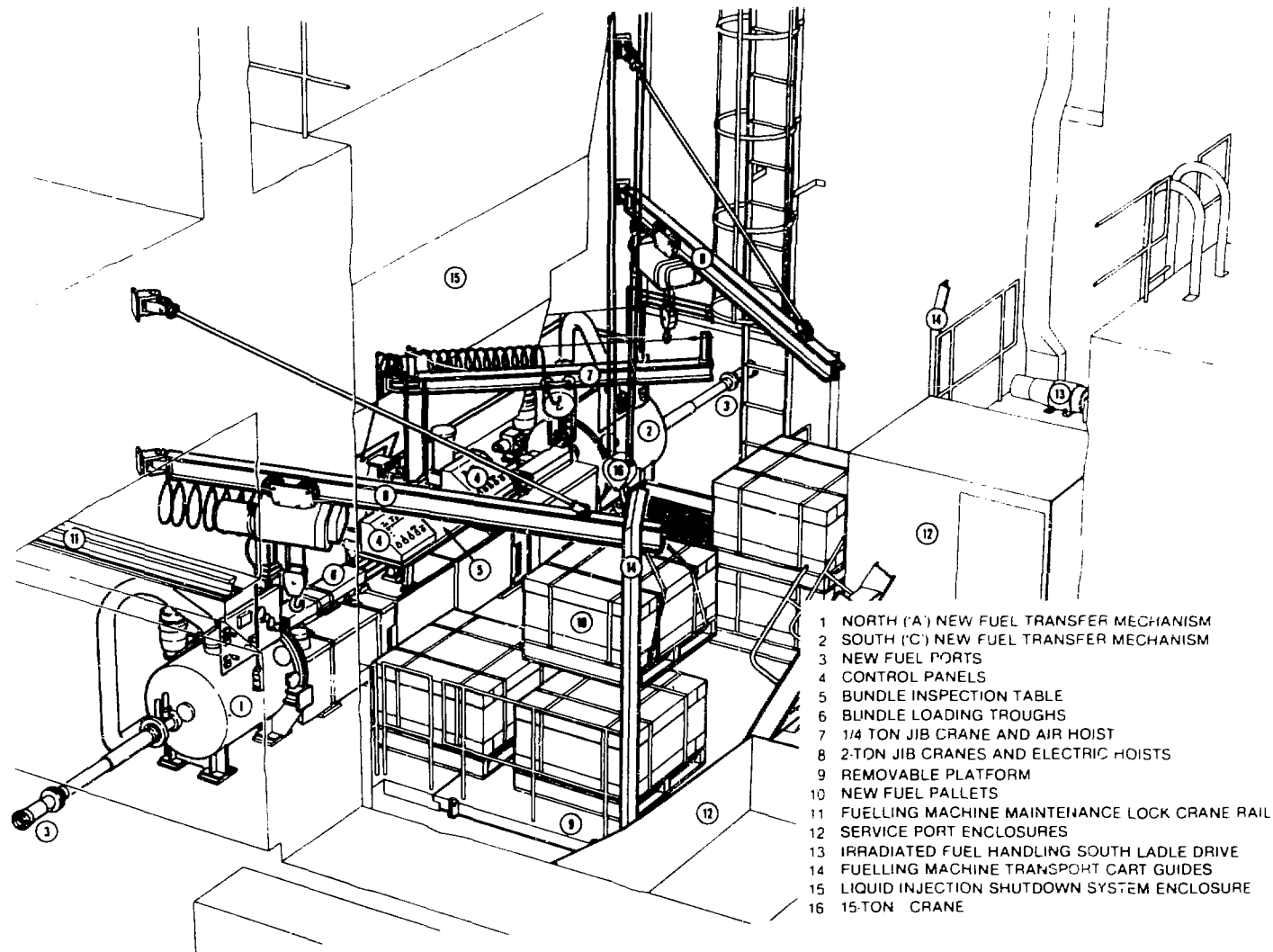
The storage area of the bays has sufficient capacity for 10 years' accumulation of irradiated fuel.

4. MATERIALS UNDER SAFEGUARDS

The materials of safeguards significance in the CANDU reactor are the natural uranium fuel and the plutonium in the irradiated fuel. Each bundle of natural uranium contains 21.3 kg of UO_2 , and the quantity of Pu in the irradiated fuel (which depends on burnup) is assumed to be 76 g. However, as this is contained in the fuel bundles at all times (except in the very rare event of fuel failure) the bundles are considered to be the unit item for accounting. Thus one can translate the IAEA values for significant quantities (75 kg of U^{235} in natural U or 8 kg of Pu) into 560 bundles of fresh fuel and 112 bundles of spent fuel respectively.

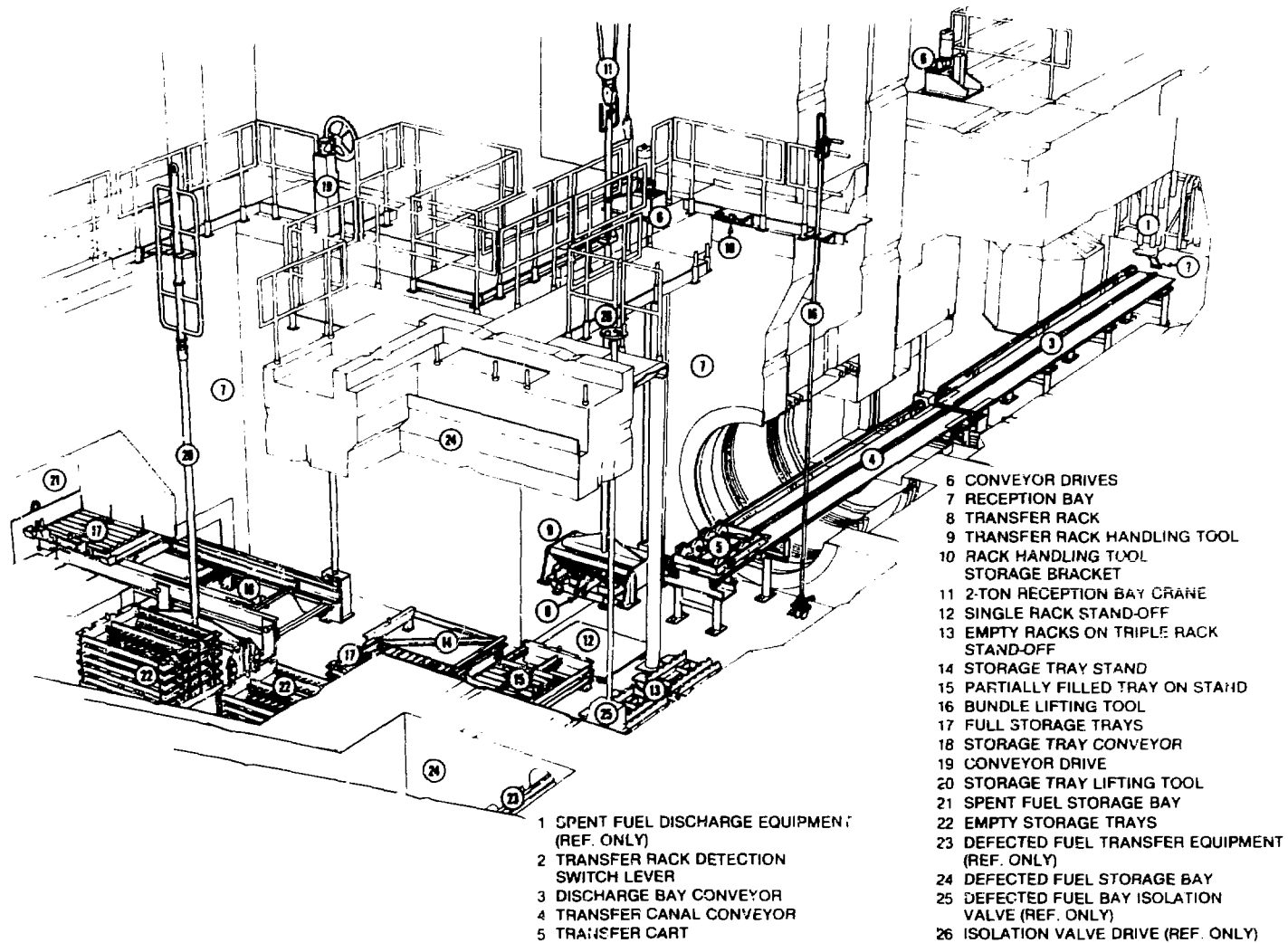
5. OPERATOR RECORDS

Each fuel bundle is stamped with an identification number in its end plate. Since the operator controls the movement of fuel to maintain flux uniformity and to maximize burnup, the location of each bundle is known to the operator at all times. These records are, in later reactors, computer generated and can be made available to IAEA inspectors and state systems of accounting. The system for independently verifying the operator records depends on a combination of measuring the fuel flow at certain key measurement points, and on containment and surveillance.



600 MW(e) NEW FUEL TRANSFER EQUIPMENT

FIG. 7.



600 MW(e) FUEL STORAGE BAY EQUIPMENT

FIG. 8.

6. IAEA SAFEGUARDS APPROACH

The safeguards approach for CANDU 600-MW(e) reactors employs television surveillance systems, photo-surveillance cameras, radiation yes/no monitors that record any high radiation fields, counters for spent-fuel bundles, an attribute verifier for spent-fuel, and spent-fuel sealing equipment (Ref. 2). The equipment location is shown in Fig. 9 and the functions of the equipment are described below.

The spent-fuel transfer surveillance area encompasses the route taken by the irradiated fuel during transfer from the reactor core to the bays. A television surveillance system is used to observe all movements of the fuelling machines in the fuelling machine rooms and fuelling machine maintenance locks. Two television cameras are on top of removable assemblies located in emergency access openings in the floor of the south and north fuelling machine rooms, and similar arrangements are provided for a second pair of cameras in the south and north fuelling machine maintenance locks. For maintenance or repair work, the cameras can be removed through the emergency access floor plugs from accessible rooms located in the basement of the reactor building.

The television control unit, located in the service building, consists of a set of time-lapse recorders (TLRs) one for each camera. There is also a television monitor for review. This arrangement permits free access to the unit by the inspectors and also enables them to have uninterrupted observation of the recorded events. This unit records pictures taken in the spent-fuel transfer area as well as pictures from cameras in the spent-fuel storage area.

Two film camera units provide the necessary coverage in the new fuel area. Both fuelling machine transfer doors are viewed. Another film camera is positioned above the exit from the spent-fuel discharge room to provide surveillance of a number of openings in the walls (such as ventilation vents and an access door).

Yes/no monitors, placed on each of the new fuel and service ports, ensure that radioactive irradiated fuel does not pass through these ports (quite an abnormal procedure) without being detected. These monitors are small thermoluminescent glass dosimeters secured with standard IAEA E-type seals.

Bundle counters near the top of each of the two discharge elevators monitor the movement of irradiated fuel between the reactor and the storage bay. Several Geiger tubes spaced to differentiate between single and double bundle transfers are positioned so as to be direction sensitive. Thus they are able to record the number of bundles passing the measurement point, the direction of the movement, and the time of the transfer. This fuel flow information is used to verify inventory records for the core and the spent-fuel bay. The electronics packages for the bundle counter are located above the spent-fuel transfer room, which is an area with good accessibility.

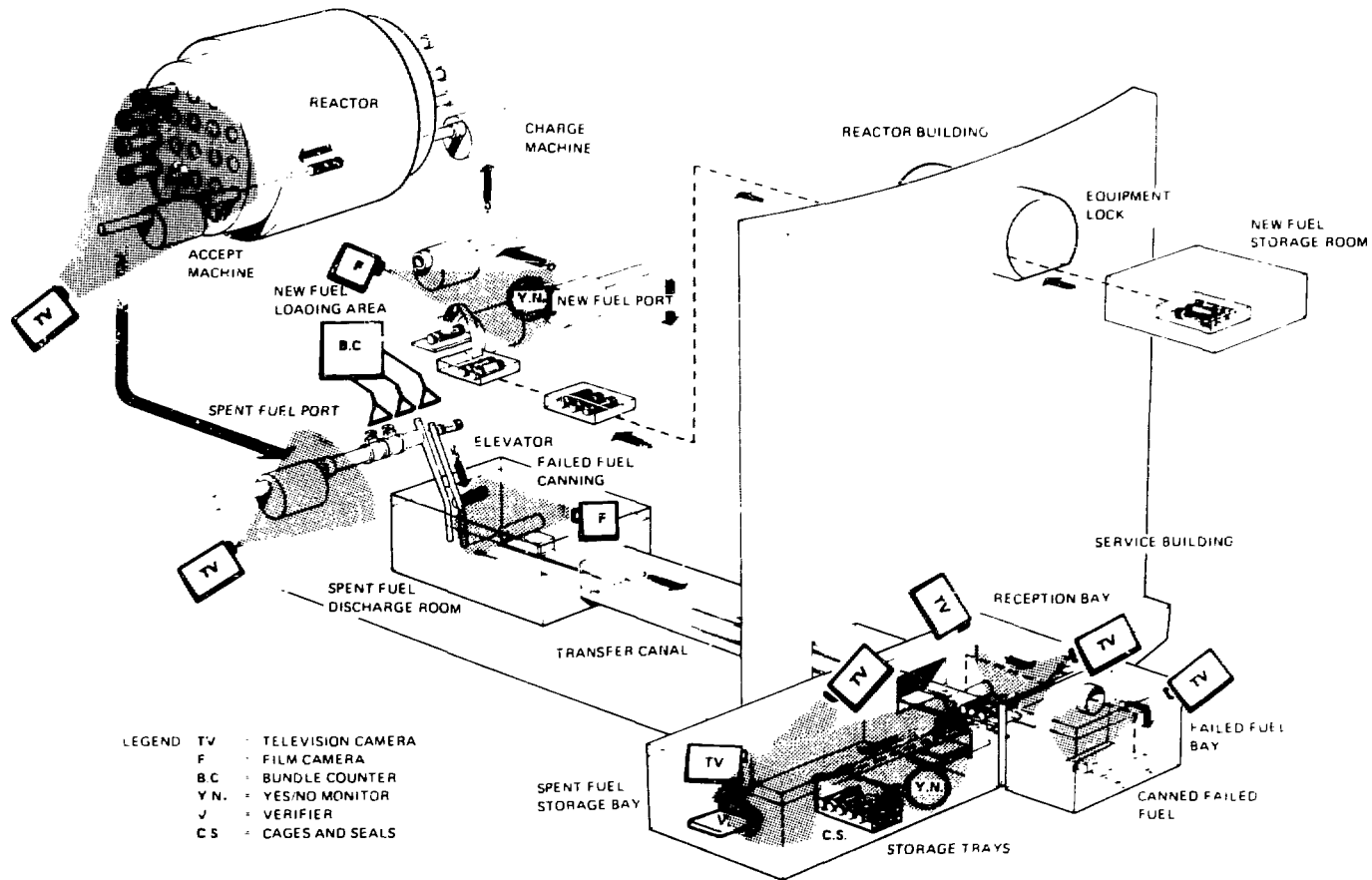


FIG. 9. LOCATION OF SAFEGUARDS INSTRUMENTATION.

Surveillance of the three compartments of the spent-fuel bay is provided by a system using five cameras. The complicated topography of the bay dictates the location of two closed-circuit television cameras (CCTV) in the receiving bay, two CCTV cameras in the main bay, and in some cases one film camera in the defective fuel bay. These cameras are positioned to observe the water surface, access doors, and other above-surface wall openings.

Radiation yes/no monitors applied to the water purification piping system would detect and record removal of irradiated nuclear material from the bay through this route.

A bundle verifier, which can verify that each fuel bundle in a loaded storage tray has been irradiated in the reactor, is available in the main storage bay. The verifier is a device, sensitive to ultraviolet (UV) light, that allows the inspector to view the Cerenkov glow produced by irradiated fuel bundles.

Tamper-indicating, spent fuel sealing units are used to enclose a stack of trays of spent fuel. Seals are used to seal a cover onto this stack of trays. This containment enables the inspector to seal nearly 1000* irradiated bundles into a single unit. This of course greatly simplifies inventory taking.

6.1 Film Camera

Standard IAEA twin Minolta cameras are used throughout the reactor where film is the surveillance recording medium.

6.2 Closed-Circuit Television (CCTV)

The CANDU 600-MW(e) safeguards approach calls for surveillance in the reactor vaults, the fuelling machine rooms, and the storage bay. The first two of these areas present significant difficulties because of high radiation fields and inaccessibility during reactor operation. Difficulties with high radiation fields are overcome by using radiation-resistant cameras containing the minimum number of electronic components. Wherever possible, electronic control functions are removed from the camera and placed in the central control station. Other requirements are a high capacity picture storage, on-the-spot review capability, high system reliability, and high tamper resistance.

*13 bundles x 4 trays x 19 high = 988.

To meet these requirements, a time lapse recorder (Panasonic model NV 8050) is connected to each camera. Experience with these units has shown that they have a very low failure rate.

In principle, however, a single failure may leave diversion paths uncovered and force the IAEA to take more time-consuming safeguards measures. Thus, a system where camera signals are fed to two (redundant) multiplexers and to two (redundant) TLRs has been designed. The pictures recorded are coded to correspond with the camera. Even though they are recorded in sequence and thus interleaved on the tape, a digital "unscrambler" puts the frames back into their correct sequence in the replay unit. A prototype is currently being reviewed in Vienna.

6.3 Bundle Counter

Bundle counting is accomplished by Geiger tubes located near the fuel flow path close to the point where the fuel bundles are discharged from the core (Ref. 3). Eight penetrations of the ceiling in the spent-fuel transfer room are provided to permit positioning of the Geiger tubes close to the spent-fuel ports. The necessary collimation and shielding were designed to comply with reactor containment specifications, operational safety requirements, and other criteria appropriate to nuclear power stations. Signals from these Geiger tubes are taken to a remote electronics package, located in an area which is accessible even when the reactor is at power. The electronics package includes logic functions which analyse the signals from the Geiger tube (i.e., duration of signals and order of appearance) and determine the number of bundles transferred, the direction of transfer, and whether the transfer is normal. This information is recorded along with the date and time of the transfer. Using this log, the IAEA inspector can independently verify the operator's fuelling records. A small "watch-dog" source of radiation is placed near each Geiger tube to enable the condition of the connecting cables and the Geiger tubes to be monitored continuously.

6.4 Cerenkov Verifier

The system views the Cerenkov glow emanating from irradiated fuel bundles. Bundles that do not have the characteristic glow will be identified as dummies or substitutes. The viewer is based on the Varo Corporation night-viewing devices, but has optics and photocathode optimized for the safeguards application. The system has been designed to be capable of operating under normal building lighting conditions by adding clear Lexan (TM) ultraviolet filters to the regular reactor bay lighting.

6.5 Containment-Sealing Equipment

As indicated, the sealing of irradiated fuel in-place in the storage bay is an important part of the safeguards approach. The stack of trays itself forms a containment shell and a rod, fixed to the bottom of the stack and reaching up to a cover placed on top of the stack, is used to lock the stack together to prevent the removal of fuel bundles (Fig. 10). A seal can be applied to assure the integrity of the containment shell.

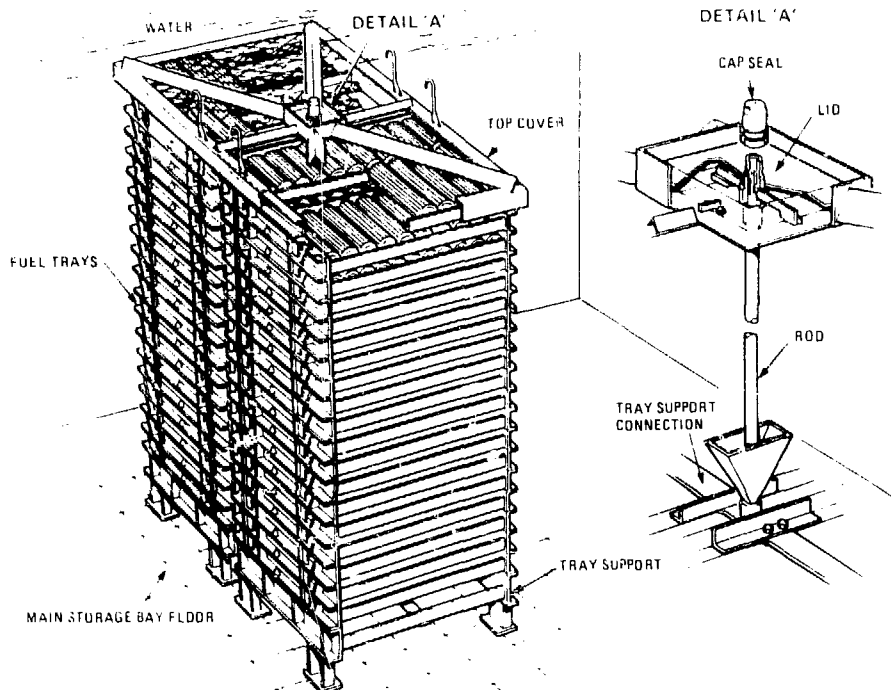


FIG. 10. SAFEGUARDS CONTAINMENT UNIT FOR
CANDU 600-MW STATIONS.

The seal is capable of being applied and verified underwater at depths of about four metres. (The top bundles in the stack are stored under this depth of water.) The seal developed for this task is a cap-and-stud type containing an ultrasonically readable element (Fig. 11 and Ref. 4).

The design for the CANDU ultrasonic cap seal uses a stainless steel wire coil as the identity element. This coil is incorporated in a cavity in the top of the seal. An important feature is that the coil is attached to the seal body such that it will be distorted if the seal is broken. Thus the identity and integrity features are linked. Since it is in water, it is directly accessible from the interrogating transducer. Signatures are recorded as time-amplitude patterns. A special readout unit provides a pulse train to excite the ultrasonic transducer and the circuit to process the reflected signals. The instrument stores reference signatures in plug-in bubble memories and the correlation between reference signals and inspection signals is calculated and displayed along with the signature itself. The measured signature is a function of ultrasonic frequency, transducer location, and timing window.

Tools to install and read the coil seal have been produced, and results obtained in a reactor test environment show that the seal has potential for Agency use.

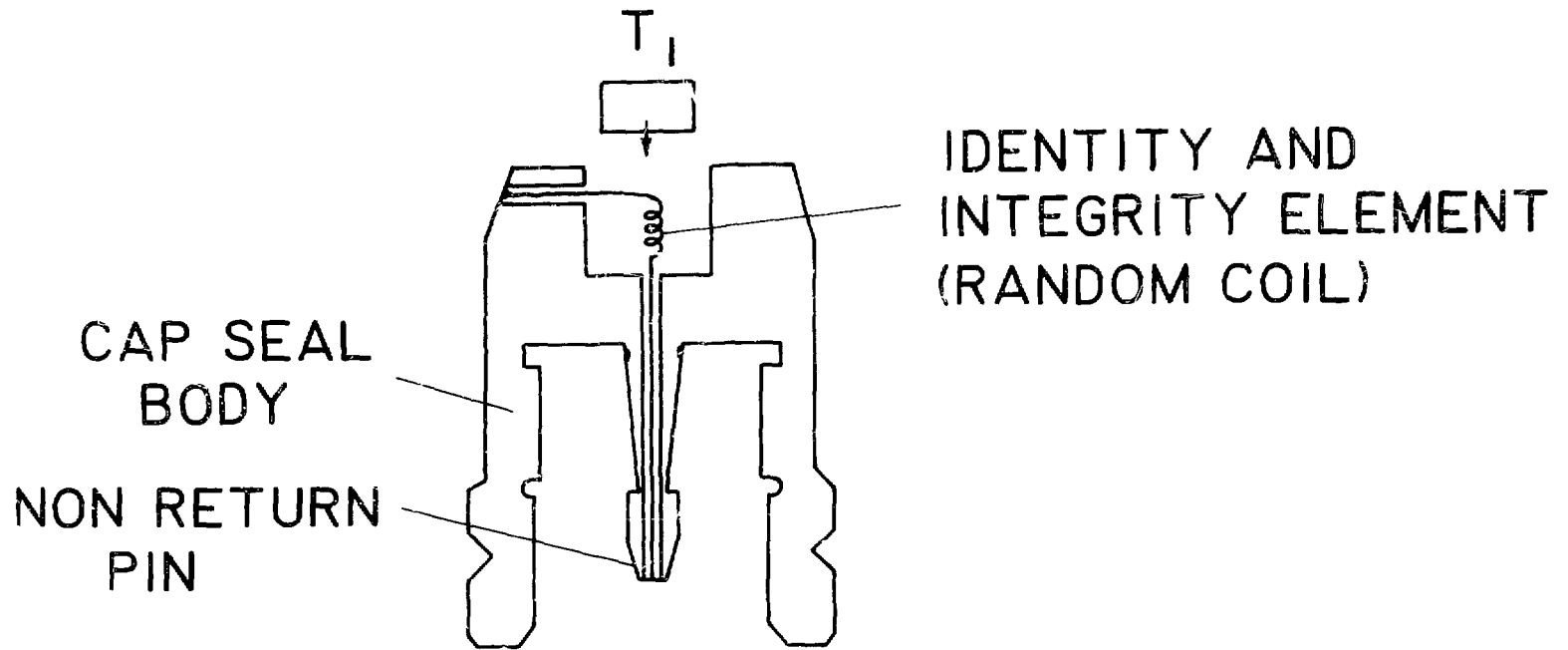
There is also an incentive to use seals which can be interrogated visually because optical inspection methods are used for verifying the integrity of the fuel storage container. Since inspectors may use telescopes or underwater video cameras, it may also be possible to use these instruments to verify the seal.

Progress on the optical seal has been in the areas of

- selecting an identity which cannot be duplicated,
- designing the seal so that its identity and integrity would be destroyed together and would be visible from the surface of the storage pool (Fig. 12), and
- specifying and procuring CCTV, telescopes, and photographic equipment suitable for laboratory evaluation of the seals.

The most promising identity developed is based on growing millimetre-sized crystals in pure aluminum using well-known metallurgical techniques. The crystals have random shapes, akin to a geo-political map.

In addition to the ultrasonic and optically verifiable seals, an underwater version of the IAEA E-type has been developed for the IAEA.



T_1 = TRANSDUCER FOR INTERROGATING
IDENTITY AND INTEGRITY

FIG. 11. CAP SEAL INCORPORATING A RANDOM COIL WHICH PROVIDES
A UNIQUE IDENTITY AND EVIDENCE OF ITS INTEGRITY

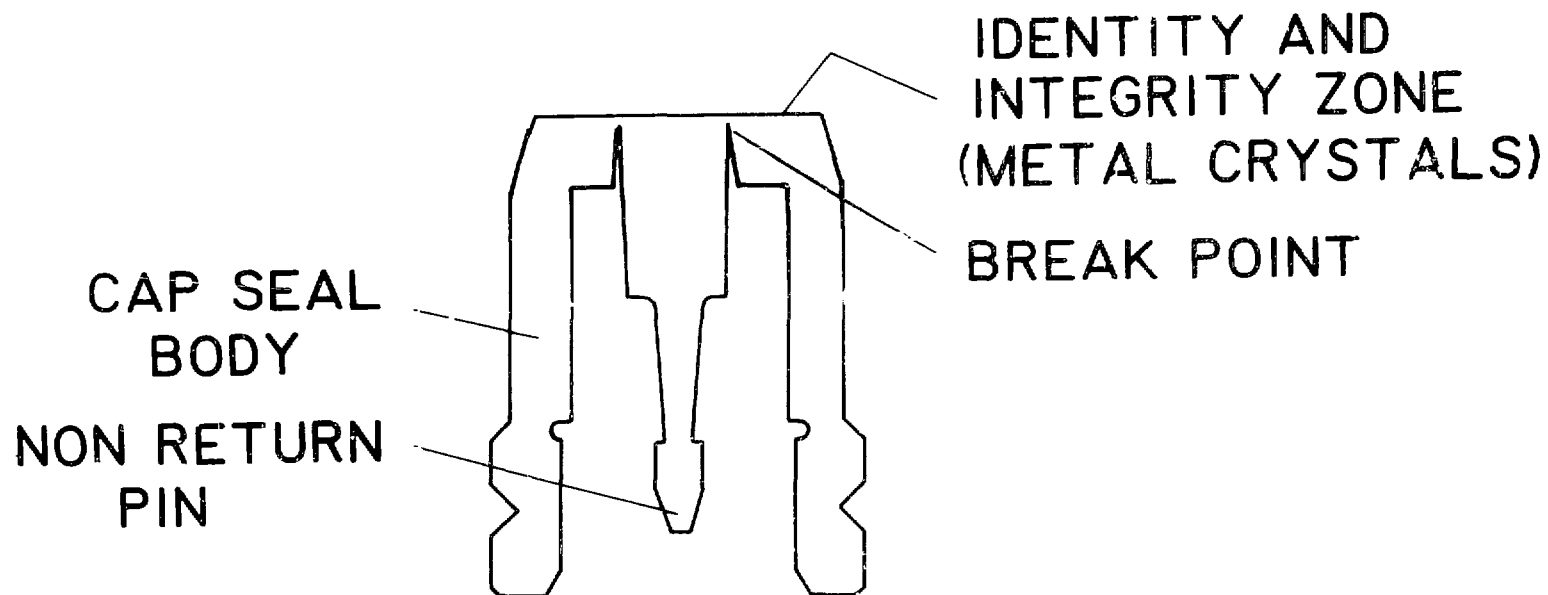


FIG. 12. CAP SEAL INCORPORATING RANDOM CRYSTAL GROWTH IN THE METAL SURFACE TO PROVIDE A UNIQUE IDENTITY AND EVIDENCE OF ITS INTEGRITY

7. IMPACT ON THE OPERATOR

In keeping with the IAEA principles of minimizing operator interference and maintaining independence of record, the installed equipment has little or no impact on the operator. IAEA cameras do require that certain lights be left on. The devices used to verify that Cerenkov radiation is being emitted from irradiated fuel mean that fuel bay lights need UV filters. Sealing and seal verification needs assistance from the operating staff who must handle the remote tooling. Most of the equipment, of course, needs a stable and reliable power source.

SUMMARY

While CANDU reactors differ from LWRs quite markedly in the way they operate, the principles of materials accounting and safeguards are equally applicable. Indeed, since CANDU fuel is not reprocessed, the relatively simple procedure of item accounting is sufficient for CANDUs. However, on-power refuelling means that automatic item counting is needed to independently confirm operator records. Surveillance and sealing techniques for spent fuel are needed for a practical system. The equipment developed has allowed the IAEA to apply safeguards at reasonable cost and with minimal interference to the utility operating the station.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 22: IAEA SAFEGUARDS EXPERIENCE AT CANDU REACTORS

The CANDU reactor facilities under IAEA safeguards will be described briefly with emphasis on the types and quantities of nuclear material found at these facilities. The nature and requirements of a typical facility attachment will be presented. Records and reporting requirements will be discussed. IAEA inspection procedures and measurement instrumentation will be described in detail, and the analysis of inspection data to verify the nuclear material inventory will be considered. The session will emphasize actual experience at CANDU reactors under IAEA safeguards.

After the session, participants will be able to:

- Describe the major activities during IAEA inspection of CANDU reactor facilities.
- Describe the records and reports prepared by the facility operator for the IAEA.
- Discuss the inspection procedures and measurement instrumentation used during IAEA inspections at CANDU reactors.
- Understand how the IAEA activities help to meet the safeguards goals of detection and deterrence.



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SESSION 22: IAEA SAFEGUARDS EXPERIENCE AT CANDU REACTORS

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I. INTRODUCTION

1. Inspection Activities and Facility Types

Reference will be given in this session to terms and concepts frequently used by safeguards people. They are also used in other sessions of this training course and they are defined and described in the references. I am, nevertheless, attaching an appendix to this session paper containing background information, i.e., definitions of subjects that are of particular relevance for my presentation of the Agency's safeguards experience, and also for my presentation in Session 31 of Agency Inspection Procedures.

As an IAEA safeguards inspector, I will refer both safeguards experiences and inspection procedures to the basic concept of inspection activities, which are those activities carried out by the Agency's inspectors both in the field and at the Vienna based headquarters. The inspection activities in the field are carried out at the nuclear facilities and are called inspections. The purposes and scopes of inspections are defined in the safeguards agreements. The inspection activities at the Vienna headquarters are the pre-inspection activities, i.e., planning, preparation of equipment and documents, and the post-inspection activities, i.e., analysis and evaluation of collected information and verification data and preparation of inspection reports and statements to be communicated to the states.

In the IAEA Safeguards Implementation Report (SIR) which each year is presented by the Secretariat to the Board of Governors (Ref. 1), the facilities under safeguards are grouped in so-called installation categories, as shown in Table 1 below.

Table 1: Installation Categories

		<u>Installation Category</u>
<u>Item Facilities</u>	- Power Reactors	A
	- Research Reactors and	
	Critical Assemblies (RRCA)	B
<u>Bulk-Handling Facilities</u>	- Conversion Plants	C
	- Fuel Fabrication Plants	D
	- Reprocessing Plants	E
	- Enrichment Plants	F
	- Separate Storage Facilities	G
	- Other Facilities	H

To the group "Power Reactors" belong both light-water reactors (LWRs) and on-load refueled reactors (OLRs) to which the CANDU reactors belong.

From a safeguards point of view all reactors, both power and research reactors, are considered as "item facilities", while all other facilities are called "bulk handling facilities".

In item facilities the nuclear material is contained in identifiable items, i.e., fuel assemblies or bundles, the integrity of which is usually preserved during their presence in the facility. The fuel items can be followed through a considerable part of the whole fuel cycle, from the fuel fabrication plant through the reactor facility, and to the reprocessing plant, or long-term storage. The nuclear material accountancy can, thus, be based on counting, identification, and nondestructive measurement of the fuel assemblies, i.e., item accounting. In bulk handling facilities, on the other hand, the nuclear material appears normally in both item and bulk or loose form, e.g., powder and pellets in a fuel fabrication plant.

2. The CANDU Reactor

Of 147 power reactors under Agency safeguards in non-nuclear weapons states in 1983, 28 were OLRs. Of these, 20 were CANDU reactors. There are three multi-unit CANDU stations, all of them situated in Canada, i.e., Pickering with 8 reactor units and Bruce A and Bruce B with 4 reactor units each. All the other CANDU facilities are single unit stations situated in Canada, the Republic of Korea, Argentina, Pakistan, and India. Furthermore, one single unit station is under construction in Romania.

From a safeguards point of view, the CANDU reactor is characterized by:

- the continuous fueling during reactor operation;
- the large number of fuel bundles in the inventory; and
- difficult access to the core for verification of the fuel inventory.

The continuous fueling means that the nuclear material, i.e., the fuel bundles, are moved within the material balance area (MBA) from the Fresh-Fuel Storage, through the Reactor Core, and into the Spent-Fuel Storage Bay, or Bays in the case of multi-unit stations. One fuel bundle weighs about 20 kg and is about 0.5 m long. A significant quantity (SQ) is made up of about 550 fresh-fuel bundles or about 110 spent-fuel bundles.

Typical flows and inventories of bundles are shown in Table 2.

Table 2: Typical Fuel Bundle Flow and Inventories in CANDU Reactors

Facility	Bundle Flow		Bundle Inventory		
	per year	per day	Fresh-Fuel Storage	Reactor Core	Spent-Fuel Storage
CANDU 600 single unit 600 MW(E) (SQ)	6,500	18	7,000 (12)	4,560 (40)	50,000 (480)
Multi-unit station 4 units 3,400 MW(E) (SQ)	23,000	70	17,000 (30)	25,000 (240)	28,000 ^{1/} (250)

^{1/} In the primary storage bay. In the intermediate/long-term storage bay the capacity is 280,000 bundles or 2500 SQs.

The CANDU design and the containment and surveillance equipment incorporated in the safeguards scheme is described in Session 21.

II. SAFEGUARDS APPROACH

1. Inspection Goals

The main inspection goal of Agency safeguards at CANDU reactor facilities is to be able to detect the diversion of one SQ of nuclear material in accordance with the following timeliness criteria:

- within 3 months for core and spent fuel;
- within 1 year for fresh fuel.

Minimum uniform criteria for the evaluation of inspection goal attainment are established within the Agency and serve the purpose of evaluating goal attainment in connection with the preparation of the annual implementation report (Ref. 1). They are called SIR-criteria and are used by the inspectors as a check list for planning purposes. The SIR-criteria list those inspection/verification activities that must have been performed in order to meet both the quantity and timeliness component of the inspection goal. The activities so listed refer--in the case of item facilities--to accountancy records and reports, confirmation of shipments, verification of fresh fuel, core fuel and spent fuel, unreported irradiation, material borrowing, and follow-up of anomalies.

2. Diversion Path Analysis

When designing a safeguards approach for a facility in general the possible diversion strategies, or diversion paths, will have to be considered. The safeguards measures through inspections and the use of containment and surveillance (C/S) devices are designed to prevent diversion along those paths. The so-called diversion path analysis is an activity that more or less constantly occupies the inspector's mind, and he carries it out in close cooperation with system and instrument experts for different types of facilities in order to establish an effective safeguards.

The safeguards measures that are implemented at a CANDU facility consider the possibility of diversion of fresh fuel from the Fresh Fuel Storage Area, and of spent-fuel bundles from the Storage Bays. Diversion from the reactor core is also considered, assuming diversion strategies other than those requiring major structural modifications of the reactor containment. When assigning priorities for implementation, it is assumed that diversion of spent fuel is more attractive to the potential diverter and, hence, more probable than diversion of fresh fuel.

The diversion path analysis identifies the method and route contemplated, and the time for the diversion of a relevant number of bundles is estimated. The time for diversion is the total time to initiate and complete the diversion over the route in question within the containment or surveillance boundary.

The complexity of the diversion path is judged using the following scale:

- A: Relatively easy, i.e., no significant plant modification is required.
- B: More difficult; minor modifications required;
- C: Very difficult or not considered feasible, i.e., significant plant modifications are required.

Some examples of diversion possibilities at a CANDU 600 facility are shown in Table 3.

Table 3: CANDU 600 Diversion Paths

Path No.	Station Area	Diversion Path Description	Time for diversion (hours)	Bundle yield per attempt	Remarks/ Complexity
1	Fresh-Fuel Storage	Removal of one box out of the store using a fork truck.	0.5	36	A
2	Reactor vault	Removal of bundles directly from the reactor pressure tubes into a flask moved out via different routes. Special equipment needed.	2 days	max. 12	Shutdown necessary C
3	Fueling Machine	Discharge of bundles from fueling machines into a flask moved into the Fresh Fuel Loading Room and out of the Main Air Lock.	1.5	2	A
4	Fueling Machine Area	As 3, but flask moved up along either of the two front stairs and out the Main Air Lock.	2	2	B
5	Reception Bay	Removal of bundles from Reception Bay, in flask crossing water surface and out via normal exit route.	8	120	A
6	Main Storage Bay	As 5, but removal from Main Storage Bay	2	24	A

Bundles could be diverted from the reactor core by using the fueling machines and having the removed bundles inserted into a flask. The flask could be of a minimum size to carry one bundle, in which case it would have to be transferred out of the containment about 110 times. Or it can take up to 12 bundles, which is considered to be a maximum practical quantity due to heat dissipation considerations. The limitation of flask size is also set by the access airlocks and the way out from the reactor vault. The bundles could be inserted into the flask remotely during reactor operation, while the removal of the flask from the vault would be performed during a reactor shutdown.

Diversion of bundles from the storage bay could be done using any type of shielded flask. The bundles could be loaded into the flask with existing handling equipment and the flask lifted up and onto a truck, also by existing loading facilities.

It is justified to ask: How well can the inspector make a diversion path analysis and what possibilities does the potential diverter have to cheat him? The difficulties facing the inspector are evident; the potential diverter is not the facility operator alone but a whole state with economical and personnel resources, as well as "power and control" capabilities. One should, however, remember that the purpose of the IAEA safeguards is not only to detect diversion but also to deter diversion if contemplated. The more diversion paths and strategies that are considered when designing a safeguards approach, the more difficult for the diverter to overcome it. The diversion path analysis is made by IAEA staff members who together possess knowledge and experience not only from safeguards inspection activities but from designing and operating nuclear facilities, and experts and consultants from different specialist disciplines are involved.

3. Safeguards Measures

As for all types of reactors, nuclear material accountancy (i.e., examination of operator's records and state reports) alone does not provide sufficient assurance that diversion can be detected by the safeguards measures implemented in a CANDU facility. Containment and surveillance measures must be used as necessary complements. The major difference between CANDU and light-water reactors is that more C/S measures are used in the CANDU safeguards scheme than for LWRs.

For accountancy purposes the CANDU facility--be it a single or multi-unit station--consists of one MBA; only where spent fuel is stored on site in a separate, intermediate/long term storage bay does the Agency contemplate a separate MBA for that bay. Within the MBA the nuclear material inventory and flow is referred to KMPs which are defined in the Facility Attachment. In a single unit CANDU facility there are 4 inventory KMPs:

KMP A	Fresh-fuel storage and handling areas
KMP B	Reactor core
KMP C	Spent-fuel storage and handling areas
KMP D	Other locations.

For the material flow there are 4 KMPs:

KMP 1	Receipt/shipment of fresh fuel
KMP 2	Nuclear production for bundles discharged from the reactor
KMP 3	Nuclear loss
KMP 4	Shipment of spent fuel

Of special importance in a CANDU reactor are, of course, the refueling records which are the basis for verification of the bundle flow. Other facility records are listed in paragraph III.1.

The bundle flow is directly verified by measurements at the output, or discharge, from the reactor core by bundle counters. They measure gross gamma radiation and can be adjusted to have a certain gross gamma count rate window. They can determine in which direction a bundle is being moved, i.e., they can detect a potential diversion of spent-fuel bundles from the bay and back into the containment.

In multi-unit CANDU stations the bundle flow is also measured when the bundles are transferred from the primary to the intermediate/long term storage bay. This is done by inspectors who count the bundles visually and also use NDA measurements.

Surveillance systems, i.e., film or TV cameras, are used to cover

- a) the reactor core, either by TV cameras installed in the reactor containment (CANDU 600), or outside the containment, overlooking accesses and penetrations, and,
- b) the spent fuel in transfer and storage bays.

For the detection of clandestine movements of spent-fuel bundles yes/no monitors are installed in certain penetrations in the reactor containment. Other penetrations may also be covered with seals.

As an extra, back-up measure, the spent fuel that is stacked on trays in the storage bay is put under seals (ultrasonic seals).

For the verification of irradiated fuel by gross gamma (attribute) check a so-called night viewing device (NVD) or Cerenkov radiation glow instrument is used. For the verification of interbay fuel transfers at the multi-unit stations gamma spectrometers are used (i.e., silica with both an Na/I and a germanium detector). For the verification of fresh-fuel bundles in opened boxes hand-held gross gamma instruments are used.

The instrumented safeguards scheme is shown in Figure 1 for a single-unit CANDU 600 station, and in Fig. 2 for a multi-unit station.

III. INSPECTION ACTIVITIES

With a fully implemented safeguards instrumented scheme, the inspection activities by the IAEA inspectors comprise the following items at a CANDU 600 facility:

1. Auditing of records and reports

- a) Examination of accounting records, i.e., the General Ledger, is examined versus
 - Receipt and shipment records, and
 - Fueling records.

Other records that are examined are:

- Fuel bundle inventory records;
 - Records on the integrated thermal power production; and
 - Reactor power records, showing reactor shutdown periods.
- b) Comparison between facility records and state Inventory Change Reports.
- c) Updating of book inventory for each material category, i.e., natural uranium, depleted uranium and plutonium, based on transfer forms.

2. Servicing of Containment/Surveillance Equipment

- a) The CCTV system recorded tapes are reviewed and replaced. Review records are filled in by the inspector over the observations from the surveillance areas, i.e., the two cameras viewing both reactor faces; the two cameras viewing the loading ports in the fueling machine area; the two cameras viewing the reception bay, and the two cameras viewing the storage bay.
- b) The film camera systems are serviced and films are replaced and taken off site for processing and later review. There is one twin camera system overlooking the fresh-fuel loading area outside the reactor vault, and one twin camera system viewing the main access area, both inside the Reactor Building.
- c) Some of the yes/no monitors located in bundle penetrations in the fresh-fuel loading area are replaced and brought to Vienna for checking.
- d) Some of the metallic seals on emergency plugs under the floor of the reactor vault are replaced and brought to Vienna for checking.

3. Verification of Bundle Flow

The two bundle counter electronic units located in the main access area inside the containment are serviced and the number of discharged bundles is recorded. The number is compared with the operator's fueling record.

4. Verification and Sealing of Spent Fuel

The unsealed spent-fuel bundles accumulated and stacked in the Storage Bay during the period from the previous inspection are counted visually and when there are two full stacks, i.e., 912 bundles (about 8 SQ), two ultrasonic seals will be attached. Provided that the surveillance systems and bundle counters had operated successfully during the period from the previous inspection and no unreported activities were observed, no further verification of the bundles is needed before sealing. If surveillance or bundle counters have failed, the bundles will have to be counted and verified by using the NVD instrument on randomly selected trays.

Already attached ultrasonic seals on stacked spent fuel are checked randomly.

Inspection activities 1, 2, 3 and 4 are carried out at each monthly routine inspection. At the annual physical inventory verification (PIV) the following inspection activities will be performed in addition:

5. Verification of Fresh-Fuel Bundles in the Fresh-Fuel Storage Room and in the Loading Area. Boxes are opened randomly and bundles identified, counted, and checked by hand-held gross gamma instruments.

At the PIV inspection the General Ledger is "closed" and the book ending inventory reconciled with the verified physical inventory.

IV. EXPERIENCE

At the four CANDU 600 facilities that are now in operation, i.e., two in Canada, one in the Republic of Korea, and one in Argentina, the Agency has implemented complete safeguards instrumented schemes. The safeguards equipment at each of these facilities consists of:

- a) TV and film camera surveillance systems covering both the reactor containment and the spent-fuel storage bay areas;
- b) Discharge bundle counters for the verification on a continuous basis of the flow of spent-fuel bundles from the reactor core and into the bay;
- c) Metallic seals and yes/no monitors placed on certain penetrations inside the reactor containment;
- d) During 1985 a back-up sealing system, in the form of underwater ultrasonic seals, will be installed on the storage stacks in the spent-fuel storage bay;
- e) Night viewing devices (NVD) are used for the attribute verification of spent fuel in the storage bay by observing the Cerenkov glow.

The surveillance systems and the bundle counters are serviced and the recorded data are reviewed at monthly routine inspections. The inventory of fresh fuel is verified by random opening of fuel boxes and counting of individual bundles and by attribute NDA once per year. The spent-fuel inventory is verified continuously over the year by the bundle counters and is maintained under surveillance (and later seals).

The inspection activities, including the examination of accounting and operating records, make it possible for the Agency to draw conclusions with respect to the inventories of declared nuclear material, and also to the detection of clandestine activities in the form of unreported production of plutonium.

At the three multi-unit stations in Canada surveillance systems have been installed and are in successful operation since the first inspection activities commenced, while the installation of discharge bundle counters

is not yet completed. Counters are installed at Bruce B and will be installed at Bruce A during 1985, and at Pickering during 1986. The rather long delay of installing the bundle counters--as well as underwater ultrasonic sealing systems--has been due to limited manpower and financial resources and also to the complexity of developing and introducing new safeguards equipment techniques. Long lead times, including time for long-term testing, is needed before reliable safeguards tools can be made available for routine use by the Agency inspectors.

The very particular feature of the CANDU reactors, i.e., the continuous material flow through the plant in an on-load refueling scheme, presents a challenge to the inspector with respect to his task to independently verify the nuclear material inventory and to detect diversion and unreported plutonium production. There are two principle ways for adopting a safeguards approach for these facilities. One is the implementation of an instrumented scheme as described above. The other is focused on the use of residential inspectors having almost unlimited, unannounced, and unrestricted access to the facilities.

Mainly on the basis of cost comparisons, the Agency has chosen the instrumented approach in establishing a complete safeguards scheme for the CANDU reactors. For the multi-unit stations in Canada, where the installation of bundle counters is not yet complete, the attainment of inspection goals has also been promoted by having inspectors frequently present at the reactor sites for the verification of the fuel bundle flow. This has been facilitated by the establishment of a field office in Toronto in September 1980.

The relatively large inspection effort that is spent for inspecting the multi-unit CANDU stations, as reported by the Agency in the SIR, is not directly comparable with the inspection effort used for light-water reactors. The interbay fuel transfers, that are verified 100% by the inspectors at Bruce and Pickering, correspond to a material flow that in other fuel cycles involves two facilities, i.e., both a reactor and a long-term storage facility. In the case of Bruce and Pickering this material flow takes place within one and the same facility and the inspection man-days are referred only to the reactor facility.

The importance of maintaining a successful surveillance of the spent-fuel storage bay areas in the CANDU reactor stations is referred to the difficulty of reverifying the spent-fuel inventory, should the surveillance be lost. The fuel bundles are laid on trays with 24 bundles on each tray. The trays are stacked up to a height of about 20 trays. The stacks are standing so tight that the trays cannot be seen and counted other than in the outer stacks. For the counting of the trays and the application of the NVD measurement of the bundles, the stacks will have to be disassembled and the trays moved. This is, of course, a time consuming and delicate handling operation for the facility operator. In order to limit the number of tray movements, the ultrasonic sealing system has been developed as a back-up measure to the surveillance system.

For the annual verification of the fresh-fuel bundle inventory the inspectors have to open boxes with bundles on a random basis, using a statistical sampling plan. This is a procedure that creates a disturbance

of the fuel-handling routines at the facility. Efforts have, therefore, been initiated to have the bundle box design modified in such a way that the bundles can be counted in an unopened box by using an NDA instrument. The idea is to place cardboard tubes alongside the bundle rows in the box and to insert an NDA detector in the tube, thereby counting the individual bundles indirectly by a gross gamma attribute verification.

The surveillance systems, both the TV and film camera systems, are subject to a continuous improvement effort. The development result will be of benefit to the surveillance system application in all types of nuclear facilities under the Agency safeguards. The TV system that was first installed in the CANDU 600 stations has been modified to meet performance criteria compatible with the well established Minolta film camera system which has been used by the Agency for many years. The hardware modifications of the TV system are being implemented at the four CANDU 600 facilities during 1985.

The Minolta film cameras will most probably be replaced successively by a high film capacity Elmo camera which has been modified and tested to meet with the Agency's specific requirements. It should, thereby, be possible to reduce the inspection frequency at the CANDU facilities from monthly to once every two months.

REFERENCE

1. The Safeguards Implementation Report for 1983, GOV/2160, 30 April 1984.

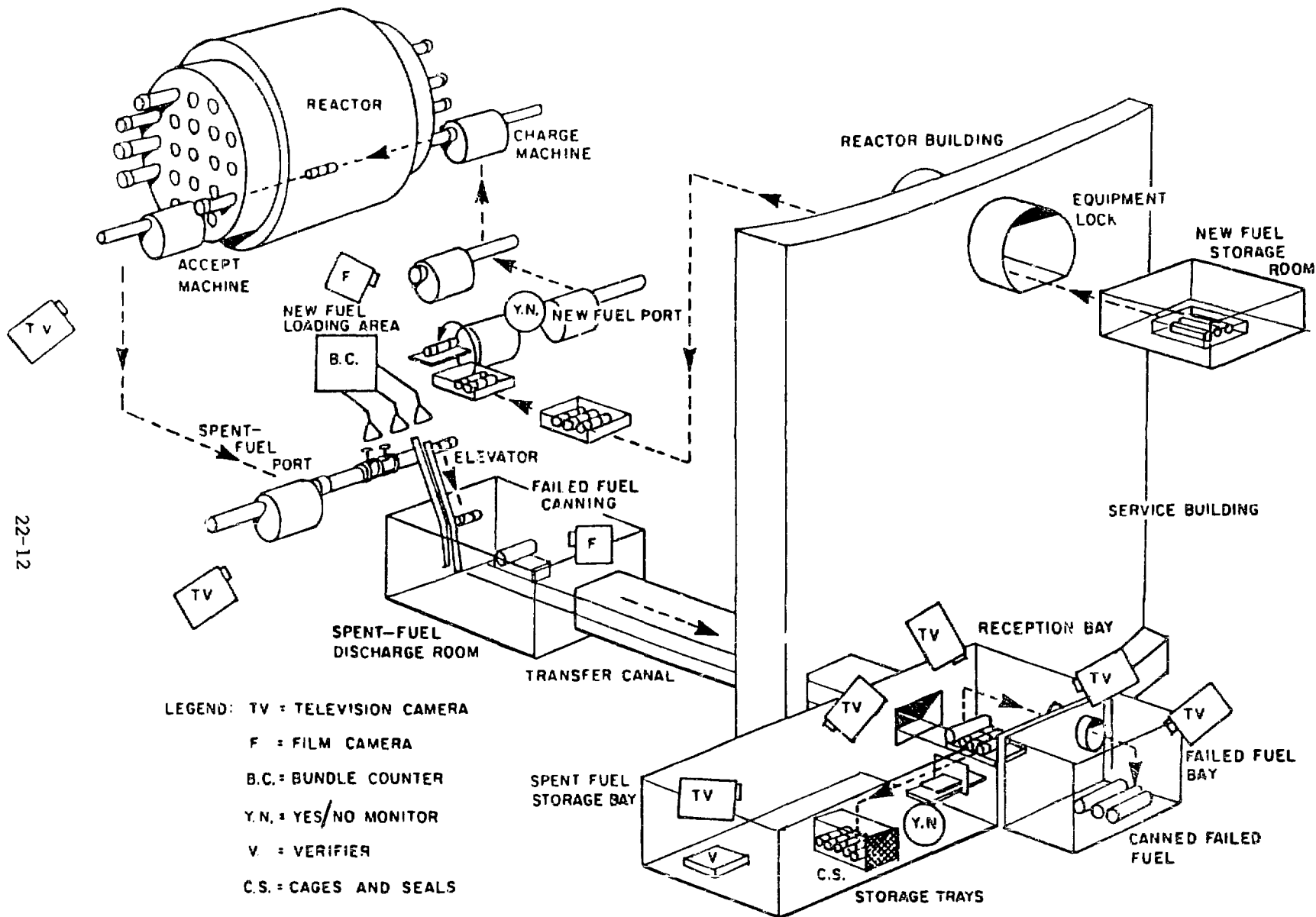


FIGURE 1. Location of Safeguards Instrumentation.

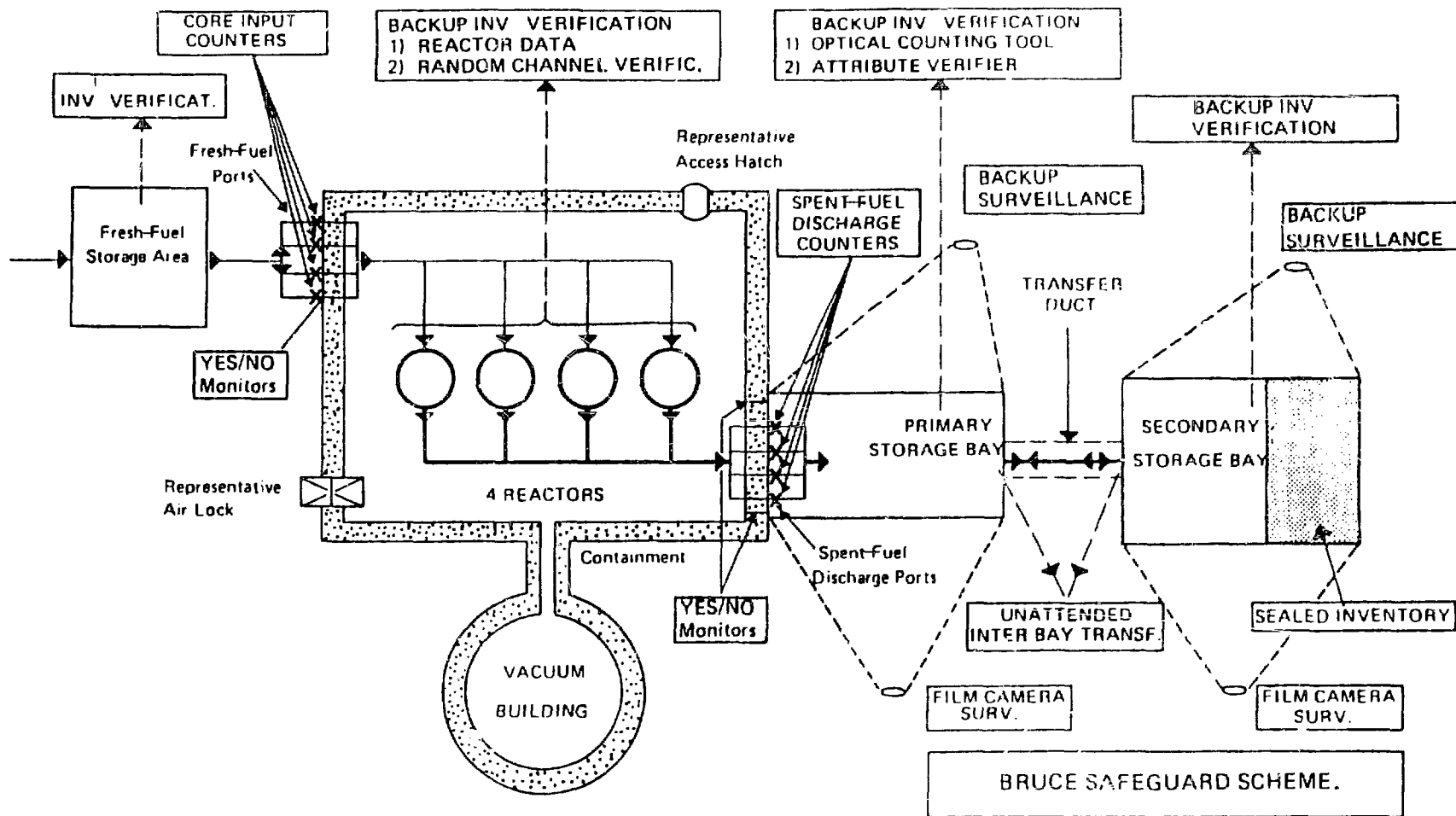


Figure 2a.

PICKERING SAFEGUARD SCHEME.

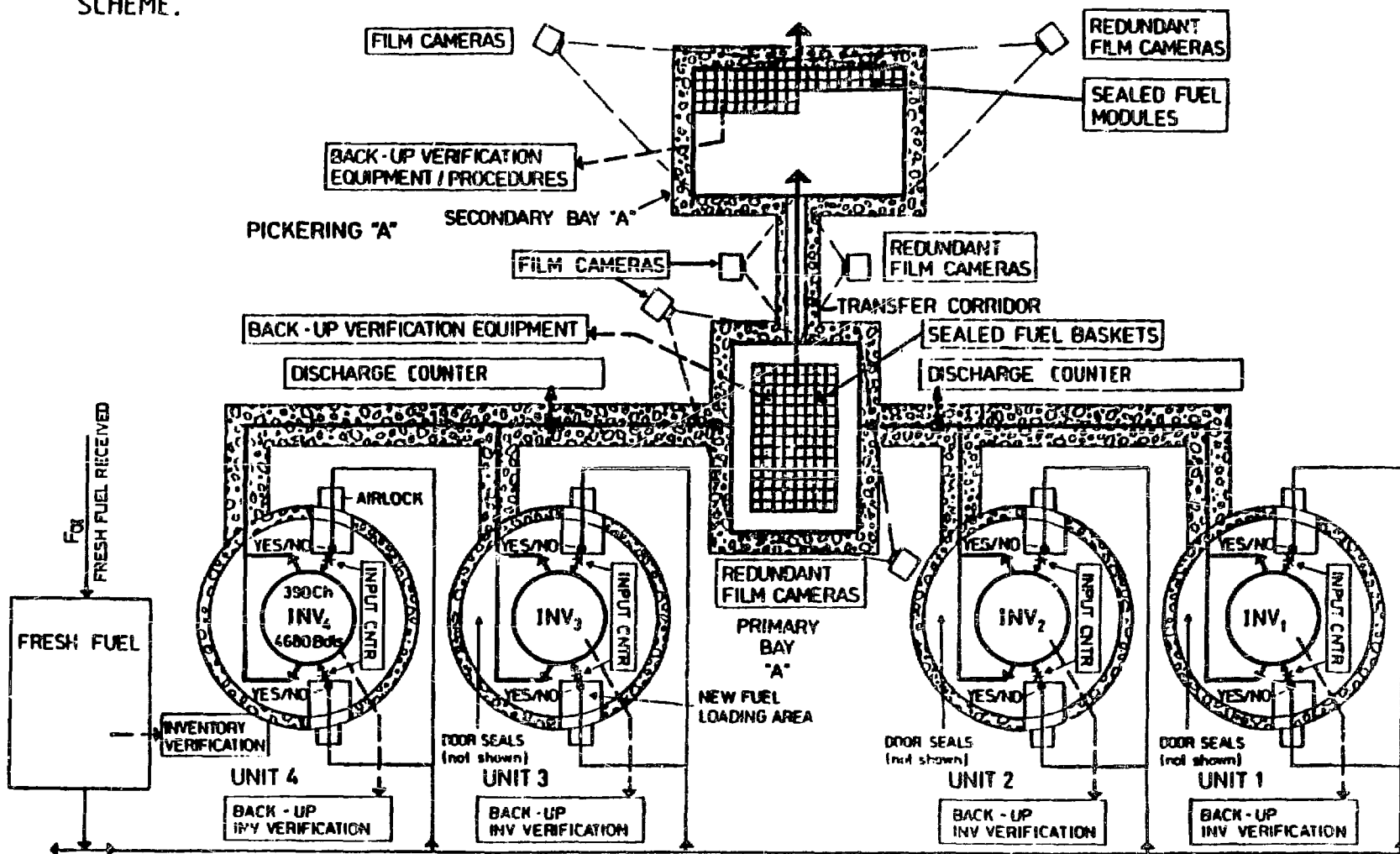


Figure 2b.

Appendix to Session 22

Background Information

Inspection Activities

Inspection activities by the IAEA are those activities carried out by the Agency's inspectors, both in the field and at the Vienna headquarters. The inspection activities in the field are carried out at the nuclear facilities and are called inspections. The purposes and scopes of inspections are defined in the safeguards agreements. The inspection activities at the Vienna headquarters are the pre-inspection activities, i.e., planning, preparation of equipment and documents; and the post-inspection activities, i.e., analysis and evaluation of collected information and verification data and preparation of inspection reports and statements to be communicated to the state.

Within the IAEA the implementation of safeguards agreements is entrusted to the three Regional Operational Divisions of the Department of Safeguards. Each division is manned by inspectors and supporting staff responsible for the direct application of safeguards.

The inspection activities carried out by the inspectors are defined and determined by the selected safeguards approach for each facility. The safeguards approach is based on the inspection goals, which, in turn, are derived from the IAEA detection goals and the technical objectives of the IAEA safeguards.

The logic behind the establishment of the instructions for the inspection activities is described in the following.

2. Purposes and Technical Objectives of Safeguards

Reference is made to the IAEA Safeguards Information Series, booklet no. 4 (Ref.1) in which the purposes and technical objectives of safeguards are defined in Chapters III and IV.

The technical objective of IAEA safeguards applied under NPT is made explicit in paragraph 28 of INFCIRC/153 (Ref.2), i.e., "...the objective of safeguards is the timely detection of diversion of significant quantities of nuclear material...and deterrence of such diversion by the risk of early detection".

Safeguards agreements based on INFCIRC/66/Rev.2 (Ref.3) do not contain explicitly formulated technical objectives. Paragraph 46 of INFCIRC/66/Rev.2 does, however, state that the purpose of safeguards inspections shall be to verify compliance with safeguards agreements. In NPT agreements the purpose of safeguards is expressed in paragraph 2 of INFCIRC/153: "...safeguards will be applied...for the exclusive purpose of verifying that such material is not diverted...."

Both provisions place the emphasis on verification, i.e., the Agency's independent verification measures through nuclear material accountancy, and containment and surveillance.

The purposes and objectives of IAEA safeguards can be summarized as:

- PURPOSES:
- to assure ----- no diversion occurs
 - to deter diversion ----- if contemplated
 - to detect diversion ----- if undertaken

TECHNICAL

- OBJECTIVES:
- timely detection of diversion of significant quantities of nuclear material; and
 - deterrence of such diversion by risk of early detection.

3. Implementation and Performance Parameters

The safeguards objectives are defined in quantitative terms by the IAEA detection goals which are a set of numerical parameters for:

- significant quantity (SQ);
- detection time; and
- detection probability.

The detection goals are not rigid requirements but serve as guidelines from which the IAEA derives the targets of actual verification activities, i.e., the inspection goals.

The parameters that constitute the detection/inspection goals are described and quantified in Ref.2, Chapter IV.1, 2, and 3, and in Ref.1, item 89, 90(87), and 91. They are summarized as follows:

a) Significant Quantity (SQ)

SQ is defined as the approximate quantity of nuclear material in respect of which, taking into account any conversion process involved, the manufacturing of a nuclear explosive device cannot be excluded.

SQ values currently in use are:

- Direct-use nuclear material
Pu, 8 kg; HEU, 25 kg
- Indirect-use nuclear material
N and LEU, 75 kg U-235

b) Detection Time

The IAEA has provisionally set the detection times to be of the same order of magnitude as the corresponding conversion times, i.e., the time required under optimal conditions to convert various forms of nuclear material into the metallic components of a nuclear explosive device.

The conversion time estimates used at present by the IAEA are:

- Pu-239 or HEU metal 7 to 10 days
- Pu in irradiated fuel 1 to 3 months
- N and LEU about 1 year

c) Detection Probability (Risk of Detection)

The overall detection probability is the probability of detecting the absence of a quantity of nuclear material. There is today no method for calculating the overall detection probability at any plant where both material accountancy and containment and surveillance measures are used. At present it is only possible to quantify the detection probability that can be achieved by material accountancy.

The IAEA aims at a 90 to 95% detection probability, and less than 5% probability of false alarm.

4. Safeguards Approaches and Safeguards Measures

The technical objectives of safeguards are translated into detection goals which are used as guidelines to establish inspection goals which are performance targets specified for a given facility. The inspection goals are required to implement the safeguards approach at that facility.

The safeguards approach constitutes the system of safeguards measures which are used to achieve the objectives and contains the concrete instructions for inspection activities.

The safeguards measures are--in general terms--methods for verification.

In IAEA safeguards, to verify is to establish the truth of statements regarding the amounts, presence, and use of nuclear material as recorded by facility operators and as reported by the state to the IAEA.

The fundamental basis on which verification rests is:

- nuclear material accountancy, and
- containment and surveillance.

Reference is made to Article 29 of INFCIRC/153 which says that safeguards approaches use "material accountancy as a safeguards measure of fundamental importance, with containment and surveillance as important complementary measures."

The verification process--constituting the inspection activities--comprises three distinct stages:

1. The examination of information provided by the state.
2. The collection of information by inspectors.
3. The evaluation of the received and collected information.

In summary, the inspection activities at a specific facility can be described in the form of, a) nuclear material accountancy activities, and b) containment and surveillance activities.

5. Nuclear Material Accountancy

In Ref.4, item 110, nuclear material accountancy is defined as "the practice of nuclear material accounting by the facility operator and the SSAC

and, in addition, the verification and evaluation of this accounting system by the IAEA."

Nuclear material accounting is in item 109 of Ref.4 defined as "the activities carried out to establish the quantities of nuclear material present within defined environments and the changes in those quantities taking place within defined periods of time. Essential elements of nuclear material accounting are material measurements, record keeping, preparation and submission of accounting reports, verification and analysis of these accounting data to determine correctness, accuracy of MUF, and evaluation of causes of MUF."

The essential accountancy activities are:

- (i) by the operator:
 - identification, counting, measurement, and keeping records of the material in the area concerned (i.e., the material balance area, MBA), including the periodical taking of the physical inventory;
 - measurement and recording of transactions from and to the MBA;
- (ii) by the state (SSAC)
 - preparing and submitting of accounting reports to the IAEA;
 - providing for inspector access to enable the IAEA to carry out its inspection activities;
- (iii) by the inspector (IAEA)
 - independently verifying nuclear material quantities and locations through:
 - a) examination of records,
 - b) verification of inventory,
 - c) verification of inventory changes (flow),
 - preparing and submitting of statements on the IAEA inspection/verification activities to the state.

All accountancy records and reports are referred to material balance areas (MBAs) which are established in a nuclear facility. The MBAs are areas into and out of which all transfers can be determined and in which physical inventories can be taken to establish a nuclear material balance. The MBAs are agreed between states and the IAEA and are defined in the Facility Attachments.

Within MBAs measurements are made at key measurement points (KMPs), which are locations where nuclear material can be measured for the determination of flow and inventory.

The inspector examination of records is an independent review, comparison, and assessment of facility accountancy and operating records and state

reports, with an intent to report and verify material quantities stated in such records and reports. Examination of records consists of:

- examining accounting records;
- comparing facility records with state reports;
- updating the book inventory; and
- examining operating records.

The term "examination" is often replaced by the term "audit" to denote these activities.

The inspector verification of inventory is carried out at an inspection of the facility by independent counting and identification, calibration, measurements and sampling, and (after the inspection) analysis and evaluation of the inspection data.

The inspector verification of inventory changes (flow) is carried out at inspections of the facility using the same verification methods as used for the inventory verifications but on nuclear material batches/items transferred into or out of the facility.

The result of a periodical physical inventory taking by the facility operator and the relation to the corresponding book inventory is fundamental to nuclear material accountancy. Any difference between the book inventory and the physical inventory is called "material unaccounted for" or MUF. The reasons for MUF may be found in measurement uncertainties, losses, or other technical causes. If the size of MUF is found to be beyond a value attributable to such identifiable causes, the possibility that a diversion has been made must be considered.

MUF is determined for a specific time period, i.e., the material balance period (MBP), which is the time between two consecutive physical inventory takings by the operator as reflected in the state's Material Balance Report (MBR).

The MBP is closed by the inventory verification. After verification of the ending physical inventory (PE), the MUF between PE and the ending book inventory (BE) is determined.

The MUF equation for an MBA is commonly written as:

$$\text{MUF} = \text{PB} + \text{X} - \text{Y} - \text{PE},$$

where

PB = the beginning physical inventory for the MBP
= the beginning book inventory (BB)

X = the sum of increases to the MBA (receipts, nuclear production, de-exemption)

Y = the sum of decreases from the MBA (shipments, nuclear loss, exemption, measured discard, accidental loss).

Since $\text{BE} = \text{PB} + \text{X} - \text{Y}$,
thus, $\text{MUF} = \text{BE} - \text{PE}$.

For a complete evaluation of the MUF it is necessary to verify all the terms of the MUF equation, i.e.,

- a) PB and PE by inspector verification of inventory, and
- b) X and Y by inspector verification of inventory changes (flow).

In carrying out the inventory and the inventory change verifications the inspectors are, for evident reasons, using statistical methods. This is stipulated in NPT agreements (Article 6.6 of Ref.2).

Sampling plans used for random selection of nuclear material items for verification (by measurements) are established on the basis of a preferred detection probability as described above.

The samples selected for verification according to the sample plan are contributed to an attribute test and/or a variables test. An example of an attribute test is the measurement of the characteristic, or "attribute" radiation emission to detect dummies amongst spent-fuel elements. An example of a variables test is chemical analysis for the determination of element (or uranium) concentration in a sample of nuclear material.

6. Containment and Surveillance

Containment and surveillance measures are important complements to nuclear material accountancy. In item handling facilities, e.g., reactors, they are in fact necessary for the verification of location, integrity, and movement of nuclear material.

Most often reference is made to the combined containment/surveillance (C/S) measures in a facility, and the equipment used by inspectors to perform C/S functions are referred to as C/S devices.

Containment means a physical barrier provided either by the facility structure (e.g., walls) or transport flasks/containers or by C/S devices, i.e., seals.

Surveillance means the collection of information through either C/S devices or inspector observation in order to detect undeclared movements of nuclear material or tampering with containment or equipment. C/S devices used for instrumental surveillance are film cameras and TV cameras and recorders (i.e., video surveillance systems). Surveillance by inspector observation is called human surveillance and may be carried out by manning key observation points continuously, or by unannounced, randomly selected frequent visits.

One of the most used applications of C/S measures is that of spent-fuel storages at reactor facilities. In this case, containment is provided by the walls of the storage bay and building, while surveillance is carried out by redundant film or TV cameras. The cameras take pictures at intervals shorter than the time needed to remove fuel. Periodic evaluation of the films or tapes would, thus, reveal any removal of spent fuel. The fuel stored in the bay may also be sealed.

Other C/S devices are monitors which can be load/position monitors, reactor power monitors, radiation passage (yes/no) monitors, or fuel assembly (bundle) counters.

The containment/surveillance inspection activities comprise:

- installation of surveillance cameras and systems,
- periodic service and maintenance of surveillance devices,
- review and evaluation of recorded surveillance data,
- attachment/detachment of seals,
- checking of seals,
- reading and recording of other C/S device data, and
- in-field testing of new C/S devices.

REFERENCES

1. IAEA/SG/INF/4 IAEA Safeguards, Aims, Limitations, Achievements, Vienna, November 1983.
2. INFCIRC/153 (corrected), Vienna, June 1972.
3. INFCIRC/66/Rev.2, Vienna, September 1968.
4. IAEA/SG/INF/1 IAEA Safeguards Glossary, Vienna, June 1980.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 23: MATERIAL ACCOUNTING AND CONTROL FOR LIGHT-WATER REACTORS AND SPENT FUEL STORAGE FACILITIES IN FINLAND

This session will address various aspects of state system of accounting and control (SSAC) in safeguarding of Finnish light-water reactors and fuel cycle activities. Emphasis is put on practical experience at operating facilities and in nuclear fuel procurement.

The following aspects are considered:

- Implementation of the agreement between Finland and IAEA as well as bilateral agreements with other countries.
- Different types of reactors and fuel assemblies.
- Cooperation with different countries supplying uranium, enrichment services, and fuel assemblies.
- Interim and long-term spent-fuel storages.
- The identification and measurement systems of nuclear materials.

After the session, participants will be able to:

- Describe the requirements concerning a material control and accounting for an LWR facility.
- Identify the major components of LWR fuel cycle, which have influence to SSAC.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 23: MATERIAL ACCOUNTING AND CONTROL FOR LIGHT WATER
REACTORS AND SPENT FUEL STORAGES IN FINLAND

Presented by Mauri Riihonen
Finnish Centre for Radiation and Nuclear Safety

1 INTRODUCTION

When we are talking about the safeguarding of nuclear materials, we must keep in mind that the meaning of that word may be different for the different organizations involved. The technical objective of IAEA safeguards is defined as

"the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or other nuclear explosive device or for purposes unknown and deterrence of such diversion by risk of early detection" (IAEA Statute).

The IAEA safeguards under the NPT is based on the existence of a State System of Accounting for and Control of Nuclear Materials (SSAC) (INFCIRC/153, Corr.) the technical objective of which is often more extensive than that of IAEA. The Finnish SSAC for example has the same objective as the IAEA and in addition, it has to assure that the nuclear material, for which Finland is responsible, is used in compliance with the provisions set forth in legislation and in licences issued on the basis of legislation as well as with the requirements of bilateral agreements Finland has signed in the nuclear field.

The objective of an international safeguards system is essentially to deter a government from making a diversion. The government tends by means of the SSAC to deter individuals or organizations within its jurisdiction from misusing nuclear materials. So the timeliness requirements and the significant quantities are stricter for SSAC. The structure and share of duties of the SSAC are strongly influenced by the responsible authority, its organization and relation to other nuclear authorities, as well as the requirements coming from international agreements.

In this paper, the structure of the Finnish SSAC is described. Emphasis is put on practical experience in the implementation of safeguards at operating nuclear facilities and in their nuclear fuel procurement. The aim is to illustrate all the activities which are the responsibility of the safeguards-organization and which therefore have influence on the structure and practical work of the SSAC.

2 NUCLEAR ACTIVITIES IN FINLAND

2.1 Reactors and Spent Fuel Storages

In Finland, there are today four operating nuclear power plant units, which were ordered between the years 1970 and 1974. No decisions have since been made concerning new nuclear power plants although the feasibility of some LWR plants to Finland is being studied. The first two units (Loviisa 1 and 2) are operated by the state-owned power company Imatran Voima Oy (IVO) and their main supplier was the Soviet export organization V/O Atomenergoexport. The plant units are PWRs of the type VVER 440. Their thermal power is 1,375 MW and the net electrical power 440 MW.

The two other units (TVO I and II at Olkiluoto) are owned by Teollisuuden Voima Oy, which has been founded by 15 Finnish industrial companies. The units are of BWR type and they were supplied by the Swedish Ab Asea-Atom. Their thermal power has been raised from the original 2,100 MW to 2,160 MW which corresponds with the net electrical power 710 MW.

The Loviisa nps is furnished with two special spent fuel storages (water pool type), which locate in the auxiliary building. In one storage the fuel assemblies are handled and stored separately, but in the other, older storage the fuel assemblies are handled and stored in baskets of 30 fuel assemblies.

At the TVO plant site a special long term spent fuel storage is under construction. The storage is of the water pool type and the total capacity is about 7 200 assemblies. The storage consists of six storage pools and an associated receiving area.

The first unit, Loviisa 1, was commissioned in 1976 and after that the Finnish nuclear power plants have been in operation for 26 reactor years in total. According to the operation reports, the load factor of the Finnish plants in 1984 was 91.9 %.

Besides the nuclear power plants, there is in Finland a research reactor TRIGA MARK II which has been supplied by General Atomic Division of General Dynamics Corporation (USA). This reactor was commissioned in 1962 and its thermal power is 250 kW.

Table 1 shows the most important specifications of the Finnish nuclear facilities.

Table 1. Specifications of nuclear facilities in Finland.

	<u>Loviisa 1&2</u>	<u>TVO I&II</u>	<u>TRIGA</u>
<u>1. Reactor</u>			
Type	PWR	BWR	open tank
Thermal output	1375 MW	2160 MW	250 kW
Maximum linear power/ heat flux	285 W/cm	108 W/cm ²	-
Number of fuel assemblies	313	500	79
Weight of U in core	38 t	89 t	14.6kg
Annual refuelling	14 t	20-25 t	-
Nominal duration of refuelling cycle	7000 h	7000 h	-
Number of control rods	37	121	4
<u>2. Fuel</u>			
Enrichment	1.6 - 3.6 %	1.9 - 3.0 %	20%
Number of rods in ass.	126	8 x 8 - 1 8 x 8 - 2 9 x 9 63 (SVEA)	1
Length of assembly	3.2 m	4.1 m	1.09m
Average burn-up	28.6 MWd/kgU	32 MWd/kgU	20-25 MWd/kgU
<u>3. Fuel Storage Capacities</u>			
Fresh fuel storage	552	140	70
Fuel pools	537	1520 (TVO I) 1560 (TVO II)	12
Spent fuel storages	720 (1.) 1560 (2.)	7200 *	72

*) Away from reactor storage (AFRS).

2.2 Nuclear Fuel Management

The nuclear fuel for Loviisa nuclear power station is procured from the Soviet Union as fuel assemblies ready for use. After use the spent fuel is first stored in water pools in the reactor halls of both units. After a cooling time of 1 to 2 years the fuel assemblies will be transferred to two adjacent spent fuel storages (water pool type) which are common to the both units. According to the plans of Imatran Voima, all the spent fuel assemblies will be returned to the Soviet Union after a cooling time of at least 5 years.

So far the uranium for the nuclear fuel of TVO I and II plant units has originated in Australia, Canada and Niger. Until now the uranium has been enriched in the Soviet Union, but according to the plans of Teollisuuden Voima Oy, a part of the uranium will be enriched in the future in France, as well. The fuel assemblies have been manufactured in Sweden and in the Federal Republic of Germany. The spent fuel of TVO plant is stored in the reactor hall pools of both units for some years. After that cooling period, the spent fuel assemblies will be transferred to the separate spent fuel storage (water pool type), which is now under construction at Olkiluoto. Pursuant to the plans of Teollisuuden Voima Oy, the spent fuel will be stored in that storage for about 40 years. There is no technical decision concerning the final disposal or reprocessing of spent fuel, but according to the decision in principle by the Council of State of Finland, Teollisuuden Voima Oy has to provide preparedness for carrying out the final disposal in Finland in the case the spent fuel cannot be transferred and disposed irrevocably outside the domestic territory.

Fuel elements for TRIGA-reactor have been procured from the United States. The spent fuel elements will be stored in the reactor hall. There is no technical decision concerning the spent fuel management for a long-term period.

2.3 Nuclear Research

There is no special nuclear research centre in Finland. However, research and development in the nuclear field is carried out by several research laboratories and power companies.

As far as light water reactors are concerned, fuel assemblies are dismantled and individual rods are withdrawn for investigations for technical safety purposes. Destructive studies have been made for fresh fuel, but the integrity of the irradiated fuel rods has been retained.

3 THE STRUCTURE OF SSAC

3.1 The Basis of SSAC

Due to the special physical features of nuclear materials (radioactive, fissionable, potential source of powerful explosives) the safe use of nuclear materials is ascertained by a strict governmental control.

The SSAC is a system, which is composed of legally binding regulations, written procedures and measures for supervising nuclear materials and activities related to nuclear materials, as well as organizations responsible for implementing the supervisory activities. In Finland the SSAC includes elements from governmental authorities to organizations and individuals possessing and using nuclear materials to be controlled.

The starting point is that all nuclear materials in Finland shall be subject to the Finnish SSAC. The system is based on the Finnish Atomic Energy Act.

In addition to national control the nuclear activities are governed by the international agreements Finland has signed in the nuclear field.

Finland signed the Non-Proliferation Treaty (NPT) in 1970. In June 11, 1971 a safeguards agreement was signed between Finland and the Atomic Energy Agency. This safeguards agreement is published in INFCIRC/155 and it is the first agreement based on INFCIRC/153 model agreement.

Bilateral agreements in the nuclear field Finland has concluded with the following states: Australia, Canada, the Great Britain, Sweden, the USA and the USSR.

The Finnish SSAC has been formed so as to accomplish also the obligations of the above mentioned international agreements.

3.2 Licensing

According to the Atomic Energy Act, a special licence is required for the construction, possession and operation of a nuclear facility as well as for the production, possession, transport, use, import, export, trade and other transfer of nuclear materials. The licences are granted by the Council of State or by the Ministry of Trade and Industry. The licences include necessary conditions for safety and safeguards and they also take into account the international agreements signed by Finland.

3.3 Safeguards Authority

The Finnish Centre for Radiation and Nuclear Safety (STUK) is the authority which in practice is responsible for supervising that the provisions set forth in the nuclear legislation and the licence conditions are observed. The supervisory role of STUK starts from reviewing the licence applications and it covers the nuclear and radiation safety of nuclear activities and installations as well as safeguards and physical protection.

As regards safeguards STUK is responsible for

- reviewing the licence application from safeguards point of view,
- establishing provisions governing the possession, transfer and use of nuclear materials,
- maintaining and developing a system of accounting for and control of nuclear materials,
- ensuring that the national safeguards objectives are met,
- ensuring that the obligations of relevant international agreements can be discharged,
- serving as the point of contact in implementing the safeguards agreement concluded with the IAEA and the bilateral nuclear agreements.

For that purpose STUK

- issues guides (YVL GUIDES) on licensing procedures as well as on nuclear material control procedures,
- approves safeguards manuals which are prepared by the license holders and which include the basic instructions for the operators' safeguards activities,
- approves activities related to nuclear materials,
- inspects facilities to ensure that the nuclear material activities are in compliance with licence conditions and other approved procedures,
- prepares and submits reports and notifications to the IAEA and the appropriate competent authorities in other countries,
- carries out research and development in the field of safeguards (currently NDA-measurements, computerization).

3.4 Safeguards Implementation

The safeguards activities of the Finnish LWRs can be divided into two separate areas: control of nuclear materials that exist in the Finnish territory and control of nuclear materials that are outside Finland but in certain sense under the control of Finland. Below, both areas will be dealt with separately and the practical actions that are taken to implement the control will be discussed.

For the safeguards purposes the following material balance areas (MBA) have been established:

- SF-A; Reactor Laboratory, Otaniemi
- SF-B; Loviisa Nuclear Power Station
- SF-C; TVO Nuclear Power Plant, Unit I
- SF-D; TVO Nuclear Power Plant, Unit II
- SF-T; Multinational transfers of nuclear material between Finland and other states (concerning nuclear material for TVO)
- SF-Z; Miscellaneous Locations of Nuclear Material Outside Facilities in Finland

The operators are responsible for the management of nuclear fuel and other materials at the facilities. All the measures shall be taken according to the procedures approved by STUK and the guides given by STUK.

4 IMPLEMENTATION OF NUCLEAR MATERIALS CONTROL AT FACILITIES

4.1 Characteristics of the Nuclear Facilities

As far as nuclear materials control is concerned, the BWR reactors, TVO I and TVO II, are two separate plants; they are physically separated, both have their own fresh fuel storages and pools for spent fuel in the reactor halls. Each unit constitutes a MBA of its own (see figure 1).

The PWR reactors (Loviisa 1 and Loviisa 2) constitute physically a so-called twin unit. The reactors themselves are located each in their own building inside Westinghouse-type containments. Near by the reactor there is a loading pool, which in practice always contains spent fuel in storage. The plant has one common fresh fuel storage and two common spent fuel storages. The fuel pools have been provided with covers that can be sealed. Due to common storage areas the Loviisa nps forms one single MBA (see figure 2).

The separate spent fuel storage being built at the TVO plant site will constitute a new material balance area (see figure 3).

4.2 Basic Provisions for Safeguards Procedures

In the fuel licence granted by the Ministry of Trade and Industry for the power companies it is stipulated that in actions concerning nuclear materials, the licensee shall comply with the procedures approved by STUK and that the licensee shall maintain a control system of nuclear materials approved by STUK.

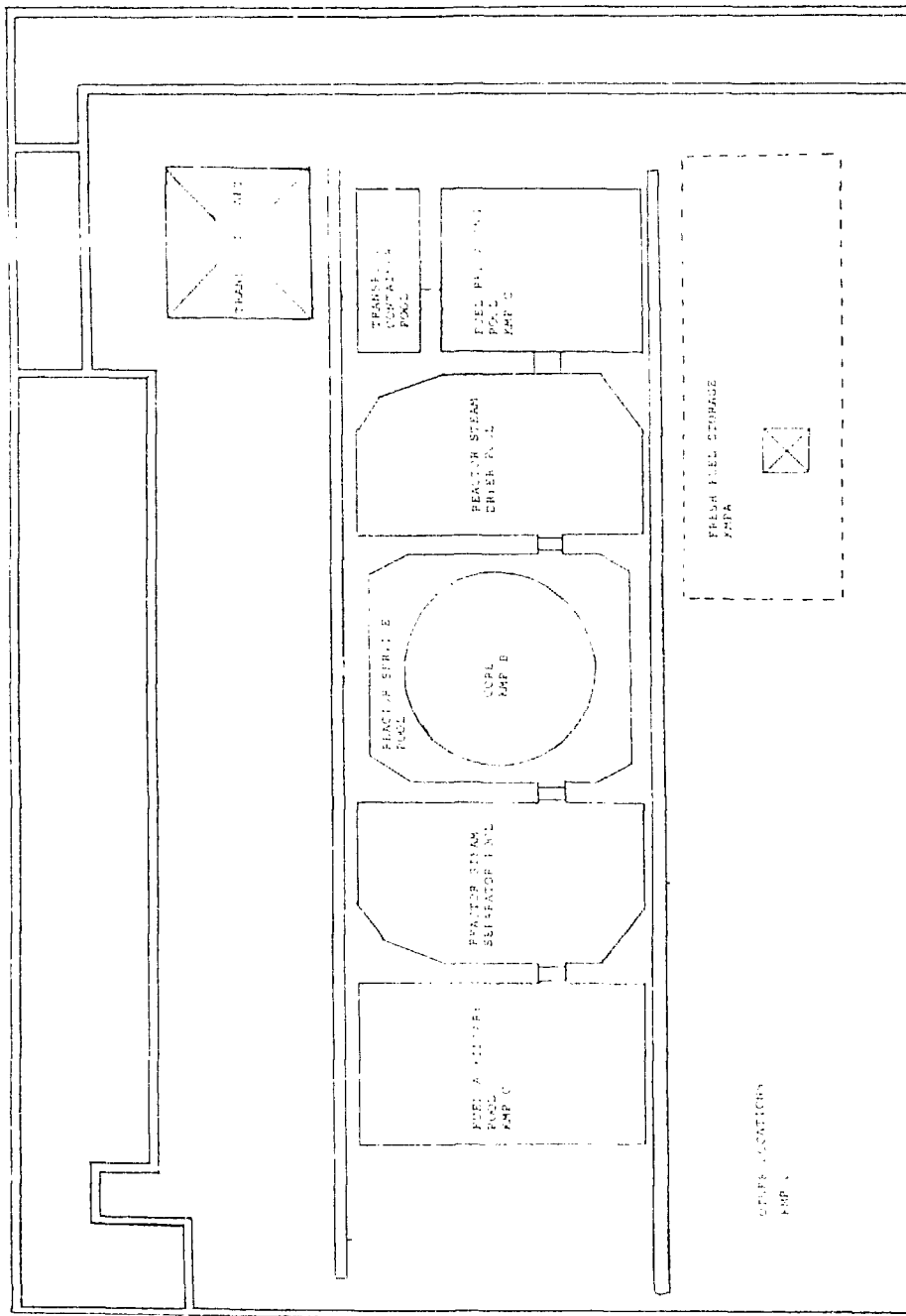


Figure 1. SCHEME OF TWO REACTOR BUILDING, IDENTICAL FOR UNITS 1 AND 2

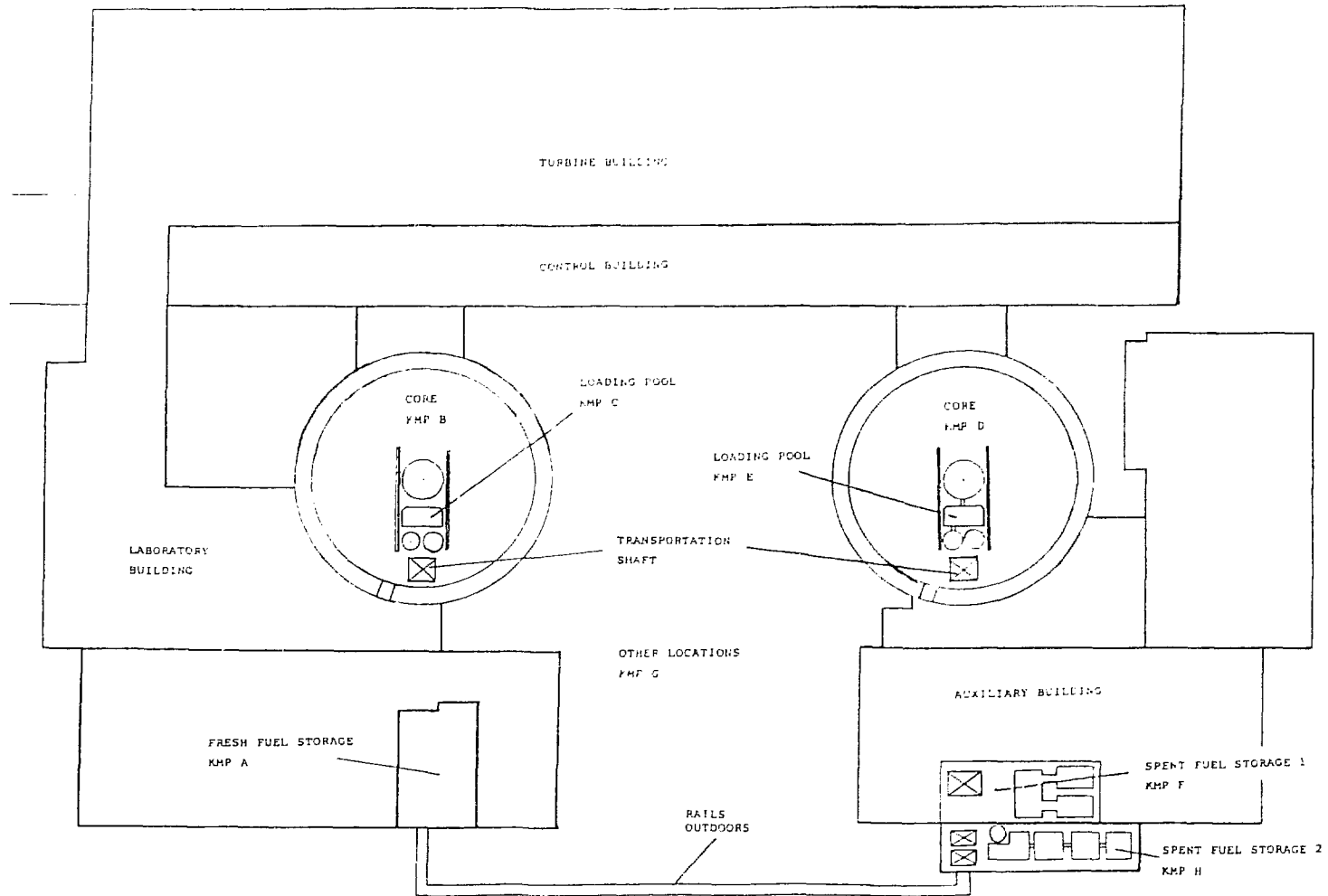


Figure 2. SCHEME OF LOVIISA NUCLEAR POWER STATION

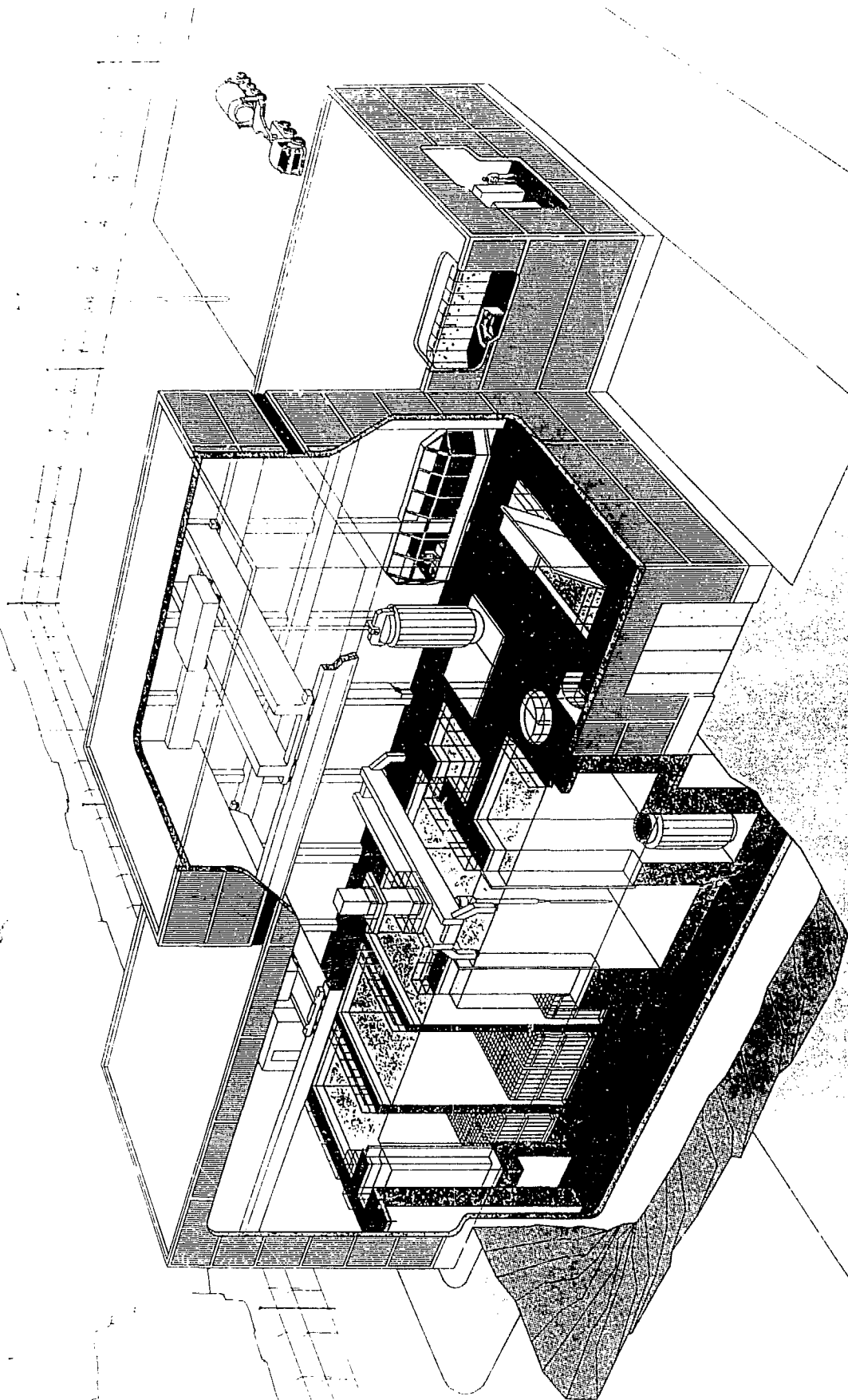


Figure 3. Spent fuel storage of Teollisuuden Voima Oy (TVO)

Accounting and reporting are the basic methods of control. The nuclear fuel is shipped to the Finnish reactor plants from abroad, and so the first data are obtained from the documents provided by the shipper. By means of control activities, it is ensured thereafter that this information is in compliance with the supplied nuclear material and that the continuity of that knowledge is retained.

The licensee is obliged to compile a safeguards manual, which shall in detail indicate all the actions concerning nuclear materials control. The manual is subject to the approval of STUK. In principle, the manual shall be written in such detail that an operator can, solely on the basis of the manual, perform all necessary actions and that STUK can supervise the operator's work.

Among his organization, the licensee nominates persons to be responsible for the control activities of nuclear materials at the plant and to act as contact persons with STUK in safeguards matters.

4.3 Design Information

Before establishing safeguards measures for a certain facility, special design information is collected and evaluated. The operator has to submit to STUK a compilation of facility design data, which at the minimum shall contain the information IAEA is asking for in the Facility Design Information Questionnaire. The facility design data are subject to approval of STUK.

4.4 Accounting

In practice the safeguards matters at the plants are taken care of by persons who are also otherwise responsible for the handling and utilization of fuel. Therefore it has been possible to flexibly combine the actions required by safeguards, including the compilation and recording of documents, with the other management of nuclear fuel without causing any unnecessary overlapping.

The accounting system kept by a LWR operator consists of the following documents:

- source documents; documents prepared and supplied by the shipper of the fuel, or, in case measurements or calculations of nuclear materials are performed, records of the actions of the operator
- general ledger, in which the licensee records all inventory changes in MBA and the cumulative total inventory specified according to each nuclear material type; it also includes reference to reports submitted to STUK

- balance book; updating after annual physical inventory taking (PIT), format is identical with material balance report (MBR)
- operating records; consist of work orders concerning the transfer and handling of nuclear fuel, power history records, documents on inventories (itemized lists, receiving inspection protocols, storage maps, history file) and the so-called logbook, in which one records chronologically all actions significant to safeguards
- reporting documents; consist of copies of the reports that have been sent to STUK.

4.5 Computerization of Nuclear Material Data Processing

The processing of nuclear material data has been automatized as much as possible. The first stage of accounting, the input of data to the computer, must still be performed manually. After that, all otherwise laborious recording work from the updating of the history file to the compilation of reports is performed utilizing the computer. The fuel management requires close observation and bookkeeping also in other respects, e.g. in the preparation of work orders and storage maps. It has been the aim to join all these actions together as effectively as possible. After all, the safeguards accounting does not add much to the fuel accounting that has to be carried out at the plant in any case. An international accounting practice, which is as uniform as possible, will make it practical for the operator to follow a certain pattern in his accounting system. (However, this can also be regarded as an inconvenient limitation.) The following is a list of the actions that are carried out with the computer:

- updating of the history file
- transfer of burn-up data from the process computer
- design of core configuration; preparation and drawing of the location map
- preparation and drawing of the storage maps on the basis of operating orders
- compilation of summaries
- preparation of ICR, PIL and MBR reports and their transfer to magnetic tape.

Of the actual accounting documents, only the general ledger is updated manually; however, the computer gives the necessary figures.

It must yet be noticed that the computer is used only as a technical aid. The official documentation is made entirely in writing. Most documents are computer print outs supplemented with signatures.

4.6 Reports and Notifications

There are three types of reports, that the operator provides to STUK: advance notifications (for example on imports and exports, refuellings, fuel dismantling operations), accounting reports (ICRs, MBRs and PILs) and special reports (on any unusual incident). The reporting and notification procedure is described in detail in the safeguards manual. The time limits for reporting have been determined in such a way that STUK can take care of the reporting to the IAEA, as stipulated in the agreement between Finland and the IAEA. STUK reserves about two weeks for the handling of the reports. As a rule, the accounting reports are transferred on magnetic tapes. For short reports only print outs are used.

4.7 Other Control Actions Taken by the Operator

Besides accounting and reporting duties, the operator is responsible for the compatibility of the actual nuclear material inventory with the information given in the book-keeping. For this purpose the operator has two procedures available: administrative procedures and the physical protection of nuclear material. The purpose with the administrative procedures (instructions, regulations, delegation of responsibilities) is to ensure that all handling of fuel takes place in a controlled manner under the supervision of the responsible persons. As supportive means one uses containments, locks, access control, seals and tv-camera surveillance.

When the nuclear fuel is used its isotopic composition is changed. The nuclear loss and production is calculated on the basis of measurements carried out with the in-core instrumentation during the operation of the reactor. The instrumentation consists of neutron flux monitors and coolant thermocouples. The isotopic composition is derived from power distribution and burnup calculations.

The nuclear loss and production are recorded when irradiated fuel is discharged from the core.

The plants have been provided with facilities for NDA measurements. Those measurements are mainly performed for purposes of fuel safety but the results can also be utilized for safeguarding purposes.

5 STUK SAFEGUARDING NUCLEAR FACILITIES

The organization of STUK is presented in Figure 4. The supervision of safety and safeguarding of nuclear facilities are carried out by the Department of Nuclear Safety. For this purpose it employs about 60 inspectors. In the Department, the Office of Fuel Cycle takes care of i.a. safeguards, supervision of physical protection and control of the nuclear fuel cycle and the fuel handling. In

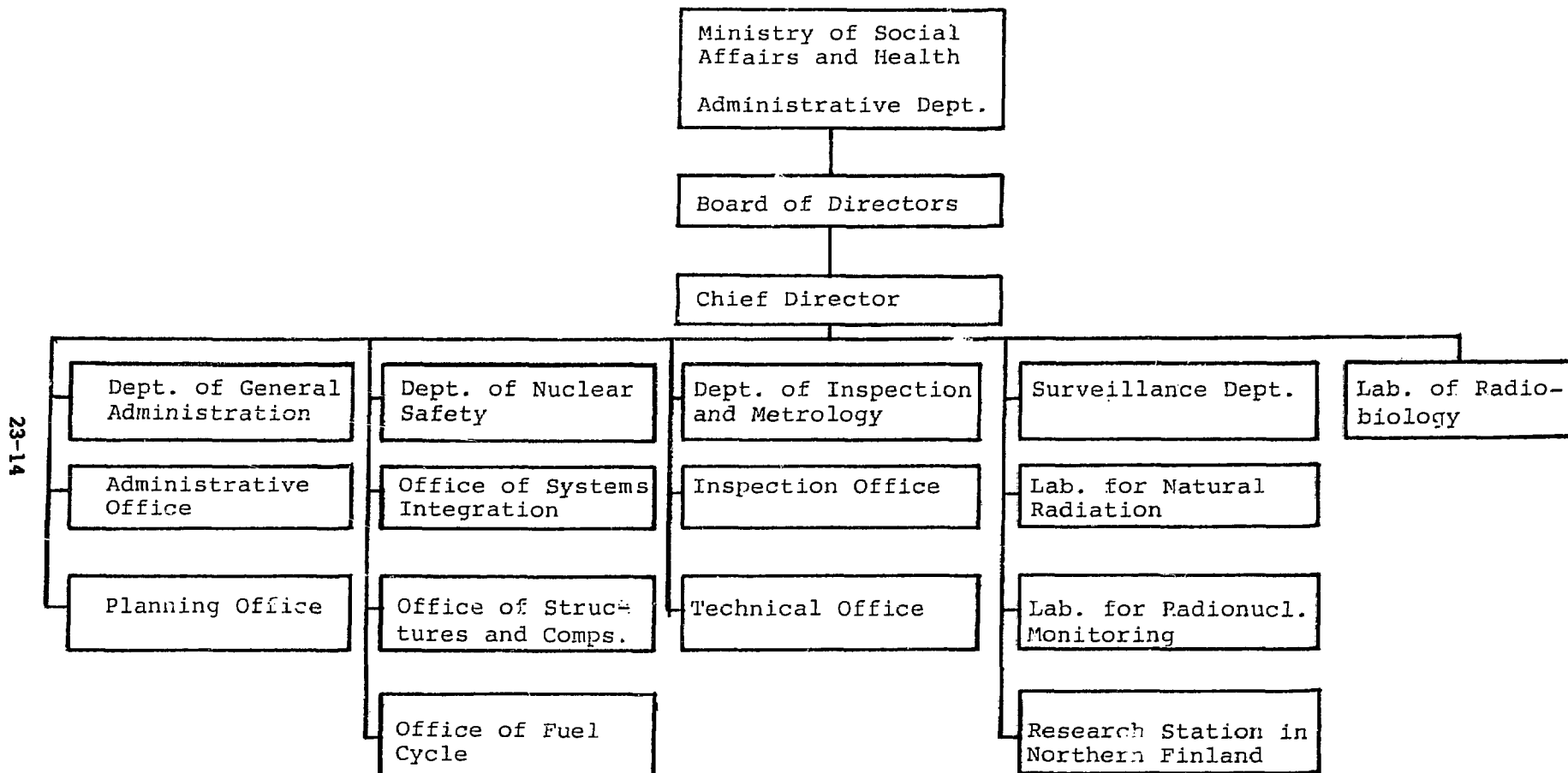


Figure 4. Organization of STUK

the implementation of the safeguards, the other activities of the Department can be utilized in many ways. Information on the operation and operating incidents of the plants is received continuously through the reporting system developed for safety purposes. Whenever convenient, inspections at the plant sites are combined and thus unnecessary overlapping is avoided. In addition, STUK has resident inspectors at the plant sites, and they can easily undertake safeguards inspections, as well.

5.1 Approval of Operators Activities

In the previous chapter it was already mentioned that operators activities related to nuclear materials are subject to the approval of STUK. The licence conditions and the guides issued by STUK include provisions and instructions for the application for approval in different situations.

As regards safeguards there are three major documents to be prepared by the operator and submitted to STUK for approval:

1. Design Information
2. Safeguards Handbook
3. Physical Protection Plan.

After careful evaluation and approval of the Design Information it is submitted to the IAEA. On the basis of that document the IAEA prepares a draft for respective Facility Attachment.

Before any nuclear material is taken to the facility the Safeguards Handbook must have been approved. The aim is that the Safeguards Handbook takes into account all the necessary details including Facility Attachments.

Physical Protection on nuclear materials and facilities is handled in Finland as a whole of its own. However, physical protection of nuclear materials is an essential part of safeguards, as well.

The safeguards performed by STUK is based on the premise that the operator is responsible for the control and accounting of nuclear materials at the plant site and STUK verifies with inspections that the operator takes care of these duties in accordance with the clauses in the licences and with the manual and procedures approved by STUK.

Figure 5 shows the communication channels in the implementation of the Finnish SSAC.

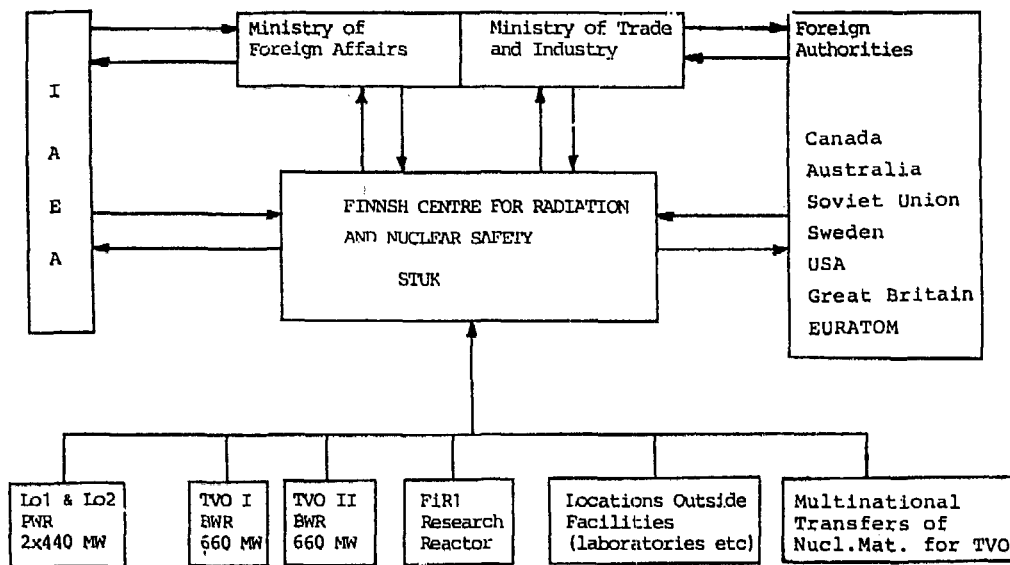


Figure 5. The communication channels in the implementation of the Finnish SSAC.

5.2 Central Accounting of STUK

All accounting documents are kept at the plant, where they shall be available for inspection at any time. On the basis of the reports submitted to it, STUK maintains a central accounting system covering the whole country.

A computerized system has been developed in STUK for handling nuclear material accounting data. The tasks of the system can be summarized as follows

- recording, checking of ICRs, PILs and MBRs (hard copies or on magnetic tapes) submitted to STUK by licencees,
- maintaining a centralized data base of safeguards data,
- preparation of ICRs, PILs and MBRs to the IAEA (as hard copies and on magnetic tapes),
- preparation of reports for inspection purposes (inquiries, accounting reports in STUK-format, excerpts of general ledgers, book inventory listings etc.),
- preparation of reports based on requirements pursuant to bilateral agreements.

More information of the computerized safeguards system of STUK can be found in ref 1.

5.3 Inspections of STUK

The inspection program of STUK comprises inspections of three kinds:

- preprogrammed inspections
- general inspections
- special inspections.

A preprogrammed inspection is performed on each material balance area once a year. The inspection encompasses the whole system: permits, up-to-dateness of the safeguards manual, organizations (possible changes), need to modify procedures on the basis of the experiences obtained, and bookkeeping. The general inspections usually deal with a certain activity, such as inventory changes, refuelling, or some other extensive handling of fuel, physical inventory taking, or an inspection by IAEA. A special inspection is carried out to clear up an unusual event or if there is reason to believe that nuclear material may be missing.

The inspections of STUK consist of an examination of records and a verification of the inventory or changes in inventory in order to ensure the consistency between records and the physical inventory. In addition, inspectors often follow fuel handling activities from the safety point of view and serving safeguards purposes as well.

5.4 Special Features of the Facilities

Physical Inventory Taking

Because of the special features of the plants, the safeguards measures may be of different types at different plants. It was already pointed out above that the TVO plant units each form a material balance area of their own, whereas the Loviisa units form one material balance area.

The physical inventory taking is carried out by the operator once a year immediately after the annual reloading. Due to operational reasons the operator is willing to close the pressure vessel as soon as possible after loading operations have been completed. The STUK inspectors are prepared to verify the physical inventory when the reactor core is still open.

At Loviisa nps the reloadings of the two units take place at different times. Therefore the fuel in one core is always inaccessible. Consequently the physical inventory taking and its verification have to be carried out in two phases. When one core has been verified it is ensured, basically by C/S measures, that its contents are not changed until verification of the second core.

The date of physical inventory taking is determined by the reloading of the second core in that calendar year.

At TVO the PIT and PIV are carried out once a year during the annual maintenance period.

Construction of the TVO Fuel

The construction of the TVO fuel assembly is such that it is possible to change the rods in the assembly. The identification number of the fuel bundle has been engraved on a handle that is easily detachable from the assembly. Therefore the physical verification of the materials merely on the basis of the identification and the item counting is not always satisfactory. For this purpose, there are measurement methods under development (will be discussed later on).

Because of the planned rod-exchange operations, it shall be possible to accomplish the bookkeeping also on a single rod basis. The source documents include rod-specific data on the fresh fuel; for irradiated fuel the material data are obtained with calculational methods. For fuel, the accounting unit is a bundle, and to rod-exchanges one applies the rebatching operation. The accounting proceeds down to the level of a single rod only when necessary, i.e. when dealing with disassembled bundles, and the bundle-specific accounting is resumed after reassembling.

Construction of Loviisa Fuel

The fuel bundles that are used in Loviisa have not been designed to be disassembled; the rods are inside a hexagonal fuel box that has been welded to end pieces. However, for purposes of technical inspections of fuel, both fresh and spent assemblies are dismantled at Loviisa plant. The accounting is managed following the same principles as at TVO.

So the Finnish nuclear power plants utilize the rebatching operation, which is not usually used by the item-counting facilities, such as light water reactors.

In the reactors of the Loviisa plant units, the control rods are withdrawn upwards from the core. At the lower end of each control rod (in total 37) there is a fuel bundle, a so-called fuel follower, which is placed in the core when the control rod is withdrawn. When the core has been loaded, the control rods are inside and their fuel followers cannot be seen. If these fuel followers are to be identified during physical inventory verification (PIV), the identification of one bundle takes about 2 hours. When one considers the fact that the reactor re-loading work is then on a critical path, the time taken by the identification is long. As far as STUK is concerned, the matter can be settled in connection with the supervision of refuelling. The IAEA has employed the random sampling method in identifying the fuel followers.

Dummy Elements

Another special feature in the reactor cores of the Loviisa plant units is the fact that altogether 36 fuel elements in the perimeter of the core have been replaced by dummy elements. The purpose of those dummies is to reduce neutron flux to the pressure vessel wall. The dummies look like fuel assemblies but can be visually distinguished from them. As safeguards is concerned, the dummies could be used as irradiation positions and reasonable control measures are to be applied. At the moment, sensible methods seem to be inspections of unirradiated dummy elements and NDA-measurements after irradiation. A method based on observation of Cherenkov glow is satisfactory for qualitative measurements.

5.5 NDA Measurements

Above it was stated that accounting is the basic element in nuclear materials control. The correctness of the accounting is supervised by means of checks, and the reported inventory changes and the physical inventory are verified at the plant sites mainly by using identification and item counting. However, in reliable control these methods are not always sufficient, for instance, when the nuclear material inventories have to be re-verified. One needs measurement methods whereby it is possible to verify the amount of nuclear material in the fuel without breaking the structure. The measurement systems of spent fuel are under extensive development; no complete system is so far available. STUK has developed and tested Non Destructive Assay (NDA) methods for about four years. The aim is to acquire the ability to carry out the NDA measurements of fresh and spent fuel at the Finnish nuclear power plants.

STUK is developing the NDA methods in cooperation with the IAEA, the Los Alamos National Laboratory (LANL) and the Nuclear Power Inspectorate (SKI) of Sweden. With the IAEA, STUK has signed a Technical Contract, and within the framework of this contract methods have been developed for the spent fuel of Loviisa (ref. 2), and starting from this year, also for the fuel of TVO. The work was begun with the PWR fuel of Loviisa because there were more reference data available. The experiences that have been obtained have made it possible to extend the scope so as to include BWR fuel of TVO. Sweden has BWR plants that are of the same type as the BWR plant in Finland, and in the cooperation between SKI och STUK one has been able to utilize the experiences and knowledge of both parties.

The development work has been concentrated on two methods: the High Resolution Gamma-Ray Spectrometry (HRGS)-method and the method based on both passive neutron and gross-gamma counting. In the neutron and gross-gamma measurements,

the basic equipment used is the ION-1 equipment developed by LANL. On the basis of the experience obtained, the development of the methods has been continued.

5.6 Control Performed by the IAEA

All contacts relating to the implementation of the safeguards agreement between Finland and the IAEA are arranged through STUK. The control measures of the IAEA are aimed at the state system, which is maintained and developed by STUK. STUK gives assistance in preparing the inspections performed by the IAEA in Finland and a representative of STUK is always present when an inspection is carried out at a plant site. In these inspections, the representative of STUK ensures that the IAEA inspectors receive all the information that is needed in the inspection and specified in the safeguards agreement, and that everything is arranged so that the inspection can be performed smoothly. On the other hand, the representative of STUK sees to it that the inspection complies with the agreements. For the assessment of the statements supplied by the IAEA on the inspections, the representative of STUK writes down all the necessary information of the inspection procedures, for instance, removed and attached seals, verifications, and maintenance safeguards equipment. It is also important for STUK that the inexperience or carelessness of inspectors does not make it necessary to repeat the inspection. For this reason, attention is also paid to the quality of the inspector's work, whenever possible.

Due to the fact that the actions of the inspectors of the IAEA and STUK are mostly the same at the facilities, it is reasonable to carry out the inspection in co-operation. However, one must keep in mind that as regards the results and conclusions of the inspection, the independence shall not be hampered. The co-operation between the Agency and STUK has been running smoothly.

6 SAFEGUARDING THE FUEL CYCLE OF FINNISH LWR'S

6.1 General

One particular feature of nuclear trade is that the supply of fuel services, such as conversion, enrichment, fuel fabrication and reprocessing are confined to few countries. From the point of view of international safeguarding this has positive consequences: the IAEA gets data on the flow of nuclear material through the cycle because of the requirements of NPT countries to supply details of all imports and exports, and the number of sensitive facilities remains low. On the other hand, for a country like Finland which has no facilities of this kind, safeguarding the fuel cycle often means quite much extra work.

Generally all transfers, processing and use of nuclear material take place under the provisions of bilateral agreements between the supplier and customer countries. These agreements provide a detailed set of safeguards obligations and requirements. So there exists additional bilateral safeguards control to the safeguards applied by the IAEA pursuant to the NPT.

The safeguards activities that are dealt with in this chapter involve nuclear materials which are not within the territory of Finland. It is obvious that such safeguards measures as physical inventory verification can not be used. The means of control are accounting and reporting, which have been established among the companies involved in the activities and the respective state authorities. In other respects it is based on reliance that other parties obey agreed procedures.

Supplier countries of nuclear technology have agreed on common criteria, so called London guidelines (INFCIRC 254), which shall be observed when considering the export of nuclear material, equipment or technology to a non nuclear weapon state. These guidelines include i.a. the following principles:

- transfer of Trigger list items should be authorized only upon formal governmental assurances from recipients explicitly excluding uses which would result in any nuclear explosive device,
- all nuclear materials and facilities identified by the agreed Trigger list should be placed under effective physical protection to prevent unauthorized use and handling,
- suppliers should transfer Trigger list items only when covered by IAEA safeguards,
- for a transfer of an enrichment facility, or technology therefore, neither the transferred facility nor any facility based on such technology shall be designed or operated for the production of greater than 20 % enriched uranium without the consent of the supplier nation,
- for retransfers of Trigger list items the same assurances as those required by the original supplier shall be provided by the recipient of retransfer.

In addition to the above mentioned requirements bilateral agreements may include provisions like:

- details of alternative safeguards arrangements which would apply if the IAEA's safeguards ceased to apply,
- provisions for suspension or cancellation of further supply and requirements to return all the supplied nuclear material,

- the termination of agreement does not release the parties from safeguards obligations under the agreement.

6.2 Safeguarding of the Fuel Cycle of Loviisa Units 1 and 2

The fuel of the Loviisa units is imported directly from the Soviet Union as complete fuel assemblies. After a storage time of at least 5 years, the spent fuel will be returned back to the Soviet Union. Spent fuel has been transported from Loviisa to the Soviet Union in the years 1981 and 1982.

While in Finland, the fuel is constantly under the supervision of the IAEA. Besides that, it is subject only to the bilateral agreement between Finland and the Soviet Union.

6.3 Safeguarding of the Fuel Cycle of TVO Units 1 and 2

Phases of Fuel Cycle

The supply of fuel to the TVO units has been organized in a way that is typical to most small countries: the services in the various phases of the cycle have been acquired from different countries. For the TVO units

- the uranium has been originally obtained from Canada, Australia and Niger
- the conversion from U_3O_8 to UF_6 takes place in Canada and in France
- the enrichment takes place in the Soviet Union and in the future also in France
- the fuel is manufactured in Sweden and in the Federal Republic of Germany.

After the spent fuel has been taken from the reactor, it will be stored at the plant site.

Figure 6 shows a scheme of the multinational transfers of uranium for the fuel cycle of TVO nps.

Application of Control

In principle, the control includes the following areas: (1) licensing, (2) accounting, (3) reporting, and (4) physical protection. The control is affected by the provisions of the countries which are involved in the fuel cycle activities. The international agreements of Finland require that the uranium for TVO power plant is under the Finnish control even outside the Finnish jurisdiction.

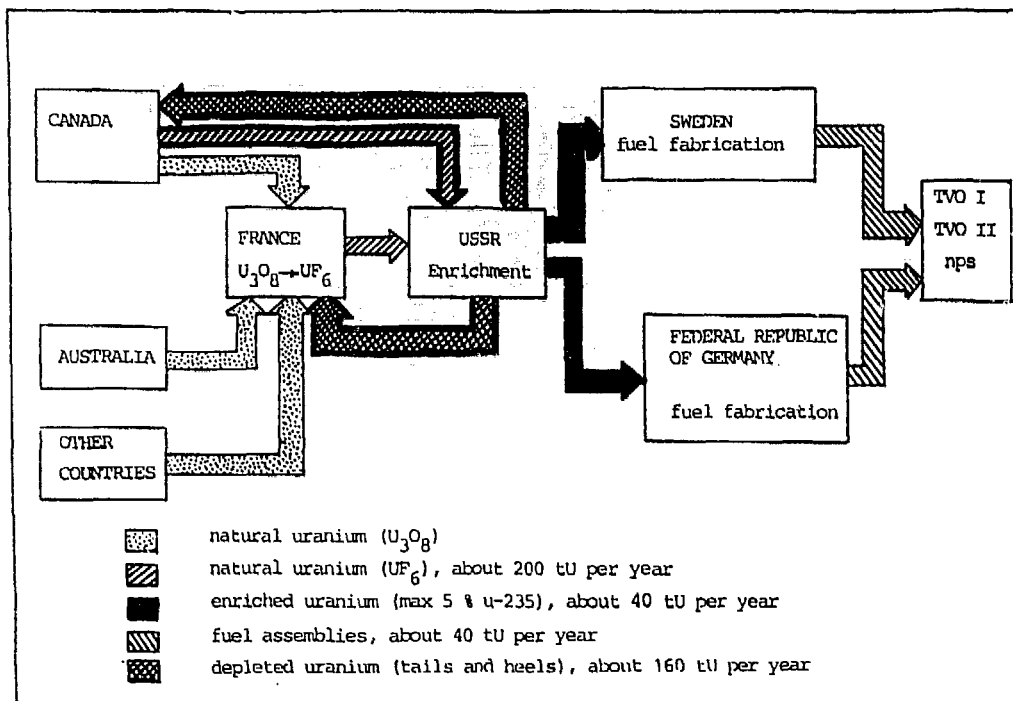


Figure 6. Multinational transfers of uranium for the fuel cycle of TVO nps.

As regards the contents and format of the accounting and reporting the general principles presented by the IAEA in INFCIRC 153 are followed. The basis is formed by the source documents supplied by the shipper with reference to the transport.

The fuel management of the TVO units can be studied in refuelling cycles. In principle, the first thing is then to acquire to the plant site those fuel assemblies which will be loaded into the reactor in the annual refuelling. On the basis of this need, one calculates the amount of raw uranium (yellow cake) that will be needed at the beginning of fuel cycle.

In each period, a detailed plan for the fuel cycle of the next period is made. Before any transports of uranium can be started, these plans must have been approved by STUK. From the point of view of safeguards, STUK compares the planned amounts of uranium to be transported with the need, estimates process losses, and reviews the licences of the power company, the transport routes and schedules, and balances. If there are any special requirements, their fulfilment is also ensured.

At the next stage, STUK asks the Ministry of Trade and Industry to make a request for the necessary international prior consents. After they have been obtained, and if there are no other requirements in the agreements, the plan will be accepted.

TVO shall notify STUK of each transport in advance. This notification is checked and information of the transport will be sent to foreign authorities according to the appropriate agreements.

After the transport has been completed, TVO will submit a confirmation report to STUK. The information contained in this detailed report shall be based on the original source documents provided by the transport companies and the consignor and the consignee. These documents are in the possession of TVO.

In addition, the agreements between Finland and Canada and between Finland and Australia require that the parties send annual reports to each other. In this way the parties ascertain that the safeguards goals have been accomplished in that year.

Special Features of Control

In conformity with their export policies, the supplier countries of uranium generally require that the uranium remains all the time under the supervision of the IAEA, even when it is in a nuclear weapon state, or is subject to some bilateral agreement concluded by the supplier country. As was said above, the uranium of TVO has so far been enriched in the Soviet Union. The Soviet Union is a NPT country, but its agreements with the IAEA do not entitle the Agency to control nuclear materials inside the Soviet Union. Therefore the Canadian and Australian origin uranium is agreed to become subject to the Canada/Finland or Australia/Finland agreement, respectively, before it is taken to the USSR for enrichment. It is also required that the transfers of Australian and Canadian uranium to the USSR take place on the basis of co-called "all in - all out" principle.

In practical safeguards work this means that the nuclear material becomes subject to a bilateral agreement, complying with its provisions, also when it is outside the territory of Finland. So STUK keeps a close watch over the transports and the balances of the TVO uranium also abroad and provide reports to foreign authorities in accordance with the agreements.

The safeguards questions relating to the enrichment of the raw uranium of TVO were discussed comprehensively with various authorities before the enrichment could be started. In the discussions it was concluded that these transports be reported to the IAEA following the principle that nuclear material transfers to and from the Soviet Union should be regarded as transactions through Finland, even if the material did not physically arrive in Finland. For that purpose the fictitious material balance area SF-T was established.

Experience

The fuel cycle of TVO and its safeguards control requires a great deal of international cooperation. This cooperation has been conducted in good atmosphere. The provisions in the bilateral agreements have, however, increased the duties of STUK in safeguards, although in many respects the goals and the practical control measures are the same as in the work performed by the SSAC anyhow.

Below, some of our experiences will be dealt with in more detail.

The countries participating in the fuel cycle of TVO report to the IAEA on transfers into or out of the country in question. Nuclear weapon states report on the basis of voluntary agreements and the non-nuclear weapons states on the basis of the NPT agreements. From the point of view of the safeguards of the IAEA, it is essential that the Agency be informed of the reception of a material consignment that has been sent from one country to another.

After the consignor has informed IAEA of a transport, this information will be kept on the so-called Export Account for so long as the report of the consignee has arrived at the IAEA and its information has been matched with the ICR data of the consignor.

In the Safeguards Implementation Reports (SIR) of the IAEA there are indications about difficulties in matching the accounting reports provided by the shipper and receiver, respectively. It is also reflected in the statements IAEA provides on the Export Accounts in accordance with the safeguards agreement.

Based on our experience the problems seem to be caused mainly by different reporting procedures of the shipper and receiver states and only rarely reports are lacking. So evidently enough information is received by the Agency but not in a proper form to be matched easily.

The accounts and reports of STUK are based on source documents of the consignee and on the confirmations given by the appropriate foreign authorities. The inspections of STUK have revealed no unclear events.

The matching of reports concerning international transfers would be facilitated if the transfers are reported according to the real physical transactions.

Furthermore, it does not improve the credibility of international safeguards, if the Export Accounts gather much data on transports that in reality have involved no difficulties.

The transit control of international transports is a very important safeguards objective from the standpoint of operators and both national and international safeguards. With the purpose of standardizing the reporting on international transports, actions have been taken both at bilateral and at multilateral levels, also in the IAEA. The IAEA has organized consultant meetings on this question in 1981 and 1984. As a result of these meetings the IAEA has given recommendations for reporting procedures. STUK has complied with the recommendations whenever applicable.

The experiences of the exchange of safeguards information between STUK and the parties of the bilateral agreements are good: the exchange of information has been carried out in the spirit of friendly cooperation.

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INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 24: IAEA SAFEGUARDS EXPERIENCE AT LWR POWER REACTORS

The LWR power reactors under IAEA safeguards will be described briefly with emphasis on the types and quantities of nuclear material found at these facilities. The nature and requirements of a typical facility attachment will be presented. Records and reporting requirements will be discussed. IAEA inspection procedures and measurement instrumentation will be described in detail. The analysis of inspection data to verify the nuclear material inventory will be considered. The session will emphasize actual experience at LWR facilities under IAEA safeguards.

After the session, participants will be able to:

- Describe the major activities during IAEA inspection of LWR facilities.
- Describe the records and reports prepared by the facility operator for the IAEA.
- Discuss the inspection procedures and measurement instrumentation used during IAEA inspections at LWR facilities.
- Understand how the IAEA activities help to meet the safeguards goals of detection and deterrence.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 24: IAEA SAFEGUARDS EXPERIENCE AT LWR POWER REACTORS

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I. Introduction

It is worthwhile to define first of all the scope and the intention of this presentation, in other words to identify the relative importance of the particular type of power reactors which we are dealing with, and secondly to outline the content and structure of this presentation.

I intend to proceed as follows:

- I. Certain introductory remarks
- II. SG objectives, political and technical
- III. Basic facility characteristics; evolution of SG approach
- IV. Inspection procedures; SG measures; verification techniques
- V. SIR criteria and inspection activities; SG products

In order to illustrate the scope of our task let me begin with a few statistical data. According to Agency's data as of end of 1984 there was a total of 464 nuclear installations (not including so-called "other locations") under IAEA safeguards; this number included 166 power reactors, i.e., 36% of the total number of nuclear installations, and approximately 33% of the overall inspection effort (man-days) was spent there [Fig. 1].

A further differentiation may be made. Out of the 166 power reactors there were 128 light-water reactors, or 77% of the total number. Out of the 128 LWRs there were 109 covered by an NPT-type agreement, and only 19 by a non-NPT type agreement (INFCIRC/66 etc.).

These statistical data can be found for example in the 1984 SIR. The data show that installation category "A", namely power reactors, is from a SG point of view about the most significant one out of 10 such categories. The IAEA has accumulated most of its safeguards experience with light-water power reactors.

We are not discussing in detail any differences between BWRs or PWRs or LWRs using MOX fuel. The basic SG approach will be the same for the different LWR types; any variations regarding SG implementation are rather due to the relative intensity than due to actual differences in safeguards activities.

All safeguards agreements concluded between the Agency and a state (or a group of states) can be traced back to two major types of documents, the so-called INFCIRC/153 (blue book) describing the structure of NPT-type agreements, and INFCIRC/66/Rev. 2, "The Agency's Safeguards System." For the pur-

pose of this presentation we will not describe different safeguards concepts and techniques for different types of SG agreements, even though under INFCIRC/66 it is not only the nuclear material but also the facility or specific items which may be subject to SG.

It may be worthwhile to draw your attention to the SIR criteria as used by the Agency for evaluation of inspection goal attainment in connection with the preparation of the annual SIR, the Safeguards Implementation Report. I will review the basic terms referring to political and technical SG objectives and then after having described inspection activities look into their compatibility with those SIR criteria.

Finally then we will have to turn to the Agency's products, namely the content of statements which are formally sent to the government of the relevant state, and secondly the SIR which is prepared annually and submitted to the Agency's Board of Governors for consideration.

II. SG-Objectives

For reasons of clarity I would like to refer you to Fig. 2 which helps to visualize the interrelationship of terms regarding concepts and methodology in safeguards.

Para. 46 of INFCIRC/66 reads:

"The purpose of safeguards inspection shall be to verify compliance with safeguards agreements and to assist states in complying with such agreements and in resolving any questions arising out of the implementation of safeguards."

Para. 2 of INFCIRC/153 reads:

SG to be applied to nuclear material "...for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other nuclear explosive devices."

This means that the IAEA is given the role of a verification system by which assurance of compliance with certain commitments can be given to the world community as a kind of confidence-building measure. A state concluding a SG agreement with the IAEA makes the commitment of not misusing nuclear material or equipment.

Under 66-type agreements "nuclear and other materials, services, equipment, facilities and information" may be subject to SG, whereas the 153-type regime will put the emphasis on the nuclear material (NM), however all NM in all peaceful activities. "Special Procedures for Reactors" are contained in INFCIRC/66 as extended in 1966 and 1968.

The legalistic provisions in an agreement are not sufficient for a practical implementation of such a verification task; Subsidiary Arrangements to the Agreement and facility-specific Facility Attachments are prepared; these Facility Attachments (FAs) provide the necessary infrastructure (MBAs, KMPs, records, reports, etc.) to satisfy the operational aspects of

IAEA safeguards. We will come back to the structure and content of FAs.

We interpret the purpose of SG as the "political objectives"; the mandate for verification implies however a technical task and the underlying "technical objectives" can be found in either type of agreement in one way or the other. The various numerical parameters such as "significant quantity" (SQ), "detection time", "detection probability", and "false alarm probability" constitute the so-called detection goals.

Our SG efforts are aimed at detecting whether 1 SQ (or more) of nuclear material had been diverted; the values for 1 SQ had been set at 8 kg of plutonium or 75 kg contained U-235 in LEU or natural U or 20 tons of thorium.

We want to be able to detect diversion within a time period equivalent to or smaller than the time necessary to convert different forms of nuclear material to the metallic components of a nuclear explosive device; time settings range from 3 weeks for HEU or separated Pu to 3 months for Pu in irradiated spent-fuel or 1 year for fresh LEU fuel. Finally, scope and degree of available SG measures are planned according to common statistical practice whereby a detection probability of 90-95% and a false alarm probability of less than 5% are assumed.

The key issue of any SG agreement is the IAEA's right for verification; the available measures to put this right into effect are nuclear material accountancy (NMA) as a SG measure of fundamental importance, and containment and surveillance (C/S) as important complimentary measures. The sequence of actions to be taken by the IAEA to prepare for SG implementation would be firstly to evaluate Design Information, secondly identify potential diversion strategies, concealment methods, and related anomalies, and thirdly design a SG approach.

III. SG-relevant Features/SG Approach

The LWR reactors are off-load fueled power reactors which definitely favours their safeguardability. In the IAEA's Glossary one reads: "(LWR) a reactor fueled while it is shut down and the generator disconnected from the power grid. The existing off-load fueled power reactors are item facilities."

The nuclear material is present in form of sealed sources (fuel rods or fuel assemblies) and a pure item accountancy system becomes applicable. The only and relatively simple task is then to verify the presence of these items and to verify their integrity. For obvious reasons the continuity of knowledge of this item integrity is desirable.

The off-load feature means that the total nuclear material inventory only becomes accessible for IAEA verification at the time of refueling, refueling intervals ranging from approx. 12 to 18 months. During refueling however there is the maximum movement of nuclear material; there may even be a complete core unloading into the spent-fuel (SF) pond (e.g., maintenance and safety inspection) but at least part of core fuel (CF) is being finally discharged to the spent-fuel storage pond. Fresh-fuel (FF) has to be moved from a usually dry fresh-fuel storage or the SF pond to the reactor core.

The amount and composition of nuclear material in FF as discrete items is measured at the fuel fabrication plant and ought to be verified there. During the residence time at the reactor, amount and composition of the fuel change and can only be verified by NDA and assessed by calculation, which is no problem as long as the integrity of the fuel assemblies is maintained. The design of a LWR usually offers favourable conditions for successful application of C/S measures.

Light-water reactors are generally fueled with low-enriched uranium (LEU) of about 3 wt% U-235. There are mainly gradual differences between BWRs and PWRs such that there are smaller items and more items at a BWR; a potentially unpleasant feature may be the presence of rods with different enrichments in one assembly (BWR). The LEU is in the form of oxide pellets encapsulated in Zirconium alloy rods, a varying number of which are arranged in fuel assemblies. A few typical fuel data for BWR, PWR, and WWER are presented in Fig. 3, according to which 6 to 7 PWR assemblies contain about 1 SQ of U-235 (1 SQ of LEU = 75 kg contained U-235).

LWRs may also operate with a core partially filled with mixed plutonium-uranium oxide fuel (MOX), either for experimental purposes or because of a regular "plutonium recycle" operation mode. There will be no assemblies containing only MOX fuel but assemblies with some fuel rods containing MOX. The safeguards effort in such a case becomes significantly greater due to the associated higher strategic value and the much shorter (3 weeks compared to 1 year for LEU) detection time.

It should be observed that some power reactors (particularly BWRs) have the option of exchanging individual fuel rods in an assembly, either to simply replace leaking or otherwise damaged fuel rods or to reconstitute fuel assemblies with pins having less burnup such that the assembly can be returned to the core for reasons of fuel economy. This practice, called rebatching, is of some SG concern, firstly since it may open a relatively easy diversion path for fuel rods and it may be difficult to distinguish on basis of optical surveillance devices between legitimate and necessary activities associated with fuel rods exchange and activities associated with diversion; secondly rebatching complicates the straightforward item accountancy concept.

In Fig. 4 you will find a list of diversion possibilities, concealment methods, related or observable anomalies, and also possible IAEA inspection activities to detect the act of diversion or the results of it, i.e., the related anomalies; this list is not exhaustive. The interested reader must be referred to the enormous amount of literature on the subject "diversion analysis". In a simple form one could say diversion means physical removal of either fresh-fuel (FF) or spent-fuel (CF or SF) or undeclared plutonium production (diversion of neutrons).

Some basic reactor data for BWR, PWR, and WWER-440 are compiled in Fig. 5 which may be useful during further discussion.

During normal operation of a reactor the nuclear material outside the core is present mostly as irradiated spent-fuel in the SF storage pond. For mainly economical reasons the operator tries to avoid having a significant number of FF assemblies stored on site, and he will schedule FF receipts for shortly before refueling. About one-fifth to one-third of the CF assemblies are then being replaced by FF. The timeliness goal for LEU fresh-fuel is

1 year and therefore the verification annually and preferably during reload is sufficient.

One SQ of LEU FF corresponds to 2500 kg (at 3%) or 3750 kg (at 2%) of total uranium; accordingly the FF added during refueling is approximately equivalent to 6.6 SQs for a BWR, 9.9 SQs for a PWR, and 8 SQs for a WWER-440. This "significant turnover" of material is a big challenge for IAEA SG and verification measures beyond mere item identification may be needed to protect against substitution of LEU by lower value material.

From Fig. 5 it is obvious that, measured in SQs, the nuclear material inventory removed from the core as SF is considerably larger than the FF inventory added. IAEA safeguards activities in between refueling operations undoubtedly focus on the SF storage pond. The requirements for timely detection are 3 months for irradiated fuel due to the presence of plutonium. In practice this means that the results of surveillance film evaluation must be available not later than about 2 weeks after the last event was recorded by the surveillance device. I was assuming an average 2 1/2 months' surveillance period. Inspections must be planned such that within the timeliness goal (3 months) the continued presence and the integrity of SF can be confirmed.

The "diversion of neutrons" mentioned above can be achieved by introducing into the reactor either uranium or thorium as "fertile" material and producing through irradiation plutonium or uranium-233 as fissile material. Relevant studies carried out by the IAEA have shown that it is more rewarding to apply this diversion scenario at a PWR than a BWR; PWR assemblies have vacancies (water rods) into which target rods with fertile material could be introduced. It has also been concluded that it would be impractical to use ex-core locations due to the neutron-flux levels being several orders of magnitude below in-core levels. Anyhow, a potential diverter's effort and risk to be detected have to be seen in relation to the target amount of approximately 8 kg of Pu. The attractiveness of this diversion scenario is questionable but eventually difficult to detect and consequently SG measures will have to attempt to keep any undeclared produced fissile material contained and inaccessible for undeclared fuel reprocessing which would be the next step.

The IAEA's SG approach for LWRs must be designed to ensure that verification activities are capable of leading (with a high probability) to the timely detection of anomalies resulting from any diversion attempt. This is achieved by nuclear material accountancy (NMA) plus C/S measures.

NMA at a LWR means item accountancy, i.e., item counting and identification and NDA assay measurements in addition to examination of records and reports. Containment is automatically provided when fuel is loaded into the reactor core. Surveillance refers to both human and instrumental observation. C/S measures help to preserve the knowledge of the inventory once verified, to monitor the flow of material, or simply to assure the integrity of containment.

LWRs are therefore inspected 4 to 6 times per year. One of these inspections normally coincides with annual core refueling and involves a complete physical inventory verification (PIV) of fresh-fuel, fuel in the core, and irradiated fuel in the storage pond. Interim inspections every 2 to 3 months are carried out for records examination, seals checking, servicing of surveillance equipment, and eventually verification and sealing of spent-fuel shipment.

IV. Inspection Activities, Verification Procedures

The "legal instrument" and the "infrastructure" which are required for IAEA's inspection activities are provided by the Facility Attachment (FA). The basis for negotiating an FA between the IAEA and a state is the Design Information (DI) provided by the latter for a specific facility. For a course like this which deals with SSAC requirements it is, I believe, essential to be aware of the provisions contained in an FA; Fig. 6 is basically a "table of contents" of an NPT-type LWR FA with the possibility for assembly dismantling and pin-exchange (66-type FAs are similar). Very briefly the most important items should be addressed (the numbers identifying codes in the FA):

- 3.1.1. an LWR constitutes 1 MBA only;
- 3.1.2. defines KMPs for flow and inventory verification;
- 3.1.3. PIT by operator on refueling or at least every 12 months;
- 5.1.1. NL and NP to be recorded asap after discharge, in any case within 12 months, calculated as of the date of discharge;
- 5.2.1. operating records (location of items, power log, burnup, periods of shutdown, operation log of fuel movement equipment, dismantling operations, etc.);
- 6. timing and format for ICRs, concise notes, MBRs, PILs;
- 7.3. [ARIE] inspection effort estimated to be spent (under certain circumstances);
- 7.4 this describes the scope of and the methods used for the 4 major activities:
 - 7.4.1. 1. auditing
 - 7.4.2. 2. verification of inventory
 - 7.4.3. 3. confirmation of flow
 - 7.4.4. 4. application of C/S;
- 7.9. (IAEA occasionally encountered problems with) health and safety rules;
- 8. (the SG "product") IAEA statements;
 - 8.1. 1. on results of inspections;
 - 8.2. 2. on conclusions drawn from verifications.

The "auditing activities" (see FA Code 7.4.1.) comprise examinations of the facility records, verification for self-consistency, and comparison of facility records and reports submitted by the state; they are carried out routinely during any inspection. The facility operator is expected (and obliged) to keep a set of records, indicating for example:

- number of FF assemblies received;
- number of SF assemblies shipped;
- number of internal transfers;
- periods of shutdowns;
- core maps;
- integrated power curve;
- operation log of major equipment;
- records for burnup and nuclear production;
- assembly dismantling, pin-exchange.

Usually a great variety of accounting records and particularly operating records is available. The inspector has to check for arithmetical cor-

rectness and consistency of different records. It must be possible to establish a book inventory total and reconcile it with the physical inventory totals. A certain degree of transparency and standardization of records is in the interest of both parties; for example records should reflect material categories and number of items (batches) should be given in addition to amounts of nuclear material.

Due to the usual delays in submitting state reports to the IAEA, the comparison of such reports with records can normally only be done for the previous inspection period, i.e., up to 2 or 3 months preceeding the date of inspection. In cases where the inspector can obtain copies of the relevant facility records he can do records/reports comparison at IAEA headquarters.

The "verification of inventory" (see FA Code 7.4.2.) is to be achieved by means of item counting, identification and integrity checks, NDA measurements, and the use of seals. After refueling the facility operator carries out a PIT followed by the IAEA's PIV before the reactor vessel is closed.

FF is assumed to be verified by variables methods at the fuel fabrication plant. At the reactor LEU FF is subject to checking assembly identification numbers, supported by a simple confirmatory (attributes) measurement, e.g., NDA of outer rods enrichment with the new portable Mini-MCA (Ge-detector plus microprocessor assisted MCA). Reliable seals that can help preserve the once-verified integrity of an FF assembly during its whole lifetime from the fuel fabrication plant through the reactor up to the reprocessing plant are not yet available. Development work is under way with ultrasonic seals which would be interrogatable and verifiable in situ; such a system would make FF-variable measurements at a reactor unnecessary.

Presently FF is either verified at the fabrication plant and sealed in shipping containers, or verified at the reactor and put under surveillance. The instrument used is the Uranium Neutron Coincidence Collar (UNCL); it can measure the U-235 content in LWR assemblies to within 1%, and (in the passive mode) confirm U-238 content to within approx. 10%. This instrument is sufficiently sensitive to detect the removal or substitution of 2-3 rods in a fresh PWR assembly and one rod in BWR assembly. Field tests demonstrated the relative non-intrusiveness of the UNCL, and the combined active and passive measurement modes giving U-235 and U-238 content make this verification rather tamper-resistant.

There are currently four LWR facilities under IAEA safeguards with FF inventories including recycled plutonium (nominally 2-4% Pu) in form of MOX fuel. From a safeguards point of view (strategic value of material, SQ, short detection time) the MOX FF constitutes the most important part of the inventory and accordingly requires a high degree of inspection effort. A typical PWR assembly with nominally 2-4% Pu loading may contain 10-20 kg of Pu. Again, the verification is ideally done at the fabrication plant, followed by C/S measures to preserve continuity of knowledge through to the reactor site. NDA methods basically use neutron coincidence counting techniques and gamma spectroscopy. The latest accomplishment seems to be the Universal FBR Assembly Counter (UFBC). For details I must however refer to the relevant literature.

The inventory of SF in the on-site storage pond is verified by item counting plus identification of serial numbers and checking the location of items against the Spent-Fuel Pond records. Qualitative NDA measurements may

be used, but are not yet general practice; the best currently available qualitative NDA test is the observation of the Cerenkov radiation. Practically no technique is presently available to measure directly the Pu content in SF. The ION-1, now authorized for limited use, employs fission chambers to measure gross neutron emission and ion chambers to measure gross gamma emission (detector under water). These "hybrid gamma/neutron" measurements can be correlated with operator's declared burnup and cooling time (time since discharge) data if available, i.e., consistency checks within a population of items become possible. If continuity of surveillance is lost between inspections then an SF inventory reverification may become necessary; otherwise only item counting is usually performed.

The verification of CF inventory is basically limited to item counting and eventually item identification or in other words checking of the core mapping as prepared by the operator. This activity requires underwater viewing devices which are mostly available on-site anyhow; subsequently surveillance is used to monitor activities in the core from the time of verification until it is resealed by the Agency.

"Verification of flow" (or better, "confirmation") refers to FF receipts and is done in connection with PIV during refueling (det. time for LEU = 1 year). Verification (LEU or MOX) is usually done at the fabrication plant, such that only a confirmation of receipts at the reactor (detach and verify seals) is needed and subsequent application of C/S measures.

If there are shipments of SF then in particular "partially filled" shipping containers should be verified and sealed, since otherwise the removal of undeclared material (in the context of undeclared Pu production) becomes possible. In this context it should be mentioned that to the ARIE figure of normally 14-16 MDI/a additional man-days are to be added to cover loading and/or sealing of partially filled SF shipping containers. The same applies to re-establishment of inventory after loss of continuity of surveillance.

The "application of C/S measures" (see Code 7.4.4.) at "strategic points" provides for:

- application, examination, and removal of IAEA seals;
- servicing C/S instruments and devices.

This activity is probably the most valuable and rewarding one at an LWR; the main areas of application are:

- sealing of the reactor biological or missile shield;
- automatic surveillance systems at the SF pond;
- surveillance of loading and/or sealing of SF containers.

The systems currently used are mainly double cap metal seals (CAPS). They can be examined on the spot. However for verification at HQ they need to be detached. Fibre optic seals (FBOS) and Vacoss electronic seals (VCOS) are still under field test; they are in situ verifiable and eventually remotely verifiable which would be important to cover non-access situations, such as seals inside the containment during reactor operation.

The surveillance systems in use are mainly photo-surveillance systems (PHSR) such as twin Minox, and in some cases TV-based surveillance systems.

The maximum capacity of the Minolta is about 3 months at around 20-minute picture taking interval, which is compatible with timeliness requirements. The Compact TV Surveillance Unit (CTVS) has been recommended for use inside containment during refueling to monitor core activities between IAEA seal removal and closing and resealing the vessel.

The evaluation of surveillance films is usually done at the IAEA HQ in Vienna. For the inspector it is essential to have a complete facility record against which surveillance can be evaluated; this need is obvious in case of SF container movements, but detailed records are also required where suspicious structure, containers, or objects capable of containing SF assemblies or rods are either present in the SF pond area or are being moved. It is important that an operator recognizes this need for adequate operating records.

V. SIR Criteria vs Inspection Activities and Statements

The IAEA prepares annually the Safeguards Implementation Report (SIR) which is an evaluation of the attainment of inspection goals. Criteria have been set in order to carry out this evaluation.

We will recall that the inspection goals for LWRs are:

- detect diversion of 1 SQ of nuclear material;
- within 4 weeks for fresh MOX fuel;
- within 3 months for irradiated fuel;
- within 12 months for fresh LEU fuel.

The SIR criteria specify:

- the "minimum probability" with which the absence of nuclear material is to be detected (quality criteria)
- the "minimum frequency" with which activities must be carried out (timeliness criteria)

In general the SIR criteria define quality and frequency with which inspection activities need to be carried out; they represent the currently achievable inspection performance, and so far they had been changed annually.

It should be noted that in case of LWRs there are some differences between these SIR criteria and those inspection activities (see codes 7.4.1 to 7.4.4) which are understood to reflect a "good safeguards"-type approach. For example the SIR criteria require only item counting and identification as minimum verification activities to attain the quality goal, instead of qualitative or quantitative NDA; also, the minimum detection probability is 50% compared to usually 95% for fresh-fuel and spent-fuel. And further, the timeliness goal may be attained if the various fuel categories are under C/S and the success of C/S measures can be confirmed according to average timeliness goals.

Summary: When the integrity of any prior "high quality verification" can be preserved through successful containment/surveillance, the inspection goal can be achieved by carrying out less frequent verifications of a lower quality (less rigorous)!

It is conceivable that under these circumstances inspection planning becomes a bit problematic; it may seem more important to accomplish a good mark in the SIR rather than "good, old fashioned" safeguards. This latter statement is on purpose exaggerating to make the dilemma more obvious.

The end "product" of IAEA SG is (apart from the SIR) a statement; para. 254 of the IAEA-Glossary reads:

"Statement, an official communication by the Agency to a state, indicating the results of an inspection carried out in the state or the conclusions the Agency has drawn from its verification activities."

There are 2 types of statements: a "summary statement" is made on the results of each inspection within 30 days; a "conclusion statement" has to be made only after a PIV, i.e., on the occasion of closing the material balance. They should address (see Code 8 of Facility Attachment) the following items:

1. records examination; quality;
2. reports to the Agency; quality and timeliness;
3. C/S results;
4. flow verification;
5. verification of material accountancy;
6. quality of operator's measurement system;
7. activities regarding MUF, SRDs, losses.

Generally, conclusion statements for item-accountancy facilities are much easier to prepare than for bulk-handling facilities; reactor facilities require major effort on C/S evaluation and do not require an elaborate statistical material balance evaluation as bulk facilities do.

REFERENCE MATERIAL

IAEA Safeguards Equipment Handbook.

ISPO-190, Nuclear Power Plant Characteristics.

ISPO-238, Relationship between SIR-Criteria and Inspection Activities.

STR-123, Safeguarding Light Water Reactors.

STR-80, Safeguarding Light Water Reactors.

Draft SIR for 1984.

IAEA/SG/INF/6, IAEA Safeguards-Implementation at Nuclear Fuel Cycle Facilities.

IAEA/SG/INF/1, IAEA Safeguards Glossary.

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Fig. 1. Statistical Data.

2. Concepts and Methodology in Safeguards.
3. LWR Nuclear Specifications.
4. Example - Diversion Analysis for LWR.
5. Typical Reactor Data for BWR, PWR, WWER440.
6. Structure and Content of a Facility Attachment.

| F I G. 1. |

I A E A A C T I V I T I E S - S T A T I S T I C A L D A T A

A.	Number of installations		
	Installation category	INFCIRC/153	INFCIRC/66/Rev.2 Total
	A. Power reactors	140	26 166
	B. Research reactors and critical assemblies	148	26 174
	C. Conversion plants	4	2 6
	D. Fuel fabrication plants	29	9 38
	E. Reprocessing plants	4	2 6
	F. Enrichment plants	4	0 4
	G. Separate storage facilities	27	2 29
	H. Other facilities	39	1 40
	I. Other locations	389	27 416
	J. Non-nuclear installations	0	1 1
	<u>Totals</u>	<u>784</u>	<u>96 880</u>

B.	a) Total Installations	==	464	----	6500 Man-Days
	"		(=100%)		(=100%)
	"				
	b) Power Reactors	==	166	----	2100 Man-Days
	"		(= 36%)		(= 33%)
	"				
	c) Light-Water Reactors	==	128		
			(= 77% of b.)		
			"		
			"		
			NPT	==	109
			non-NPT	==	19

FIG. 2.

SAFEGUARDS CONCEPTS AND METHODOLOGY

CONCEPTS:

1) Purpose of SG	- assurance of non-diversion - deterrence	} "political objective"
2) Scope of SG	- INFCIRC/66/Rev.2 - INFCIRC/153	
3) Concept of Verification	- diversion hypothesis - concealment methods - observable anomalies	
4) Detection Goals	- significant quantity (SQ) - timely detection - detection probability	} "technical objectives"
5) Inspection Goals	- accountancy verification goal (AVG) - timeliness goal (range 4 weeks to 12 months) - credibility - practicality - efficiency - equability	} "performance objectives"

METHODOLOGY:

- a) Safeguards measures
 - nuclear material accountancy (NMA)
 - containment and surveillance (C/S)
- b) Diversion analysis
 - diversion possibilities
 - concealment methods
 - anomalies
 - design SG approach

in context of design review and subsequent design information verification
- c) Performance of inspections
- d) Safeguards effectiveness evaluation

| F I G. 3. |

L W R N U C L E A R S P E C I F I C A T I O N S

<u>Specification</u>	<u>BWR</u>	<u>PWR</u>	<u>WWR-440</u>
Conversion Ratio	0.6	0.6	n.a.
Burnup (Mwd/t)	26 000	30 000	28 000
U-235 per assembly (kg)	4.3	12	4
U-235 per fuel rod (g)	85	60	30
Pu per spent-fuel assembly (kg)	2	4	1.2
Pu per spent-fuel rod (g)	40	20	10

EXAMPLE, DIVERSION ANALYSIS FOR LWR

Diversion	Concealment methods	Anomalies	Inspection activities
This row applies to all the diversions listed below.	Falsification of documents	Inconsistencies in documents	NMA, Acc. records examination, inventory verification
1. Removal of fresh-fuel assemblies from storage	Substitution with dummies	Incorrect composition and/or enrichment	NMA, inventory verification, NDA measurements
2. Removal of irradiated fuel assemblies from the core	Substitution with fresh-fuel	C/S devices indicate anomaly	C/S; ASVS during reloading, C/S; Sealing of reactor vessel
		Incorrect composition or irradiation status of fuel	NMA, NDA, inventory verification
3. Unreported irradiation of fertile material (Pu rods production)	Replacement of guide tube fillers with U-containing material	Unreported shutdowns, presence of unreported fertile material (low enriched or natural uranium in fuel rods)	as in case 2
4. Removal of spent-fuel assemblies from storage pond	Substitution with dummies	C/S devices indicate anomaly	C/S; ASVS
		Incorrect composition or irradiation status of fuel	NMA; NDA, inventory verification
	Substitution with borrowed fuel	Assemblies missing in another MBA	Simultaneous inspection
5. Removal of spent-fuel from container (or during transit)	Substitution with dummies	Incorrect loading	C/S 1.11 - Observation of loading or verification plus sealing of partially filled container

| F I G. 5. |

T Y P I C A L R E A C T O R D A T A

	B W R	P W R	WWR-440
Gross thermal power	1380 MW	2200 MW	1375 MW
Net electrical power	439 MW	700 MW	413 MW
Conversion ratio	0.375 (initial)	0.6	Not available
Fuel assemblies	440	157	349 hexagonal
Core loading at power	78 200 kg U (2.5% avg.enr.)	70 100 kg U (3.1% avg.enr.)	42 000 kg U (3.5% avg.enr.)
Burnup at equilibrium	22 000 Mwd/t	31 500 Mwd/t	28 600 Mwd/t avg.
Refueling interval	1/4 core annually	11 months	1/3 core after 7000 hrs
Rods per assembly	49 in 7 x 7 array	204 in 15 x 15 array	126 in triangular array
Assemblies added at reload	100 (20 000 kg U, 6.6 SQs)	53 (24 000 kg U, 9.9 SQs)	150 (18 000 kg U, 8 SQs)
Rate of plutonium buildup as spent-fuel	160 kg/yr (<u>20 SQ</u>)	220 kg/yr (<u>27 SQs</u>)	140 kg/yr (<u>17 SQs</u>)
Control	97 cruciform control rods. Control curtains in 1st core.	53 control rod clusters, 45 full length (3.61 meter AL), 8 part length (91 cm AL), 20 rods per cluster of 80% Ag, 15% In and 5% Cd alloy in 304 SS cladding. 816 burnable poison rods, made of borosilicate glass in SS tubes. One instrumentation sheath per assembly.	73 of 349 positions are hexagonal absorber tubes, 250 cm active length. Water traps for fast neutron inside. Burnable poison in assemblies. Boron in the primary coolant to compensate for burnup and temperature effects of reactivity.

| FIG. 6. |

STRUCTURE OF FACILITY ATTACHMENTS

<u>Code No.</u>	<u>Description of content</u>
1.	<u>Identification of facility</u>
1.4	Description
2.	<u>Information on the facility</u>
2.2	Changes in the information on the facility to be provided in advance
3.	<u>Safeguards measures</u>
3.1	Accountancy
3.1.1	MBAs
3.1.2	KMPs
3.1.3	PIT
3.2	Containment and Surveillance
4.	<u>KMP specifications</u>
4.1	Flow KMPs
4.2	Inventory KMPs
5.	<u>Records system</u>
5.1	Provisions for accounting records
5.2	Provisions for operating records
6.	<u>Reports system</u>
6.1	Provisions for ICRs
6.2	Provisions for concise notes
6.3	Provisions for MBRs
6.4	Special reports
7.	<u>Inspections</u>
7.1	Mode of routine inspections
7.2	MRIE
7.3	ARIE
7.4	Scope of inspections
7.4.1	Examination of records; records vs. reports comparison
7.4.2	Verification of the inventory
7.4.3	Verification of the flow
7.4.4	Application, examination, removal of seals; servicing C/S devices
7.4.5	Interim inventory verification
7.5	Use by Agency of equipment for independent measurements
7.6	Persons to whom request for services to be addressed
7.7	Contacts at the facility
7.8	Services and/or charges
7.9	Health and safety rules
8.	<u>Agency statements</u>
8.1	Summary statement
8.2	Conclusion statement



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 25: EXPERIENCE AND PROBLEMS SAFEGUARDING REACTOR FACILITIES

Participants and panel members will be divided into five subgroups. Each subgroup will develop a list of questions for the panel. Then, following a short break, the panel will be convened to respond to subgroup questions and questions from the floor. Discussion will focus on the experience of the IAEA, facility operators, and state authorities safeguarding reactor facilities. Material from Sessions 18 through 24 will provide the basis for the questions and discussions. Full participation by course attendees will be encouraged.

Willi Theis, IAEA, leads subgroup discussion.



Lars Wredberg (back to camera) leads subgroup discussion.



Panel Members: Left to Right-
Lars Wredberg, IAEA; Willie Theis,
IAEA; Leroy Norderhaug, NRC;
Dick Schneider, EXXON; and
Steve Stilwagen, SONGS.





INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 26: INTRODUCTION TO SAN ONOFRE NUCLEAR GENERATING STATION

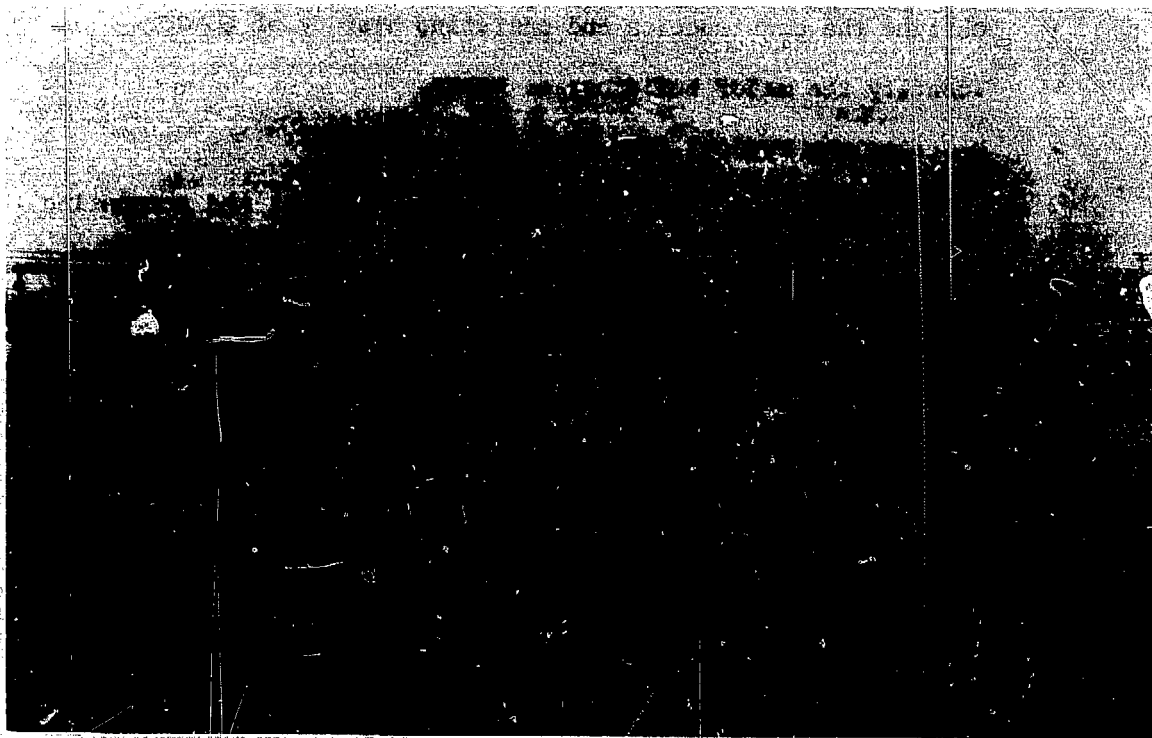
The history and major features of the San Onofre Nuclear Generating Station (SONGS) will be described. Reactor unit 2 is now under IAEA safeguards. A major goal of this session will be to give participants sufficient knowledge of SONGS to provide a background for the plant tours and the description of the nuclear material control and accounting system to be given in Session 27.

After the session, participants will be able to:

- Describe the reactor types and generating capacities at SONGS.
- Outline the construction and operating history of SONGS.
- Identity the major buildings at SONGS.
- Describe the typical fuel inventory at SONGS.
- Better understand the station's material control and accounting system which will be described in Session 27.
- Better appreciate the plant tours that will be conducted during Sessions 28 and 29.



SSAC course headquarters in California; Marina Inn, Dana Point, California.



SONGS units 2 and 3 viewed from Training Center across Interstate Highway 5.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 26: INTRODUCTION TO SAN ONOFRE GENERATING STATION

E. Arnold Hakkila
Los Alamos National Laboratory

The San Onofre Nuclear Generating Station (SONGS), operated by Southern California Edison Company (SCE), consists of two 1100-megawatt pressurized water reactors supplied by Combustion Engineering and one 436-megawatt electric pressurized water reactor supplied by Westinghouse. All three reactors are operating at full capacity. SONGS is located just south of San Clemente, California, on the Camp Pendleton Marine Base. The generating station is built next to a beach that is part of the California State Park System. Surfers continuously use the beach and ocean in front of the plant. All three reactors draw cooling water from the ocean. A briefing by SONGS Engineering and Operations personnel covering plant characteristics, fuel management, and safeguards introduced the attendees to the facility.

Unit 2, one of the Combustion Engineering supplied reactors, is under IAEA safeguards. Because of the experience of the SONGS staff with IAEA safeguards procedures and the knowledge of the IAEA inspectors with the SONGS facility, this unit was selected as a case study for the 1985 course.

TECHNICAL DATA ABOUT SAN ONOFRE NUCLEAR GENERATING STATION UNITS 1, 2 AND 3

	UNIT 1	UNIT 2 (Data is identical for Unit 3)		UNIT 1	UNIT 2 (Data is identical for Unit 3)
CONTAINMENT STRUCTURE					
Houses reactor and reactor coolant system					
Inside diameter:	140 ft.	150 ft.	Primary Coolant		
Inside height:	140 ft.	170 ft.	Coolant flow rate	78,000,000 lb/hr	148,000,000 lb/hr.
Containment wall thickness:			Inlet temperature	553°F	553°F
Concrete:	3 ft. at sides 1½ ft. at roof (shielding)	4½ ft. at sides 3¾ ft. at dome (containment)	Outlet temperature:	597°F	611°F
Steel:	1 in (containment)	¼ in (liner plate)	Operating pressure:	2.085 psig	2.235 psig
Nominal internal net free volume:	1,210,000 ft³	2,305,000 ft³	Water and Fuel Volume		
Containment designed for internal pressure of:	46.4 psig	60 psig	Volume of water:	666,100 in³	1,159,236 in³
			Volume of fuel:	387,500 in³	597,544 in³
			Ratio-water/fuel (cold):	1.72	1.94
REACTOR					
Reactor Vessel					
Height:	37 ft	43 ft 6 in.	STEAM AND POWER CONVERSION SYSTEM		
Diameter (inside):	11 ft 10 in	14 ft 4 in	Steam Generator		
Weight (dry):	348 tons	452 tons	Type	Vertical U tube with integral steam separator	Vertical U tube with integral steam separator
Core			Capacity	1,902,000 lb/hr each with 25% moisture at 710 psia	7,565,000 lb/hr each with 25% moisture at 900 psia
Fuel height:	120 in.	150 in	Number	3	2
Equivalent diameter:	111 in.	136 in	Steam Turbine		
Total weight:	145,800 lb. UO ₂ (58 MTU) (equivalent to 25,000,000 barrels of fuel oil)	223,900 lb. UO ₂ (89.5 MTU) (equivalent to 42,000,000 barrels of fuel oil)	Type	Tandem-compound, 4-flow, 456,000 kW at 1.5" Hg absolute, 1800 rpm	Tandem-compound, 6-flow, 1,181,055 kW design, valves wide open, at 1.5", Hg abs., 1800 rpm
Number of fuel rods:	28,260	51,212	Throttle Conditions		
Number of subassemblies:	157	217	Pressure	680 psia	855 psia
Flux-fast:	1.3 x 10 ¹⁴ neutrons/cm²-sec.	2.5 x 10 ¹⁴ neutrons/cm²-sec.	Temperature	500°F	526°F
Flux-slow:	1.8 x 10 ¹³ neutrons/cm²-sec.	5.6 x 10 ¹³ neutrons/cm²-sec.	Flow	5,350,000 lb/hr	14,381,000 lb/hr
Fuel cladding in core:	22,000 lb (stainless steel)	64,092 lb. (Zircaloy -4)	Turbine cycle heat rate	10,211 Btu/kWh	9,895 Btu/kWh
Number of fuel regions:	3	3	Generator		
Multiplication factor, cold (K):	1.25	1.17	Voltage	18,000 volts	22,000 volts
Equilibrium cycle burnup for discharged fuel (average):	31,100 MWh-D/MTU	34,350 MWh-D/MTU	Rating (power factor = 0.9)	500,000 KVA	1,252,000 KVA
Neutron Control			Stator cooling:	Hydrogen at 60 psig	Water cooled
Number of control rods:	45	91	Rotor cooling:	Hydrogen at 60 psig	Hydrogen at 60 psig
Composition of rod:	Silver-Indium-Cadmium	Boron carbide & Silver-Indium-Cadmium	Net Plant Electrical Output	436,000 kW	1,100,000 kW
Soluble Absorber					
Type	Borax (Boric Acid)	Boron (Boric Acid)	MAJOR EQUIPMENT		
Maximum concentration-Full power (no rods, K = 1.0) hot, clean:	1,465 ppm	719 ppm	Main Condenser		
Refueling (rods in) cold:	2,900 ppm	2150 ppm	Type	Tandem, single-pass, single pressure, surface condenser	Tandem, single-pass, dual pressure, surface condenser
Heat Transfer			Tube material	Copper-Nickel & Titanium	Titanium
Reactor thermal power:	1,347 MW	3,390 MW	Number of Tubes	22,016	52,936
Average heat flux:	143,350 Btu/hr-ft²	182,400 Btu/hr-ft²	Condensing Surface	150,000 ft² each	340,892 ft²
Maximum heat flux:	483,000 Btu/hr-ft²	428,000 Btu/hr-ft²	Number	2	2
Ratio of max./avg.:	3.22	2.35	Reheater:		
Power density per foot of rod:	15 kW (max.)	12.5 kW (max.)	Number	4 (single stage)	2 (two stage)
Fuel			Outlet pressure	115 psia	176 psia
Type:	Sintered UO ₂ pellets	Sintered UO ₂ pellets	Outlet temperature	453°F	495°F
Density, UO ₂ :	10.41 grams/cc	10.38 grams/cc	Total flow	3,927,000 lb/hr.	9,970,353 lb/hr.
UO ₂ dimensions in fuel:	0.383 x 120 in.	0.325 x 150 in.	Heat source	349,064 lb/hr main steam	536,886 lb/hr. (first stage) 591,680 lb/hr. (second stage)
Average enrichment:	4.0%	2.35%	Feedwater Heaters		
Clad thickness:	0.0165 in. (stainless steel)	0.025 in. (Zircaloy -4)	2 parallel paths		
Fuel rod spacing (center to center):	0.556 in.	0.5063 in.	Number (both paths)	10	13
			Type	Horizontal U tube, double pass	Horizontal U tube, double pass



San Onofre Nuclear Generating Station (SONGS).



**INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS**



June 3 - 21, 1985

Session Objectives

**SESSION 27: NUCLEAR FUELS MANAGEMENT, PROGRAM ADMINISTRATION, AND IAEA
INTERFACE AT THE SAN ONOFRE NUCLEAR GENERATING STATION**

The nuclear material control and accountability system at the San Onofre Nuclear Generating Station (SONGS) will be described in detail. The discussion will consider the safeguards organizational structure, fuel-handling procedures and documentation, nuclear material records and reports, and physical inventory procedures. The interactions between SONGS and the two safeguards authorities, the NRC and the IAEA, will be described. The impact of safeguards requirements on the facility will be considered.

After the session, participants will be able to:

- Describe the material control and accountability system at SONGS.
- List the material balance areas and key measurement points at SONGS.
- Describe the records and reports generated at SONGS to satisfy national and international safeguards requirements.
- Summarize personnel, training, and equipment requirements associated with nuclear material safeguards at SONGS.
- Discuss the impact of IAEA safeguards on facility operations.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



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SESSION 27: NUCLEAR FUELS MANAGEMENT,
PROGRAM ADMINISTRATION, AND IAEA INTERFACE
AT THE SAN ONOFRE NUCLEAR GENERATING
STATION

Scott W. Wylie
Southern California Edison Company

I. THE LEGAL BASIS AND REGULATORY GUIDELINES FOR SOUTHERN
CALIFORNIA EDISON'S ACCOUNTABILITY OF SPECIAL NUCLEAR
MATERIAL (SNM)

The legal basis for Southern California Edison's Nuclear Material Accountability Program (the Program) is found in Title 10, Parts 70 and 75 of the Code of Federal Regulations (CFR). Part 70, "Domestic Licensing of Special Nuclear Material," provides the legal requirements associated with the handling of SNM by nuclear power plants not under agreement with the International Atomic Energy Agency (IAEA). For those nuclear power plants that are under agreement with the IAEA, Title 10, Part 75, "Safeguards on Nuclear Material Implementation of US/IAEA Agreement," establishes the legal requirements. This part contains program requirements related to the following:

- o general provisions.
- o installation information.
- o material accountability and control.
- o reports.
- o installations designated for IAEA safeguards.

There are other documents related to the control of SNM that provide detailed instructions regarding various aspects of the program. Two of these documents are NUREG/BR-0006 and NUREG/BR-0007. NUREG/BR-0006, "Instructions for Completing Nuclear Materials Transaction Reports (Form 741)," and NUREG/BR-0007, "Material Balance Report, Physical Inventory Listing and Concise Note Forms (Form 742)," provide detailed instructions for filling out forms 741 and 742, respectively. These are the two reports required by 10CFR75 as the means of accounting for changes in inventory and the material balance of Special Nuclear Material.

Because 10CFR75 establishes the requirements regarding the control of SNM as it applies to the US/IAEA Agreement, all installations of the United States Eligibility List that are participating in the Program must conform to its requirements.

By the fact of compliance with 10CFR75, Southern California Edison is performing the required interactions with the NRC. The major interaction involvement is that of delivery of information, reports, and other communications.

II. DEFINITIONS

Agreement - The Agreement between the United States and the International Atomic Energy Agency for the Application of Safeguards in the United States. The term refers to both the principal text of the agreement, consisting of 90 articles, and to the protocol thereto. (Reference: 10CFR75.4 [b])

Facility Attachment - That portion of the Subsidiary Arrangements to the principal text of the Agreement that pertains to a particular installation that has been identified pursuant to Article 39(b) thereof. (Reference: 10CFR75.4[g])

III. THE FACILITY ATTACHMENT

10CFR75 requires that a Facility Attachment be developed for each nuclear power plant that is operating under the US/IAEA agreement. The Facility Attachment documents the determinations referred to in 10CFR75.11 and contains such other provisions as may be appropriate. See Appendix A for the text of 10CFR75.11.

The Facility Attachment for San Onofre 2, designed to meet the requirements of 10CFR75.11, was prepared by the IAEA and sent to the U.S. for formal acceptance.

The IAEA has identified the following items as required safeguards measures and delineated them in the Facility Attachment:

- o accountancy.
- o containment and surveillance.

Appendix B contains the specific key points under each of these major safeguards measures.

The Facility Attachment requires that if there is a need, on the part of SCE, to break a safeguards seal or interfere with the operation of safeguards instruments, the IAEA shall be informed in advance by the fastest means. This notification shall include the (probable) date on which the operation will take place. If a seal is removed in the absence of an Agency Inspector, without the operator being able to inform the Agency in advance, a special report must be prepared.

The Facility Attachment also specifies requirements associated with the following:

- o records system.
- o reports system.

Appendix C lists the specific items under each of the above requirements.

The IAEA will make formal statements regarding inspections at San Onofre 2. A summary statement will be made on the result of each inspection within 30 days of its completion. 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.

IV. SOUTHERN CALIFORNIA EDISON'S COMPLIANCE WITH THE REQUIREMENTS OF 10CFR75

Southern California Edison's vehicle for meeting 10CFR75 requirements is Procedure NFS-10 of the Nuclear Fuel Supply Quality Assurance Procedures. NFS-10, "Procedure for Nuclear Material Accountability," provides instruction in the following areas:

- o organization and responsibility.
- o facilities description.
- o record description and flow.
- o nuclear material accounting system.
- o measurements and statistics.
- o maintenance and control.

Each of the subject areas is included in the procedure to meet the requirements of 10CFR75, and the guidelines provided in NUREGS/BR-0006 and BR-0007.

A. Organization

Southern California Edison Company has three organizations involved in the Nuclear Material Accountability Program. They are:

- o site.
- o fuel supply department.
- o nuclear engineering.

1. The Site Organization. The Site (San Onofre Nuclear Generation Site) Organization is responsible for the physical handling involved in the receipt, shipment, storage, and onsite movement of Special Nuclear Material. Some specific responsibilities include:

- o inspecting and accepting new fuel.
- o obtaining flux distribution and power measurements.
- o maintaining records and initiating reporting forms for the Site.

2. The Fuel Supply Department. The Fuel Supply Department has six areas of responsibility in terms of the program. They are:

- o maintenance of files of records of all receipts, shipments, and movements of Special Nuclear Material.
- o maintenance of inventory status of SNM.
- o maintenance and operation of a material accounting system of sufficient detail and accuracy to satisfy SNM control requirements and furnish data for all reports.
- o issuance of nuclear material reports required by the NRC company management, or other agencies.
- o publication, maintenance, and control of NFS-10.
- o audit of material measurements and measurement systems and controls of both fuel fabricators and reprocessors following the recommendations of American National Standards Institute (ANSI) publication N15.8.

3. Nuclear Engineering. Nuclear Engineering has the responsibility of nuclear analyses and core inventory determination. Some of the specific responsibilities include:

- o providing fuel assembly burn-up rate data.
- o maintaining records of computer calculations for engineering review and historical trend records.
- o evaluating and providing results of in-core data analyses.

Figure 1 depicts the SCE Organization structure. It shows the relationships between the generating Site and the nuclear material management function. As the figure shows, nuclear materials management, as a function of the Fuel Supply Department, is organizationally independent of the generating station.

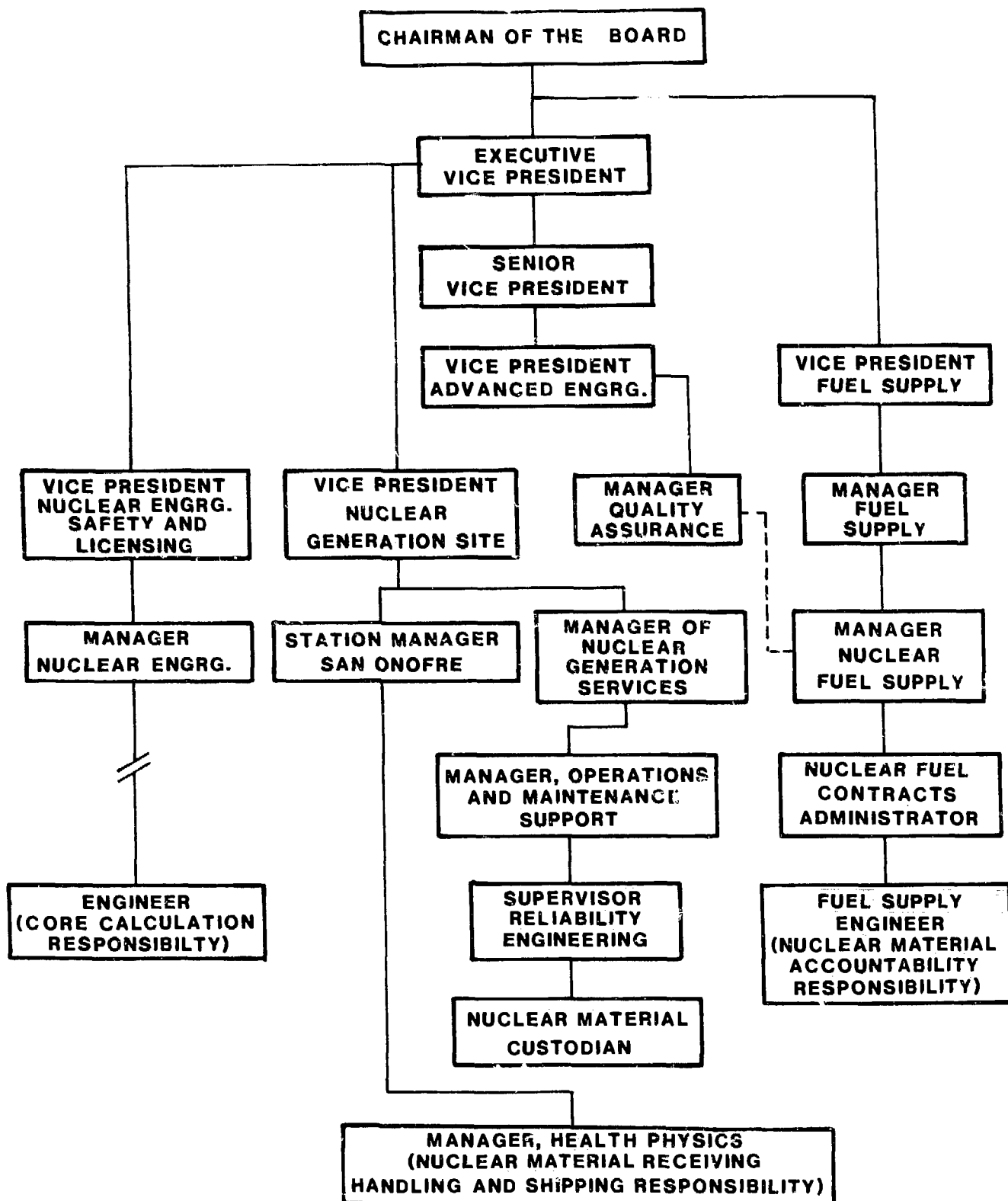


FIGURE 1
SCE ORGANIZATION STRUCTURE FOR NUCLEAR MATERIAL CONTROL RESPONSIBILITY

V. THE NUCLEAR MATERIAL ACCOUNTING SYSTEM

The Nuclear Material Accounting System described in NFS-10 is designed to meet the requirements of paragraphs 75.31 through 75.37 of 10CFR75. The titles of the paragraphs are:

- o 75.31 General Requirements.
- o 75.32 Initial Inventory Report.
- o 75.33 Accounting Reports.
- o 75.34 Inventory Change Reports.
- o 75.35 Material Status Reports.
- o 75.36 Special Reports.
- o 75.37 Disclosure of Reports to IAEA.

Southern California Edison's Nuclear Material Accounting System produces records that are of sufficient detail to furnish data for the compilation of all the NRC required reports. The material accountability records consist of journal entries, backed up by supporting source documents, and ledgers to form a double-entry system. Journal entries are prepared for removal, addition, or internal transfer of any reportable weight of material for which Edison is responsible. Source information supporting journal entries consists of receipts, transfer reports, Form NRC/DOE-741, "Material Transaction Reports," computer code calculations of in-core transmutations, and supported adjustments.

At SCE, the application of the Nuclear Materials Accounting System is through the establishment of Item Control Areas (ICAs). Each ICA is a location defined by physical boundaries and/or organizational responsibility over which exact material accountability is maintained.

The Item Control Areas associated with San Onofre Unit 2 are the Unit 2 Fuel Handling Building and Reactor Cavity Pit, and the Neutron Source Storage Area. Figure 2 shows the flow of SNM through the Item Control Areas.

Any time material is moved from one ICA it must go to another, and a transfer record is made. The information is then recorded in a journal and posted to the two or more ledgers involved. Each ICA is represented by at least one account ledger. Most ICAs are subdivided into a number of accounts to maintain specific identities such as fuel batch, type of material, and type of responsibilities. Appendix E contains a listing of the accounts that support the Nuclear Material Accounting System.

The material in each ICA is inventoried annually by inspection, counting, weighing, or whatever physical measurement method is appropriate. Fuel and products in the reactor core must be inventoried by certified records.

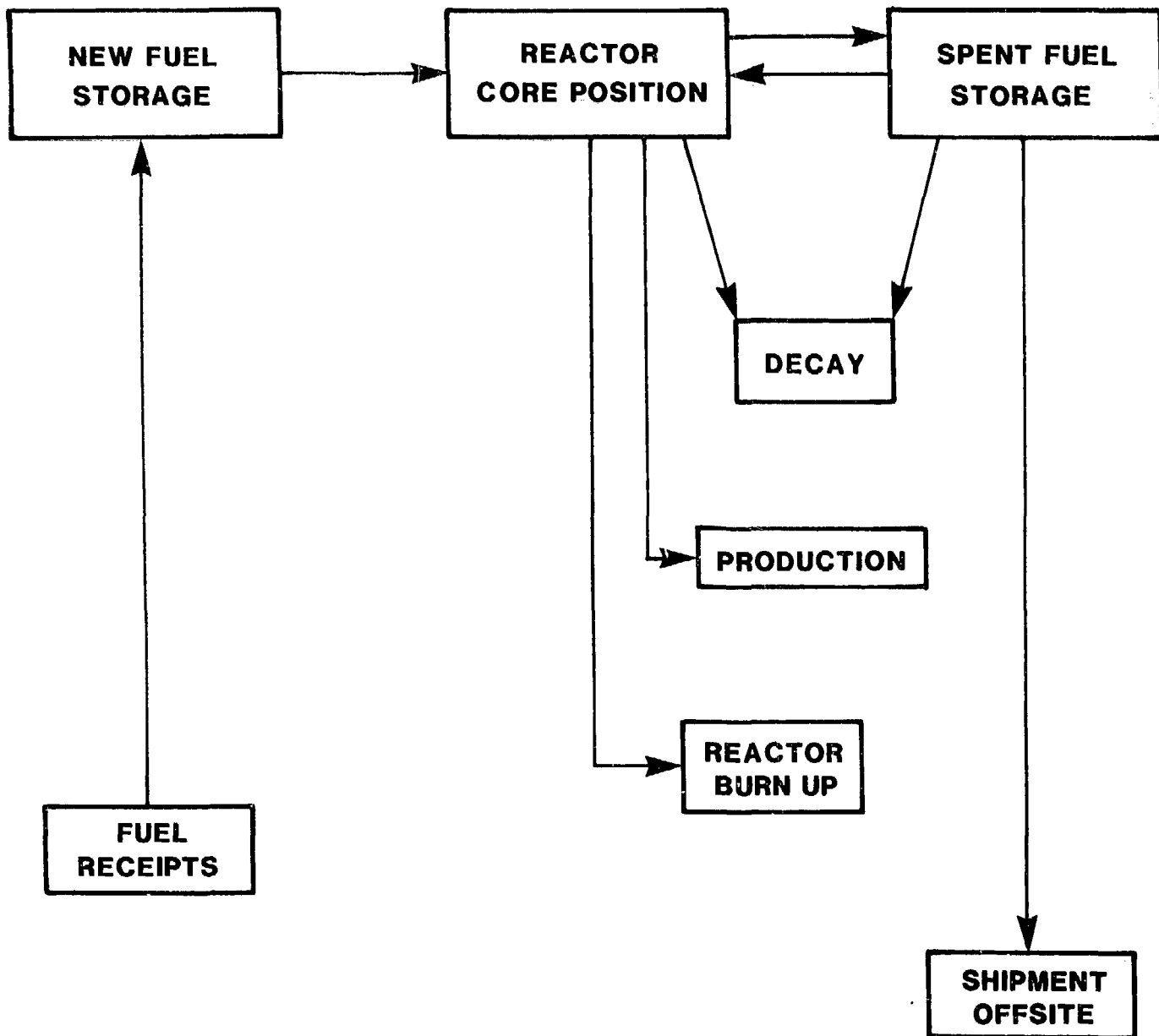


FIGURE 2
FLOW OF SNM THROUGH THE ITEM CONTROL AREAS

The Nuclear Material Accountability System utilized by SCE satisfies the IAEA accountancy safeguards measures given in the Facilities Attachment of SONGS Unit 2.

VI. THE ACCOUNTABILITY PROCESS

The steps taken by Southern California Edison to control nuclear material containing more than 0.5 gram of Special Nuclear Material will now be presented through the use of a scenario.

The scenario begins with the shipper loading Special Nuclear Material (fuel assemblies for example) on a vehicle and transporting it to San Onofre Unit 2. The first step in the accountability process is the shipper's transmittal of the required copies of Form NRC/DOE-741 to Southern California Edison's Fuel Supply Department.

Form NRC/DOE-741 is required by the Nuclear Regulatory Commission (NRC) to report any Special Nuclear Material or source material inventory changes including shipment, receipt, burn-up, production, on-site losses, on-site gains, inventory differences, or any other inventory adjustment involving one gram or more of contained SNM (U-235, U-233, plutonium, or any combination thereof), or one kilogram or more of source material (depleted uranium, natural uranium, or thorium).

The next step in the scenario is the arrival of the transport vehicle at SONGS. Upon arrival of the SNM on-site, the Nuclear Material Custodian, after the verification of the quantity of SNM, fills out Form FS13, "Internal Material Transfer Form." After the form is completed, one copy is sent to the Fuel Supply Department.

Form FS13 is the official record that source and SNM has been received on-site, shipped off-site, or moved to a different Item Control Area. It is also the source document for the Fuel Supply Department to post the transaction to the Nuclear Materials Accounting System.

In addition to filling out the FS13 form, the Nuclear Material Custodian completes a Fuel Storage Record Form. This form is used by the Site to record the history of SNM movements between ICAs. It is also used to record the current, specific location of each fuel assembly.

The accountability control of SNM continues all the while material is on-site. The Site and Fuel Supply Department have responsibilities with respect to this control.

A. Site Responsibilities

The following are the Site's responsibilities in terms of the continuing control of SNM:

- o Performs physical inventories of SNM and the results are reported on the inventory report.

- o Retains copies of all forms so long as the SNM is in Edison's possession plus five years subsequent to shipment off-site.

B. Fuel Supply Department Responsibilities

The following are the Fuel Supply Department's responsibilities in terms of the continuing control of SNM:

- o Core Calculation Reports - Computer calculations of changes in isotopic inventory due to consumption and/or production of the various isotopes in the reactor are run for accountability information to establish the core status at each refueling and at least every six months. Additional computations may be made within these periods for special purposes. The details of core burnup calculations are described in Section VI for each reactor. A computer printout of the calculated burnup values by individual assembly is filed for detailed information.
- o Decay Calculations - The NRC Instructions for preparation of the semiannual Material Status Report require the calculation and reporting of decay in isotopes Pu-238 and Pu-241. Fuel Supply calculates the Pu-241 decay in irradiated fuel. A computer program is used to calculate the data or the decay may be hand calculated using the factors given in the NRC Instructions. Decay of Pu-238 is not applicable to SCE's inventory.
- o Inventory Reports - The Inventory Reports submitted to Fuel Supply Department (Nuclear Fuel) are compared to the book inventory for each Item Control Area. If a difference is found, Fuel Supply Department (Nuclear Fuel) and the Site Nuclear Material Custodian involved work together to resolve the difference.
- o Form NRC/DOE-742, "U.S. Nuclear Regulatory Commission Material Status Report." - This report is issued, as required by the Nuclear Regulatory Commission, as of each refueling and at least every six months. All information necessary to complete form NRC/DOE-742 is contained in the Accounting System and files of transfer records.
- o Record Retention - Nuclear Material Transaction Reports (Form NRC/DOE-741) pertaining to material receipts, core calculation reports and inventory reports are to be retained as long as the material is in Edison's possession plus five years subsequent to shipment off-site. Forms NRC/DOE-741 pertaining to shipments off-site are to be retained until the Nuclear Regulatory Commission publishes retention requirements for off-site transfer records.

Figure 3 shows the complete document flow for the receipt and on-site transfer of SNM.

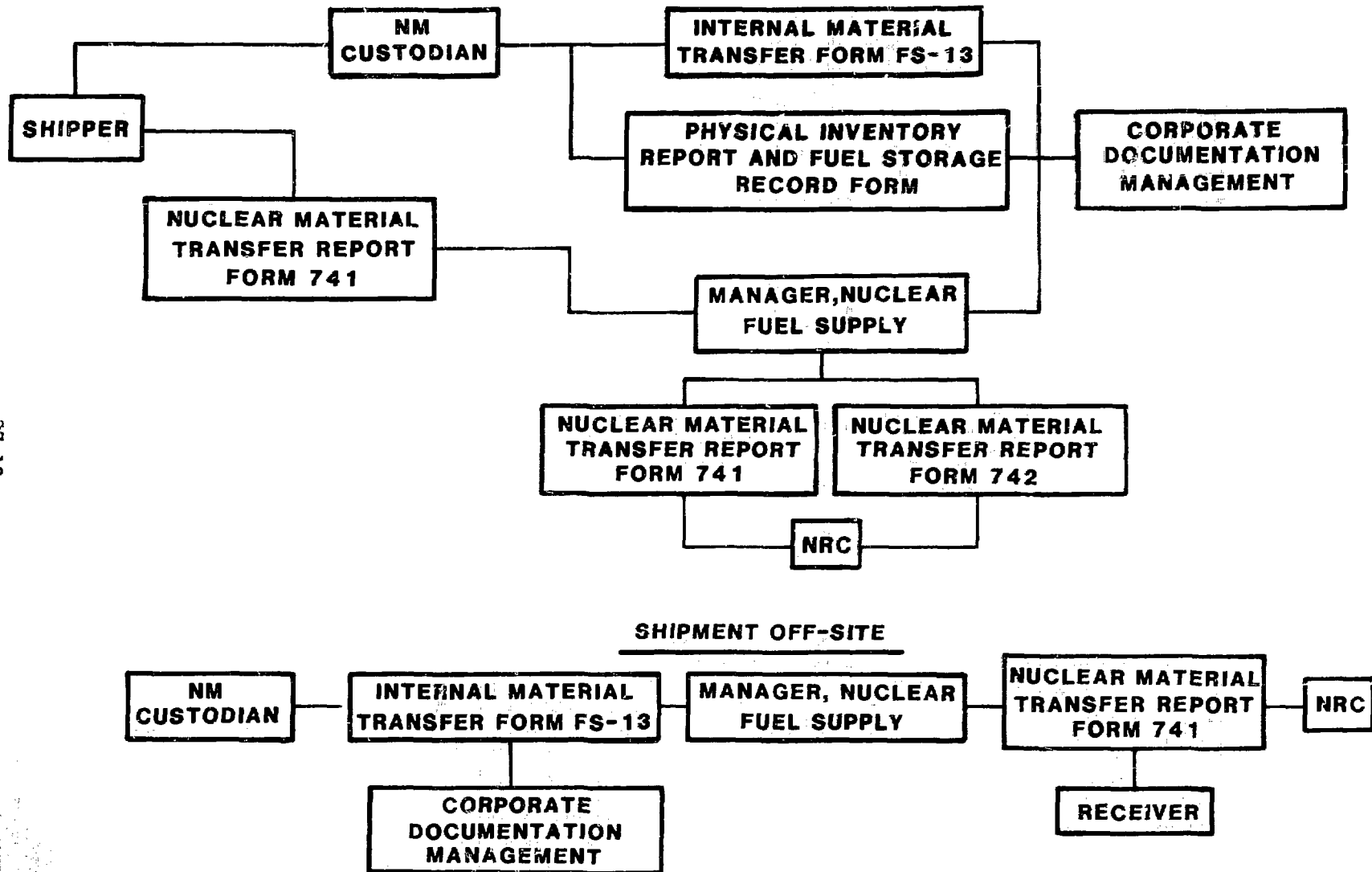


FIGURE 3
DOCUMENT FLOW FOR THE RECEIPT AND OUTSIDE TRANSFER OF SNM.

APPENDIX A

10CFR75.11

(a) Each licensee subject to the provisions of this part shall submit installation information in response to a written request from the Commission, with respect to any installation which the Commission indicates has been identified under the Agreement and in which the licensee carries out licensed activities. (The Commission request shall state whether the installation has been identified under Article 39(b) of the principal text of the Agreement or Article 2(a) of the Protocol.) The licensee shall submit such information to the Commission within the period, which shall be at least 45 days, specified in the Commission's request.

(b) Installation information includes:

(1) The identification of the installation, stating its general character, purpose, nominal capacity (thermal power level, in the case of power reactors), and geographic location, and the name and address to be used for routine purposes;

(2) A description of the general arrangement of the installation with reference, to the extent feasible, to the form, location and flow of nuclear material and to the general layout of important items of equipment which use, produce, or process nuclear material;

(3) A description of features of the installation relating to material accounting, containment, and surveillance; and

(4) A description of the existing and proposed procedures at the installation of nuclear material accounting and control, with special reference to material balance areas established by the licensee, measurement of flow, and procedures for physical inventory taking. (As part of this description, the licensee may identify a process step involving information which it deems to be commercially sensitive and for which it proposes that a special material balance area be established so as to restrict IAEA access to such information.)

(c) Each licensee shall thereafter submit to the Commission information with respect to any modification at the installation affecting the information referred to in paragraph (a) of this section. Such information shall be submitted:

(1) With respect to a modification of a type described in the license conditions: At least 70 days before the modification is scheduled to be completed, except that in an emergency or other unforeseen situation a shorter period may be approved by the Commission.

(2) With respect to any other modification relevant to the application of the provisions of the agreement: At the time the first inventory change report is submitted after the modification is completed.

APPENDIX A (continued)

10CFR75.11

(d) The information specified in paragraphs (a) and (c) of this section shall be prepared on Form N-71 or other forms supplied by the Commission (including appropriate IAEA Design Information Questionnaire forms). The information shall be sufficiently detailed to enable knowledgeable determinations to be made in the development of Facility Attachments or amendments thereto, including:

(1) Identification of the features of installations and nuclear material relevant to the application of safeguards to nuclear material in sufficient detail to facilitate verification;

(2) Determination of IAEA material balance areas to be used for IAEA accounting purposes and selection of those strategic points which are key measurement points and which will be used to determine flow and inventory of nuclear material;

(3) Establishment of the nominal timing and procedures for taking of physical inventory of nuclear material for IAEA accounting purposes;

(4) Establishment of the record and reports requirements and records evaluation procedures;

(5) Establishment of requirements and procedures for verification of the quantity and location of nuclear material; and

(6) Selection of appropriate combinations of containment and surveillance methods and techniques at the strategic points at which they are to be applied.

(e) The licensee's detailed security measure for the physical protection of an installation shall be included in the installation information only when and to the extent specially requested by the Commission.

APPENDIX B
SAFEGUARDS MEASURES

Accountancy

1. Material Balance Area and Identification Codes
2. Strategic Points Which are Key Measurement Points
 - a. for determination of nuclear material flow
 - b. for determination of the physical inventory
 - c. physical inventory taking

Containment and Surveillance

1. Strategic Points for the Application of Containment and Surveillance Measures
2. Installed Agency Instruments and Devices
 - a. seals to ensure the containment of the reactor vessel
 - b. cameras for surveillance of fuel movements into or out of the reactor containment including the fresh and spent fuel storage areas
 - c. seals on shipping casks with spent fuel

APPENDIX C
IAEA REQUIREMENTS

Records System

1. Specific Provisions for Accounting Records
 - a. inventory and batch changes and the time of recording
 - b. Measurement results used for determination of the physical inventory and time of recording.
 - c. adjustments and corrections, time of recording
2. Specific Provisions for Operating Records
 - a. Operating data used to establish changes in the quantities and composition of nuclear material.
 - b. calibrations
 - c. sequence of the actions taken in preparing for and in taking the physical inventory
 - d. Actions taken in order to ascertain the cause and magnitude of any accidental or unmeasured loss.
3. Location and Language of Records
4. Retention Period for Records

Reports System

1. Specific Provisions for Inventory Change Reports
 - a. contents
 - b. timing or frequency of dispatch
2. Specific Provisions for Concise Notes
 - a. concise notes explaining the inventory changes
 - b. concise notes describing the anticipated operational program: subject and timing of dispatch

APPENDIX C (continued)

IAEA REQUIREMENTS

3. Specific Provisions for the Material Balance Reports (MBRs)
 - a. contents
 - b. Physical inventory listings will accompany the MBRs.
 - c. time or frequency of dispatch
4. Special Reports
 - a. specification of circumstances requiring submission of special reports
 - b. contents, as appropriate

Inspections

1. mode of routine inspections
2. applicable formula and procedure for determination of maximum routine inspection effort
3. indication of the actual routine inspection effort under ordinary circumstances
4. indication of the scope of routine inspections under ordinary circumstances
5. arrangements for the use by the Agency of equipment for independent measurements
6. services and charges

APPENDIX E

ACCOUNTS IN SUPPORT OF THE NUCLEAR MATERIAL ACCOUNTING SYSTEM

1. On-Site Inventory Accounts
 - a. new fuel-storage areas
 - b. reactor cores
 - c. irradiated fuel-storage areas
 - d. neutron source storage/Health Physics sources less than 5 microcuries total plutonium
2. Ledgers for Material in Off-Site Storage Areas
3. Non-Inventory Accounts
 - a. transactions
 - b. production
 - c. burn-up
 - d. material accounted for
 - e. process of accidental losses
 - f. decay



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 28 & 29: TOURS OF THE SONGS TRAINING CENTER AND SONGS UNIT 2

The San Onofre Nuclear Generating Station (SONGS) operates an extensive Training Center to provide the specialized training required by SONGS personnel. Course participants will tour the Training Center to observe the reactor control room simulator and training laboratories for machinists, health physics technicians, chemical laboratory technicians, and electrical technicians. All participants will have an opportunity to use the laser disc retrieval system (SeeVu); this unique instrument uses 40 000 recorded plant photographs to provide a detailed video tour of any part of the SONGS facility.

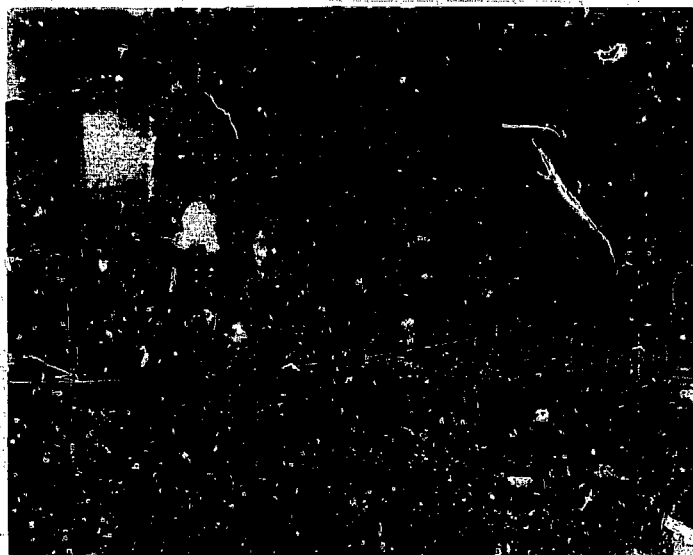
Participants will tour selected parts of the three reactor units of the San Onofre Nuclear Generating Station (SONGS). The tour will illustrate the size and complexity of a modern nuclear power station and the problems encountered when applying safeguards controls at a reactor facility. The tour will emphasize the fuel-handling and storage areas. Participants will have the opportunity to observe the surveillance instrumentation installed as part of the IAEA inspection program at SONGS.

Control room simulation in SONGS
Training Center.



Instrumentation training lab in
SONGS Training Center.

Machine shop training lab in SONGS
Training Center.





INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSIONS 28 & 29: TOURS OF THE SONGS TRAINING CENTER AND SONGS UNIT 2

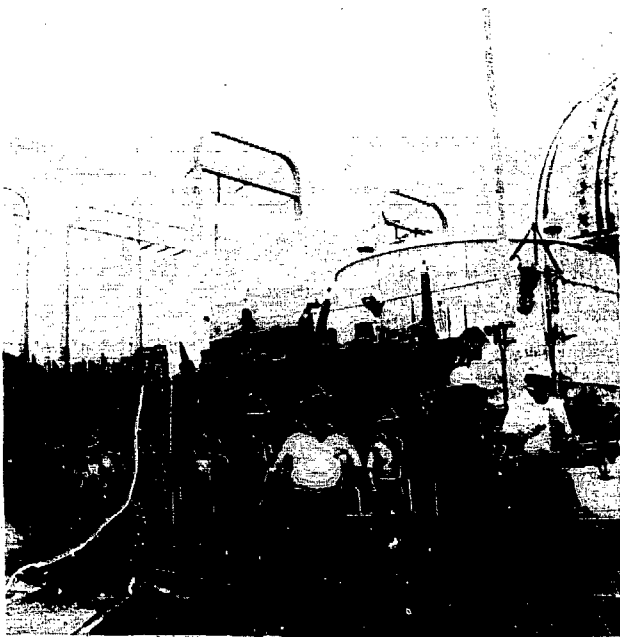
E. Arnold Hakkila
Los Alamos National Laboratory

On June 16 and 17, the participants toured the SONGS Training Center and the Combustion Engineering 1100-megawatt nuclear reactor under IAEA safeguards, Unit 2.

The Training Center is designed to provide technical training and recertification in all areas of expertise required to operate the three units at the San Onofre Nuclear Generating Station. Many of the courses offered by the center are certified for college credit. The Training Center includes classroom and laboratory facilities in technical areas such as electronics, chemistry, and instrument maintenance. All types of equipment on site that require maintenance are available in the Training Center for hands-on training. The Center also includes a control room simulator for Units 2 and 3. Reactor operators can obtain training and recertification on the simulator. The Center has a laser disk video system, containing over 40,000 pictures, that enables one to walk through any area of Unit 2 or 3 and see the facility without actually entering it. The participants were given tours of the Training Center. This is a very extensive training facility and the participants were very impressed as most had never seen a facility of this size operated by a nuclear power station.

Prior to the tour of Unit 2, safety and security training lectures were given to all people taking the tour. The preparation also included a whole body count for all participants. The tour was designed to familiarize the participants with safeguards and security aspects of the reactor. It included a tour of the reactor control room, the spent-fuel storage pool, and the fresh-fuel receiving and storage areas. Participants were able to stand by the spent-fuel pool and observe the Cerenkov glow from the spent-fuel bundles in the pool. They were also able to observe the Minolta surveillance cameras installed by the IAEA to monitor all fuel movements in the Fuel Handling Building.

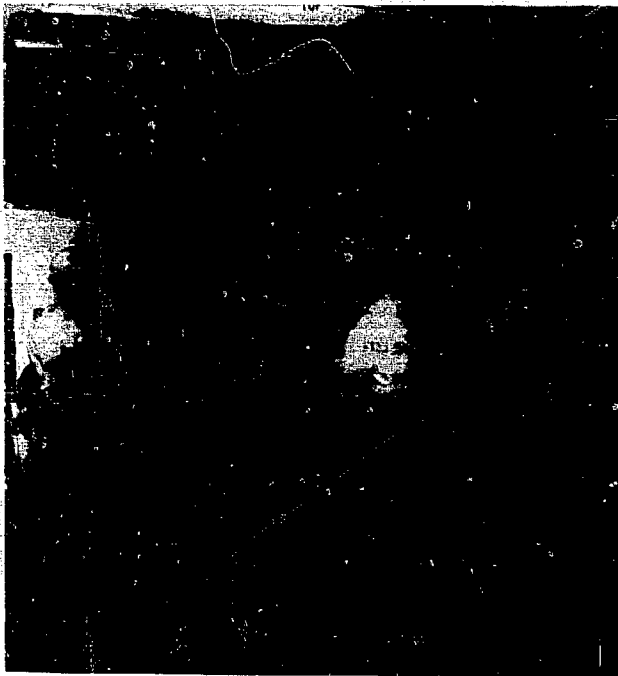
The tour included brief stops and explanations at the switch yard, cooling water inlet and cleanup area, and the turbine building. During the tour, participants had a good opportunity to observe the extensive security, administrative control, and radiation safety procedures in the station. These tours were high points of the SSAC course.



Tour group heading for fuel shipping/receiving area.



Viewing spent-fuel storage pond.



Discussions with SONGS security officer.



Monitoring for contamination after visiting fuel handling building.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 30: US NUCLEAR REGULATORY COMMISSION INSPECTION PROCEDURES AT THE SONGS

The US Nuclear Regulatory Commission (NRC) is the safeguards authority for the US state system of accounting and control of nuclear material (SSAC). The NRC conducts regular inspections at all commercial nuclear facilities in the United States. The general requirements of the US SSAC are discussed during Session 4. The specific requirements for nuclear power stations and SONGS will be discussed here. NRC inspection procedures at SONGS will be described in detail. The session will emphasize methods for analyzing inventory data, available measurement techniques, and procedures for seals. The statistical analysis of inventory and inspection data will be discussed.

After the session, participants will be able to:

- Describe the major requirements of the US SSAC at the SONGS facility.
- List the records and reports supplied by the reactor operator to the NRC.
- Describe the inspection and verification activities carried out by the NRC at SONGS.
- Understand data analysis procedures used by the NRC.
- Discuss the measurement instruments used by the NRC at SONGS and other nuclear power stations.
- Discuss the goals and effectiveness of the NRC inspection program at power reactor facilities.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 30: US NUCLEAR REGULATORY COMMISSION INSPECTION PROCEDURES AT THE SONGS

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References:

1. Code of Federal Regulations, Title 10, (10 CFR) Parts 0 to 199, Revised as of January 1, 1985. Available from the Superintendent of Documents, US Government Printing Office, Washington, DC 20402.
2. Instructions for Completing Nuclear Material Transaction Reports, NUREG/BR 0006 Revision 2, effective March 1, 1985. Available from the Office of Nuclear Material Safety and Safeguards.
3. Instructions for Completing Material Balance Report, Physical Inventory Listing and Concise Note Forms, NUREG/BR 0007, effective January 1, 1980. Available from the Office of Nuclear Material Safety and Safeguards.

I. INTRODUCTION

The US Nuclear Regulatory Commission (NRC) is the safeguards authority for the US state system of accounting and control (SSAC) of nuclear materials. The NRC conducts regular inspections at all commercial nuclear facilities in the United States.

The NRC inspection program is designed to assure that effective, on-going control measures are maintained for safeguarding special nuclear material (SNM). The program has been segmented into individual modules or procedures, each emphasizing a different aspect of the licensee's safeguards program. The routine program assigns an inspection frequency for each separate procedure which will vary based upon the strategic nature of the material possessed and the enforcement history of each licensee.

For a nuclear power plant such as SONGS, the routine inspection program is described in a single inspection procedure or module; several modules (procedures) are used to guide the inspection program for LEU fuel fabrication facilities. In addition, several special purpose procedures are utilized as events warrant.

<u>Procedure Number</u>	<u>Title</u>	<u>Frequency</u>
85202	Material Control and Accounting- Reactors (LEU Fuel Fabrication Facilities)	Triennial
85203	Facility Organization and Management Controls	Biennial
85205	Facility Operation and Internal Controls	Biennial
85207	Measurement Systems	Biennial
85209	Measurement Control and Statistics	Biennial
85211	Physical Inventory	Biennial
85215	Records and Reports	Biennial
85301	MC&A for Imports and Exports	As Events Warrant
30703	Management Meetings - Entrance and Exit Interviews (Special Purpose Procedures)	Each Visit
92700	On-site Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities	As Events Warrant
92701	Follow-up on Inspector Identified Problems and Unresolved Items	As Events Warrant
92702	Follow-up on Items of Non- compliance and Deviations	As Events Warrant
92703	IE Bulletin, Confirmatory Action Letter, and Generic Letter Follow-up	As Events Warrant
92713	Independent Inspection Effort for Safeguards Inspections	As Events Warrant
92718	Inventory Differences Action Guidelines	As Events Warrant

The NRC program normally schedules a single routine inspection visit for power plants and approximately four routine visits per year for fuel fabrication facilities. Individual modules are completed during those visits although some may require more than one visit to complete. The manpower allocated for an inspection visit at a given plant or facility is based on prior inspection experience and the level of effort anticipated by the regional office.

Nuclear Power Reactors

Specific material control and accounting requirements for power reactors in general, and SONGS in particular, are found in 10 CFR Part 70. The requirements cover three main areas:

- Documentation of the transfer (receipt or shipment) of SNM, or any adjustments to the SNM inventory holdings,
- Reporting SNM inventory holdings, and
- Reporting lost or stolen material or threats to do so.

Documentation of transfers (required by 10 CFR 70.54) is accomplished by completing and distributing the DOE/NRC Form 741 according to the instructions set forth in NUREG/BR-0006 (Reference 2). This form must be used to document any SNM inventory changes, including shipments, receipts, burnup, production, on-site gains or losses, inventory difference, or any other inventory adjustment involving one gram or more of contained uranium-235, uranium-233, or plutonium, or any combination thereof.

Semiannual reports to reflect inventory holdings as of March 30 and September 30 of each year are required (10 CFR 70.53) to be filed within thirty days after the end of the period covered by the report according to the instructions set forth in NUREG/BR-0007 (Reference 3). If licensees are reporting pursuant to 10 CFR 75.33 which imposes special reporting and other requirements on licensees under IAEA safeguards, reports are to be dispatched as soon as possible and, in any event, within thirty days of the start of the physical inventory taken as part of the licensee's material accounting and control procedures. (Form DOE 742A and 742B referred to in Reference 3 are not required for privately owned SNM.)

Title 10 CFR 70.52 requires each licensee who possesses one gram or more of contained uranium-235, uranium-233, or plutonium to report immediately to the appropriate NRC Regional Office any incident in which an attempt has been made (or is believed to have been made) to commit a theft or unlawful diversion of such material.

LEU Fuel Fabrication Facilities

In addition to the above mentioned requirements and as a point of review for the NRC's requirements for LEU fuel fabrication facilities as was discussed in Session 4, the following summary is provided:

The CFR is the most general document, giving a broad framework for the safeguards program. In response to CFR requirements, licensees have, over the past several years, submitted written programs addressing separate areas. The 70.51 plans were implemented during 1974 and cover tamper-safing, inventory frequency, record keeping requirements, and

ID/LEID evaluation. The 70.58 plans were implemented during 1976 and cover such topics as organizational structure, separation of functions, the material balance area/item control area (MBA/ICA) structure, management reviews and audits, scrap controls, and inventory procedures. The 10 CFR 70.57 program was implemented during 1978-1979 and covers measurement quality control. Collectively, the licensee's submittals to these three CFR paragraphs comprise a description of his fundamental nuclear material control program.

The safeguards inspection program for fuel fabrication facilities is designed to assure compliance with program requirements drawn from three sources: Title 10 of the Code of Federal Regulations, the NRC License Amendments or Conditions imposing special safeguard constraints on the activities authorized by the license, and the licensee's Fundamental Nuclear Material Control (FNMC) plan submitted in accordance with 10 CFR Part 70, paragraphs 70.51, 70.57, and 70.58.

On February 25, 1985, the NRC established a new CFR regulation, Part 74, to replace the Part 70 requirements discussed above for certain facilities. Part 74 applies to those fuel fabrication facilities (such as Exxon Nuclear Company at Richland, Washington) who possess only uranium enriched to 10% or less in the isotope U-235. Fuel fabrication facilities possessing more than 10 kilograms of U-235 in uranium enriched to greater than 10% continue to be regulated by paragraphs 70.51, 70.57, and 70.58.

Licensees possessing only low strategic significant quantities of SNM, i.e., Category III facilities (such as Exxon), are required by 10 CFR 74.31 to submit to the NRC no later than August 26, 1985, a new FNMC plan to describe their system capabilities to:

- Confirm the presence of SNM,
- Resolve indications of missing material, and
- Aid in the investigation and recovery of missing material.

II. TEXT

Each of the individual inspection procedures describes specific areas for the inspector to examine. The purpose of these procedures is to make the inspection both uniform and comprehensive for all privately owned material held throughout the United States.

A. Number 85102: Material Control and Accounting - Reactors

The inspection program for nuclear power reactors reviews the licensee's records as they relate to inventory, receipt, and shipment of SNM in support of the reports (DOE/NRC 741 & 742) submitted. The inspector reviews the licensee's records of other SNM inventory adjustments such as burnup, production, and radioactive decay. He also determines whether the licensee has limited his possession and use of SNM to the location and purpose authorized by his license.

The program determines whether the licensee has prepared, maintained, and

implemented an adequate and effective program and procedures to control and account for the SNM in his possession. The inspector conducts a random spot check of new fuel, irradiated fuel in the spent-fuel pool, SNM-bearing sources and test specimens, etc, and compares actual fuel element history records and location documents with the information noted on loading diagrams, transfer forms, or other fuel-handling and accounting records.

B. Number 85203: Facility Organization and Management Controls

This procedure is the first in a series of five used for routine inspection of fuel fabrication facilities. It examines the management structure of the licensee's organization and the safeguards functions within that structure, the authority and responsibilities assigned to the management positions, and the reporting channels established by written procedures. The licensee's safeguards program must ensure independence of action to assure that production pressures will not hinder sound safeguards decisions.

Due to frequent process changes (caused, for example, by technical advances or new business endeavors) the licensee's FNMC program is a dynamic activity requiring frequent revision and continuous management oversight and audit. This module also reviews those aspects of the management of the material control system instituted to keep it attuned to current process innovations and technological improvements.

C. Number 85205: Facility Operation and Internal Controls

This aspect of the LEU fuel fabrication plant inspection program focuses on the facility operation within the constraints of the NRC license, the MBA/ICA structure, and the FNMC plan.

The inspector assures that licensee procedures are established, maintained, and followed for tamper-safing containers of SNM and that such procedures describe the control of access to and accountability of tamper-safing devices.

An important area to be explored during the inspection is the verification of the significance of shipper/receiver differences. This segment of the program compares the measurement capabilities of both parties. Thus, this procedure verifies the control over shipment and receipt of SNM at the licensed facility.

The provisions of 10 CFR 70.58 (and 74.31) require the licensee to have an internal control system designed to protect SNM from loss and theft. For LEU facilities this procedure examines the control exercised over internal SNM transfers and the documentation thereof.

D. Number 85207: Measurement Systems

Another area that is examined at an LEU fuel fabrication plant is the licensee's system of measurements to assure strict accountability of the SNM in his possession. All aspects of measurement are reviewed. This segment of the inspection program assures the credibility of the licensee's material accountability statements. The review includes such topics as measurement techniques, standards, calibration, training of personnel, and generation of the basic measurement error data ultimately used to calculate the limit of error associated with inventory differences and shipper/receiver differences.

The inspection program provides for the review of measurement methods and the periodic overcheck of measurement techniques and practices through observation and independent sampling and measurement.

E. Number 85209: Measurement Control and Statistics

Essential to credible measurements is a measurement quality control program. This inspection procedure covers all aspects of measurement control to include the proper operation of programs for monitoring measurement equipment performance and evaluation of the validity of the random and systematic error estimates, whether introduced by sampling technique, analytical method, or data reduction.

The licensee's inventory difference (ID) and its associated limit of error (LEID) are subject to the constraints of 10 CFR 70.51(e). This segment of the inspection program verifies that the inventory difference and its associated limit of error are appropriate and accurate. It includes a review of measurement techniques, standards, calibration, training of personnel, and generation of error data. It further examines the licensee's measurement control program and method of statistical treatment of measurement variances to assure the credibility of the accountability measurements used to calculate the limit of error associated with inventory differences and with shipper/receiver differences.

F. Number 85211: Physical Inventory

An important segment of the inspection program is an assessment of the adequacy of the physical inventory performed by the licensee. This procedure includes item spot checks performed by the inspector, which coupled with the check of the measurement systems through sample taking and independent analysis provides assurance of the quality of the licensee's inventory records. Examination of the scheduling, training, and performance of those taking the inventory plus the licensee's reconciliation of the book inventory to the physical inventory are reviewed.

G. Number 85215: Records and Reports

An essential part of a good accounting system is the records and reports that are used to document material status and control material transfers. Consideration of the licensee's system of records and a detailed review of source and subsidiary records for consistency internally as well as with reports submitted to the NRC are guided by this procedure.

H. Number 85301: MC&A for Imports and Exports

In the implementation of this procedure, the inspector verifies that imports and exports of SNM are conducted in accordance with the applicable regulations and the conditions specified in the import/export license. The inspector assures that the transfer reports (DOE/NRC Form 741) have been prepared and distributed, that the integrity of all containers and seals is checked upon receipt, and that any shipper/receiver differences are reviewed and evaluated and that timely investigative and corrective actions are taken to reconcile differences which are statistically significant.

I. Number 30703: Management Meetings - Entrance and Exit Interviews

This module is performed whenever an inspector visits a plant and it consists of two elements. Upon arrival onsite, the inspector briefs licensee management as to the overall scope and schedule for the visit. The inspector asks to be informed of any nonreportable safeguards-related events such as minor process changes which have occurred since the last inspection.

Prior to leaving the site, the inspector briefs the licensee as to the inspection findings.

J. Number 92700: On-site Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities

A significant inventory difference or a handling mishap with the potential for a loss of material accountability, for example, would prompt the inspector, guided by this module, to verify the nature, impact, and cause of the nonroutine event, actions taken or planned by the licensee, and other information of significance.

K. Number 92701: Follow-up on Inspector Identified Problems and Unresolved Items

During a routine inspection the inspector may detect a possible problem as indicated perhaps by an anomaly noted in a log book or test record. While an actual problem may not exist, the matter must be resolved to assure that the facility will not be operated in a manner inimical to safeguards principals.

L. Number 92702: Follow-up on Items of Noncompliance and Deviations

This procedure, used on the inspection following one in which items of noncompliance were identified, requires the inspector to assure that the licensee responded in a timely manner to the notice of violation and that the measures taken to correct the item and avoid further items of noncompliance were effected as described and within the time period specified in the licensee's reply.

M. Number 92703: IE Bulletin, Confirmatory Action Letter, and Generic Letter Follow-up

The inspector is instructed to ascertain whether the information submitted by a licensee in response to an IE bulletin sent for action is technically adequate, satisfies the requirements established in the bulletin, and that the action taken by the licensee is consistent with the intent of the bulletin, confirmatory letter, generic letter, or information notice.

N. Number 92713: Independent Inspection Effort for Safeguards Inspections

Approximately 20% of the inspector's time on-site during the inspection visit is set aside for the inspector to examine areas outside the defined inspection program. In the course of a material control and accounting inspection, the inspector may observe a questionable practice whether in the area of physical security, radiation contamination control, or operational safety. This procedure requires the inspector to spend some time in an unstructured review of activities related to other disciplines.

O. Number 92718: Inventory Differences Action Guidelines

In the event that a licensee generates a significant inventory difference or the limit of error of the inventory difference exceeds regulatory constraints, he is required by 10 CFR 70.53(b) to promptly report that fact to the NRC regional office. The guidance of this procedure specifies thresholds for immediate reporting to NRC headquarters, as well as documentation of appropriate licensee actions which could include process shutdown and reinventory.

III. INSPECTOR TRAINING

The objective of the NRC training program is to ensure that inspectors meet minimum knowledge and qualification standards. Through this program inspector qualifications and training are maintained in general conformance to standards for comparable job activities in the nuclear industry. Safeguards inspectors are trained in the region where the inspector is assigned, at NRC headquarters, or the NRC's Training Center in Chattanooga, Tennessee, and, particularly for specialized training, at selected DOE laboratories such as Los Alamos National Laboratory.

Training conducted at the Regional Office includes:

- Regional Orientation
- Code of Federal Regulations
- Safeguards Inspection Procedures
- Regulatory Guides
- NRC/IE Manual
- Industry Codes and Standards
- Safeguard Inspection Accompaniments (on-the-job training)

Specialized training required of all inspectors assigned to inspect fuel facility material control and accountability systems includes:

- Radiation/Contamination Protection
- Fundamentals of Inspection
- Basic Statistics
- Chemical Assay
- Fundamentals of Nondestructive Assay of Fissionable Material Using Portable Instrumentation

Supplementary training is provided for individual inspectors depending upon the requirements of their specific inspection assignments and includes:

- Advanced Neutron Workshop on Nuclear Material Assay
- Advanced Gamma-ray Spectroscopy for Nuclear Material Accountability
- Advanced Statistics

Inspectors are also encouraged to attend appropriate workshops and seminars to maintain an understanding of the state-of-the-art in safeguards concepts, techniques, and equipment.

IV. SUMMARY

The NRC's safeguards inspection program is a dynamic program--changing as necessary to best insure that the inspection objectives and goals are accomplished. This evolving program may be influenced in many ways: regulation changes, better equipment, more efficient accounting practices, and so on. The program includes periodic training and retraining to assure that the NRC safeguards inspectors are equipped with the latest in inspection equipment and techniques.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 31: IAEA INSPECTION PROCEDURES AT SONGS

The history, legal basis, and objectives of IAEA inspections at US nuclear power stations will be described briefly. The planning and performance of these inspections will be described in detail. The inspection activities to be discussed will include the records examination, the comparison of facility records with state reports submitted to the IAEA, flow and inventory verifications, the design of statistical sampling plans, and the performance of independent verification measurements. The session will address the principles of material balance and MUF evaluation and the content and format of the summary statement and report prepared on inspection activities. IAEA inspections at SONGS will be presented as an example of LWR power plant inspections conducted throughout the world; variations in inspection procedures at different reactors will be discussed.

After the session, participants will be able to:

- Describe the objectives and major activities of IAEA inspections at SONGS and other LWR power stations.
- Describe the measurement instrumentation and data analysis procedures used at LWR inspections.
- Discuss the use and purpose of seals and video surveillance equipment at LWR power stations.
- Understand the Agency's evaluation activities and summary statements.
- Discuss the scope and adequacy of IAEA inspection activities at LWR power stations.
- Appreciate the problems faced by IAEA inspectors.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 31: IAEA INSPECTION PROCEDURES AT SONGS

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I. INTRODUCTION

Under the voluntary safeguards agreement between the Agency and the United States of America (INFCIRC/288, Ref. 1) selected facilities are subject to Agency safeguards inspection during limited time periods. This is due to the limited manpower resources that are allocated by the Agency for the inspection of facilities in the nuclear weapon states, and also to the wish to have the inspection programme alternating between all the facilities that are covered by the Agreement. At present, there are four facilities which are being inspected by Agency inspectors in the USA; one of them is the San Onofre Unit No. 2 Pressurized Light-Water Reactor Station.

The starting date of the Agency safeguards at San Onofre was 31 July 1983, which is the date of the initial inventory, i.e., the inventory of nuclear material at the facility on which the accountancy verification activities are based. The first inspection at the facility was carried out on 9 and 10 November 1983, and it has been followed by 10 more inspections up till May 1985. The last inspection is planned to take place in November 1985, by which time a 2 year inspection period will be terminated.

The inspection activities at the San Onofre facility have been carried out in a very positive spirit of cooperation between the facility staff, officials of the US NRC, and the Agency inspectors. Without the assistance from the facility staff it would have been much more difficult and time consuming to implement the different measures which are necessary for carrying out a successful inspection programme.

During the time of Agency safeguards at San Onofre the inspection goals have been met, and the Agency has been able to draw conclusions with respect to the detection (or nondetection) of diversion of nuclear material and of unreported plutonium production.

The verification of fresh-fuel assemblies with a neutron collar NDA instrument, which took place in October 1984, has contributed to the Agency's further experience of the practical implementation of such NDA measurement procedures. The same is expected to be the case with the planned ION-1 measurements of spent-fuel assemblies later this year.

II. SAFEGUARDS APPROACH AND INSPECTION ACTIVITIES

1. LWR Safeguards Approach in General

Most research reactors as well as commercial power reactors are, from a safeguards point of view, considered as "item facilities." The nuclear fuel is contained in identifiable items, i.e., fuel assemblies, the integrity of which is usually preserved during their presence in the facility. The fuel assembly items can also be followed through a considerable part of the whole fuel cycle, from the fuel fabrication plant, through the reactor facility, and to the reprocessing plant or long-term storage. The nuclear material accountancy can, thus, be based on counting, identification, and nondestructive measurement of the fuel assemblies, i.e., item accounting. In addition, containment and surveillance (C/S) measures are used.

When designing a safeguards approach for a facility the possible diversion strategies (or diversion paths) will have to be considered. The safeguards measures through inspections and C/S devices are designed to prevent diversion along those paths.

The characteristics of an LWR from a safeguards point of view are:

- The reactor is shut down, normally once per year, for refueling;
- There are three categories of nuclear material, or fuel assemblies, i.e., fresh-fuel (FF) in dry or wet storage, core fuel (CF) in the reactor vessel, and spent-fuel (SF) in the storage bay.
- The SF is stored for a shorter or longer time in the bay before it is shipped to a reprocessing plant or long-term storage facility.

The safeguards measures should be designed for the detection of the diversion of one fuel assembly. In certain LWR facilities there is equipment for the dismantling of the SF assemblies into fuel rods, in which case the verification activities should take that into account.

The detection time for FF is set to one year. For CF and SF the detection time is set at 3 months. This means that the inventory of CF and SF assemblies should be verified once per year provided that the CF and SF is kept under successful containment/surveillance; otherwise more often. The service and frequency of surveillance equipment for SF should be such that the time period from starting a surveillance film and until it is replaced, processed, and reviewed should be 3 months maximum.

Fuel assemblies can be diverted from the dry storage area, as well as from the storage bay. Spent-fuel assemblies can be diverted from the storage bay. Fuel assemblies in the reactor vessel core can be diverted only when the reactor is shut down and the reactor vessel is opened.

At the annual physical inventory verification (PIV) fuel assemblies are identified and counted. The fuel assemblies that are making up the new core are verified by the inspectors when they are moved from the storage bay and into the reactor vessel. FF assemblies are verified by NDA measurements when they are received at the plant, or when they are moved from the dry storage and into the bay. SF assemblies are verified by NDA in the storage bay.

When the refueling is terminated and the reactor vessel lid is put back in place, seals are placed on the structure above the vessel so that the vessel lid cannot be removed without breaking the seals. Metallic Type E seals are used. The seals are attached by the Agency inspectors when the vessel is closed, and they are detached by the inspectors before the vessel is opened again.

During the refueling period, when the reactor vessel is open, film cameras are installed in the reactor containment for the surveillance of the reactor vessel area. The cameras are installed by the inspectors at the same inspection as the seals on the vessel top are detached. The cameras are removed at the end of the refueling period when new seals are attached to the reactor vessel.

The storage bay area is constantly under surveillance by a film camera system and any observed movements of fuel flasks are compared with the operator's records.

At routine inspections the inspectors examine the facility accounting records and compare them with the state's reports for inventory changes, i.e., receipts of FF and shipment of SF, and plutonium production.

2. Safeguards Measures at San Onofre Unit 2

The following nuclear material accountancy measures and containment and surveillance measures have been implemented at the San Onofre Unit 2:

- (i) The facility records, i.e., general material ledger and records of power history, burnup, nuclear loss, and plutonium production, are examined at routine inspections and compared with state reports of transactions (inventory changes) and material balance.
- (ii) The inventory of core fuel has been verified in connection with the refueling and the reactor vessel is kept under Agency seals during reactor operation.

The fuel assemblies loaded into the core were identified and counted using the operator's underwater TV system, and for the spent fuel an attribute NDA verification was carried out with a

Cerenkov verifier.

Four metallic seals are attached to structures on the vessel top which necessarily have to be removed before the vessel can be opened.

- (iii) The inventory of spent fuel will be verified 1/ by NDA methods and it is kept under surveillance.

The NDA measurements will be carried out with an ION-1 instrument which has been developed at Los Alamos National Laboratory in cooperation with the Agency and was first used by the Agency at some other USA reactor facilities. The surveillance in the spent-fuel storage bay is by a twin Minolta film camera unit.

- (iv) The fresh fuel was verified after arrival on site and before it was moved into the storage bay.

The fuel assemblies were measured with an N-collar instrument also of Los Alamos National Laboratory design. In the storage bay the fresh-fuel assemblies were kept under surveillance by the twin camera unit mentioned above until the refueling took place.

- (v) The opened reactor vessel was kept under surveillance by a twin film camera unit installed in the reactor containment during the refueling period.

3. Inspection Activities at San Onofre Unit 2

The routine inspections at the facility are scheduled to take place every second month and the inspection activities comprise:

- (i) Examination of records and comparison with reports. This activity takes place at the Southern California Edison headquarters at Rosemead.
- (ii) Servicing of the film cameras in the storage bay area.

Two days are normally used for these activities.

In addition the following inspection activities are carried out; whenever possible the routine inspection activities (i) and (ii) are scheduled to take place in connection with these additional activities, i.e.:

- (a) When the reactor is shut down for refueling and the reactor vessel is to be opened, inspectors have to visit the facility for the removal of the seals on top of the vessel and for the temporary installation of a film camera unit in the containment.
- (b) When the reactor vessel lid is put back in place after a

1/ The verification is planned to take place during the second half of 1985.

refueling shutdown, another visit is necessary for attaching the seals and for the removal of the camera unit.

- (c) When the core has been refueled a physical inventory verification (PIV) is carried out. At the PIV the core inventory and the fuel in the storage bay are verified by identification, counting, and Cerenkov NDA.
- (d) The fresh fuel is verified immediately before it is moved from its dry storage position and into the bay.
- (e) The spent fuel in the storage bay will be verified by ION-1 NDA within half a year after the refueling.

III. EXPERIENCE

1. Inspection Statistics

The inspection programme which has been carried out at the San Onofre facility from November 1983 up to the end of March 1985, i.e., during 16 months, has comprised 10 inspections during 41 inspection man-days ^{1/} that were spent both at the plant and at the Rosemead headquarters. In total 10 inspectors have visited the facility. The inspection statistics are shown in Table 1.

^{1/} Inspection man-day is defined in item 249 of Reference 2.

Table 1: Inspection Statistics from November 1983 to March 1985

(a) Inspection No.	Date	Number of Inspectors	Man-days	Inspection Activities ^{1/}
1	831109, 10	4	8	DI, PR, RE
2	831208, 09	1	2	SE, PR, DI
3	840213	1	1	RE
4	840723, 24	1	2	RE, PR
5	840928 - 841003 ^{2/}	3	11	RE, ND/FF, IN
6	841022, 23	2	4	SE, IN, SC
7	841203	1	1	SC
8	850120 - 28 ^{2/}	2	6	PIV, RE, SC
9	850225, 26	1	2	SE, SC
10	850328, 29	2	4	RE, SC
Total			41	

(b) Inspector	A Fac. Off.	B	C	D	E	F	G	H Insp. Ass.	I Insp. Ass.	K SH ^{3/}	
Man-days	5	8	6	4	3	2	2	2	7	2	= 41
Number of Inspections	2	3	2	2	2	1	1	1	3	1	

In addition, one visit at the plant in May 1984 by the Director SGOB and the Operations Section Head.

- ^{1/} Legend: DI = Design Information Verification
RE = Records Examination
IN = Installation of Camera Unit
SC = Service of Camera Unit
SE = Attachment/Detachment of Seals on Reactor Vessel Top
PR = Preparation for NDA Measurements and/or Camera Unit Installation
ND/FF = NDA Measurement of Fresh Fuel
PIV = Physical Inventory Verification
- ^{2/} Period during which inspection took place.
- ^{3/} Section Head at the first inspection.

- (c) • Average time between inspections: 1.6 months
- Number of calendar days used at the plant for inspection activities including SCE training: 15
- Number of calendar days used for examination of records and reports at the SCE Rosemead HQ: 6
- Number of inspections with only one inspector: 5
- Number of inspections with two inspectors: 3
- Number of inspections with three or four inspectors: 2
- Average number of inspectors per inspection: 1.8
- Average number of man-days for a 12 month period: 31

2. Inspection History

When the Agency safeguards started at San Onofre, i.e., on 31 July 1983, the reactor unit No. 2 was in operation in its first fuel cycle, i.e., there was no spent fuel in the storage bay. The reactor had produced, at the time of the initial inventory, nearly 200,000 MW(Th) days.

At the first inspection in November 1983 the Design Information was verified as far as possible with the reactor in operation. At the inspection a presentation to and a discussion with the operator took place on (a) the purpose of IAEA safeguards, (b) the planned inspection activities, (c) Design Information matters, (d) Facility Attachment matters. The plant was described by the operator and possible camera unit locations were considered in the storage bay area. At the Rosemead headquarters the facility records were examined and compared with reports. The layout of a General Ledger for the nuclear material on the inventory was discussed.

As soon as a reactor shutdown took place and the reactor containment could be entered, the reactor vessel top was sealed by an inspector on 8th December 1983 using 3 Agency E-type metallic seals. On the same occasion the Design Information verification was completed. A detailed presentation of the planned NDA measurements of both fresh- and spent-fuel assemblies was given to the operator and the necessary preparations were identified. Furthermore, the camera box location in the storage bay area was determined and the time and cost needed for the necessary preparation and installation work was discussed.

At the inspection in July 1984 an unpleasant experience happened to the inspector who visited both the plant and the headquarters alone. Whilst visiting a bank in Los Angeles the inspector was robbed of his briefcase which contained not only his personal belongings but also the inspection working papers. This occurrence illustrates the "inspector life" and also, to a certain extent, what may lie behind the

Agency's policy of normally sending out inspection teams with two inspectors on the inspection missions.

The inspection activities from the fourth inspection were repeated at the fifth inspection at the end of September 1984. In addition, the fresh-fuel assemblies were verified with the N-collar, using a random sampling plan. Out of a total of 72 assemblies, 7 were measured. The enrichment was determined and the quantity of U-235. The total quantity of the fresh-fuel inventory was determined with an accuracy of less than 2%. In addition, the camera unit was installed in the storage bay area.

On the 21st October 1984 the reactor was shut down for refueling. The seals on the vessel top were removed by inspectors during the morning of the following day and a camera unit was installed in the reactor containment for temporary surveillance during the refueling period.

The physical inventory verification was carried out on 26th January 1985 by counting and identification of fuel assemblies, using the operator's underwater TV system. The spent fuel was also verified by a Cerenkov instrument.

Seals were attached on the reactor vessel on 26th February 1985 and the camera unit was removed from the reactor containment. The reactor was started again in early April.

3. Problem Areas

- (a) There are 19 inspectors of the Division of Operations B of the Agency's Department of Safeguards that are designated for carrying out inspections in the USA. All these inspectors are also designated in Canada, and some of them in other states as well. In order to establish an efficient and economical travel programme for these inspectors many factors have to be taken into account. With an optimal use of the inspectors it is not possible to have only a few of them assigned for one particular facility, which would have been desirable from the facility operator's point of view considering the training and familiarization aspect.

At San Onofre the training procedure established by the operator for getting the inspectors qualified for the entering of the plant has caused certain problems and maybe also misunderstandings. In order to qualify the inspector has to pass a training and examination programme during one full day. This qualifies him for escorted access to the plant during one year, i.e., the programme does not have to be repeated if he returns to the facility within one year. As can be seen from Table 1(b), 10 different inspectors visited the facility during a period of 16 months. Of these, 6 made more than one visit, and only 3 could profit from the one-year-effective training.

The Agency prefers to have the inspectors passing as brief a

training/preparation as possible and does not request unescorted access.

- (b) As in most of the LWR facilities, there are difficulties in obtaining an exact and definitive date for the physical inventory verification in advance so that the travels can be scheduled in the most optimal way. Because of the long distance from Vienna to California late changes in the planned PIV date may cause changes in travel arrangements which could have a cost impact. This problem has, however, not been more stressed at San Onofre than at other LWR facilities located a long distance from Vienna.
- (c) Because of programme coordination by the Agency the finalization of the San Onofre Facility Attachment took an unnecessarily long time. The Design Information as of September 1983 did not result in a final Facility Attachment draft until December 1984. The Agency is determined to reach an improvement of the Facility Attachment preparation procedure with the next generation of US facilities to come under inspection.

REFERENCES

1. Agreement between the United States of America and the International Atomic Energy Agency for the Application of Safeguards in the United States of America, INFCIRC/288, December 1981.
2. IAEA Safeguards Glossary, IAEA/SG/INF/1.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 32: INTRODUCTION TO EXXON NUCLEAR FUEL FABRICATION FACILITY.

The Exxon Nuclear Company operates an LWR fuel fabrication facility in Richland, Washington, which makes fuel assemblies for both pressurized-water and boiling-water reactors. The main characteristics of the facility will be described in detail from a number of viewpoints. These include the physical layout, material flows, process steps, and key measurement points and methods. The plant's physical layout and operation will be related to the general safeguards requirements for low-enriched uranium.

A one ton-a-day version of the Exxon Nuclear plant is developed to provide a reference or "Model Plant." The Model Plant is used to provide a quantitative example of the application of materials accounting to this type of bulk-handling facility. A material balance model is described that illustrates the relationships between measurement quality, material measurability, and process capabilities. The material in the session will help the participants understand the basis of the material accounting measures that are required for a bulk-handling facility. The Model Plant physical arrangement is also used to describe the internal material control areas into which the Model Plant is divided to meet US safeguards requirements for material control.

After the session, participants will be able to:

- Describe the major characteristics of the Exxon Nuclear fuel fabrication plant.
- Discuss the relationships between plant layout, material flows, process steps, and key measurement points.
- Better understand how the basic safeguard measures for low-enriched uranium are applied to a bulk-handling facility.
- Better understand the discussion of the Exxon Nuclear plant's nuclear material control and accounting system given in Sessions 33 and 34.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 32: INTRODUCTION TO EXXON NUCLEAR FUEL FABRICATION PLANT

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I. INTRODUCTION

The Exxon Nuclear low-enriched uranium fuel fabrication plant in Richland, Washington produces fuel assemblies for both pressurized water and boiling water reactors. Those fuels are supplied to light water reactors in the United States, Europe and Asia. The plant also supplies pellets to the Company's German affiliate, Exxon Nuclear GmbH, which operates a fuel assembly plant at Lingen in the Federal Republic of Germany. The Richland plant has been in operation since 1971 and the Lingen plant since 1979. The Richland plant has a capacity for processing UF₆ through to finished fuel assemblies of about 700 metric tons of uranium per year.

The Richland plant was the first U.S. bulk-handling facility selected by the IAEA for inspection under the U.S.-IAEA Safeguards Agreement. The plant was under IAEA inspection from March 1981 through October 1983.

This text for Session 32 provides a written description of the plant layout, operation and process. The text also includes a one ton-a-day model (or reference) plant which was adapted from the Exxon Nuclear plant. The Model Plant provides a generic example of a low-enriched uranium (LEU) bulk-handling facility. The Model Plant is used to illustrate in a more quantitative way some of the key safeguards requirements for a bulk-handling facility. The information on the Model Plant supplements the texts for Sessions 33 and 34.

The rest of the text is presented in the following order:

- (1) General Description;
- (2) Safeguards Considerations; and
- (3) Model Plant.

II. GENERAL DESCRIPTION

A. Plant Geographical Location

The Exxon Nuclear fuel fabrication plant is located in the City of Richland in the southeastern part of the State of Washington. The location of the plant within the Pacific Northwest is shown in Figure 1. The City of Richland is in the semi-arid region of Eastern Washington and is located on the juncture of the Columbia and Yakima Rivers.

22-3

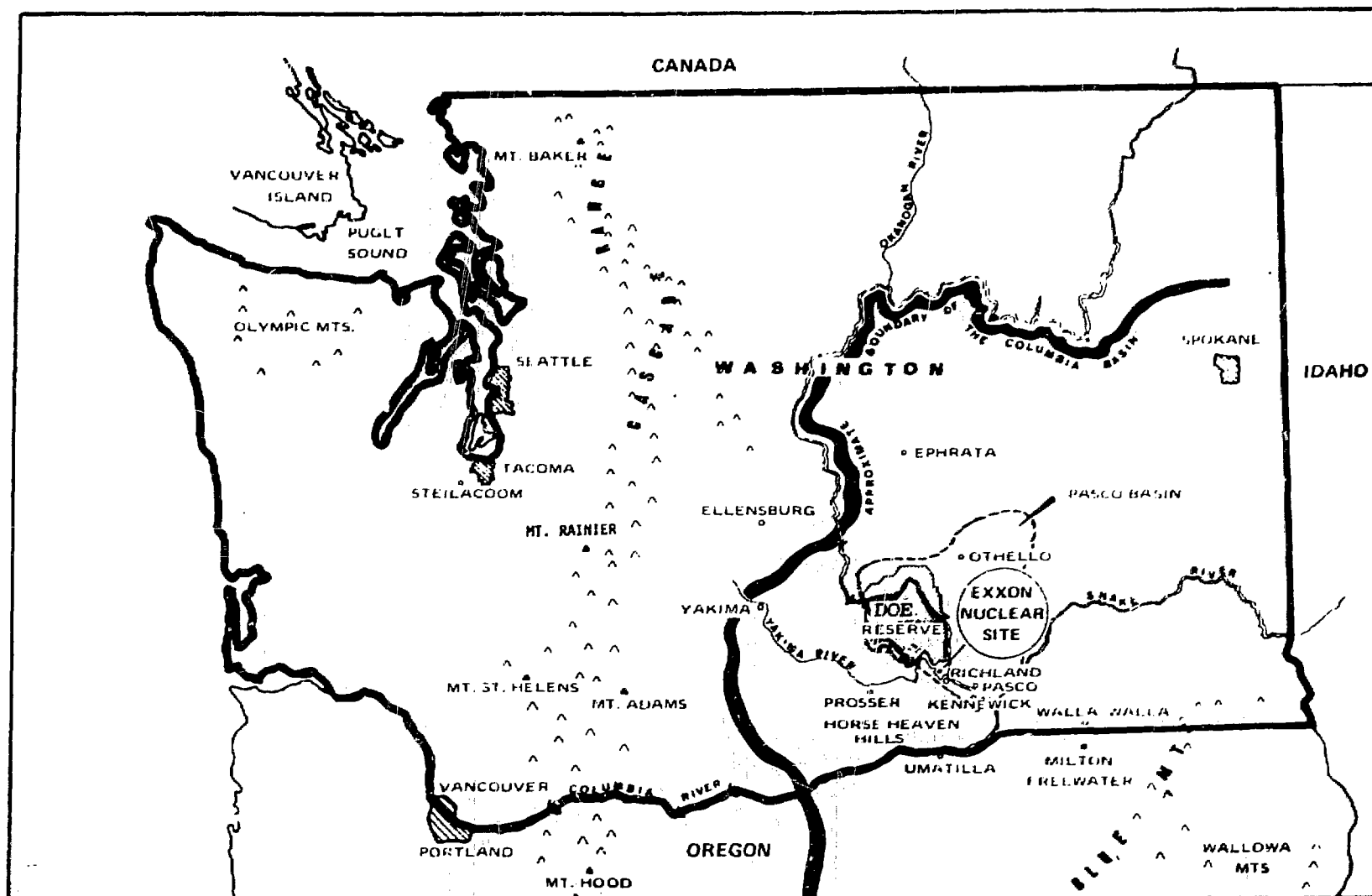


Fig. 1
Plant Location in State

B. Plant Site Arrangement

The plant layout and structures are shown in the artist's rendition in Figure 2. The figure shows the five solar evaporation ponds which are used to concentrate and contain the liquid waste from the conversion process. Figure 2 also shows the chain link fence and lighted perimeter which encloses the material access area and facilitates physical protection measures designed to control entry and exit from the material access area.

The various structures are identified by function in Figure 3 which gives an overhead view of the site. Figure 3 shows the office buildings in the upper part of the figure. The lagoon system is shown on the right side of Figure 3. The main process building, the UO_2 Facility, is shown in the center of the figure. The Special Fuels Building is shown to the left of the UO_2 Facility. The Special Fuels (also called Specialty Fuels) Building houses the process lines for making nuclear absorber and other special fuels.

The four internal material control areas, Nuclear Material Control Areas (NMCA's), into which the plant area is divided for internal control of nuclear materials is also shown in Figure 3. The four control areas are:

- (1) Logistics (NMCA-1);
- (2) UO_2 Shop (NMCA-2);
- (3) Specialty Fuels (NMCA-3); and
- (4) Engineering Laboratory (NMCA-4).

The Logistics NMCA includes the shipping and receiving staging areas, the UF_6 cylinder weighing and storage area, the solid waste storage area, and part of the Radioactive Materials Warehouse. The UO_2 Shop NMCA includes the UO_2 Fuels Plant, the lagoon system, and part of the Radioactive Materials Warehouse. The Specialty Fuels NMCA includes the Specialty Fuels Building and part of the Radioactive Materials Warehouse. The Engineering Laboratory NMCA includes the Engineering Laboratory Operation Building (ELO) and the Test Facility (FCTF).

C. Material Flows

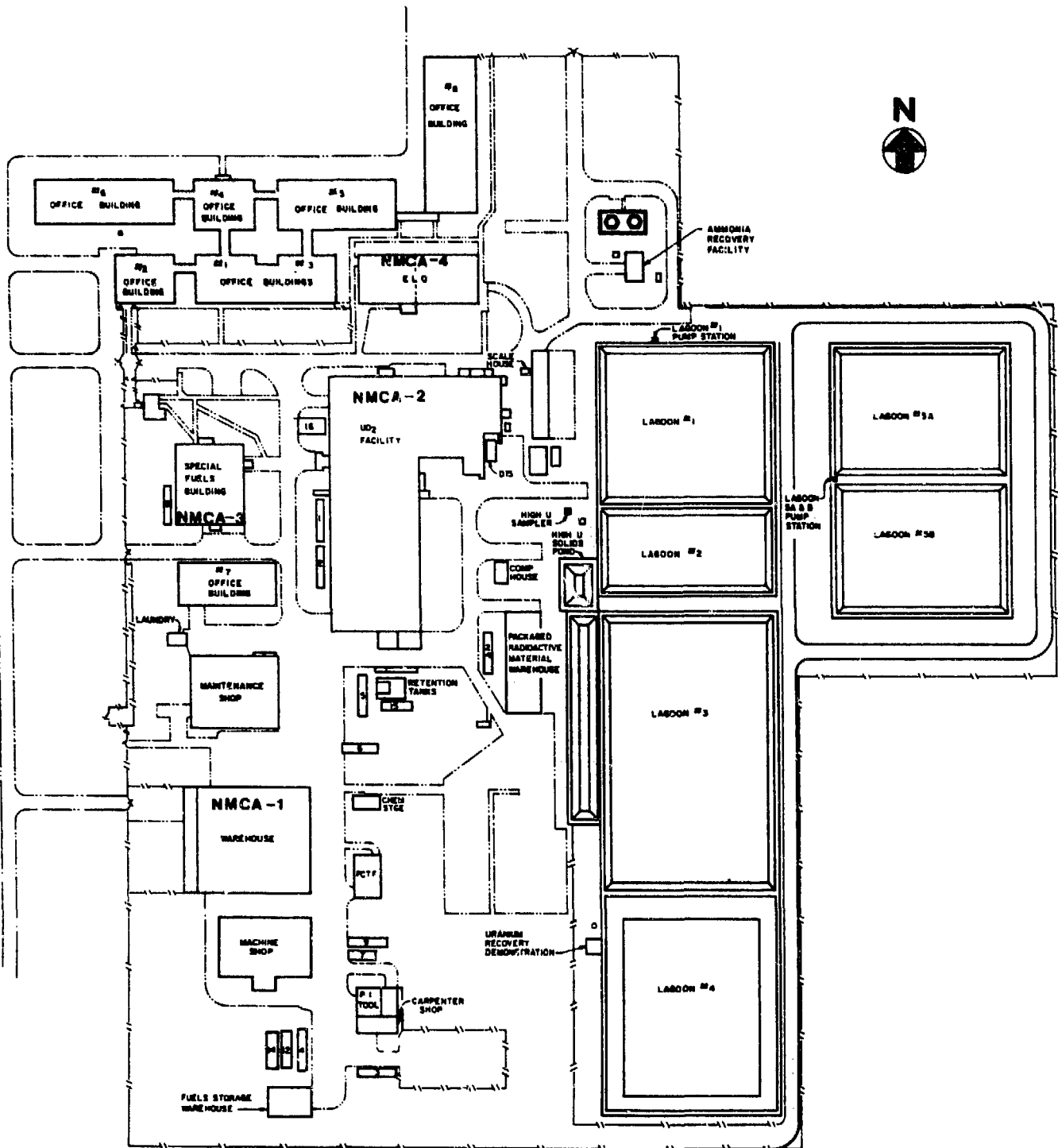
The feed to the plant is UF_6 which is transported to the plant by truck. The UF_6 is contained in cylinders (30-B cylinders) which are transported in overpacks. The trucks carrying the UF_6 enter the plant through the main gate (shown in Figure 3 adjacent to Office Building 2) and proceed to the UF_6 cylinder storage area between the UO_2 Facility and Lagoon 1. The overpacks are taken off the truck, opened, and the 30-B cylinders removed from the overpacks and weighed. Each cylinder contains about 1400 kilograms of uranium as UF_6 and about 42 kilograms of U-235 when the nominal enrichment is 3.0 percent U-235. After weighing, the cylinders are placed in storage on the UF_6 cylinder storage pad.

The cylinders are next transferred to the UO_2 Facility where the UF_6 is converted to UO_2 which is formed into pellets, sintered, ground to size, and loaded into fuel rods. After sealing and completion of finishing steps and final inspection, the rods are assembled into bundles and the finished bundles are packed in



Fig. 2
Artist's Rendition of Exxon Nuclear Plant Site

Fig. 3
Exxon Nuclear Plant Site



shipping containers (two bundles per container). A typical pressurized water reactor bundle will contain about 400 kilograms of uranium and a boiling water reactor bundle will contain about 200 kilograms of uranium.

The shipping containers of fuel bundles are loaded on trucks at the Shipping Staging Area (between the Maintenance Shop and the NMCA-1 Warehouse, shown in Figure 3). The fuel bundles are trucked to light-water reactors in the United States or to seaports (or airports) for shipment to Europe or Asia.

The pellets which are produced for the Lingen plant are packed in special pellet shipping boxes, placed in shipping containers and transported by truck to a seaport (or airport) for shipment to Germany.

The liquid waste from the conversion process is treated to remove all but the last traces of uranium, measured for uranium content (U and U-235), transferred to ammonia recovery, and then to the lagoons for storage and solar evaporation.

The contaminated handling materials such as gloves, cloths, paper towels, etc., are placed in 55-gallon metal barrels (drums) and measured for uranium content. Depending on the uranium content, they are sent either to a burial grounds or to on-site storage for later recovery of the uranium. The other form of solid waste, contaminated HEPA (air) filters, is treated in a similar manner. The filters are stored in the Radioactive Materials Warehouse and the waste barrels are stored on the Waste Barrel Pad located to the left of Lagoon 3 and 4 (see Figure 3).

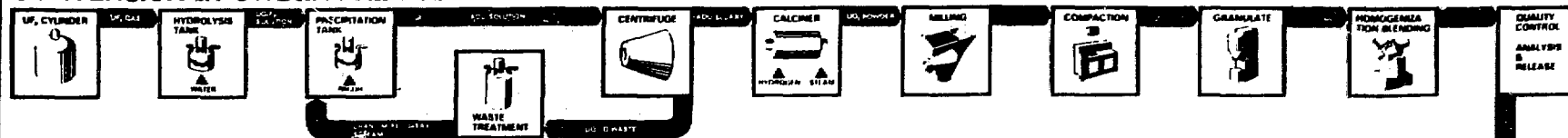
For the preparation of nuclear absorber fuels, UO_2 powder is transferred to the Specialty Fuels Building from the UO_2 Facility. The UO_2 powder is blended with a nuclear absorber (usually Gd_2O_3) and made into finished pellets and loaded into fuel rods. After finishing steps are completed, the nuclear absorber rods are loaded into specified positions in the fuel assemblies along with the regular (non-absorber) rods.

D. Processing Steps

A detailed diagram of the various process and quality inspection steps which are carried out in a typical low-enriched uranium fuel fabrication process is shown on Figure 4. A summary of the main process steps and locations of the key measurement points for the Exxon Nuclear plant are shown in Figure 5.

As shown in the figures, UF_6 is vaporized by heating, reacted with water to form uranium fluoride, and the uranium precipitated with ammonium hydroxide as ammonium diuranate (ADU). The ADU is separated from the slurry by centrifuging, placed in a rotating screw dryer, dried, and then calcined (heated) in a reducing atmosphere to form UO_2 powder. The calcined powder is homogenized by mechanically mixing and prepared for pressing by milling, compacting and granulating. The prepared powder is pressed into pellets. The pellets are ground to size by wet grinding and treated to remove moisture and loaded into fuel rods. The fuel rods are sealed by welding on end caps. After finishing steps are carried out, the rods are assembled into fuel bundles, placed in shipping containers, and shipped to light-water reactors.

CONVERSION & POWDER PREPARATION



PELLET FABRICATION



ROD PREPARATION



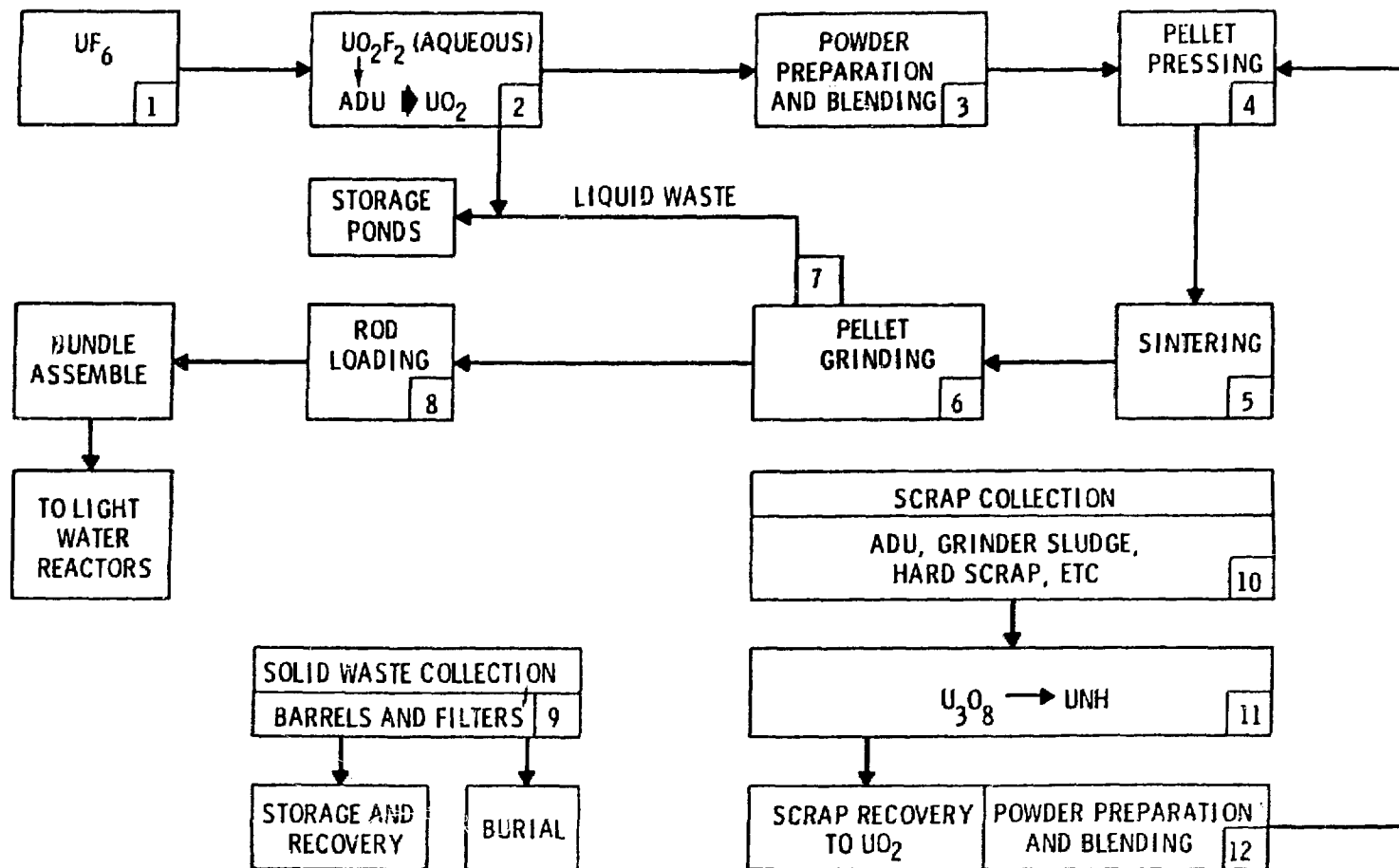
FUEL ROD FABRICATION



FUEL ELEMENT ASSEMBLY



Fig. 4
Flow Diagram
of a
Typical Fuel Manufacturing Process



X KEY MEASUREMENT POINTS

Fig. 5
Process Flow Diagram For Model Plant

The ammonical solution from the last centrifuge is further treated to remove uranium, measured for uranium content, and transferred to ammonia recovery and the storage lagoons as noted before.

The scrap generated during processing (grinder sludge, dirty powder, hard scrap of rejected pellets, and ADU from equipment cleanouts) is converted to U_3O_8 and the U_3O_8 dissolved in nitric acid. The resulting uranyl nitrate solution is sent to scrap recovery where it is precipitated with ammonium hydroxide, converted to UO_2 , and processed through to finished fuel in the normal process line.

The handling of solid waste barrels and filters was described in the previous section.

III. SAFEGUARDS CONSIDERATIONS

A. General

Safeguards measures for a low-enriched uranium fuel fabrication plant include materials accounting and material control. They also include the physical protective measures which are taken for materials of low strategic significance.

Materials accounting measures include the measurement and recording of the quantities of nuclear material received, shipped, discarded, and on inventory. Materials accounting measures for a bulk-handling facility include the maintenance of a measurement system which meets measurement uncertainty (quality) standards set by safeguards authorities. They also include the reporting to the safeguards authorities all quantities of nuclear material received, shipped, discarded, and on inventory.

Material control measures include the use of tamper-indicating seals to protect the prior measurement of the contents of an item, maintaining current knowledge of all discrete items, and maintaining a proper number of internal control areas to localize losses and to maintain custodial control. Formal transfers, including signatures and formal records, of the movement of material from one custodian to another are a part of custodial control measures.

For the physical protection of material of low strategic significance (e.g., low-enriched uranium), the main requirements include keeping nuclear material in a material access area where personnel access to the material is controlled by a system of barriers, sensors, personnel identification, and guard activities so as to provide for early detection of unauthorized access or unauthorized activities. Physical protection measures are also required for nuclear materials in-transit. These include maintaining current knowledge of the material in-transit and transporting the material in tamper-indicating sealed containers.

B. Material Accounting Measures Unique to Bulk-Handling Facilities

Materials accounting measures which are unique to a bulk-handling facility stem from the need to account for nuclear materials in bulk form. In this form, they can only be accounted for

by measuring the quantities (e.g., grams or kilograms of materials). Also, those quantities change when materials are processed and change form and are placed in various containers. Thus, accounting in a quantitative way requires that material balances be established in which the quantities entering and leaving are measured and recorded and that periodically the quantities present in the area be measured (inventoried) and compared to the quantities which should be present. This type of comparison requires knowledge of the measurement uncertainty involved so that loss can be distinguished from measurement uncertainty. Thus, the material balance accounting measures used in a bulk-handling facility require that a system of measurements be established for measuring all the various forms and quantities of nuclear material and that the measurements uncertainties of those measurements be estimated. It is also necessary to establish a system for properly combining and propagating those measurement errors so that they can be applied to distinguish loss from measurement uncertainty. This subject is discussed in more detail in Session 33. However, to make the plant description consistent with the safeguards measures taken for materials accounting, a "model plant" or reference plant is developed. This is used to illustrate the measurements, the measurement errors, and the combined measurement uncertainty for the Model Plant uranium material balance for a six-month accounting period. The main features of the Model Plant are described next.

IV. MODEL PLANT

A. Process Assumptions

The Model Plant operates at a rate of one tonne uranium per day for five days a week and 20 days a month. The nominal enrichment is 3.0 weight percent U-235. Feed to the plant is UF_6 and the plant product is finished fuel assemblies. The production of nuclear absorber fuel and the production of pellets for the Lingen plant are not included. Process steps include vaporization of UF_6 , reaction with water to form uranyl fluoride, precipitation with ammonium hydroxide as ADU, isolation of the ADU by centrifuging, and treatment of the centrate and other liquid waste to give a plant liquid waste loss of 0.3 percent of input. The ADU is dried, heated in a reducing atmosphere to yield UO_2 and mechanically mixed to provide homogenized lots of UO_2 which are prepared for pressing. The UO_2 powder is pressed into pellets. The pellets are sintered at a high temperature and ground to size. The finished pellets are weighed into fuel rods. After rod finishing steps are complete, the rods are assembled into fuel bundles. The fuel bundles are packaged and shipped to light water reactors.

The plant is assumed to generate solid waste trash and contaminated air filters at the rate of 0.2 percent of throughput. It is assumed that liquid wastes are generated at the rate of 0.3 percent of throughput so that the combined loss rate is 0.5 percent of throughput. This loss rate is chosen to illustrate the importance of waste measurements.

B. Six-Month Material Balance Model

The uranium material balance model for the plant process is shown in Table I. The material balance is formed around the process with the inputs consisting of UF_6 in cylinders and the outputs consisting of UO_2 pellets in rods and liquid and solid wastes. Inventory is assumed to consist entirely of measurable items of green powder and pellets, sintered pellets, U_3O_8 , hard scrap, dirty powder, ADU and grinder sludge.

The material balance period is assumed to begin with the completion of a physical inventory of the process area in which all items are measured and the total quantity of uranium in the material balance area (MBA) is determined. That quantity, the physical inventory (or what was found), becomes the ending inventory for the completed accounting period and the beginning inventory for the new accounting period. The process is then operated for a six-month period. The quantity of uranium in each addition is measured and added to the beginning inventory and quantity of uranium in each removal is measured and subtracted from it. This quantity, beginning inventory plus additions minus removals, is termed the book inventory. It represents the quantity of uranium which "should be" present. At the end of six months, the process is cleaned out, all the material is converted to measurable items and the physical inventory is taken. The quantity of uranium found in the MBA by the physical inventory is compared to the book inventory to obtain the difference between the book inventory ("what should be") and the physical inventory ("what is"). That difference is termed material unaccounted for and is denoted by the acronym MUF.

The process material balance shown in Table I illustrates the number of items and variety of materials which must be measured to form an accurate material balance. It also indicates number and variety of measurement errors which must be combined to determine the measurement uncertainty of the difference between the book inventory and the physical inventory, i.e., MUF.

The quantities of uranium contained in typical items are also shown in Table I. A 30-B cylinder of UF_6 contains about 1400 kilograms of uranium while a typical fuel rod contains only about 2.5 kilograms. Green UO_2 powder and U_3O_8 are stored in critically safe barrels holding 300-350 kilograms of uranium. Grinder sludge, ADU, hard scrap, and ADU are stored in 5-gallon buckets with contained quantities of uranium ranging from 10-17 kilograms. Green and sintered pellets are present at inventory time in sintering boats, each having about 18 kilograms of uranium in the form of green or sintered pellets. Finished pellets are stored on pellet trays holding about six kilograms of uranium.

Typical uranium compositions (percent uranium) are also shown in Table I for the various uranium compounds and process forms.

It should be noted that in actual practice, both the uranium element MUF and the U-235 MUF and their associated measurement uncertainties are determined. However, to illustrate the principles involved in the simplest way, only the uranium material balance is used in this text.

TABLE I

SIX MONTH URANIUM MATERIAL BALANCE MODEL FOR EXAMPLE PLANT

<u>Material Balance Component</u>	<u>Material Class</u>	<u>Percent Uranium</u>	<u>Number of Items</u>	<u>Kgs U per Item</u>	<u>Total U per Type</u>
<u>Additions</u>					
UF ₆	1	67.60	86	1400	120,400
<u>Removals</u>					
Rods (UO ₂ pellets)	2a	88.10	47,850	2.5	119,625
Waste Barrels	2b	--	960	0.2	192
Filters	2c	--	500	0.1	50
Liquid Wastes	2d	50 ppm	6	60	360
UF ₆ Heels	2e	67.60	86	2.0	172
<u>Inventory⁽¹⁾</u>					
Green Powder	3a	87.6	50	350	17,500
Green Pellets	3b	87.6	20	18	360
Sintered Pellets (on trays)	3c	88.10	1,000	6	6,000
Sintered Pellets (in boats)	3d	88.10	20	18	360
U ₃ O ₈ Powder	3e	84.8	14	300	4,200
Hard Scrap	3f	88.10	40	21	840
ADU	3g	60.0	40	10	400
Grinder Sludge	3h	80.0	20	12	240
Dirty Powder	3i	86.0	20	17	340

1 Quantities present for both beginning and ending inventory.

C. Measurements

The key locations where the flow of uranium through the plant and the quantities on inventory are measured are denoted by the numbered square boxes in Figure 5. The measurements and locations are described in Table II and the uranium assay measurements are summarized in Table III. Sampling methods are summarized in Table IV.

The various measurement methods are described briefly below.

1. Gravimetric Method. This method is used to determine the weight percents of uranium in UF_6 (outside laboratory), in UO_2 powders and pellets, and in scrap. For powders and pellets, five to ten grams of sample are loaded into a previously ignited, tared crucible, weighed, and the UO_2 ignited to U_3O_8 in a muffle furnace at $900^\circ \pm 25^\circ C$. The weight percent uranium calculation depends on the sample weights before and after ignition, the impurity content as determined from the spectrographic analysis and the calculated U_3O_8 to U gravimetric factor which includes the atomic weight of uranium as computed from isotopic concentrations measured with the mass spectrometer.

2. Titration Method. The Davies-Gray titration method is used to determine uranium in liquid waste, scrap, and pellet standards. An excess of ferrous sulfate is used to reduce U(VI) to U(IV) in a phosphoric acid medium containing sulfuric acid. Excess ferrous ion is destroyed with nitric acid using Mo(VI) as a catalyst. The titration is made potentiometrically with standard potassium dichromate in a sulfuric acid solution. The primary variables used in the calculation include the weight and normality of $K_2Cr_2O_7$, the volume of the titrant, the equivalent weight of U, and the sample weight.

3. Isotopic Composition. The mass spectrometer is used to determine isotopic composition (percent U-235). A solid sample deposited on a filament is thermally ionized. The ions travel through electrical and magnetic fields that accelerate and separate the ions into beams, each beam consisting of ions having the same mass-to-charge ratio. Separation of the ions is explained by an equation expressing the mass-to-charge ratio as a function of the magnetic field strength, the radius of curvature of the ion path, and the accelerating voltage. The magnetic field is varied to focus a specific ion on the detector. The detector output is recorded electronically and isotopic content is calculated from the voltages of the ion beams.

4. Fluorimetric Method. The uranium concentration of low-level liquid waste is determined using the fluorimetric method. Samples are fused with NaF-LiF and the amount of uranium determined by measuring the amount of fluorescence when activated with ultraviolet radiation. Samples are purified via solvent extraction where interfering materials are present.

5. Solid Waste Measurement. Uranium contaminated trash in 55-gallon barrels and contaminated HEPA filters are measured by an NDA system consisting of four sodium iodide (NaI) detectors and associated electronics and a barrel rotating fixture. The barrel (or filter) is rotated at about five rpm to provide an average count from the barrel independent of the radial location

TABLE II

DESCRIPTION OF KEY MEASUREMENTS FOR MODEL PLANT

Key Measurement Points	Measurement Description
(see Figure 5)	
1	Each cylinder of UF ₆ is weighed upon receipt and the cylinder tare weight is used to determine the net weight of UF ₆ .
(1)	Percent of uranium and U-235 are determined for each cylinder or for each group of cylinders with the same nominal composition using sealed samples taken at the enrichment plant and witnessed by an Exxon Nuclear employee or authorized agent.
(1)	After UF ₆ removal, the cylinder is weighed to determine the net weight of any residual heel using the cylinder tare weight.
2	The uranium concentration, enrichment, and volume of liquid wastes are measured using Lagoon 2 as a batching tank.
3	Each barrel of UO ₂ is weighed and the barrels are tared individually. The homogenized powder is sampled during filling of the barrels for measurement of percent uranium and U-235.
4	Each boat is weighed with the boats tared individually for green pellet inventory.
5	The loaded boats containing the same pellets, as at Measurement Point 4, are weighed for inventory of sintered pellets.
6	The loaded pellet trays are weighed; each tray is individually tared for sintered UO ₂ inventory.

TABLE II (Continued)

Key Measurement Points	Measurement Description
7	Centrifuged grinder water is measured in Lagoon 2, the Accountability Lagoon, along with other conversion wastes.
8	Pellet samples are sent to the analytical laboratory. Percents of uranium and U-235 are determined. The weight of the UO_2 in each rod is determined by weighing the fuel column stack before inserting into the rod. Accountability is maintained thereafter on a piece count basis.
9	Solid wastes (filters, solid wastes in barrels) are contained as a heterogeneous mass and measured for total U-235 by NDA.
10	All containers of dirty powder, ADU scrap, and grinder sludge are individually weighed, sampled, and assayed. The percent of uranium factor for hard scrap is the same as for sintered pellets.
11	Blended lots of U_3O_8 are sampled for percent uranium and U-235. Each barrel of U_3O_8 is weighed and barrels are tared individually.
12	Same as Measurement Point 3.

TABLE III
SUMMARY OF URANIUM METHODS

<u>Measurement Point</u>	<u>Analytical Method</u>	
	<u>Element</u>	<u>Isotope</u>
(see Figure 5)		
1 UF ₆	Gravimetric	Mass Spectrometer
2 Liquid Waste	Fluorimetric, Titration	Mass Spectrometer
3,12 UO ₂ Powder	Gravimetric	Mass Spectrometer
8 UO ₂ Pellets	Gravimetric	Mass Spectrometer
10 Dirty Powder, ADU	Gravimetric or Titration	Factor ⁽¹⁾
11 U ₃ O ₈	Gravimetric	Factor ⁽¹⁾
10 Grinder Sludge	Gravimetric or Titration	Factor ⁽¹⁾
9 Solid Waste	Factor	NDA

¹ Factors are based on traceability to starting materials.

TABLE IV
MODEL PLANT SAMPLING METHOD

<u>Key Measurement Point</u>	<u>Materials</u>	<u>Description</u>
(see Figure 5)		
1 UF ₆ Receipts	UF ₆	Sampled at enrichment plants. See UF ₆ sampling procedures, TID-7029, "Selected Measurement Methods for Plutonium and Uranium in the Nuclear Fuel Cycle," USAEC, 1963.
2,7 Liquid Waste Lagoon	Centrifuged Grinder Water Centrate and Mop Water Recovery	Sampled in Lagoon 2 by boat. Nine locations are sampled each at three depths.
3,12 UO ₂ Green Powder	UO ₂ Powder	Thief or scoop sampled during filling. Three samples are assayed from each UO ₂ lot.
11 U ₃ O ₈	+ U ₃ O ₈ Powder	
8 Sintered Pellets UO ₂ Product	UO ₂ Pellets	Random samples of whole pellets are taken for percent U and U assay. Five pellet samples per lot are taken for percent U and two for percent U-235.
10 Scrap Inventory	Grinder Sludge, ADU, Dirty Powder, U ₃ O ₈	All scrap is sampled by scoop after mixing of container contents by tumbling the container or by homogenizing before filling the container.

of the uranium. Lead shields around the detectors provide vertical and horizontal collimation to flatten out the system response due to variations in source location in the vertical direction.

6. Volume Measurements. The volumes of liquid waste are measured in Lagoon 2, the Accountability Lagoon, using a depth stick to measure the volume transferred from the Accountability Lagoon to one of the storage lagoons. A typical batch transfer is about 300,000 gallons. The depth stick is calibrated by dimensional measurements of the lagoon so that liquid depth can be related to volume.

7. Mass Measurements. UF₆ cylinders are weighed on the UF₆ cylinder scale having a capacity of 4000 kilograms (kg). All other weighings except fuel pellet columns and powder barrels are weighed on scales of 50 kg capacity. Fuel pellet columns are weighed on digital readout scales of 10 kg capacity and powder barrels are weighed on a digital readout scale of 800 kg capacity.

D. Error Parameters, Measurements and Combined Measurement Uncertainty (Sigma MUF)

The complexity of carrying out materials accounting at a bulk-handling facility is further illustrated by the number of measurements, recordings, error parameters, and calculations involved in computing the measurement uncertainty of the material balance in terms of the measurement uncertainty of MUF. The measurement uncertainty of MUF is expressed as the standard deviation of MUF (σ_{MUF} or σ_{MUF}).

The kind, variety and magnitude of the measurement errors which must be estimated and controlled to provide a basis for the sigma MUF calculation are given in Table V which shows the error parameter values for the Model Plant material balance for uranium element. The number of measurements made by each measurement method during the balance period and the type of items and quantities for which each measurement applies are shown in Table VI.

The last step in including the concept of measurement uncertainty in the application of materials accounting in a bulk-handling facility is the calculation of sigma MUF. This calculation is illustrated in Table VII which shows the calculations for the more important contributors to sigma MUF.

As illustrated in Table VII, the measurement uncertainty of the uranium MUF is computed by combining the measurement errors (random and systematic errors) associated with the measurement of each item comprising the material balance. The procedure used for the Model Plant, is to propagate the errors by method; e.g., weighing, sampling, and uranium assay for all components of the material balance and then combine the totals for each kind of measurement (e.g., weighing, sampling, analysis) into a total for all the measurements comprising the material balance. The measurement errors are propagated as variances (square of the standard deviations) and combined by adding variances (assuming the methods to be independent).

TABLE V

ERROR PARAMETER VALUES FOR MODEL PLANT URANIUM MATERIAL BALANCE

<u>Method</u>	<u>Description</u>	<u>Number</u>	<u>Component and Class⁽²⁾</u>	<u>σ, % RSD U (or as noted)⁽¹⁾</u>		
				<u>Random</u>	<u>S.T. System</u>	<u>L.T. System</u>
Weighing	Scale 1 UF ₆ Scale	W ₁	<u>Additions</u> --UF ₆ Full Cylinders, Class 1 <u>Removals</u> --UF ₆ Heels, Class 2e	0.40 kg	0.60 kg	0.15 kg
Weighing	Scales 2-13	W ₂ -W ₁₃	<u>Inventory</u> --Class 3a, 3b, 3c, 3d, 3e, 3f, 3g, 3h, 3i	8 g	--	6 g
Weighing	Scales 14,15	W ₁₄ , W ₁₅	<u>Removals</u> --Rods, Class 2a	0.30 g	--	0.20 g
Weighing	Powder Barrel Scale	W ₁₆	<u>Inventory</u> --Class 3a and 3e	200 g		150 g
Sampling	Sampling by Scoop or by Thief	S ₁	<u>Inventory</u> --Scrap, Class 3g, ADU	6.0	--	3.0
Sampling	Sampling by Scoop or by Thief	S ₂	<u>Inventory</u> --Scrap, Class 3h, Grinder Sludge	3.0	--	2.0
Sampling	Sampling by Scoop or by Thief	S ₃	<u>Inventory</u> --Scrap, Class 3i, Dirty Powder	3.0	--	0.5

TABLE V (Continued)

<u>Method</u>	<u>Description</u>	<u>Number</u>	<u>Component and Class(2)</u>	<u>σ, % RSD U (or as noted)(1)</u>		
				<u>Random</u>	<u>S.T. System</u>	<u>L.T. System</u>
Sampling	Dip Samples From Boat	S4	<u>Removals</u> --Liquid Waste, Class 2d	5.0	--	3.0
Volume	Liquid Depth Stick	V1	<u>Removals</u> --Liquid Waste, Class 2d	3.0	--	2.0
U-Assay	Gravimetric	U1	<u>Receipts</u> --UF ₆ Input, Class 1 <u>Removals</u> --UF ₆ Heels, Class 2e	0.013	--	0.005
U-Assay	Gravimetric	U2	<u>Removals</u> --Class 2a	0.02	--	0.015
U-Factor	Gravimetric	U3	<u>Inventory</u> --Class 3a, 3b	--	0.30	0.015
U-Factor	Gravimetric	U4	<u>Inventory</u> --Class 3c, 3d, 3f	--	0.030	0.10
U-Factor	Gravimetric	U5	<u>Inventory</u> --Class 3e, U ₃ O ₈	--	0.10	0.10
U-Assay	Gravimetric	U ₆ -U ₈	<u>Inventory</u> --Scrap, Class 3g, 3h, 3i	0.04	--	0.10
U-235 Assay	Passive NDA	U-235 ₁	<u>Removals</u> --Waste, Class 2b, 2c, Barrels and Filters	15.0	--	20.0 ⁽³⁾

TABLE V (Continued)

<u>Method</u>	<u>Description</u>	<u>Number</u>	<u>Component and Class(2)</u>	<u>σ, % RSD U (or as noted)(1)</u>		
				<u>Random</u>	<u>S.T. System</u>	<u>L.T. System</u>
U-Assay	Fluorimetric	U ₉	<u>Removals</u> --Liquid Waste, Class 2d	10	--	8

¹ RSD denotes relative standard deviation. S.T. System and L.T. System denote short-term and long-term systematic errors. Weighing errors for UF₆ cylinders are given as absolute standard deviations for gross weights of uranium. The tare is assumed to be constant. The other weighing errors are given as absolute standard deviations for net uranium weight.

² The material classes correspond to those given in Table I.

³ Illustrative composite value. Actual errors are calculated for each Class 2b and 2c from calibration error equations.

TABLE VI

MEASUREMENTS FOR MODEL PLANT SIX-MONTH URANIUM MATERIAL BALANCE

<u>Measurement</u>	<u>Material</u>	<u>Class</u>	<u>Method Number</u>	<u>Batch Size kg U</u>	<u>Total kg U by Method</u>	<u>Measurements by Method</u>		<u>Total Affected by L.T. System Error</u>
						<u>Random</u>	<u>S.T. System</u>	
Weighing	UF ₆	1	W ₁	1400	120,400	86	17 ⁽¹⁾	86 Items
Weighing	UF ₆ Heels	2e	W ₁	2.0	172	86	17	86 Items
Weighing	Pellet Columns (Rods)	2a	W ₁₄	2.5	59,813	23,925	--	23,925 Items
Weighing	Pellet Columns (Rods)	2a	W ₁₅	2.5	59,813	23,925	--	23,925 Items
Weighing	Green Powder	3a	W ₁₆	350	35,000	100	--	50 Items (2)
Weighing	Hard Scrap	3f	W ₂	21	420	20	--	10 Items (2)
Weighing	ADU	3g	W ₂	10	400	40	--	20 Items (2)
Weighing	Dirty Powder	3i	W ₂	17	340	20	--	W ₂ = $\frac{10 \text{ Items (2)}}{40 \text{ Items (2)}}$
Weighing	Hard Scrap	3f	W ₃	21	420	20	--	10 Items (2)
Weighing	ADU	3g	W ₃	10	400	40	--	20 Items (2)

TABLE VI (Continued)

Measurement	Material	Class	Method Number	Batch Size kg U	Total kg U by Method	Measurements by Method		Total Affected by L.T. System Error
						Random	S.T. System	
Weighing	Dirty Powder	3i	W ₃	17	340	20		W ₃ = $\frac{10 \text{ Items (2)}}{40 \text{ Items (2)}}$
Weighing	Green Pellets	3b	W ₄	18	360	20	--	10 Items (2)
Weighing	Green Pellets	3b	W ₅	18	360	20	--	10 Items (2)
Weighing	Sintered Pellets (on trays)	3c	W ₆	6	6,000	1,000	--	500 Items (2)
Weighing	Sintered Pellets (on trays)	3c	W ₇	6	6,000	1,000	--	500 Items (2)
Weighing	Sintered Pellets (in boats)	3d	W ₈	18	360	20	--	10 Items (2)
Weighing	Sintered Pellets (in boats)	3d	W ₉	18	360	20	--	10 Items (2)
Weighing	Grinder Sludge	3h	W ₁₁	12	240	20	--	10 Items (2)
Weighing	Grinder Sludge	3h	W ₁₂	12	240	20	--	10 Items (2)

TABLE VI (Continued)

<u>Measurement</u>	<u>Material</u>	<u>Class</u>	<u>Method Number</u>	<u>Batch Size kg U</u>	<u>Total kg U by Method</u>	<u>Measurements by Method</u>		<u>Total Affected by L.T. System Error</u>
						<u>Random</u>	<u>S.T. System</u>	
Weighing	U ₃ O ₈	3e	W ₁₆	300	8,400	28	--	14 Items (2)
Sampling	ADU	3g	S ₁	10	800	80	--	--
Sampling	Grinder Sludge	3h	S ₂	12	480	40	--	--
Sampling	Dirty Powder	3i	S ₃	17	680	40	--	--
Sampling	Liquid Waste	2d	S ₄	60	360	54	--	360 kgs
Volume	Liquid Waste	2d	V ₁	60	360	6	--	360 kgs
U-Assay	UF ₆	1	U ₁	1400	120,400	17	(full- heels)	120,228 kgs
U-Assay	UF ₆ Heels	2e	U ₁	2.0	172	17		
U-Assay	UO ₂ in Rods	2a	U ₂	1200	119,625	500	--	119,625 kgs
U-Factor	UO ₂ Unsintered	3a,3b	U ₃	1200	35,720	--	30 (lot-to-lot) (variation)	--
U-Factor	UO ₂ Sintered on Inventory	3c,3d	U ₄	1200	14,400	--	12 (lot-to-lot) (variation)	--

TABLE VI (Continued)

<u>Measurement</u>	<u>Material</u>	<u>Class</u>	<u>Method Number</u>	<u>Batch Size kg U</u>	<u>Total kg U by Method</u>	<u>Measurements by Method</u>		<u>Total Affected by L.T. System Error</u>
						<u>Random</u>	<u>S.T. System</u>	
U-Factor	U ₃ O ₈	3e	U ₅	300	8,400	--	28 (lot-to-lot) (variation)	--
U-Assay	ADU	3g	U ₆	10	800	80	--	--
U-Assay	Grinder Sludge	3h	U ₇	12	480	40	--	--
U-Assay	Dirty Powder	3i	U ₈	17	680	40	--	--
U-235 Assay	Waste Barrels	2b	U-235 ₁	0.2	192	960	--	192 kgs
U-235 Assay	Filters	2c	U-235 ₁	0.1	50	500	--	50 kgs
U-Assay	Liquid	2d	U ₉	60	360	54	--	360 kgs

¹ The short-term systematic error for UF₆ cylinder weighing is assumed to affect all cylinders weighed on a given day. For the 86 cylinders and heels, five cylinders are weighed per day for 17 days and one cylinder weighed during one day.

TABLE VII

EXAMPLE OF SIGMA MUF CALCULATIONS FOR MAIN ERROR CONTRIBUTORS
TO THE SIX-MONTH URANIUM MATERIAL BALANCE

Measurement	Material	Class	Method Number	Items or Quantities ⁽¹⁾ Affected by Types of Error			σ^2 , kg ² U by Method ⁽²⁾		
				Random	S.T. System	L.T. System	Random	S.T. System	L.T. System
Weighing	UF ₆	1	W ₁	86	17	86	13.76	153.36	--(3)
Weighing	UF ₆ Heels	2e	W ₁	86	17	86	13.76	153.36	--
Weighing	Sintered Pellets (Rods)	2a	W ₁₄	23,925	--	23,925	0.002	--	22.90
Weighing	Sintered Pellets (Rods)	2a	W ₁₅	23,925	--	23,925	0.002	--	22.90
Weighing	Green Powder	3a	W ₁₆	100	50(2)	4.0	--	--	
Weighing	U ₃ O ₈	3e	W ₁₆	28	Total W ₁₆	28(2) 78(2)	1.12	--	273.78
Sampling	Liquid Waste	2d	S ₄	54		360 kg	6.0	--	116.64
Volume	Liquid Waste	2d	V ₁	6	--	360 kg	19.44	--	51.84
U-Assay	Sintered Pellets (Rods)	2a	U ₂	500	--	119,625	1.14	--	321.98

TABLE VII (Continued)

Measurement	Material	Class	Method Number	Items or Quantities ⁽¹⁾ Affected by Types of Error			σ^2 , kg ² U by Method ⁽²⁾		
				Random	S.T. System	L.T. System	Random	S.T. System	L.T. System
U-Assay	Green Powder	3a	U ₃	--	30 lots ⁽⁴⁾	--	--	367.5	--
U-Assay	U ₃ O ₈	3e	U ₅	--	28 lots ⁽⁴⁾	--	--	2.52	--
U-Assay	Waste Barrels	2b	U-235 ₁	960	--	192 kgs	0.864	--	--
U-Assay	Filters	2c	U-235 ₁	500	--	50	0.112	--	--
							Total U-235 ₁		2342.56
U-Assay	Liquid Waste	2d	U ₉	162	--	360	5.12	--	829.44

- 1 Unless otherwise noted as kilogram quantities, the numbers shown refer to the number of items affected by random, short term systematic, or long term systematic errors.
- 2 Combined measurement error variance for measurement method and material class. Individual measurement error values used in the calculation were given previously in Table V.
- 3 Long term systematic weighing errors for UFG cylinder and heels are assumed to cancel.
- 4 Long term systematic analytical errors are assumed to cancel for inventory items.

An example calculation is shown in Table VII for the weighing errors associated with the weighing of pellet columns. The random error variance for weighing all the pellet columns during the material balance period on one of the two rod loading scales is computed by multiplying the number of weighings (23,925, from Table VI) by the weighing random error variance (the square of the standard deviation of 0.3 grams, from Table V). That calculation results in a random error weighing variance for one rod loading scale of 0.002 kg^2 of uranium. The systematic error variance for weighing the 23,925 pellet columns on the scale is computed by multiplying the square of the number of weighing by the systematic error variance (square of the systematic error standard deviation from Table V of 0.2 grams). This results in a systematic error variance for weighing on one of the rod loading scales of 22.90 kg^2 of uranium.

A summary of the total measurement uncertainty for the Model Plant uranium balance is shown in Table VIII. The summary in Table VIII includes all the measurements and errors which contribute to MUF not just those highlighted in Table VII. As shown in Table VIII, the Model Plant has a relatively low sigma MUF value of 0.058 percent of input, that value reflects the high quality measurements made on feed and product and the relatively small amounts of difficult-to-measure scrap materials present on inventory such as ADU because of the conversion of such scrap materials to a more measurable form such as U_3O_8 .

E. Material Control Areas for Model Plant

For safeguards reporting purposes the entire plant site is treated as a single material balance area. Emphasis is given to reporting transactions which change the plant book inventory such as shipments and receipts and to periodically reporting the status of the material balance in terms of MUF and inventory holdings. Inventory changes such as receipts and shipments are reported in Nuclear Material Transaction Reports (Form DOE/NRC-741) to the U.S. NRC and via NRC to the IAEA. Similarly, the status of the material balance is reported periodically (every six months) through Material Balance Reports (Form DOE/NRC-742) and Physical Inventory Listings (Form DOE/NRC-742C).

To meet U.S. safeguards material control requirements for isolating losses, providing custodial control, and maintaining current knowledge of discrete items, the plant reporting MBA is divided into a number of smaller material control areas. The transactions and inventory holdings in those smaller areas are not reported separately to the NRC and the IAEA but are maintained as internal plant information.

Two types of internal control areas are maintained: material (or mass) balance areas (MBAs) and item control areas (ICAs). In processing areas where material is handled in bulk form, material balance areas are maintained in which a separate book inventory is kept of the quantities of element and U-235 isotope present in the MBA. For storage areas or processing areas where uniquely identified and usually sealed containers are handled, item control areas are maintained in which records of each item and

TABLE VIII

SUMMARY OF MODEL PLANT SIGMA MUF CALCULATIONS

<u>General Measurement Method</u>	<u>Total Error, σ^2 kg² U</u>		
	<u>Random</u>	<u>S.T. Systematic</u>	<u>L.T. Systematic</u>
Weighing	31.66	306.72	355.90
Sampling	50.39		116.84
Volume	19.44		51.84
U-Assay	21.23	370.02	3,580.7
Subtotal	122.72	676.74	4,105.28
Total Sigma ² MUF = 4,904.74 kg ² U			
Total Sigma MUF = 70.03 kg U			
Sigma MUF as Percent of Input = 0.058% of Input			

the associated quantity of element and U-235 isotope are maintained. Each control area has a designated custodian and usually one or more alternate custodians who are responsible for the custody of material in their areas.

The division of the plant into material control areas is shown in Figure 6. As shown in the figure, the UO₂ Plant area is divided into four MBAs and two ICAs. The first three MBAs in the UO₂ Plant enclose the production process and divide it into three processing MBAs. The fourth MBA encompasses the Analytical Laboratory. The powder storage area (ICA-3A) represents the main storage area for bulk quantities of UO₂ and U₃O₈ whereas ICA-2 (Rod Storage and Bundle Assembly) represents both the storage and processing of sealed items.

Shipping and receiving makes up ICA-1 since only identifiable items are shipped or received. The storage of various scrap, waste and archive items is in a group of ICAs (ICA-3B to 3K) located in the warehouse, trailers and the waste barrel pad.

It should be noted that material control area structure for the actual plant is different than that described for the Model Plant. The material control area structure described for the actual plant in Section II, represents a recent change whereas the material control structure for the Model Plant represents the previous plant arrangement. The subject of material control areas is discussed further in Session 33.

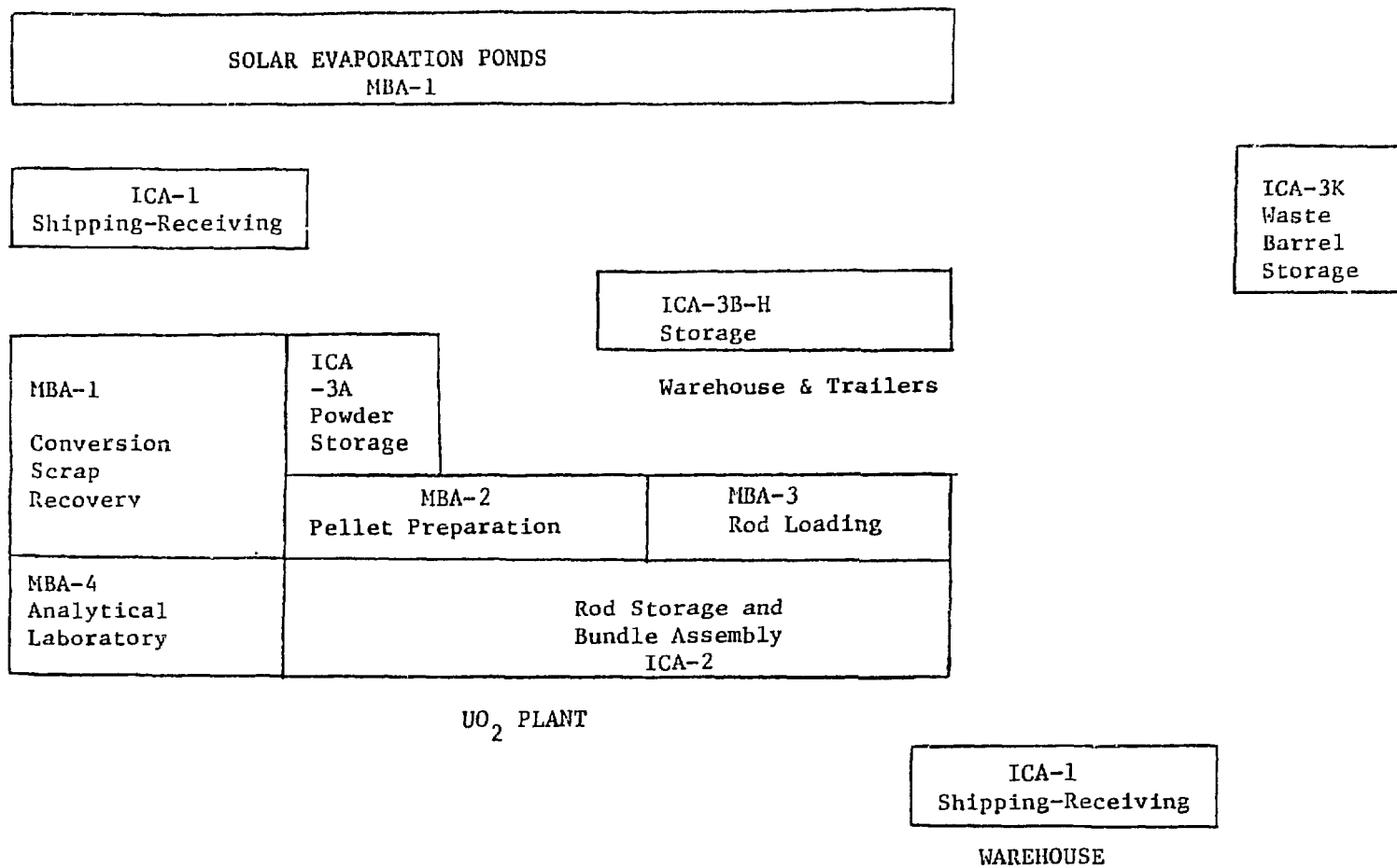


Fig. 6
Inventory Locations



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSIONS 33 & 34: NUCLEAR MATERIAL CONTROL AND ACCOUNTING AT EXXON NUCLEAR

The nuclear material control and accounting system at Exxon Nuclear will be described in detail. Subjects to be discussed will include:

- The basis of the MC&A system
- The nuclear materials accounting systems (NMRS and NICS)
- Physical inventory taking
- IAEA inspection experience
- Safeguards organization
- Measurements and measurement control
- MUF evaluation
- Accounting forms and reports
- Use of tamper-indicating seals

The general requirements for material accounting and control in this type of a bulk-handling facility will be described. The way those requirements are met for the subject areas shown above will be illustrated using a reference (Model Plant) version of the Exxon Nuclear plant. The difference between the item-accounting procedures used at reactor facilities and the bulk-accounting procedures used at fuel fabrication facilities will be discussed in detail.

After the session, participants will be able to:

- Describe the major features of the material control and accounting system (MC&A) at Exxon Nuclear.
- Contrast the MC&A system at Exxon Nuclear with the MC&A system at SONGS.
- Discuss the differences between item-accounting and bulk accounting procedures.
- Understand how the Exxon Nuclear accounting system is designed to meet the safeguards requirements of the IAEA and the US state system.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 33: MATERIAL CONTROL AND ACCOUNTING AT EXXON NUCLEAR, I

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I. INTRODUCTION

This presentation describes the design approach used at the Model Plant to meet U.S. material control and accounting (MC&A) requirements. The underlying principles involved in meeting each requirement are illustrated by the approaches implemented at the Model Plant.

II. REQUIREMENTS FOR MATERIAL CONTROL AND ACCOUNTING

The general requirements for material control and accounting for the U.S. national safeguards system are given below for bulk handling facilities.

1. Measure all quantities of nuclear material received, shipped, discarded, and on inventory.
2. Maintain a formal accounting system for recording all internal and external transactions and measurements.
3. Conduct physical inventories.
4. Maintain material control areas.
5. Maintain current knowledge of discrete items.
6. Provide to the safeguards authorities reports of all external transfers, material status, and all usual events or discrepancies.
7. Maintain a measurement control program for estimating and controlling measurement errors.
8. Provide for audits, verification activities, and performance evaluation by safeguards inspectors.

Many of the above general requirements are also common to Euratom and IAEA safeguards.

However, since a main objective of the course is to illustrate how national system requirements are met, the general requirements of the U.S. system are used for purposes of illustration. The detailed accountability requirements for U.S. plants are given in Title 10, Parts 70, 74 and 75 of the Code of Federal Regulations. This session will illustrate only the more universal features of those detailed requirements.

The implications of the general requirements and the approaches taken in the design of the Model Plant (MC&A) system are discussed for each general requirement in the next section.

III. MATERIAL CONTROL AND ACCOUNTING DESIGN CONSIDERATIONS

A. Measurements

1. Completely Measured Material Balance. The requirement that all quantities of nuclear material received, shipped, discarded, and on inventory be measured has a number of important implications. First, it requires a completely measured material balance. This, in turn, requires a measurement process for each type of item comprising the material balance. It also requires of bulk handling facilities that the engineering design be such to ensure that all quantities be included in the material balance, e.g., that there are no unmeasured effluents and that all of the holdup inventory can be measured.

In addition, the U.S. and IAEA safeguards systems require that the measured material balance be of a certain minimum quality in regard to measurement uncertainty.

a. Receipts and Shipments. For the Model Plant, receipts are cylinders of essentially pure UF_6 and product shipments are fuel assemblies of also essentially pure UO_2 . For these items, extremely high quality measurements (weight and assay of pure compounds) can be made.

b. Waste Discards. The measurement of waste discards presents a more difficult problem. Waste materials are typically difficult-to-measure material forms. Further, serious consideration must be given to the process and facility design to ensure that all discards are included in the measurement system.

For the Model Plant, liquid wastes transferred to the solar evaporation lagoons are measured by the volume and concentration of each batch volume transferred through the Accountability Lagoon. Solid wastes transferred to burial or retained waste are measured for U-235 content by passive gamma counting using calibration standards that simulate the waste materials.

The completeness of the measurement system is established in the following manner:

- All discharges to the atmosphere are monitored for uranium;
- There is only one transfer line to the lagoons;
- There are no floor drains in the process building;
- All liquid vessels have double containment and all liquid effluents are monitored for low level uranium concentrations;
- All potential liquid discharges to the ground are monitored by a system of wells;
- Potential tracking losses on shoe covers and protective clothing are monitored at the laundry; and
- All equipment which is removed for burial is checked for uranium by radiation measurements and wiping tests.

c. Inventory. As was noted earlier, each inventory item for the Model Plant is a discrete measurable item. The bulk of the inventory is based on measurements made in advance of the inventory. The inventory cleanout items represent the quantities

of ADU, dirty powder, filters, etc., which are removed in the cleanout of process equipment and associated plenums and ducts. Each item such as bucket of ADU or dirty powder is weighed and assayed for inclusion in the inventory. The small quantities of material which are not removed by rigorous cleanouts are not included in the inventory, but are treated as material unaccounted for (MUF).

The problem of system completeness also applies to the inventory of material held up in equipment and duct work. To assure the completeness of cleanouts, each piece of process equipment is inspected after cleanout. Duct work and plenums are examined to ensure that material has been removed for inclusion in the inventory and to ensure that there are no buildups of material downstream of the cleanout points.

2. Quality of the Measured Material Balance

a. Sigma MUF Criteria. For U.S. bulk handling facilities, quantitative criteria for material balance accounting are expressed in terms of materials unaccounted for or MUF. By way of review, MUF is illustrated in Figure 1. As shown in Figure 1, if there are no unmeasured losses and no unmeasured inventory (hold up) and if there are no measurement uncertainties, MUF would be zero. However, measurement uncertainties are inherent to measurement processes. Thus, even under ideal conditions of no unmeasured losses and no unmeasured inventory, MUF will not be exactly zero. Although measurement uncertainties are inherent, they can be minimized by employing high quality measurements.

In the U.S., the need for a certain minimum measurement quality is specified by numerical values for the limit of error of MUF or LEMUF.

This is illustrated in Figure 2. For a low enriched uranium fabrication plant, the U-235 LEMUF is required to be less than 0.5 percent of throughput for the six-month plant material balance. This compares to an IAEA suggested standard of accountability of 0.6 percent of throughput for LEMUF (2° MUF). The alarm level for MUF is LEMUF. Thus, the observed MUF should be less than LEMUF or an investigation of the cause of MUF exceeding the LEMUF is required.

b. Evaluation of Sigma MUF. In theory, to design the measurement and accounting system to achieve an acceptable LEMUF is a complex problem involving a number of factors. These include measurement processes, measurement errors, number of measurements, accounting practices, process aspects, and error propagation considerations.

In practice, the problem is more commonly one of establishing a measurement system to meet both manufacturing and accountability needs and then evaluating the LEMUF of that system versus safeguards numerical criteria for LEMUF.

To illustrate some of the system design considerations applicable to a low-enriched uranium fabrication plant, an example is developed and shown in Table I.

MUF = MATERIAL UNACCOUNTED FOR

Material Balance Area

- $MUF = \text{Beginning Inventory} + \text{Receipts} - \text{Removals} - \text{End. Inventory}$
- $\text{Book Inventory} = (\text{Beginning Inventory} + \text{Receipts} - \text{Removals}) = \text{"What Should Be"}$
- $\text{Physical Inventory} = \text{Ending Inventory} = \text{"What Is"}$
- $MUF = \text{Book} - \text{Physical or} = (\text{"What Should Be"}) - (\text{"What Is"})$
- If No Losses, Perfect Measurements: $MUF = 0$
- $MUF \neq 0$; Could Be Losses (Buildup), Measurement Uncertainty or Theft

Fig. 1
MUF

LIMIT OF ERROR OF MUF (LEMUF)

- $LEMUF = 2 \sigma_{MUF}$
- Limits of MUF Due Solely to Measurement Uncertainties
- $\sigma^2_{MUF} = \sigma^2_{Bl} + \sigma^2_{Rec} + \sigma^2_{Rem} + \sigma^2_{El}$
= Propagation of All Measurement Errors For All Items
(Quantities) in the Material Balance Equation
- $LEMUF < 0.5\%$ of Throughput for Uranium Fabrication
- Alarm Point: $MUF > LEMUF$

Fig. 2
LEMUF

TABLE I
EXAMPLE URANIUM LEMUF EVALUATION

Component	% Input	Measurement	Combined Error, % σ	σ , % Input	% Variance of MUF
<u>Receipts</u>					
UF ₆	100	Weight-Assay	0.04	0.04	4.8
<u>Removals</u>					
1. UO ₂ Rods	99.5	Weight-Assay	0.04	0.04	4.8
2. Liquid Waste	0.3	Volume-Concentration	20	0.06	10.9
3. Solid Waste	0.2	NDA	30	0.06	10.9
<u>Inventory</u> (Beginning and Ending)					
1. UO ₂ Pellets	10	Weight-Factor	0.06	0.006	0.1
2. UO _x Powders	10	Weight-Factor	0.1	0.01	0.3
3. Difficult-to - Measure Scrap	5	Weight-Factor	3	0.15	68.1

$$U, {}^0\text{MUF} = 0.182\%$$

$$U, \text{LEMUF} = 0.364\%$$

The first two columns in Table I represents a simplified material balance for the plant expressed in strata of like items and quantities expressed as a percentage of input (or throughput). The third column shows the measurement methods for determining the quantities of uranium element in each item. The fourth column shows the expected relative error for the entire stratum, that is the combined random and systematic error of the stratum expressed as a relative standard deviation. The fourth column shows the relative measurement uncertainty as a percentage of plant input. The last column shows the percentage contribution of each stratum to the variance of MUF (σ^2_{MUF}). The propagated relative σ_{MUF} of 0.182% is the root mean square of the values in Column 5, e.g.,

$$\% \sigma_{\text{MUF}} = \sqrt{(0.04)^2 + (0.04)^2 + (0.06)^2 + (0.06)^2 + \dots}$$

The example σ_{MUF} evaluation in Table I illustrates several concepts important to the design of the accountability system. The first is the concept of identifying the individual contributors to σ_{MUF} and evaluating the relative impact of each stratum or measurement process on σ_{MUF} . The second is the concept of the relationship between the process and σ_{MUF} . This latter point is illustrated by the contribution to σ_{MUF} from waste and scrap measurements. (Note: The large waste errors are chosen for illustration only.) The measurement errors of those items comprise about 90 percent of the variance of MUF. It is seen from Column 2 that their impact is directly related to the percentage of waste and difficult-to-measure scrap generated by the process. The adverse effect of those items on σ_{MUF} could be reduced by process improvements which reduce the waste losses and amount of difficult-to-measure scrap. For example, conversion of the difficult-to-measure scrap to a more measurable form such as U_3O_8 would significantly reduce the σ_{MUF} .

c. U σ_{MUF} and U-235 σ_{MUF} . In the course, the uranium element MUF and σ_{MUF} are emphasized. This is not to imply that enrichment measurements and enrichment control are not important, but rather to emphasize that the basic measurements and accounting features for fuel fabrication deal first with uranium element and secondly with the enrichment.

The U.S. criterion for LEMUF is for the U-235 LEMUF not the U element LEMUF. However, whether the enrichment measurement errors actually enter into the U-235 LEMUF is highly dependent on processing and accounting booking practices. Since fuel fabrication does not change the enrichment but only blends enrichments, U-235 measurement errors generally affect U-235 MUF accounting only when enrichment blending takes place. Thus, if the enrichment of the feed UF_6 is maintained throughout the process such that the accounting records can be based validly on the enrichment of the starting UF_6 , then the U-235 measurement errors do not contribute to the U-235 LEMUF. In this case, the relative U-235 LEMUF is the same as the relative U element LEMUF. However, if the measured enrichments of feed, product, and inventory are

booked into the accounting records, even though no enrichment blending has taken place, then those enrichment measurement errors will contribute to the U-235 LEMUF. In this case, the relative U-235 LEMUF will be larger than the relative U element LEMUF.

In the Model Plant arrangement selected for this course, the isotopic integrity of about 85 percent of plant input is maintained at the starting feed enrichment and only about 15 percent of throughput is affected by enrichment measurement errors. To assure that the isotopic integrity is maintained, mass spectrometric measurements are made on each lot of UO₂ powder and pellets. Those measurements are treated as verification measurements and are not booked into the accounting records unless a statistically significant difference is noted (e.g., evidence that blending took place). The 15 percent of throughput processed through scrap recovery where blending takes place is also measured for enrichment. Those measurements are booked into the accounting records and those enrichment measurement errors contribute to the U-235 LEMUF.

B. Accounting Requirements and Model Plant System

1. General Description. The U.S. national system requires that a formal accounting system be maintained for recording all receipts, shipments, waste discards, inventories, and internal transfers. To meet those requirements, the Model Plant employs a computer based system to account for low enriched uranium by project (reactor reload), enrichment, and plant location. The separate accounting by project account or reactor reload is for commercial reasons and is not a U.S. national system requirements. A given reload can consist of as many as four enrichments. The account for that reload would consist of four separate project/enrichment accounts.

In concept, the system maintains two plant material balances. One part, the Nuclear Material Reporting System (NMRS) maintains a material balance for the plant as a whole. The other part, the Nuclear Inventory Control System (NICS) maintains an internal material balance by individual material control areas. The NICS keeps two types of inventories. For material balance areas (MBAs), running book inventories are kept for the quantities of U-element and isotope for each project/enrichment account. For the material in item control areas (ICAs), a current inventory file is maintained of each item and its associated identification and quantity.

At the completion of a physical inventory, the NICS calculates a MUF for each material balance area and for each project/enrichment account.

Both the external and internal accounting systems are activated by the movement of material. The external system is activated by plant receipts and shipments. The internal system activated by material transfers among the internal material control areas. For the internal material movements, a transfer document is completed and processed by the computer based system. The system makes the appropriate transactions, for example, the

additions or removals of quantities for an MBA or the creation or deletion of an item in the ICA. The system processes many such transactions daily.

For most of the items comprising the material balance, the individual item weights are used. As an example for kilograms of UO_2 in a bucket, the measured weights are converted by the computer based system to weights of uranium element by applying a specific element factor based on the uranium assay of the item or lot or by supplying a U-element factor. Conversion to quantities of U-235 is done by application of an enrichment factor.

To facilitate the application of the U-element and isotope factors, tables are maintained in the computer based system. Entry to the U-element factor table is by the material composition code and to the isotope factor table by the project/enrichment account code.

Experimental data for the U-element factors are accumulated in separate records, evaluated statistically, and updated as appropriate from the statistical analysis. The project/enrichment accounts are identified using the nominal enrichment. This is the factor used in the isotope table until all isotopic measurements are completed, at which time a weighted average isotopic composition is computed and the table updated.

The Nuclear Material Reporting System (NMRS) represents the official records of nuclear material for the Model Plant. This system maintains all plant receipts and shipments, physical inventory and MUF by project and enrichment. Entries are made by means of the NRC 741 reports and physical inventory cards from the associated NICS system. The system generates all reports used for plant accounting, and for the NRC and customers.

The Nuclear Inventory Control System (NICS) is a computer based system which maintains internal control of all nuclear material within the Model Plant; material moving into the plant and then into each Material Control Area (MCA). The system also handles all processing for physical inventories reconciliations and inventory summarizations. Inventory data files are communicated to the NMRS system.

2. NICS Transactions

a. Material Movement Within the Plant. Several material transfer forms are used to record movement of material from area to area. These forms are collected daily and keypunched. The daily material transfers are entered into the processing program designed to edit the transactions and update material balances and inventories.

b. Receipts and Shipments. All external receipts and shipments (to and from the plant) are recorded on this form. The receipt contains the container identification, project, nominal enrichment, material composition, element factor and weight and the isotopic factor and weight. The element and isotope weights are recorded to the nearest gram. The individual items on this form are entered into shipper/receiver ICA-1. The container number identification is recorded for shipments and the container is deleted from ICA-1 when shipped.

c. Location Transfers. Transfers of all material among the MCAs are made on this form. This form contains the date, shipper MCA, receiver MCA, container, project, enrichment, material composition, gross and tare weights.

If the transfer is from an MBA, all pertinent information must be included on the form. However, if the transfer is from an ICA, only the container need be listed since the container already exists on the inventory file.

These transfers are again collected for keypunching and entered into the main processing program of the NICS system.

d. Material Record Card and Conversion Factors. Two basic data records are used in conjunction with the location transfer forms. As each item of material is generated, the applicable weight data, material composition, item identification number, project and nominal enrichment are recorded on a material record card. The material record card is attached to the container. When a container (or other similar item) is transferred, the data on the record card are entered on the location transfer form which is processed into the computer based system via keypunching of the data and submittal of keypunch cards. The corresponding data for the U-factor are selected from the memory for each appropriate material composition such as green UO₂ or sintered UO₂. For items such as ADU and grinder sludge, which require a U-assay for each item, the laboratory result is entered into the system along with the location transfer form. The isotopic factors are also stored in the memory for each project/enrichment. The use of element and isotopic factors tables is described later.

e. Project Transfers. Frequently, material is classified under a particular project and needed for another project with like enrichment. In this case, the material transfer to the new project is made on this form. Similarly, a project transfer is made of any material going into the scrap processing plant.

f. MUF Transactions. At the end of a material balance period, the MUF is calculated and entered into the system. The MUF is calculated by comparing the MBA book balance with the MBA physical inventory. Those MUFs are consolidated by project and enrichment and passed to the NMRS system.

3. Factor Tables. Only the gross and tare weights are required on the material transfer documents. In order to calculate the element and isotopic weight on the transfer, assignments of uranium and U-235 factors are made from the element and isotopic factor tables. The element factor table is keyed by the material composition code. This is a three-character code which describes the form the material is in; powder, pellets, rods, hard scrap, etc. The first character is U.

a. Element Factor Table. An example portion of the element factor table follows:

<u>1</u>	<u>2</u>	<u>3</u>
PD	0	.8760
SP	0	.8810
RD	1	.8810
UF	1	.6760

The first column shows the last two characters of the material composition code. The second column is a flag to indicate to the program whether or not the material composition code in question requires a unique or standard factor. If the flag is zero, then the table value is used to calculate uranium. If the flag is one, then a unique uranium factor must be recorded on the transfer document.

b. Isotopic Factor Table. An example portion of the isotopic factor table follows:

<u>1</u>	<u>2</u>	<u>3</u>
0000	E200	.02000
0100	E310	.03095
0222	E278	.02782

Column 1 is a project code. Column 2 is the nominal enrichment. Column 3 is the specific U-235 factor used to convert uranium weight to U-235 weight. The U-235 factor is used for all material compositions of the same project and enrichment.

c. Percent U and U-235 Factor Log. The factor log is a record of all percent U and U-235 analyses made for each lot of material for a given project and nominal enrichment. The data in the factor log are used to compute weighted average isotopic factors and to compute pellet lot U-element factors for pellets loaded into fuel rods.

4. Data Base Files. The system data base files consist of an inventory and transaction sequential tape file.

a. Inventory File. The inventory file contains all MBA balances by project and enrichment, as well as the entire file of items in the ICAs. Each item in an ICA has associated with it the following information.

- Project
- Enrichment
- Material Composition
- Shipper MCA
- Document Report Number
- Seal Number
- Date of Transaction
- Gross and Tare Weights
- U and U-235 Factor

b. Transaction File. The transaction file contains all location transfers for a given period of production, usually a month. Also on this file are the project transfers and MUF entries for a given material balance period.

c. Factor Tables. The element and isotopic factor tables are maintained on a direct access device accessible to all programs in the system.

5. Bundle Assembly. Summarizing the nuclear material in assemblies is accomplished by use of the pre-punched rod card. When the rod is created and recorded on a location transfer, the uranium factor for the lot from which the rod came is read from the factor log record and recorded on the location transfer. During computer processing of these transfers, rod cards are punched. These cards are organized at the rod storage ICA.

During bundle assembly the rod cards are pulled from storage and assembled into bundle decks according to the rod loading bundle map. The bundle identification is punched into these cards in preparation for input to the bundle composition program.

The bundle composition program produces a summary of the nuclear material content of the bundle. It also produces reports to assist in Quality Control audits and bundle shipments. A copy of the summary accompanies the shipment to the customer.

Location transfer cards are punched by the bundle composition program to create an item listing of the bundles. The rod card decks are also used to create location transfer cards to delete the rods from the rod inventory listing.

6. NICS Processing. A summary of the processing which takes place in the NICS system is shown below.

- Process all location transfers, maintaining MBA balances and ICA inventory.
- Produce ICA physical inventory listings.
- Process rod physical inventory data.
- Process MBA physical inventory records.
- Calculate MBA MUF.
- Produce complete transaction file listing.
- Process all physical inventory reconciliation transactions and produce ending inventory summaries.
- Program interface with NMRS to supply ending inventory, MUF and project transfers.

C. Physical Inventories

1. Requirements. For low enriched uranium, a physical inventory is required at six-month intervals. A completely measured physical inventory is required. For the required inventory, process cleanouts are carried out and all such materials removed and converted to measured items. Only tamper-safe (sealed) items may be included in the physical inventory on the basis of previous measurements. Items which are not sealed or items with broken seals even though previously measured must be verified by re-measurement to be included in the physical inventory. Other U.S. requirements for physical inventories include:

- Written instructions for each inventory;
- Reconciliation of the book and physical inventory;
- Documentation of inventory findings; and

- Calculation of MUF and LEMUF within 30 days after the start of physical inventory and reporting of planned actions if MUF exceeds LEMUF or if LEMUF exceeds LEMUF limits ($>0.5\%$).

2. Model Plant Physical Inventory

a. General. The plant inventory is taken over a two-three day period by inventory teams. The inventory is taken by material control area (MBA or ICA) with one or more teams assigned to a material control area. The inventory teams are two-person teams. One person on each team is from the custodial organization (e.g., Operations or Shipping and Receiving) and one from an independent organization.

b. Time. The physical inventories for the Model Plant are taken near the end of March and near the end of September. The inventories are scheduled just before a normal production break, e.g., a weekend, to minimize material movement for the reconciliation process.

When the exact date of the inventory has been determined, the inventory date is formally announced to all affected organizational components. The announcement specifies any State and/or IAEA requirements that must be met. Additionally, internal procedures for equipment cleanout, measurements, sealing procedures, and general inventory counting instructions are given.

c. Inventory Package. Prior to the inventory, the inventory coordinator prepares an inventory package for each inventory team to use in taking the inventory. The packages for teams doing the inventory of an MBA contain a set of inventory stickers and a set of physical inventory sheets. The packages for teams doing ICA's contain inventory stickers, a current computer listing of the ICA holdings, and an ICA Physical Inventory Write-In Form.

d. Tagging the Inventory. In taking the inventory, each inventory item (container, bin, stack) is physically tagged with an adhesive sticker which is uniquely numbered in a sequential order. For the Model Plant, an inventory is typically composed of between 20,000 and 25,000 items so the series of 0 through 25,000 is used in numbering the tags, allowing for some overage. In addition to the unique numbering, the stickers (also referred to as inventory tags) are color coded. One color is used for an inventory, the color is then changed for the succeeding inventory. For example, if a monthly physical inventory frequency is used, then yellow could be used in January, blue in February, orange in March, green in April, etc. This color coding provides traceability from one inventory to the next and helps avoid confusion when trying to determine if all containers have been inventoried at the time of counting.

The approximate number of items expected to be present in each control area is estimated before the inventory so that the proper sequence of inventory sticker numbers can be assigned to each area.

e. Preparation for Inventory. Progress of the pre-inventory cleanouts, MBA container weight and enrichment verifications and tamper-safing of containers is monitored as the inventory

date approaches. Additionally, sampling with analytical determination of uranium element and isotope content is required for all items which were not previously measured.

For the Model Plant, a full cleanout of process equipment, vessels, plenums, and duct work is required to achieve a "completely measured" material balance. Material removed during the cleanout is placed in containers and sampled, weighed, and sealed. Also, any filters removed during the cleanout and any solid waste drums resulting from cleanout must also be measured to be included in the inventory.

Many of the items present at inventory time were measured at the time they were created. Those which could be sealed at time of measurement were sealed so that they could be included in the inventory at the previously measured values. The contents of the items which could not be sealed at the time of the original measurement, such as pellets in open sintering boats, are verified at time of inventory by re-measurement of one or more properties to be included in the inventory. Similarly, items with broken seals must be verified by re-measurement. Also, in preparing areas for the inventory, previously measured items which were not tamper-safed are verified by remeasurement and either tamper-safed or placed in tamper-safed containers. Pellets on trays are verified by re-weighing and placed in sealed bins prior to the inventory. Fuel rods are item identified, counted and placed in sealed storage bins. These activities facilitate the efficiency and safeguards effectiveness of inventory taking.

f. Inventory Hold Period. Hold periods are established in which no material movements are permitted between material control areas. After inventory taking has been satisfactorily completed in a material control area, material movement within the area may be permitted. However, movement between control areas is restricted to facilitate the reconciliation process.

g. Reconciliation. After physical inventory taking (PIT) is complete, the inventory is reconciled by resolving all noted discrepancies such as missing items which must be located or documents of their disposition found. Transaction documents for write-in item must be prepared and processed. All variances between the previously listed data for items such as seal number, project, enrichment and weight and the data recorded by the inventory teams must be resolved or verified as correct. Items with broken or missing seals must be verified by remeasurement procedures.

h. Summarizing and Reporting. After the inventory has been reconciled, the ending inventory for the entire plant is summarized by the computer. This report shows the identity of each inventory item and the quantity of element and isotope contained. The report also summarizes the inventory by material type, e.g., depleted, natural and enriched uranium.

The ending inventory is compared to the book inventory to determine the plant MUF. The MUF and inventory composition and totals are reported to plant management, the U.S. NRC and to the IAEA via the NRC.

D. Material Control Areas

The U.S. national system requires that internal material control areas be established for physical and administrative control of nuclear materials. The material control areas for the Model Plant were shown in Figure 6 of the text for Session 32.

The material control areas for the Model Plant were selected on the basis of process, administrative, accounting, and physical considerations. The operation of each MBA is the responsibility of one of the Plant supervisors. Each MBA has defined physical boundaries and each MBA represents a natural grouping of like processing and handling operations. Lastly, the natural flow of material between MBAs involves the transfer of discrete measurable items so that flows into and out of the MBA can be accurately accounted for.

The item control area structure is designed to provide maximum inventory and administrative control over all materials not in an immediate processing status and over all items amenable to item control. ICA-1 (Shipping and Receiving) is established so that all materials entering and leaving the plant are under item control upon receipt and prior to shipment. The storage ICAs are established to place all bulk materials not in an immediate processing status under item control.

The Rod and Bundle ICA represents the ideal case for item control since each rod and bundle can be treated as previously measured sealed quantities.

The computer based accounting system maintains a running book inventory of each MBA and an item listing of each item in each ICA. The computer data base is updated daily and an item listing is printed out daily for all ICAs except Rod and Bundle assembly which is printed out at inventory time. The concept of having most of the inventory present as discrete previously measured items, each of which is identified by location quantity and serial number on the item listing, greatly facilitates inventory taking.

The combination of the item control area arrangement and measurement of items at the time of creation and the subsequent inclusion of items placed in item control on the perpetual inventory listing represents, in a sense, a form of advance inventory taking.

E. Current Knowledge of Discrete Items

The U.S. regulations for safeguards require that current knowledge of discrete items be maintained. This requirement is to aid in the detection and investigation of theft should it occur. The requirement is met at the Model Plant by maintaining a separate record (separate from the item) of all discrete items which are expected to exist for more than 14 days. The item control program is audited periodically on a sampling basis to determine if all items of record are indeed present and to assure that the items are posted to the records within the required time period.

F. Safeguards Reports

U.S. safeguards regulations require several routine reports and several types of special reports of discrepancies. The routine reports are the Nuclear Material Transaction Reports (NRC Forms 741) and the Material Status Reports (NRC Forms 742). See References 1 and 2.

The Material Status Reports (Forms 742) are submitted every six months (March 31 and September 30) and state the quantities of nuclear materials received, produced, possessed, transferred, disposed of or lost.

Special reports are required of any material accounting discrepancies. Any loss of nuclear materials such as loss of a discrete item or an accidental loss or theft or attempted theft must be reported immediately to the USNRC, Region V Office. Material balance discrepancies such as in the event that the observed MUF exceeds the LEMUF or other action limits of the U.S. national system must also be reported to NRC and appropriate actions taken to resolve the discrepancy.

For the Model Plant used in the course, routine reports are also submitted to the Region V Office (San Francisco, California) of NRC. That office conducts the routine inspections of the plant for the U.S. national system. Waste discards are reported monthly, and MUF reports are submitted following the required six-month inventory and for the three-month inventories which are taken by the Model Plant for its own purposes. The LEMUF and a MUF measurement bias report is also submitted after the six-month inventories.

The Model Plant is also assumed to be under IAEA safeguards so that reporting requirements include the submission of a Physical Inventory Listing (PIL) as part of the Form 742 submission. Also, as part of the reporting requirements for IAEA safeguards, the Model Plant reports all inventory changes using batch designations and other instructions for inventory change reporting on Form 741.

G. Measurement Control Program

The U.S. national system requires that a formal measurement control program be established and implemented to control, estimate, and document measurement errors. The measurement control program for the Model Plant is discussed in the next session.

H. Safeguards Inspections

1. Inspections by NRC. Under U.S. national system requirements, the Model Plant is inspected several times a year by inspectors from the Region V Office (San Francisco). The inspections are unannounced and are conducted usually by two inspectors. The inspections are to assure compliance with the Regulations and License Conditions for the facility. The detailed requirements are incorporated in the Fundamental Material Control Plan for the Model Plant. The Fundamental Material Control Plan is a procedures description document which describes the safeguards accountability procedures used by the

Model Plant. The inspectors also obtain samples for verification of certain key measurements. Those samples are sent to the New Brunswick Laboratory which is a U.S. government laboratory.

At the end of each inspection, the inspectors conduct an exit interview with plant personnel to discuss the results of their inspections. A formal inspection report is also prepared and a copy sent to the Manager of Licensing.

To accommodate the unannounced inspections, picture badges are kept at the main Badge House and those badges are issued to the inspectors upon presentation of their credentials and recognition by Plant Security personnel. The inspectors have access to all parts of the plant and all key personnel are made available for discussions with the inspectors. Usually, the inspectors are accompanied by plant safeguards personnel in inspections of plant areas.

2. Inspections by IAEA. The plant is also inspected by the IAEA. Inspections are conducted periodically (bi-monthly) to verify the Model Plant's stated quantities for flow items and to conduct records audits. Those inspections are usually carried out by two to three inspectors.

The physical inventory is verified annually by the IAEA. Those physical inventory verifications (PIVs) are usually conducted by a team of five to six inspectors over a six to seven day period.

For the PIV, the inspectors use a random sampling plan and a combination of NDA and destructive analysis to verify the Model Plant's statement of inventory quantities in each inventory area. As a basis for the verification, the IAEA is provided with copies of the plant's physical count sheets immediately after the plant has completed its physical inventory taking (PIT) of each area. Those sheets provide an itemized listing of each item and its nuclear material content that is present in each inventory area. The IAEA uses those sheets to randomly select samples for independent measurement to verify the plant's statement of the inventory quantities for each area. Since the inventory sheet represents a provisional, unreconciled inventory, a final reconciled itemized listing of each inventory item and its amount of contained element and isotope is sent to the IAEA so that they can verify that there are no unexplained changes between the original provisional listing and the final listing.

I. Safeguards Organization

1. Requirements. Safeguards regulations for material control and accounting in the U.S. require that those functions be independent of manufacturing and other functions which utilize the nuclear material and have direct custody of the material. Regulations also require that a single individual at an organizational level sufficient to assure independence of action be responsible for the overall planning, coordination and administration of the MC&A function.

2. Safeguards Organizational Structure. The components having primary safeguards responsibilities for the fuel fabrication plant are shown in Figure 3. A further breakdown of organizational components with safeguards responsibilities is shown in Figure 4.

The President of Model Plant Company has the ultimate responsibility for the safeguarding of SNM. He has delegated major program elements applicable to the fuel fabrication plant to three components within the company. These are the Licensing Department, the Nuclear Fuels Department, and the Finance Department. In addition, technical assistance is provided by the Research and Technology Department.

The Manager of the Licensing Department has overall program definition responsibility. This includes establishing the objectives and criteria for the program; defining, implementing, and providing assistance as necessary in solving the technological problems related to SNM control; managing the development programs in safeguards; establishing and maintaining primary safeguards liaison with regulatory agencies; and providing for overall program management and audit. He is assisted by the Senior Specialist, Nuclear Materials Safeguards, who acts for the Manager of Licensing in safeguards accountability matters. Technological assistance is provided to the Manager of Licensing by the Research and Technology Department. Assistance in the areas of statistics and computer program development is provided by the Statistical Consultant and by the Systems Analyst, respectively, who also assist the Manager of Licensing in other safeguards accountability matters.

The General Manager of the Nuclear Fuels Department has line responsibility to carry out the established safeguards requirements dealing with material custodianships, physical inventory, and the transfers of SNM. He also has the responsibility for providing the data and information of safeguards importance specified by the Manager of Licensing and of keeping him informed of safeguards-related problems and developments. In providing these data, he is responsible for controlling their quality and for generating the additional information needed to provide assurance of this quality.

In carrying out his function, the General Manager of the Nuclear Fuels Department has delegated the responsibility for designating MBA and ICA custodians to the Managers of Material and Purchasing, Process Engineering, and through the Manager of Manufacturing to the Managers of UO₂ Shop Operations, and Quality Control. The responsibility of the Nuclear Fuels Department General Manager to provide analytical data and supporting information is carried out through the Manager of Manufacturing by the Manager of Quality Control. Analytical data is provided by the Manager of the Analytical Laboratories who reports to the Manager of Quality Control.

The Manager, Finance, has the responsibility for keeping the nuclear materials accounting records; for auditing of SNM custodian accountability activities; for establishing and

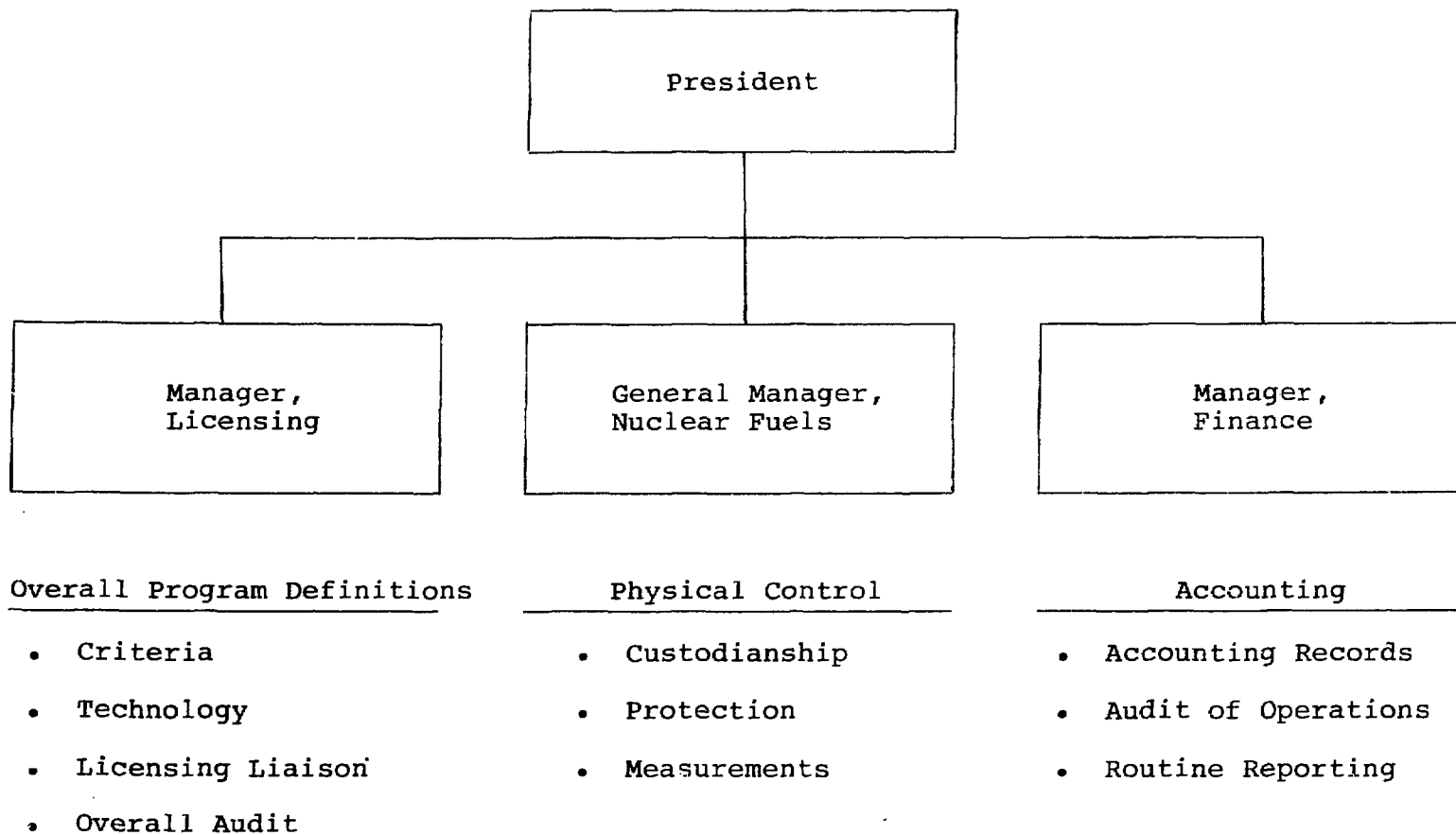


Fig. 3
Primary Safeguards Responsibility Components

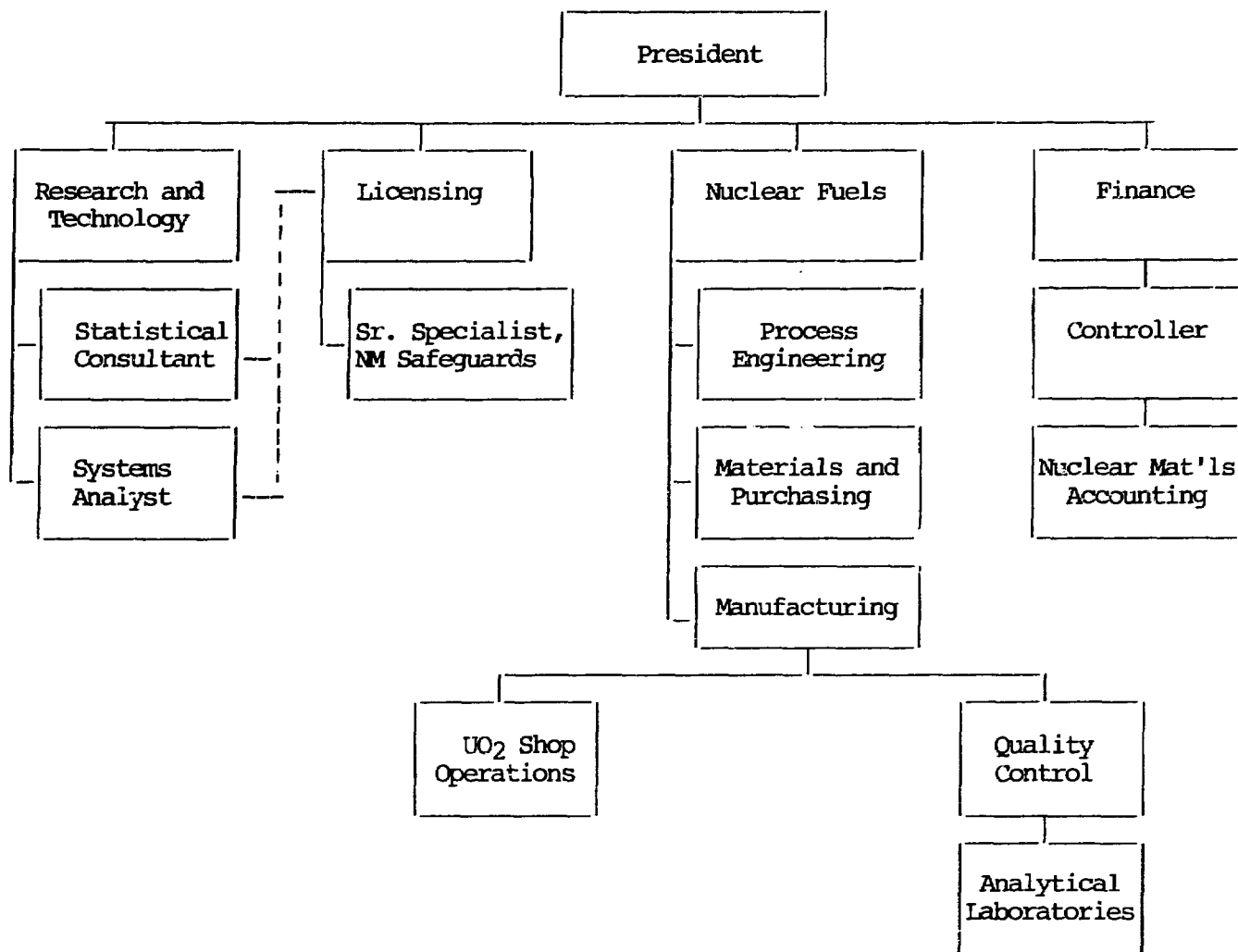


Fig. 4
Organizational Substructure of Components and Safeguards Responsibilities

maintaining an accounting procedure manual; and for handling routine accounting contacts with Region V, U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Walnut Creek, California. In performing the auditing function, he has the authority to witness measurements, make independent measurements, and inspect internal MBA and ICA records.

The Manager, Finance, carries out his safeguards responsibilities through the Accountant, Nuclear Materials.

J. Background Material

For technical background material on U.S. materials accounting requirements and for guidance in developing materials accounting procedures and techniques, the student is referred to the Division 5 Regulatory Guides Materials and Plant Protection. The table of contents and copies of those guides are available as follows:

Requests for single copies of the latest revision of a regulatory guide, the only version currently in print, should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control. Earlier revisions may be examined at the Commission's Public Document Room, 1717 H Street N.W., Washington, D.C. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

IV. REFERENCES

1. NUREG/BR-006, Revision 2, Instructions for Completing Nuclear Material Transaction Reports (Form DOE/NRC 741, 741A and 740M).
2. NUREG/BR-0007, Instructions For Completing Material Balance Report, Physical Inventory Listing, and Concise Notes (Forms DOE/NRC 742, 742C and 740M).



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



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SESSION 34: MATERIAL CONTROL AND ACCOUNTING
AT EXXON NUCLEAR, II

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I. INTRODUCTION

In this session the measurements and the associated measurement control program used at the Model Plant are described. The procedures for evaluating MUF and sigma MUF are also discussed. The use of material composition codes and their role in IAEA safeguards under the US/IAEA Safeguards Agreement are described. In addition, the various accounting forms used at the plant are described and the use of tamper-indicating seals is discussed.

II. MEASUREMENTS

A. Sampling

1. Solid Materials. Uranium oxide powders and similar solid materials are sampled with a scoop after the material has been homogenized by mechanically mixing the material. For the main material flow of powder/pellet lots, tonne lots of UO_2 are homogenized by mixing in a large mechanical mixer. Those lots are then carried through pressing, grinding, and rod loading as integral units of uniform material. The uniformity of that material is evidenced by the very small (negligible) within lot sampling error found from the analyses of powder and pellet samples from the same lot.

Pellet samples are taken with tweezers from different parts of a pellet lot.

Dirty powder and dried grinder sludge are mixed by tumbling the container before sampling with a scoop. Wet grinder sludge and ADU which are taken directly from the centrifuge bowl as a wet uniform mass are sampled with a scoop. If sampled after standing, such that the material has separated into a solid and liquid phase, the material is mixed with a paddle before sampling with a scoop.

2. Liquids. The main liquid sampled for accountability at the Model Plant is the liquid waste from the UF_6 to UO_2 conversion process. The centrate from the ADU precipitation process is pumped to Lagoon 2, the Accountability Lagoon. That lagoon is used as a batch measuring basin for measuring the liquid waste prior to transfer to one of the larger storage lagoons. The Accountability Lagoon is mixed by a continuously operating circulating pump and by the action of the wind flowing over the shallow basin surface.

The liquid phase of the Accountability Lagoon is sampled at inventory time and before a batch transfer to one of the storage lagoons. Samples are taken from a boat at nine points which represent the nine equal areas of the lagoons. Samples are taken at three depths at each of the nine points and analyzed for uranium (ppm) and U-235 by mass spectrometer.

B. Weighing

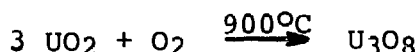
The principal bulk measurement method used at a fuel fabrication plant is weighing. The inputs, outputs, and most of the inventory in the plant are based on weight measurements. All weighings are made on digital read-out scales. The scales are load-cell, electronic scales having capacities ranging from 10 kg for the rod loading scales to 4000 kg for the UF₆ scales.

For the most part, all net weights which are recorded are based on individually determined tare weights. The number of scales in use at the Model Plant and the magnitude of typical weighing errors were shown in Table V of Session 32.

C. Analytical Measurements

1. Uranium Element Analysis

a. Gravimetric Method. High purity forms of uranium are ignited in air at 900° + 25°C to form U₃O₈. The weight of U₃O₈ is converted to uranium using the stoichiometric factor for U₃O₈. The method is not specific for uranium and corrections must be made for any non-volatile impurities founded in the ignited U₃O₈. The basic chemical reaction is illustrated below for the analysis of UO₂.



In the procedure used at the Model Plant, samples of UO₂, U₃O₈, and ADU are weighed in Vicor crucibles and ignited in air at 900°C for four hours. After ignition, the crucibles are placed in a desiccator and after cooling, re-weighed. From the weight of U₃O₈ corrected for impurities, the weight of uranium in the sample is calculated using the factor for U₃O₈. The non-volatile (metallic) impurities are measured by emission spectroscopy on a sample of the U₃O₈. The correction for impurities is made on the basis of the oxide forms present in the ignited sample material (U₃O₈).

The known or measured enrichment of the sample is used to determine the atomic weight of uranium to be used in calculating the factor for U₃O₈.

The method gives very precise results and is widely used in the industry for the assay of relatively pure UO₂, UO₃, UF₆, UF₄ and UNH.

A more detail discussion of the analytical methods used at the Model Plant is given in Session 26 of the October 17 to November 4, 1983 course text, LA-10135-C.

b. Titration Method. The Davies-Gray titration method is used to determine uranium in medium-level (100 ppm) liquid waste solutions, solid scrap materials, and uranium pellet standards. The method is based on the reduction of uranium (VI) with ferrous iron in a concentrated phosphoric-sulfuric acid solution. The excess ferrous iron is oxidized to ferric iron with nitric acid using molybdenum (VI) as a catalyst. The uranium (IV) which is not oxidized by the nitric acid is then quantitatively titrated back to uranium (VI) using standard potassium dichromate. The titrations are performed with an automatic titrator with a digital readout for the volume of titer used to reach the end point. The instrument automatically titrates to the potentiometric end point. The method is quite specific for uranium, has a relatively high precision, and is widely used in the industry for the analysis of both product grade and scrap materials.

c. Fluorimetric Method. The uranium concentration of low-level liquid waste solutions is determined by the fluorimetric method. The method is based on the specific fluorescence (5550 Å) of uranyl salts when excited by ultraviolet light (3650 Å). These two properties, excitation region and visible fluorescence region are quite specific for uranium.

The fluorescence is intensified when uranium is fused in a sodium or lithium fluoride lattice. In a fused sodium fluoride-lithium fluoride (2% LiF) lattice, it is possible to detect the presence of as little as 0.001 microgram of uranium. In addition, the total fluorescence produced is a linear function of the amount of uranium present.

These three properties - specific excitation and fluorescence regions, linear dependence, and extreme sensitivity - make possible the quantitative determination of uranium on a microgram scale. The analytical procedure used consists of fusing the uranium with a fixed amount of sodium-lithium fluoride in a platinum dish and measuring with a fluorimeter the total fluorescence of the flux and dish when excited by ultraviolet light.

A reflection type fluorimeter is used to measure the fluorescence of the melt from the same surface that is irradiated with ultraviolet light. A schematic of the instrument is shown in Figure 1. The instrument employs a mercury lump as an ultraviolet source and a phototube to measure the fluorescence. Filters are used to isolate the ultraviolet and fluorescence regions.

As yet, no other element has been found to give a measurable fluorescence at the same wave length as uranium. Interference to the method is caused, however, by the presence of foreign materials which "quench" or decrease the fluorescence of uranium. Known quenching materials include Fe, Ca, Cr, Co, Cu, Mg, Mn, Ni, Pb, Pt, Si, Zn and HNO_3 .

The quenching effect may be reduced or eliminated by dilution or by "spiking" the blank and sample dish with a known amount of uranium after the fluorescence of the blank and sample dish have been measured. Since the quenching material reduces

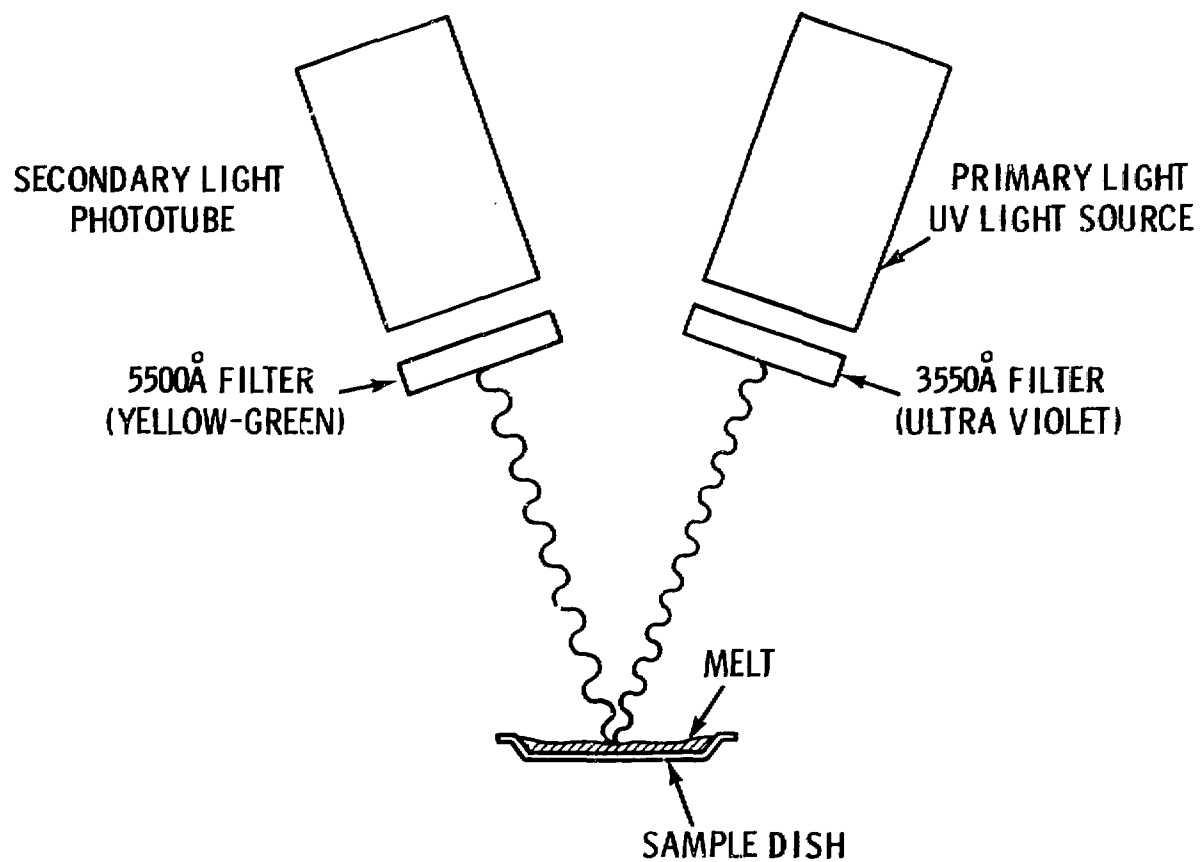


Fig. 1
Fluorimeter Schematic

the fluorescence of the "spike" to the same degree as the sample, the quenching effect can be accurately corrected for. The quenching effect may also be eliminated by chemical separation of the uranium from the quenching material. The hexone solvent extraction of uranium has been widely used as a chemical separation method to prepare samples for fluorimetry. The fluorimetric method is described in more detail in Session 26 of the October 17 to November 4, 1983 course text, LA-10135-C.

2. U-235 by Mass Spectrometry. This is the principal method used at the Model Plant to determine the isotopic composition of uranium. A thermal emission mass spectrometer is used in which a solid sample is thermally ionized from a heated filament. The ions are accelerated in a vacuum by an electrical field through a magnetic field. This action separates the ions into beams of ions having the same mass-to-charge ratio. The electrical field is held constant and the magnetic field is varied to focus a specific ion beam on the detector. This is shown in Figure 2 which shows the ion beams being bent by the magnetic field with the mass 238 ion beam focused to enter the defining slit and be recorded by the detector.

The isotopic composition of a sample is determined by mounting a dissolved portion of the sample on the rhenium or tantalum sample filaments on the source "hat" of the spectrometer and dried. After drying, the "hat" and sample filaments are attached to the source, the system closed and the source pumped down to the required vacuum. The sample is thermally ionized and the ion beam focused on the defining slit. The ion beams for U-234, U-235, U-236 (if present) and U-238 are sequentially focused on the defining slit and create a response on the electron multiplier which is amplified and recorded on a strip chart or a digital recording system. The ion beams for U-234, U-235 and U-236 are measured as voltage ratios to the U-238 voltage. Usually, six scans are made for each sample and the average voltage ratio for each mass ratio is used in the final calculations. The average ratios are corrected for the mass discrimination effect which is due to the effect different masses have on the electron multiplier. From the corrected isotopic abundance ratios, the atom percent and weight percent of each isotope are calculated. For accounting purposes, only the U-235 isotope is used. However, it should be noted that it is necessary to measure all of the uranium isotopes (U-234 and U-236) to obtain an unbiased value for U-235.

The thermal emission mass spectrometer used at the Model Plant provides highly accurate enrichment values for safeguards accounting. A more detailed discussion is given in Session 26 of the October 17 to November 4, 1983 course text, LA-10135 C.

3. Emission Spectroscopy. Trace impurities in uranium powders and pellets are measured by emission spectroscopy. This method is also the basis for the impurity correction used in conjunction with the gravimetric method for uranium assay.

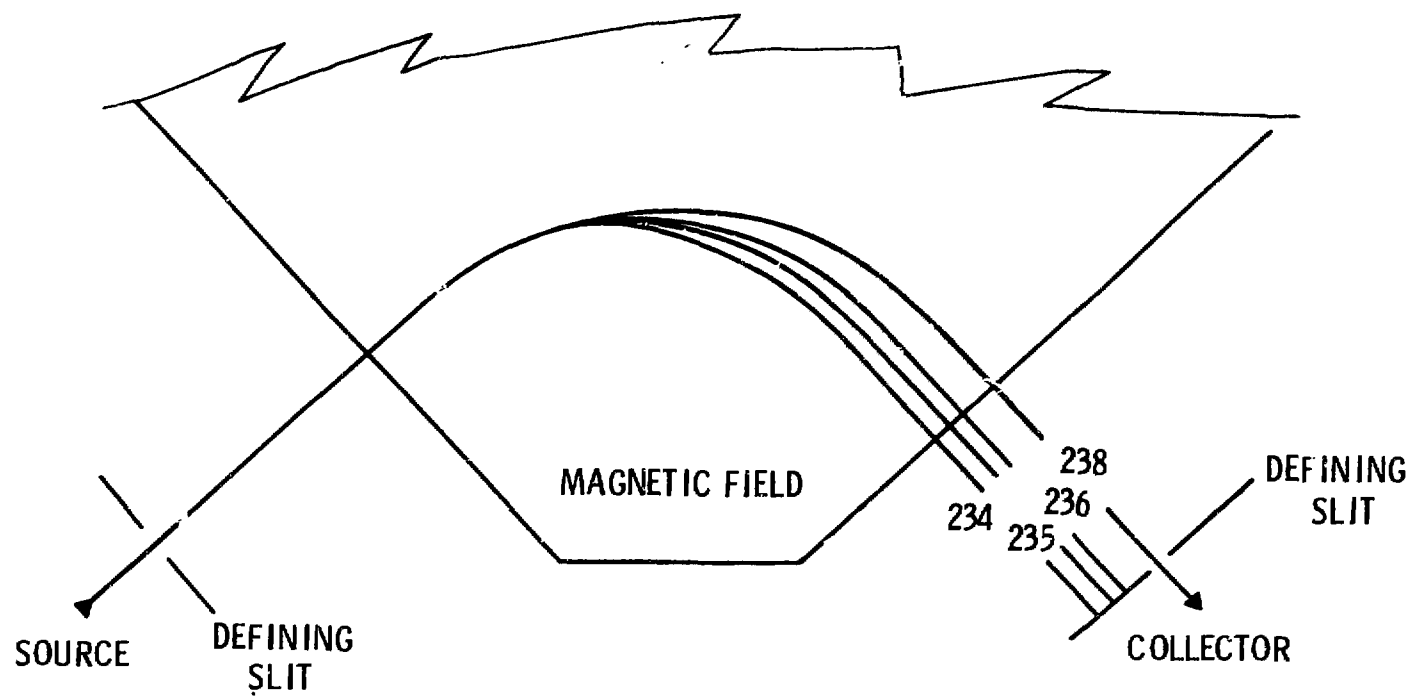


Fig. 2
Mass Spectrometer Schematic

When the atoms of an element are excited in an arc or spark, they emit light of a wavelength which is characteristic of the element. Thus, the observed wavelength of the light emitted is a qualitative measure of the presence of an element and brightness of the spectral lines is a measure of their concentrations.

The general approach used in emission spectroscopy is to disperse the emitted light with a prism or a diffraction grating so as to be able to record the specific wavelengths on a photographic plate or a phototube. When the light is recorded on a photographic plate, the intensity of the emitted light is measured by the degree the plate is darkened. The darkness of the emitted lines is measured with a densitometer. If a phototube is used the intensity of the emitted light is indicated directly by the output of the phototube. An instrument using photographic film to record the spectrum is termed a spectrograph and the direct reading instrument a spectrometer (or quantometer). The spectrograph is the generally a more versatile instrument, particularly for qualitative work since the whole spectra can be recorded. The spectrometer is more accurate but is limited to number of elements that can be analyzed without changing the instrument.

The Model Plant uses a spectrometer to determine 39 elements in a quantitative way. The samples are excited by an arc (or spark), the emitted light dispersed by a grating onto a number of defining slits set to receive specific wavelengths, and through the defining slits to phototubes which measure the intensity of the light emitted at the wavelength covered by the defining slit. A schematic of the spectrometer is shown in Figure 3.

The analysis of impurities in uranium materials is complicated by the uranium spectrum which is replete with lines. As a result, special techniques have been developed to separate, either during the excitation process or prior to excitation, the impurities from the uranium. One widely used technique which is successful for some of the more volatile impurity elements is the carrier distillation method. Carriers such as gallium oxide or silver chloride-lithium fluoride are added to reduce the excitation of uranium and to distill the more volatile impurities into the arc where they are excited in preference to the uranium. For refractory elements such as the rare earths, a solvent extraction method is used to extract the uranium away from those elements, so that they are excited free of uranium. A more detailed discussion of emission spectroscopy is given in Session 26 of the October 17 to November 4, 1983 course text, LA-10135-C.

C. Non-Destructive Assay Method

1. Waste Assay Counter. Solid waste materials of contaminated handling materials such as gloves, cloths, etc., and contaminated HEPA (High Efficiency Particulate Absorbers) filters are measured for U-235 content using a passive gamma ray counting system. The counter uses four sodium iodide (NaI) detectors,

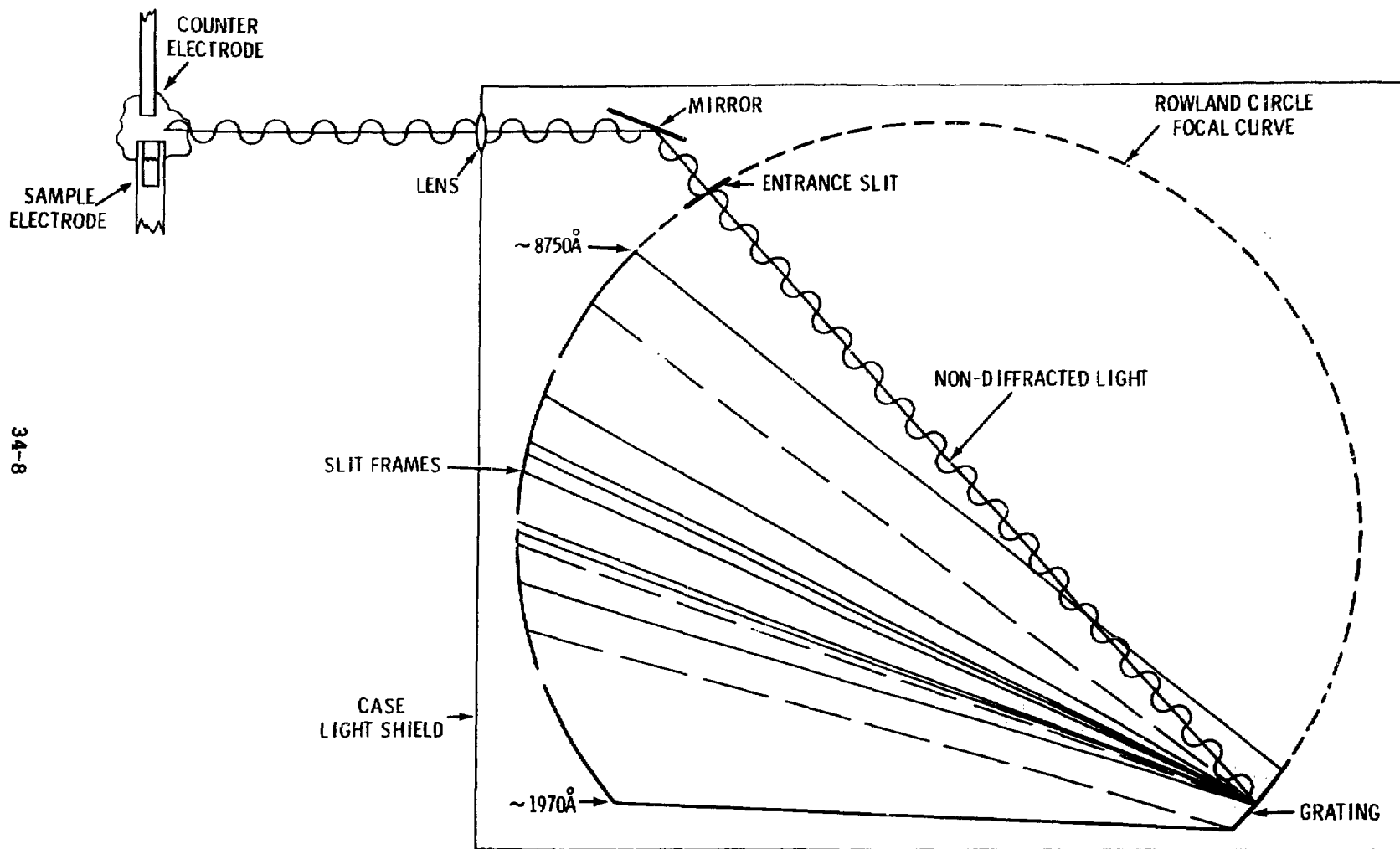


Fig. 3
Emission Spectrometer Schematic

with lead shields around the detectors to collimate the gamma rays, the associated electronics, and a rotating platform to rotate the 55-gallon barrels (drums) which hold the contaminated trash and the HEPA filters. The rotating platform rotates at about five revolutions per minute to average any radial distribution of the uranium in the containers.

A block diagram of the waste assay counter is shown in Figure 4. The signals from the four detectors are amplified by the preamplifiers and amplifiers and summed together by the summing amplifier to provide one composite signal. This composite signal provides the input to two single channel analyzers (SCAs). The SCAs have upper and lower level voltage controls (window adjustments) that can be adjusted to provide an output only if the amplitude of the input pulse is within a specified amplitude. One SCA is adjusted to cover an energy range of 155 keV to 215 keV, which brackets the 185 keV gamma ray produced by U-235. The second SCA is adjusted to cover an energy range of 230 to 290 keV and measures the signal due to Compton scatter. The outputs from the SCAs are totaled by two gated scalers which are controlled by a timer. The scaler value for Compton scattering is subtracted from the scaler value representing the U-235 peak area (185 keV) to obtain a value proportional to the quantity of U-235 in the container.

The system is calibrated with standard waste barrels and HEPA filters. The standard waste barrels contain known quantities of U-235 in low-enriched uranium in a dry sawdust matrix. For the HEPA filter standards, plastic coupons containing known amounts of U-235 in low-enriched uranium are placed in unused HEPA filters. The number of coupons per filter standard ranges from six coupons for the two gram (U-235) standard to 101 coupons for the 30 gram standard.

Based on chemical leaching experiments, the calibration curve for HEPA filters gives unbiased results on contaminated HEPA filters. The calibration curve for waste barrels, by contrast, gives biased low results on waste barrels containing typical process handling materials. As a result, extensive chemical leaching experiments were carried out to determine a bias correction curve for typical process waste barrels.

The choice of a passive gamma ray counting system using sodium iodide detectors for measuring the U-235 content of solid waste at the Model Plant is based on several practical considerations. First, the uranium energy spectrum for most of the waste materials is such that the 185 keV peak for U-235 can be resolved readily with a sodium iodide detector. As a result, a high resolution germanium crystal detection system which has both a higher initial and operating cost is not required. An active system using neutron interrogation is more sensitive and would be less subject to changes in the density of waste materials. However, the presence of nuclear absorbers in fabrication plant contaminated trash restricts the use of active systems. We have

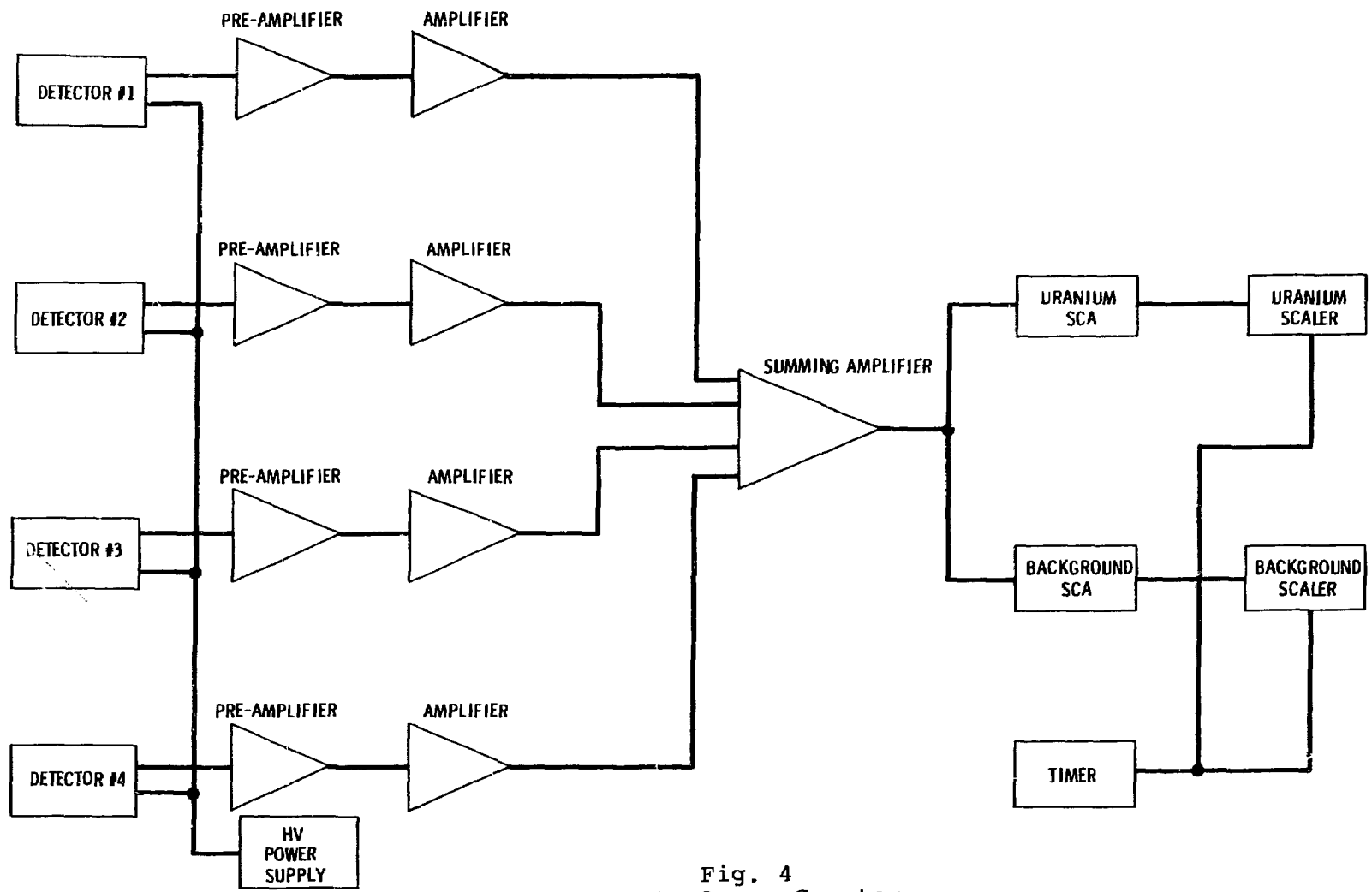


Fig. 4
Waste Assay Counter

also observed that some HEPA filters contain a nuclear absorber which prohibits their assay by neutron activation.

2. SAM-2 Enrichment Meter. The enrichments of uranium powders and pellets are measured in their process containers using the SAM-2 enrichment meter. Those measurements are used principally as enrichment control overchecks and are generally not booked into the accounting records. As noted earlier, mass spectrometry is the principal safeguards accountability measurement for U-235.

The instrument is used to measure the enrichment of an "infinitely thick" source. The counter is positioned about two inches below the bottom of a powder bucket or two inches from the side of a powder barrel to reduce the sensitivity between distance variations. The meter is calibrated with enrichment standards in containers having the same thickness as the process containers. The instrument is set so that the scaler reads directly in enrichment units for a two-minute counting time. For two-minute counting times, the instrument has a precision at one standard deviation of about 0.02 percent absolute.

3. Active Rod Scanner. The Model Plant is required by U.S. safeguards to perform a non-destructive scan of each fuel rod produced to verify the assigned nuclear material content. This requirement is met by scanning each rod with the active fuel rod scanner to measure the fissile content of each fuel rod. (Note: In the actual plant, fuel rods containing a nuclear absorber are also verified using a passive rod scanner. However, for the Model Plant nuclear absorber rods are not included. Thus, only the active rod scanner is described for the Model Plant.)

The active fuel rod assay system uses a californium-252 neutron source to induce fission in the U-235. A moderator assembly provides biological shielding of the californium-252 source and moderates the fast neutrons to an energy spectrum comparable to the neutron energies encountered in a reactor.

The system uses the fission product delayed gammas to verify the enrichment uniformity of the fuel rod and the fission product delayed neutrons to measure the fissile content of each fuel rod.

The system processes two fuel rods simultaneously. Storage racks are provided on the inlet and outlet and fuel rods are automatically loaded onto and off of the drive system.

When operated at a drive speed of 18 ft/min, the system can detect a single pellet, approximately 0.3 inches long, which differs from the average enrichment by about 22%, with 95% confidence, while providing a false reject signal for about 0.25% of the acceptable fuel rods.

When operated at a speed of 18 ft/min, the system precision for total fissile assay is about 0.6% at one standard deviation.

The rod assay system is used as a Quality Control check for enrichment control and as a safeguards overcheck on total fissile content. It is not used for accountability purposes. The

accountability values for fuel rods are based on the weights of UO_2 loaded into each rod and the assay of the corresponding pellets for percent uranium and U-235.

The NDA instruments used for safeguards at the Model Plant are discussed in more detail in Session 27 of the October 17 to November 4, 1983 course text, LA-10135-C.

III. MEASUREMENT CONTROL PROGRAM

A. Purpose and Coverage

For safeguards purposes, measurement control programs are typically carried out to meet three objectives. The first is to insure the control and quality of accountability and verification measurements. The second is to provide an experimental basis for the estimation of the random and systematic errors of measurement for calculating sigma MUF and evaluating shipper/receiver differences. The third is to provide documented evidence that safeguards measurements have met quality criteria. This section describes the measurement control program for the Model Plant.

The measurement control program encompasses all elements of the measurement processes used to determine quantities of uranium element and U-235 isotope in plant receipts, shipments, waste discards, and inventory.

The program is directed at the individual elements of the measurement processes rather than at the measurement components of the plant material balance. For each element of the measurement process, such as weighing, NDA and analytical measurement, a program of standards, replicate measurements, calibrations, and statistical analysis is applied. All data generated in the program are documented and subject to continual review. The program also includes special experiments to estimate weighing and sampling errors and the potential matrix bias arising from the passive gamma measurement of U-235 in solid wastes.

The measurement elements for the Model Plant are shown below. As shown, the control program covers five measurement elements and from one to five measurement applications for each of the five elements.

1. Fuel Fabrication Measurement Elements

- Mass - Weighing of UF_6 cylinders, pellet columns, trays, boats and buckets.
- Analytical - Gravimetric, fluorimetric, and volumetric uranium assays and U-235 by mass spectrometry.
- NDA - Total U-235 by passive gamma in solid wastes; barrels and HEPA filters.
- Sampling - Sampling of powders, sludges whole pellets, and liquid wastes.
- Volume - Volume of liquid wastes in the Accountability Lagoon.

B. Technical Requirements

1. Reference Standards. There are three general requirements for reference standards - traceability, quality and

representativeness. Reference standards should be traceable to a national or international system of measurement. They should also meet the quality objectives of the measurement program. The tolerance or uncertainty associated with a reference standard should be smaller than the uncertainty objective of the measurement. A desirable, but not always attainable goal, would be for the uncertainty of the reference standard to be of the order of five to ten times smaller than the uncertainty goal for the measurement method. Lastly, the reference standards should be representative of the measurement process and range of application; or in the case where the reference standard is not used directly, it should be possible to derive from the reference standard secondary standards or working standards which are representative.

2. Frequency of Standard Measurements. Standards should be measured with a frequency sufficient to detect, in a timely manner, out-of-control situations so as to minimize the number of items which may have to be re-measured. When used for bias estimation, the standard frequency should be sufficient to provide a precise estimate of a possible bias.

3. Calibration Standards. Calibration standards should be traceable to primary standards, cover the range of application of the method, and be representative of the materials and items undergoing measurement.

4. Experimental Design. The measurement control program should be designed to provide experimental data from which the random and systematic errors of measurement can be estimated.

C. Mass Measurements

1. Reference Standards. An example of the mass standards used at the Model Plant is shown in Table I. Also, included in the program are sets of NBS Class S weights for the analytical balances.

2. Measurement of Mass Standards. Each scale is zeroed at first use on each operating shift and a control standard weighed. Once each week, at a random time, a Quality Control technician weighs a known standard in the working range on each scale.

3. Mass Calibrations. Prior to use, each new scale is evaluated with known standards over a several day period before the scale is certified for use. Each scale is calibrated monthly by the Quality Control gauge technician and each scale is calibrated annually by an instrument technician.

D. Analytical Measurements

1. Reference Standards. The analytical reference standards used at the Model Plant are shown in Table II.

2. Measurement of Analytical Standards. Standards are run using each analytical technique with a minimum frequency of two per week, except when the method is not in use. Under continued operation, this produces 52 results for each analytical technique employed in measuring uranium and U-235 during each six month

TABLE I
EXAMPLE OF MASS STANDARDS

<u>Scale Type</u>	<u>Standards</u>	<u>Certification</u>
1. Powder, Boat, & Tray Scales	Sets of 5, 10, 20, 25 kg Metal Weights	NBS Class C. Certified by Metrology Lab. Re- certify 18 Months
2. Rod Loading Scales	Set of 0.6, 0.7, 2.4, 3.6, 4.9, 5.7 kg Metal Rods	Certified by Metrology Lab. Recertify 18 Months
3. UF ₆ Scale & Powder Barrel Scale	Set of (5) 50, 250, 500, & (3) 1000 lb Metal Blocks	Certified by Metrology Lab. Recertify 18 Months
UF ₆ Scale	<u>Replica Mass Stds</u> 2 Cylinders De- pleted U Tare Approximately 1650 Pounds Full Approximately 6400 Pounds	Certified at Diffusion Plants versus NBS Weights Recertify 2 Years

TABLE II
ANALYTICAL STANDARDS

Measurement	Standards	Certification
% U-Gravimetric	Sintered UO ₂ pellets	Depleted UO ₂ pellets certified by measurements made by New Brunswick and Ledoux Laboratories.
% U-Low-Level Solutions	Parts per million solution standards	Aliquots of NBS 950 A, 960, or their successors are weighed, dissolved with HNO ₃ , and diluted to volume and/or weight.
% U-General	NBS-950 A or 960	NBS Standard Reference Materials.
% U-235 by Mass Spectrometer	NBS U010, U015, U020, U050, U500, U100, U150, U200	NBS certified isotopic standards of standard reference material U ₃ O ₈ .

material balance period. The schedule provides bias estimates with a level of precision sufficient to detect real biases that are very small relative to random error limits. By running standards on a weekly schedule, timely indication of lack of control is given.

3. Calibration of Analytical Measurements

a. Fluorimetric (Low-Level Waste and Grinder Water). Each sample analyzed has a unique calibration which is derived by spiking the sample with a known quantity of dissolved NBS U_3O_8 or NBS metal. A calibration curve is not required.

b. Titration (Davies-Gray Method). When the Davies-Gray method is used for accountability analyses, the titrant and automatic recording titrator are calibrated against aliquots of the uranium low-level standards. The relationship between the volume of potassium dichromate titrant and uranium amounts is determined over the range of uranium amounts expected in sample aliquots. A minimum of four standard values within this range are used and each is assayed a minimum of three times. Re-calibration is performed annually or whenever the titrant is changed or standards data indicate a need for re-calibration.

c. Gravimetric analyses. The accuracy of the gravimetric uranium analysis is directly related to weights and ignition to stoichiometric U_3O_8 . Balances are serviced annually by the manufacturer, calibrated monthly by plant personnel, and checked daily with a Class S-1 weight to assure proper operation. Non-volatile impurities remaining in the U_3O_8 are measured by emission spectroscopy. An impurity correction is made for each sample

An extensive study, using NBL 97 UO_2 standard and a Graeco-Latin square statistical design, was performed to determine that the ignition conditions produced stoichiometric U_3O_8 .

d. U-235 Analysis. Multiplier correction factors are determined using NBS isotopic standards (NBS UO_{10} , UO_{15} , etc.). No other calibration is required. The factors are reestimated on an annual basis. More frequent estimation is not required because biases in the mass spectrometer are controlled through the running of standards.

E. NDA Measurements

1. Calibration Standards. The waste assay counter described in Section II-C is calibrated annually. Five clean (unused) waste barrels containing known amounts of U-235 in sawdust and five unused (clean) HEPA filters containing from 6 to 101 plastic coupons filled with known amounts of U-235 serve as standards. In addition, chemical leaching experiments are carried out on selected barrels to provide a quantitative estimate of any bias produced by the process material matrix. The results of the leaching experiments are used to develop a matrix bias correction equation to correct for the matrix effect.

2. Measurement of Standards. Two of the standard barrels (or filters when filters are measured) are measured at the beginning of each period that process waste containers are to be

measured. Once the system meets operating criteria, then the two standards are measured at a minimum frequency of once per hour until measurement of the process barrels or filters is complete.

F. Volume Measurement

The main volume measurement used at the Model Plant is the measurement of liquid waste volumes in the Accountability Lagoon. That lagoon was calibrated for volume measurements by measuring the dimensions of the empty lagoon with surveying instruments and measuring tapes which could be checked against a gauge block traceable to NBS.

The liquid depth is measured with a permanently located depth marker with graduated markings. The depth stick is calibrated in relation to the lagoon bottom, by measuring the liquid depth when the lagoon is partially full over the entire surface of the lagoon. The average liquid depth obtained in this way is used to establish a relation between "true" liquid height and the graduated marker.

The liquid depth (or height) is measured just before and just after a transfer is made from the Accountability Lagoon to one of the larger storage lagoons. The liquid depth is also measured as part of the physical inventory.

The measured liquid depths are converted to volume (gallons) by an equation based on lagoon dimensions which relates the observed depth on the marker to lagoon volume. A typical volume measurement before a transfer is about 400,000 gallons.

G. STATISTICS

1. General. The scope of the statistical aspects of the measurement control program is shown in Table IV. As shown in the table, statistical techniques are applied to all elements of the measurement processes to estimate measurement errors, set control limits, derive calibration relationships, and estimate biases. In addition, comprehensive statistical evaluations are made periodically of the overall program and error propagation methods applied using current error estimates to calculate the plant LEMUF. The more important details of the statistical application are described next.

2. Statistical Applications

a. Calibrations. For solid waste barrels and filters, a quadratic equation is developed to relate net counts to U-235 content. For the mass discrimination factor on the mass spectrometer, an average correction factor per mass unit is developed after a statistical evaluation is done to assure that there are no non-random effects.

b. Control Limits. Control limits are developed for routine measurements. The general philosophy is to control operation at the 0.05 limit and declare out-of-control if the 0.001 limit is exceeded or if repeated measurements exceed the 0.05 limit.

TABLE IV
SCOPE OF STATISTICAL PROGRAM

<u>Weighing</u>	<u>Sampling</u>	<u>Overall</u>
Control Limits	Scrap resampling program data for error estimation	Comprehensive annual measurement review
Use statistical control to keep scales within error limits - No bias corrections	Lot-to-lot variation of powder and pellets	Update error parameters each 6 months
Estimate error parameters from routine data & experiments	U ₃ O ₈ experiments for sampling error	LEMUF calculations for each 6 months
Use S/R data for UF ₆ weighing errors	Liquid waste experiments - LE and BIAS estimation	Monitor all program data
<u>Analytical</u>	<u>Volume</u>	
Control Limits	Volume calibration and error parameters	
Mass Spectrometer Calibration		
Bias Estimation & Test of Significance		
Random Error from Routine Data		
<u>NDA</u>		
Control Limits		
Calibration Equation		
Standards Data for Instrument Drift		
Chemical Recovery vs. Counting for Matrix Bias		

c. Estimation of Random Errors. Replicate measurements data are analyzed by the method of paired differences to provide estimates of the random error variance. The variance among the paired differences is an estimate of twice the appropriate measurement error variance.

d. Estimation of Systematic Error

1. Weighing. The primary purpose of weighing standards is to detect when adjustments are needed and not to estimate biases. Systematic weighing errors are estimated by special weighing experiments in which a set of weights (usually unknown weights) is weighed in all combinations on a number of scales. Standard weights are then assigned these standards on the basis of the consensus data. A systematic error component, described by the variance of the population from which the systematic error for a given scale was selected at random, is then estimated by the square of the average difference between the observed weights for the standards (weighed singly and in combinations) and the corresponding assigned weights based on the consensus data.

For the balances used in rod loading, the sensitivity is such that the systematic error variance can be estimated from the quality control data on the weighing of known standards. An analysis of accumulated standards data is used to estimate the systematic error variance. Downloading experiments are also carried out which provide additional data for error estimation.

For the UF₆ scale, shipper/receiver data are used to obtain a realistic estimate of the systematic error. The statistical techniques for estimating both the long-term and short-term systematic error variances from shipper/receiver data are described in Reference 6 of Session 28 of the October 17 to November 4, 1983 course text, LA-10135-C.

2. Analytical. The primary function of the standards program is to give an early signal of possible problems with the analytical technique rather than to create data that will form the basis for making bias corrections. However, recognizing that small biases will occur, the standards data for each analytical technique are analyzed at the end of a material balance period to estimate the bias that existed for that analytical technique during the material balance period. The estimated bias is the average difference between the observed measurements on the standard and the assigned standard value.

3. NDA. Systematic errors in the solid waste measurements are estimated from the calibration data and from the leaching experiments described earlier.

4. Sampling. Systematic errors in sampling heterogeneous materials are estimated from special experiments.

5. Volume. Systematic errors in the volume measurement for liquid waste are estimated directly from the dimensional calibration data.

6. Application to Overall Program. Regarding statistical applications to the overall measurement control program, the activities shown below are carried out.

- Annually, a comprehensive measurement review is performed in which all the standards data and applicable shipper/receiver data are evaluated statistically. Error estimates are developed for key accountability measurements and yearly trends analyses are made. The measurement review is documented and a detailed report issued.
- Error parameters for the sigma MUF calculations are updated each six months and used in the sigma MUF calculation which is performed at the end of the six-month inventory.
- All data generated in the measurement control program are monitored routinely and appropriate statistical techniques applied to detect trends and identify possible problem areas.

H. Documentation

All data generated in the program are recorded as permanent records and a formal system of documentation followed. Examples of the various records and reports associated with the program are shown in Table V.

A more detailed discussion of the measurement control program is given in Session 28 of the October 17 to November 4, 1983 course text, LA-10135-C.

IV. MUF EVALUATION

A. Determination of MUF

At the Model Plant, a physical inventory is taken every six months. Each material control area of the plant is inventoried to provide a total plant inventory and an inventory of each ICA and MBA. The measured inventory of the plant (as a whole) is compared to the plant book inventory and a MUF is computed for each material type, e.g., depleted, natural and enriched uranium. For enriched uranium, a MUF is computed for each internal MBA for uranium and U-235. A uranium element MUF is also computed for depleted and natural uranium for each internal MBA.

No MUFs are generated in ICAs. However, each ICA must be completely reconciled so there are no unaccounted for items. If a listed item cannot be found or reconciled by documentation showing it was processed or shipped, it must be reported immediately to plant management and the safeguards authorities and an investigation initiated immediately.

B. Sigma MUF Calculation

In conjunction with the inventory reconciliation activities and MUF computations, sigma MUF (σ_{MUF}) values for the plant uranium and U-235 MUFs are calculated. Those calculations use the actual material flows, measurement errors, and number of measurements made by each method to calculate the total measurement uncertainty (σ_{MUF}) of the plant uranium and U-235 MUFs. In

TABLE V
EXAMPLE OF RECORDS AND REPORTS

Records	Reports
CUSUM and Control Charts	Sigma MUF Reports
Out-of-Control Documentation	Measurement Reviews
Calibration Data and Calculations	Bias Adjusted MUF
LEMUF Data and Calculations	Standard and Reference Reports
Quality Control Weekly Standard Weighing Data	Audit Results
Standards Preparation Data	Special Experiments
Mass Standards Calibration and Certification	Error Parameters
Measurement Review	

accordance with U.S. safeguards requirements, new estimates of the measurement errors are determined for each six month accounting period. Those estimates are based on measurement control program data and shipper/receiver difference data that are generated during the six month accounting period.

C. Comparison of MUF to σ MUF

The plant wide uranium and U-235 MUFs are compared to two times the corresponding σ MUF value. In the U.S., twice σ MUF is termed the LE or limit of error or the limit of error of MUF (LEMUF). If the observed MUFs for uranium and U-235 are less in absolute value than their respective LEMUFs, they are judged to be acceptable and no investigative actions are undertaken. If the uranium MUF exceeds its LE, but the U-235 MUF is less than the LE, an investigation is carried out and the regional office of the NRC is informally appraised of the status of the investigation. However, if the U-235 MUF exceeds its LE an investigation is initiated to determine probable causes and the NRC is formally notified that U-235 MUF has exceeded its calculated limit of error. The report to the NRC includes a statement of probable reasons for the excess MUF and the actions taken or planned to resolve the problem.

For plant purposes, the plant cumulative MUF since startup is computed and compared to the estimated cumulative limit of error.

The limits of errors for uranium and U-235 are converted to relative percentages of throughput and compared to the U.S. safeguards limit of 0.5 percent of throughput. As was shown earlier, the example uranium LE for the Model Plant is well below the LE limit of 0.5 percent.

D. Reporting of Performance

After the MUFs and LEMUFs have been determined, a MUF-LEMUF report is prepared. That report shows the uranium and U-235 MUFs and LEMUFs for enriched uranium in both absolute values and as a percentage of throughput. The report shows all the quantities of element and isotope which comprise the plant material balance and the material in-process. The report also shows the cumulative uranium and U-235 MUFs since plant startup.

The LEMUF part of the report shows the total measurement uncertainty for the uranium MUF as a variance with a breakdown which shows the variance (kilogram ^{2} U) of each source of error--weighing, sampling, analytical and NDA. The error sources for the uranium MUF are also shown by random and systematic error components. The main variance contributors are also shown by measurement method e.g., UF₆ scale, powder barrel scale, etc.

The U-235 variance (kilogram ^{2} U-235) is shown by random and systematic error components and by the variance contribution of feed product, and inventory.

V. ACCOUNTING FORMS, REPORTS AND COMPOSITION CODES

A. Accounting Forms and Reports

The accounting forms and reports used at the Model Plant are listed in Table VI. They are described in detail in Session 22 of the October 17 to November 4, 1983 course text, LA-10135-C. Their use is described briefly next.

The first two forms in the table are for entering and changing the data base. Form #3 is for documenting receipts and shipments and provides the basis for completing the DOE/NRC Form 741. Form #4, the Location Transfer Form, is used to document all internal movements of materials. The Project/Enrichment Transfer form is used to record transfers of material from one project to another and from one enrichment to another (blending). The weights of UF₆ cylinders are printed by the scale printer on the UF₆ Cylinder Weighing Ticket form. Form #7 is used to enter data into the Nuclear Materials Reporting System (NMRS). Form #8 is submitted to the Analytical Laboratory when sample analyses are requested. The Material Record Card which must accompany the nuclear material container shows the container number, material composition, project number, enrichment, lot number, gross and tare weights, date the item is created, and the operator's initials. The seal logbook sheet, Form #10, is used to record the container number, seal number, date of application, and signatures of the persons applying the seal. The Rod Fabrication Follower Card is used to record the pellet column weight and serial number of each rod produced.

The MIMS Report (Report #1) is an output report of the computer-based accounting system (NJCS). It shows the plant book inventory by MBA and ICA. It shows the book inventory of the MBAs by quantities of element and isotope for each project and enrichment. For the ICAs, it shows the ICA listing of items on the book for each ICA. The report does not routinely include individual rods and waste barrels and filters. Those reports are printed only upon request.

The Material Balance Ledger (NMRS) is a computer-based system in which all plant shipments and receipts are recorded. At inventory time, the physical inventory is recorded in that ledger.

The MUF and Discard Summary Report is used to report shipments, receipts, beginning and ending inventory, and waste discards by government and non-government ownership. This report is sent to the NRC Regional Office as an attachment to the MUF-LEMUF Report. The Possession Limits Report is prepared weekly to show amounts of depleted, natural, and enriched uranium and thorium in inventory versus the authorized possession limits for the plant. That report is issued for internal distribution to assure that possession limits are not exceeded and that licensing activities to increase possession limits are initiated on time.

TABLE VI
ACCOUNTING FORMS AND REPORTS

Forms

1. Request for Changes in NICS and NMRS Data Base
2. Corporate Data Base Input Form
3. Receipt/Shipment
4. Location Transfer
5. Project/Enrichment Transfer
6. UF₆ Cylinder Weigh Ticket
7. Nuclear Materials Reporting System Data Input
8. Analytical Request and Report Form (with label)
9. Material Record Card
10. Tamper-Indicating Seal Logbook
11. Rod Fabrication Follower Card

Reports

1. Material Inventory Maintenance System (MIMS)
2. Material Balance Ledger (NMRS)
3. MUF and Measured Discard Summary
4. Possession Limits
5. DOE/NRC Form 741 - Nuclear Material Transaction Report
6. DOE/NRC Form 742 - Material Status Report

The DOE/NRC Form 741 is used to report all shipments from and receipts to plant site. The DOE/NRC Form 742 is used to report material status after the completion of a six month accounting period. The Form 742 report is a fairly large report, containing some 60-80 pages when the Physical Inventory Listing is included.

B. Material Composition Codes

An example of the Material Composition Codes used at the Model Plant is shown in Table VII. Those codes are used for internal transfers and for identifying the composition of inventory items. The codes are also used at the Model Plant for identifying batches on inventory change reports for IAEA safeguards when completing DOE/NRC 741 Forms and when completing the Physical Inventory Listing for the 742 Form report. The Model Plant material composition codes are translated in the Plant Facility Attachment for IAEA safeguards to equivalent NRC composition codes and to IAEA composition codes. Thus, the reports submitted to the U.S. safeguards information system for transmittal to the IAEA can be translated from the Model Plants Material Composition Codes to the IAEA codes.

The material composition codes for all containers except UF₆ cylinders, fuel rods, bundles, and liquid waste are also shown on the container serial number. The containers are identified with sequentially numbered labels consisting of the last two letters of the composition codes in Table VII and a serial number, e.g.. HS-12300 would be the label for Hard Scrap container number 12300. A container of U₃O₈ would be labeled CU-00001 for the first container of U₃O₈.

VI. TAMPER-INDICATING SEALS

Two types of seals are used at the Model Plant. One is the Type E seal consisting of two metal parts that, when snapped together, form a closed box about the knot on the wire passing through the hasp. The seal is of the type described in Regulatory Guide 5.15. This seal is used primarily for long-term storage of items. A serial number is stamped on the outside of each of the two metal parts.

The second seal is a pressure sensitive paper seal. It is normally used to seal items for short-term storage. Each paper seal consists of two identical parts so that each container or bin can be sealed at two places. The seal in current use is supplied in rolls of 1,000 seals. It is a fully opaque paper seal with adhesive backing. The Exxon Nuclear Company name and a serial number are pre-printed on the seal.

The seals are purchased from a commercial vendor and are stored under lock by nuclear materials accounting. A record is kept of all seals purchased and of their transfer to the seal custodians who use the seals or issue them to other operators. The seal custodians sign the seal inventory logbook maintained by

TABLE VII
URANIUM MATERIAL COMPOSITION CODES

<u>Material Description</u>	<u>Composition Code</u>
Full Cylinders of UF ₆	UUF
Heel Cylinders of UF ₆	UUH
U ₃ O ₈	UOU
Prepared UO ₂ Powder	UPD
Calcined UO ₂ Powder	UCP
Sintered Pellets	USP
Fuel Rods	URD
Fuel Bundles	UUD
Hard Scrap	UHS
Grinder Sludge	USL
Ammonium Diuranate	UAD
Dirty Powder	UDP
Solid Waste Barrel (Drum)	UDW
Solid Waste Filter	UFW
Liquid Waste	ULW
Lagoon	ULG

nuclear materials accounting when they receive the seals. The seal custodians also maintain their own seal logbooks to keep a record of each seal in their possession and how each seal is used. At a minimum the seal logbook contains the serial number of the container to which it is applied, the seal number, the date applied and the signatures of the person(s) who applied the seal.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 35: SAFEGUARDS INSPECTION EXPERIENCE AT US NUCLEAR FACILITIES

Participants and panel members will be divided into five subgroups. Each subgroup will develop a list of questions for the panel. Then, following a short break, the panel will be convened to respond to subgroup questions and questions from the floor. The session will allow the participants to probe deeper into the US and IAEA safeguards activities at US facilities under IAEA safeguards. The interactions between the IAEA, the state authority, and the plant operator can be questioned in depth. Full participation by course attendees will be encouraged.



Panel members: Left to Right-Lars Wredberg, IAEA; Willie Theis, IAEA; Leroy Norderhaug, NRC; Dick Schneider, EXXON; and Steve Stilwagen, SONGS.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSIONS 36-38: WORKSHOP ON SAFEGUARDS SYSTEM DESIGN FOR REACTOR
FACILITIES

The purpose of the workshop is to provide participants with an opportunity to design the main features of the materials accounting and control system for a power reactor. To maximize individual participation, students will be divided into four groups, with each group independently tasked to design a state system of accounting and control (SSAC) for a postulated reference nuclear power program. The reference power plant will be a 1000-MWe pressurized-water reactor (PWR) in a nonnuclear weapons state. The plant receives fabricated fuel elements from another independent state. Spent fuel is 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.

Each design group will consider the problem of developing a safeguards approach and an SSAC for the reference facility, using the facility design information that is provided. The safeguards approach will consider materials accounting and control and inspection strategies for the fresh-fuel storage area, the reactor, and the spent-fuel storage pool.

Items to be addressed by the groups in their SSAC design considerations will include:

- 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 key measurement points.
- 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.

The objective is to develop an appropriate SSAC for the reference facility for use in negotiating a facility attachment with the IAEA representatives that will satisfy both domestic (SSAC) and international safeguards requirements.

Each design group will have one or more course staff assigned to assist the discussion and answer questions. Each group will prepare a summary report of their SSAC design and choose a rapporteur to present this report during session 38.



Workshop subgroup discusses problem with Dick Schneider, EXXON.



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION OF STATE SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

SESSION 36: WORKSHOP ON FACILITY SAFEGUARDS SYSTEM DESIGN

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I. INTRODUCTION

The safeguards approach to light water reactors (LWRs) is predicated on the assumption that nuclear material exist in discrete, identifiable units (fuel assemblies) whose physical integrity is maintained throughout their residence time at the nuclear reactor facility. The quantity of the material in these fuel assemblies is not measured by the facility operator, but instead is based on data supplied by the fuel fabricator and on calculations of production and loss resulting from fuel burnup. Under these circumstances, safeguards is applied through the use of "item accountancy" coupled with containment and surveillance measures. The main emphasis is on accounting for each individual fuel assembly, and in this way verifying the integrity of original fuel assemblies and thereby verifying the quantity of nuclear material contained therein.

Physical inventory of nuclear material within a LWR facility is established by counting assemblies and identifying serial numbers of fuel subassemblies. Seals and surveillance cameras are used to maintain continuity of safeguards knowledge during time intervals between inspections.

The main features of the IAEA safeguards approach at LWRs are the following:

- auditing of facility records and comparison with reports submitted to the Agency,
- periodic closing of material balance by the operator (usually annually) by taking physical inventory,
- independent verification of nuclear material by the Agency, usually by item counting and serial number identification, and
- application of containment and surveillance measures to maintain knowledge of nuclear material.

II. FACILITY DESCRIPTION

The reference power plant for this materials control and accounting (MC&A) system design is a 1,000 MWe pressurized water reactor (PWR). 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 facility for disassembly or repairs of fuel assemblies.

A. The Reactor

A normal reactor core loading comprises approximately 51,000 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 20 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 20,000 megawatts days per metric ton (Mwd/Te) with a maximum burnup of approximately 30,000 Mwd/Te.

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 540 kg including cladding and end-fittings. Representative characteristics of fresh PWR are summarized in Table I.

The irradiated fuel is strategically important because each PWR fuel pin contains about 15 g of plutonium (equivalent to about 3.5 kg of plutonium per fuel assembly).

B. 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.

C. Spent-Fuel Storage Pool

The spent-fuel storage pool has a capacity to store ~800 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. These alternatives include disassembly and recanning of fuel rods and storage in high density racks.

TABLE I

REPRESENTATIVE CHARACTERISTICS OF FRESH PWR FUEL

Fuel material	UO_2
Pellet diameter	0.9 cm
Pellet height	1.5 cm
Active fuel rod length	3.7 m
Cladding material	Zircaloy-4
Cladding thickness	0.06 cm
Assembly array	17 x 17 (or 15 x 15) open structure
Number of assemblies	~220
Total number of fuel rods	~51,000
Total amount of fuel (UO_2)	~100 Te
Design fuel burnup	32,000 Mwd/Te
Fresh fuel assay	3.2% ^{235}U (less in initial load)
Spent fuel assay (design)	0.9%
Refueling sequence	One-third of fuel per year

D. Facility Attachment

The provisions necessary to implement the facility safeguards are incorporated in Subsidiary Agreements and Facility Attachment. A Facility Attachment should contain

- (1) a short description of the facility,
- (2) a provision to submit to the IAEA changes in the information on the facility,
- (3) accountancy measures for the facility,
- (4) provisions for containment and surveillance measures,
- (5) specific provisions and criteria for termination of and exemption from safeguards of nuclear material,
- (6) a detailed description of the records and reports system,
- (7) a description of the mode and scope of IAEA routine inspections, and
- (8) provisions for administrative procedures concerning the facility.

A typical facility attachment for the IAEA is included as Appendix A.

III. STATE'S SYSTEM OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIAL

The establishment of a State System of Accounting for and Control of Nuclear Material (SSAC) can serve a useful function in all IAEA Safeguards Agreements, whether or not such a system is explicitly required. A Safeguards Agreement conforming to INFCIRC/153 (Corrected) is required to provide that "...the State shall establish and maintain a system of accounting for and control of nuclear material subject to safeguards under the Agreement...", the basic elements of which are set forth in paragraph 32 of that document.

A. Objectives

A State system of accounting for control of nuclear material should have both a national objective and an international objective. The national objective would be to account for and control nuclear material in the State and to contribute to the detection of possible losses or unauthorized use or removal of nuclear material. The international objective would be to provide the essential basis for the application of IAEA safeguards according to the provisions of the Agreement between the State and the IAEA. It is up to each State to decide whether or not to have one combined system or an independent system for each facility to pursue these different objectives.

B. Guidelines

An IAEA document (IAEA/SG/INF/2) provides a series of guidelines for the organization and function of an SSAC designed to assist the States in establishing and maintaining their system. To establish an SSAC, the State should designate a nuclear material accounting and control Authority with the responsibility for the State obligations under IAEA Safeguards Agreements. The "Authority" should establish the requirements of accounting for and control of nuclear material. These requirements should enable the IAEA to establish in a timely fashion whether there has been any diversion of "significant quantities" of nuclear material.

C. Requirements

The requirements for SSAC are established by the State authority, for example, in the U.S. by the Nuclear Regulatory Commission (NRC). In establishing requirements, the State's system should recognize:

- (1) the facility operator's need for data on material flow through the facility for criticality control, health and safety, materials control, as well as safeguards; and
- (2) the IAEA's requirements to verify whether material has not been diverted from the facility.

It is possible to design a State's system of accounting for and control of nuclear material that satisfies both the needs of the operator and IAEA without compromising any of the State's regulatory requirements. As examples, the following items are identified by the U.S. NRC as necessary for the State's system of accounting for and control of nuclear material and can be readily 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 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.

D. Material Balance Areas for Reactors

For purposes of nuclear materials accounting a reactor may be treated as a single material balance area (MBA), as shown in Fig. 1, with the following flow key measurement points (KMPs):

- KMP#1 - receipt of fresh fuel to the reactor
- KMP#2 - nuclear material loss and production in the reactor
- KMP#3 - shipments of spent fuel from the reactor
- KMP#4 - rebatching.

The following item accounting KMPs also may be defined:

- KMP A - fresh fuel in the fresh fuel storage pool or in dry storage
- 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).

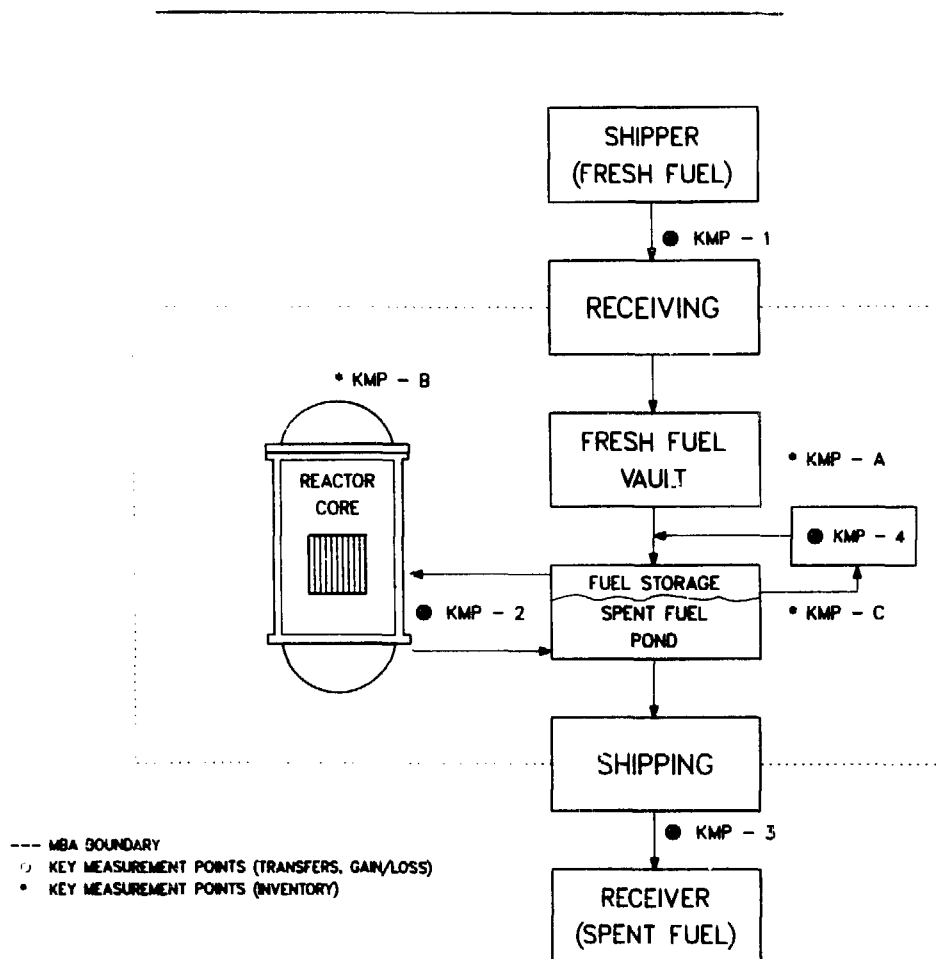


Fig. 1. Nuclear material flow at a LWR facility.

IV. 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 activities and examination of 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 knowledge of the fresh and spent fuel in the respective storage ponds is maintained through a complementary system of containment and surveillance.

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.

TABLE II

IAEA GOAL QUANTITIES AND DETECTION TIMES FOR NUCLEAR REACTORS

<u>Type of Material</u>	<u>Significant Quantity</u>	<u>Detection Goal</u>	<u>Detection Time</u>	<u>PIT Frequency</u>	<u>Inspections</u>
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

V. SAFEGUARDS SYSTEMS DESIGN APPROACH

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.

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 Agency's 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. Core movement records.
7. Inventory change reports.
8. Physical inventory listings.
9. Material balance reports.
10. Special reports.

The safeguards systems design should address the following points as a minimum.

- 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 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 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.
- An estimate of the required inspection manpower and capability.

The State's safeguards system design should also incorporate a program for assuring compliance with the requirements of the system. Such a program should have the primary objective of ensuring that the capability and performance by each facility in the discharge of its responsibility for accounting for and control of nuclear materials satisfy the requirements of the Agreement with the Agency. The term "capability" above refers to the actual presence at the facility of the records, equipment, trained personnel, documented procedures, and other resources; and "performance" refers to the use of the capability for accounting and control of nuclear materials at the facility. This objective of ensuring compliance can be achieved only through a comprehensive audit and inspection program by the SSAC Authority.

VI. WORKSHOP

The workshop session of this training course will make use of both the Facility Attachment (Appendix A) and the Design Information Questionnaire (DIQ) to formally propose a safeguards design for the reference power plant. The DIQ presented as Appendix B is partially completed. The participants of the workshop will present their safeguards system design descriptions in the format of the DIQ as answers to questions #55 and #57.

APPENDIX A

FACILITY ATTACHMENT

Model M-02 (D) for LWR with spent fuel stored for a year or longer,
with pin exchange.

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 18

Page No. 1

Date: 20 June 1985

Code	General Part Reference Codes	Agreement Reference (Articles)

1.	43(a)	<u>Identification of the facility</u> Facility identification code: AIE-
1.1		<u>Name, owner and operator</u> Atlantis Nuclear Power Station No. 4
1.2		<u>Geographic location</u> About 200 Km east of the city of Seroza
1.3		<u>Postal address</u> Atlantis Electric Power Co. Cape Nese, Atlantis
1.4		<u>Description</u> The power station consists of one direct cycle light water reactor, 1000 MWe. Refuelling Interval: Normally 12-18 months No. of assemblies Discharged: 60-100 Normal Wt of Fuel Assembly: 414 kg of U Maximum Fresh Fuel Enrichment: 3.1% No. of Fuel Assemblies in Case: 217 Fresh Fuel Storage Capacity: 180 Spent Fuel Storage Capacity: 800 Fuel rods may be discharged from fuel assemblies between refuelling operations.
1.5		<u>Maps and plans</u> Attached herewith.

Facility Attachment No. 6: Atlantis Nuclear Power Station No. 4

Page No. 2

Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
2.	3.1.1	43,44, 46(a)	<u>Information on the facility</u> This facility attachment is based on the Design Information provided by the Government of Atlantis as of July 1982.
2.1		8(c)	<u>Location of information</u> Identical sets of the information provided on the facility are kept at the Agency, at the facility, and at the Atlantis Nuclear Regulatory Bureau.
2.2	3.1.3	45	<u>Changes in the information on the facility to be provided in advance</u> (With reference to the relevant paragraphs of the design information questionnaire) 14 - Any change in the rated thermal output for continuous operation; 10 - Any change in the access routes to the reactor area; 24,25,27 - Any change in the design of the reactor fuel; 23 - Any change in the nominal enrichments of the fuel; 31 - Any change of the method of identifying individual fuel assemblies; 41 - Any change of the refuelling equipment or methods; 43 - Any change in connection with the reactor vessel or its cover influencing access to the core; 44 - Introduction of new irradiation positions inside the reactor vessel; 52 - Introduction of new loop heat removal equipment; - Installation of any fuel assembly decladding or dissolution equipment; 50 - Any change in the routes of the shipping cask for irradiated fuel within the facility; 54 - Any change in the health and safety procedures affecting the conduct of inspection.

Facility Attachment No. 6: Atlantis Nuclear Power Station No. 4

Page No. 3

Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
2.2.1			Any change in the parts of the design information to be submitted when change has been completed.
3.			<u>Safeguards measures</u>
3.1		29	<u>Accountancy</u>
3.1.1		46(b)	<u>Material balance areas and their identification codes.</u> The Atlantis Nuclear Power Station No. 4 consists of one material balance area.
3.1.2		46(b) 98K 98S	<u>Strategic points which are key measurement points (KMPs) (for their specifications see Code 4).</u>
	6.3		(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.* KMP 3 - Shipment of nuclear material, exemption, plutonium decay KMP 4 - Rebatching
	3.5 3.7		(b) For determination of physical inventory: KMP A - Fresh fuel storage and inspection area KMP B - Reactor core KMP C - Spent fuel storage, racks and shipping casks KMP D - Other locations of nuclear material at the facility
3.1.3		46(c)	<u>Physical inventory taking (PIT)</u> Nominal timing: once a year As soon as possible after each refuelling and before the reactor is closed again Procedures: Item counting and identification

*Fuel removed from the reactors shall be considered as discharged if it remains out of core for longer time than the duration of a routine refuelling shutdown.

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Page No. 4

Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
3.2		29 46(f) 98S	<u>Containment and surveillance</u> Strategic points for application of containment and surveillance measures: - Reactor - Access routes to the reactor hall, including the spent fuel pond
3.2.2		73(d) 73(e)	<u>Installed Agency instruments and devices:</u> (a) Seals to ensure the containment of the reactor vessel; (b) Cameras 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. 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	6.1	11 35(a)	<u>Specific provisions and criteria for termination of safeguards on nuclear material</u> None.
3.4	6.2	36	<u>Specific provisions and criteria for exempting nuclear material from safeguards</u> None.
3.5	3.7 7.1	12(a)	<u>Specific provisions and criteria for withdrawal of nuclear metal from safeguards</u> None.

Code 4

Agreement References (Articles): 46(b), 55, 90
Specifications for key measurement points

Code 4.1 KMPs for the flow of nuclear materials

KMP	Inventory change	Description of a typical			Material description	Measurement basis
		batch	item			
1	Receipt	<u>For fuel assemblies:</u> One fuel assembly	One fuel assembly	<u>For each fuel assembly:</u> 1) Identification number 2) Weights of total and fissile uranium and the chemical composition based on shipper's data.	BQ2F BV2F	N
		<u>For fuel pins:</u> A number of separate fuel pins with the same nominal initial total and fissile uranium content, received in one consignment.	One fuel pin	<u>For each fuel assembly:</u> 1) Identification number 2) Weights of total and fissile uranium and the chemical composition based on shipper's data.	BQ ₂ F DQ2F DV2F	N N N
		<u>For small quantities of nuclear material (each less than 0.01 effective kilogram):</u> 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.	QS0B	N
4	Rebatching	<u>For fuel:</u> 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

Code 4.1 KMPs for the flow of nuclear materials (continued)

KMP	Inventory change Rebatching	Description of a typical		Source data	Material description	Measurement basis
		batch	item			
2	Nuclear production Nuclear loss (burn-up)	<u>For fuel:</u> All fuel assemblies and separate fuel pins discharged together [for each batch].	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	<u>For each fuel pin:</u> 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
3	Nuclear loss	<u>For loss:</u> All fuel assemblies and separate fuel pins shipped together.	One fuel assembly One fuel pin	1)-2)-3) As above 4) Nuclear loss of total plutonium (Pu 241 decay) for each fuel assembly and separate fuel pin.	BQ1G BV1G DQ1G DV1G	M
3	Shipment	<u>For fuel assemblies:</u>	One fuel assembly	<u>For each fuel assembly:</u> 1) Identification number 2) Weights of total fissile uranium for irradiated fuel. 3) Weight of total fissile Pu as calculated to allow for nuclear loss and production for irradiated fuel. 4) Burn-up and data of discharge for irradiated fuel. 5) Isotopic composition, if available.	BQ3G BV1G BQ2F BV2F	M N

Code 4.1 KMPs for the flow of nuclear materials (continued)

KMP	Inventory change	Description of a typical		Source data	Material description	Measurement basis
		batch	item			
3	Shipment	<u>For separate fuel pins:</u> 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	<u>For each fuel pin:</u> 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	<u>For small quantities of nuclear material (each less than 0.01 effective kilogram:</u> Any number of such quantities shipped in one calendar month to the same recipient or, if a physical 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.	QS0B QS0J	N M
	Exemption	The same as for shipments at KMP-3 or receipts, respectively.				
	Accidental loss or gain					
1	De-exemption					

Code 4.2 KMPs for the physical inventory of nuclear materials

KMP	Description of a typical		Source data	Material description	Measurement basis
	batch	item			
A B	<u>For fuel assemblies:</u> One fuel assembly	One fuel assembly	<u>For each fuel assembly:</u> 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	<u>For fuel pins:</u> A number of separate fuel pins with the same nominal initial total and fissile uranium content	One fuel pin	<u>For each fuel rod:</u> 1) Identification number 2) Weights of total and fissile uranium and the chemical composition based on shipper's data	DQ1F DV1F	N
C	<u>For fuel assemblies:</u> 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 N
	<u>For separate fuel pins:</u> A number of separate fuel pins with the same nominal initial total and fissile uranium content	One fuel	<u>For each fuel pin:</u> 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
D	<u>For small quantities of nuclear materials (each less than 0.01 effective kilogram:</u> Any number of such quantities	Not applicable	1) Weight of compound 2) Weights of total and fissile uranium based on shipper's data	QS0B QS0J	N N

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Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
5.		46(d), 51	<u>Records system</u>
5.1		56	<u>Specific provisions for accounting records</u>
5.1.1		56(1), 98D	<u>Inventory and batch changes (for the specifications of source data see Code 4.1 above), time of recording</u> - 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 (KMP-4). - Nuclear production and nuclear loss (uranium burn-up) (KMP 2): For fuel assemblies and fuel pins, upon discharge from the reactor.* For small quantities of nuclear material, upon shipment. - Nuclear loss (KMP 2): If considered appropriate.
	6.3, 6.2		- Exemption/de-exemption (KMP 3/KMP 1): Upon transfer of nuclear material out of/into the facility respectively.
	3.5		- Accidental loss or gain (KMP to be determined after the event): Upon determining the amount of the loss or gain.
5.1.2	4.2	54(b), 98D	<u>Measurement (item counting and identification) results used for determination of the physical inventory (for the specification of source data see Code 4.2 above), time of recording.</u> - For 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.

*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.

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Page No. 10

Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
5.1.3		56(c)	<u>Adjustments and corrections, time of recording.</u> - Shipper/receiver difference (KMP 1): Not relevant. - MUF Normally identical to zero. - Corrections Whenever errors have been found.
5.2	2.1.2	56	<u>Specific provisions for operating records</u>
5.2.1		56(a)	<u>Operating data used to establish changes in the quantities and composition of nuclear material.</u> - 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; and (b) The estimated burn-up (in MWd/tU) 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		56(b)	<u>Calibrations</u> Not required.
5.2.3		56(c)	<u>Sequence of the actions taken in preparing for and in taking the physical inventory.</u> - All physical inventory KMPs: Dates and description of the actions taken and the results obtained. An itemized list of nuclear material inventory after completion of inventory taking by the operator but before commencement of verification by the Agency.

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Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
5.2.4		56(d)	<u>Actions taken in order to ascertain the cause and magnitude of any accidental or unmeasured loss:</u> - Dates and description of the actions taken and the results obtained.
5.3		50	<u>Location and language of records</u> At the facility.
5.4		51	<u>Retention period for records</u> Five years. (After shipment of fuel out of the facility)
6.		46(d),57	<u>Reports system</u>
6.1	3.4.1	61(a),62 65	<u>Specific provisions for inventory change reports (ICRS)</u>
6.1.1	10		<u>Contents.</u> The recorded entries to be reported are those types 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 assemblies will be reported using Codes RP and RM.* Forms R.01.1/c will be completed as specified in paragraphs 1-23 of Code 10, General Part, to report shipments of irradiated fuel assemblies, in respect of which the isotopic composition is available.
6.1.2			<u>Timing or frequency of dispatch.</u> 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.

*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.

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Page No. 12

Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	

6.2	3.4.1	* 62	<u>Specific provisions for concise notes</u>
6.2.1		62(a)	<u>Concise notes explaining the inventory changes:</u> - 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		62(b)	<u>Concise notes describing the anticipated operational programme; subject and time of dispatch:</u> - 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. - 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. - 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	61(b),65	<u>Specific provisions for material balance reports</u>
6.3.1	10		<u>Contents</u> 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.

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Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)

6.3.2	10	98D	<u>Physical inventory listings (PILs) to be attached to MBRs:</u> 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			<u>Timing or frequency of dispatch:</u> Within 30 days of the physical inventory taking under Code 3.1.3 above.
6.4	3.5.1	68	<u>Special reports</u>
6.4.1	3.5	66,89(b)	<u>Specification of circumstances requiring submission of special reports.</u> (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.*
6.4.2			<u>Contents, as appropriate.</u> - Date when the incident or circumstance occurred; - 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.

*In respect of seals on shipping casks, this requirement applies only while the cask remains in the facility.

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Date: 20 June 1985

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
7.			<u>Inspections</u>
7.1		78,80	<u>Mode of routine inspection</u> Intermittent.
7.2		78	<u>Applicable formula and procedure for determination of maximum routine inspection effort</u> Article 80(a) of the Agreement.
7.3		76,79	<u>Indication of the actual inspection effort under ordinary circumstances</u> 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 de- scribed in the information provided in re- spect 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 a year and two irradiated fuel shipments, including no more than one involving casks that are not completely filled; and/or
			(d) That if irradiated fuel shipments were made, there must be a means of correlating the identity and number of fuel assemblies at the receiving facility. 25 man-days per year.
7.4		72,73	<u>Indication of the scope of routine inspections under ordinary circumstances</u>
7.4.1			<u>General:</u> - Examination of the records, certification for self-consistency and consistency with reports.

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Code	General Part Reference (Codes)	Agreement Reference (Articles)

7.4.2		<u>At inventory KMPs:</u> - Verification of the inventory, e.g. by item counting, identification and integrity checks, nondestructive measurements.
7.4.3		<u>At flow KMPs:</u> - Verification of inventory changes, e.g. by item counting, identification and integrity checks, nondestructive measurements of fresh and irradiated fuel, including the use of seals on containers of fresh and irradiated fuel.
7.4.4		<u>At strategic points for containment and surveillance (reactor hall and access routes thereto):</u> - 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		73(d) <u>Arrangements for the use by the Agency of equipment for independent measurement</u> Specific arrangements for the use of equipment to be made as the need arises.
7.6	9.4	73(a) <u>Duplicates and additional samples</u> Not relevant.
7.7		85,86 <u>Persons to whom a request for any operation or for services at the facility should be addressed</u> The representative of the ANRB.
7.8		<u>Contacts at the facility</u> The representative of the ANRB.

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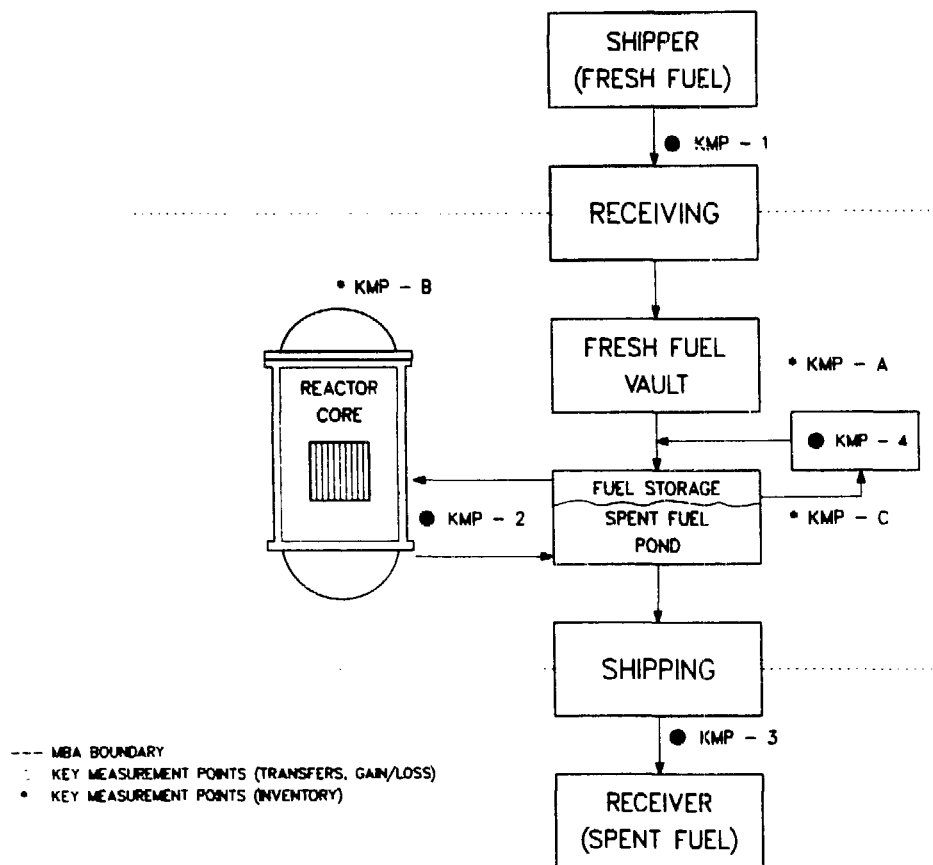
Code	General Part Reference (Codes)	Agreement Reference (Articles)	
7.9		14,86	<u>Services and charges</u> Services provided by the operator free of charge:
7.9.1			- Health and safety services (protective clothing, dosimeters); - Office space for the Agency's inspectors; - Power supply for the Agency instruments; - Personnel for handling the fuel assemblies during their measurements; - Available equipment for handling the fuel assemblies during their measurements. - Personnel to escort and facilitate audit activities.
7.9.2			<u>Services provided by the operator with the charges to the Agency, as quoted below:</u> - 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			<u>Mode of reimbursement of the expenses charged to the Agency:</u> - By cheque after receipt of the invoice by the Agency.
7.10	3.2	44	<u>Specific facility health and safety rules and regulations to be observed by the Agency's inspectors</u> As specified in paragraph 42 of the Design Information provided by the Government of Atlantis dated July 1984.

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Code	General Part Reference (Codes)	Agreement Reference (Articles)	
8.	4.1.3	30,88	<u>Agency statements</u>
8.1		88(a)	A summary statement will be made on the result of each inspection within 30 days of its completion.
8.2		88(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.



Nuclear material flow at the Atlantis Power Station No. 4.

APPENDIX B

DESIGN INFORMATION QUESTIONNAIRE

Atlantis Nuclear Power Station No. 4

CONFIDENTIAL

INTERNATIONAL ATOMIC ENERGY AGENCY
DEPARTMENT OF SAFEGUARDS AND INSPECTION

DESIGN INFORMATION QUESTIONNAIRE *

IAEA USE ONLY

The purpose of this document is to obtain the facility design information required by the Agency in order to discharge its safeguards responsibilities. It will also serve as a check list for examination of design information by Agency inspector(s). If, in any area, insufficient space is available add further sheets to the extent necessary.

IAEA USE ONLY	
COUNTRY	
COUNTRY OFFICER	
TYPE	
DATE OF INITIAL DATA	
VERIFICATION	
LAST REVIEW AND UPDATING	

* Questions which are not applicable may be left unanswered.

ALL FACILITIES

GENERAL INFORMATION			
1. NAME OF THE FACILITY (incl. usual abbreviation)	Atlantis Nuclear Power Station No. 4 (ANPS-4)		
2. LOCATION AND POSTAL ADDRESS	Atlantis Electric Power Co. P. O. Box 1000 Cape Nese, Atlantis Located 200 Km east of the city of Seroza		
3. OWNER (legally responsible)	Atlantis Electric Power Co.		
4. OPERATOR (legally responsible)	Atlantis Electric Power Co.		
5. DESCRIPTION (main features only)	Pressurized light water reactor, slightly enriched uranium (av: 2.5%), 1 operating unit on-site		
6. PURPOSE	Commercial power generation		
7. STATUS (planned; under construction; in operation)	Operational First fuel receipt: November 15, 1982 Initial criticality: April 10, 1983		
8. CONSTRUCTION SCHEDULE DATES (if not in operation)	Start of Construction	Commissioning	Operation
	(Facility in operation)		
	N/A	N/A	N/A
9. NORMAL OPERATING MODE (days only, two-shift, three shift; number of days/annum, etc.)	3 shifts; 330 on-power days per year		
10. FACILITY LAYOUT (structural containment, fences, access, nuclear material storage areas, laboratories, waste disposal areas, routes followed by nuclear material, experimental and test areas, etc.)	DRAWING(S) ATTACHED UNDER REF. Nos. (See Attachments 10.1-10.8)		
11. SITE LAYOUT (site plan showing in sufficient detail; location, premises and perimeter of facility, other buildings, roads, railways, rivers, etc.)	DRAWING(S) AND/OR MAPS ATTACHED UNDER REF. Nos. (See Attachment 11.1)		
12. NAMES AND/OR TITLE AND ADDRESS OF RESPONSIBLE OFFICERS (for nuclear material accountability and control and contact with the Agency. If possible attach organization charts showing position of officers)	See Attachment 12.1 Contact: Plant Engineer Atlantis Electric Power Co. P. O. Box 1000 Cape Nese, Atlantis		

GENERAL REACTOR DATA									
13. FACILITY DESCRIPTION	GENERAL FLOW DIAGRAM(S) ATTACHED UNDER REF. Nos. See Attachment 13.1-13.6								
14. RATED THERMAL OUTPUT, ELECTRICITY OUTPUT (for power reactors)	<table> <tr> <td>Gross thermal</td> <td>3338 MWt</td> </tr> <tr> <td>Gross electrical</td> <td>1134 MWe</td> </tr> <tr> <td>Net electrical</td> <td>1084 MWe</td> </tr> <tr> <td>Station use</td> <td>3.5%</td> </tr> </table>	Gross thermal	3338 MWt	Gross electrical	1134 MWe	Net electrical	1084 MWe	Station use	3.5%
Gross thermal	3338 MWt								
Gross electrical	1134 MWe								
Net electrical	1084 MWe								
Station use	3.5%								
15. NUMBER OF UNITS (REACTORS) AND THEIR LAYOUT IN THE NUCLEAR POWER PLANT	• 1 reactor unit • Reference Attachment 10.1								
16. REACTOR TYPE	Combustion Engineering Pressurized Water Reactor								
17. TYPE OF REFUELLING (on or off load)	Off-load refueling								
18. CORE ENRICHMENT RANGE AND PU CONCENTRATION (at equilibrium for on-load reactors, initial and final for off-load reactors)	See Attachment 18.1 (3.2% → 0.9%)								
19. MODERATOR	Pressurized light water								
20. COOLANT	Pressurized light water								
21. BLANKET, REFLECTOR	Reflector is water. There is no blanket.								

NUCLEAR MATERIAL DESCRIPTION	
22. TYPES OF FRESH FUEL	INITIAL LOAD -- Low-enriched uranium (operational) [Between 2-3% enriched] RELOAD -- Low-enriched uranium (operational) [Between 3-4% enriched]
23. FRESH FUEL ENRICHMENT (U-235) AND/OR PU CONTENT (average enrichment per each type of assembly)	See Attachment 23.1 As in 22 above [Indicates enrichment of all fuel pins, if they are different]
24. NOMINAL WEIGHT OF FUEL IN ELEMENTS/ ASSEMBLIES (with design tolerances)	See Attachment 24.1 [Provides weights of fuel assemblies and fuel pins]
25. PHYSICAL AND CHEMICAL FORM OF FRESH FUEL (general description)	<u>Operational Fuel</u> • Physical form: sintered pellets • Chemical form: UO_2 • Poisoned rods: Yes • Radioactivity: 200 $\mu\text{Ci/kg}$

NUCLEAR MATERIAL DESCRIPTION

26. REACTOR ASSEMBLIES*
(indicate for each type)

- types of assemblies;
- number of fuel assemblies, control and shim assemblies, experimental assemblies in the core, in blanket zone(s);
- number and types of fuel rods/elements**
- average enrichment and/or Pu content per assembly;
- general structure;
- geometric form;
- dimensions;
- cladding material

DRAWING(S) ATTACHED UNDER REF. Nos.

See Attachments 26.1-26.3

27. DESCRIPTION OF FRESH FUEL ELEMENTS

- physical and chemical form of fuel;
- nuclear material and fissionable material and its quantity (with design tolerances);
- enrichment and/or Pu content;
- geometric form;
- dimensions;
- number of slugs/pellets per element;
- composition of alloy;
- cladding material (thickness, composition of material, bonding)

DRAWING(S) ATTACHED UNDER REF. Nos.

See Attachments 27.1-27.2

* Assembly is the combination of elements or handling units such as cluster or bundle.
** Element is the smallest contained fuel unit.

NUCLEAR MATERIAL DESCRIPTION									
28. PROVISION FOR ELEMENT EXCHANGE IN ASSEMBLIES OF EACH TYPE (indicate whether this is foreseen to become a routine operation)	- Fuel assemblies are designed to be non-destructively disassembled and rebuilt. - The facility was designed to accommodate element exchange. - Element exchange is routine. - See Attachment 28.1 - Reference Attachment 13.4								
29. BASIC OPERATIONAL ACCOUNTING UNIT(S) (fuel elements assemblies, etc.)	DRAWING(S) ATTACHED UNDER REF. Nos See Attachment 29.1 Reference Attachment 26.2 [Fuel assemblies: 1 batch = 1 item. Fuel pins: 1 batch $\equiv$ all fuel pins of equal initial fuel content and enrichment status]								
30. OTHER TYPES OF UNITS	See Attachment 30.1 Reference Attachment 26.2 and 31.1								
31. MEANS OF NUCLEAR MATERIAL/FUEL IDENTIFICATION	<table> <thead> <tr> <th><u>Method</u></th> <th><u>Area</u></th> </tr> </thead> <tbody> <tr> <td>Direct visual</td> <td>Fresh fuel storage</td> </tr> <tr> <td>Underwater camera</td> <td>Core</td> </tr> <tr> <td>Underwater camera</td> <td>Spent fuel storage</td> </tr> </tbody> </table>	<u>Method</u>	<u>Area</u>	Direct visual	Fresh fuel storage	Underwater camera	Core	Underwater camera	Spent fuel storage
<u>Method</u>	<u>Area</u>								
Direct visual	Fresh fuel storage								
Underwater camera	Core								
Underwater camera	Spent fuel storage								

NUCLEAR MATERIAL DESCRIPTION

32. OTHER NUCLEAR MATERIAL IN THE
FACILITY
(each separately identified)

The core contains 2 Pu-238 beryllium neutron sources for initial and subsequent startups. The sources are supported from the fuel assembly upper end fitting and are contained within the control rod assembly guide tubes. The sources are accounted for by an item count. Fresh startup sources are stored in the fresh fuel storage area and irradiated sources are stored in the spent fuel pool. Ex-core instrumentation includes neutron detectors (i.e., fission chambers) which contain gram amounts of U-235. These also are accounted for by item count.

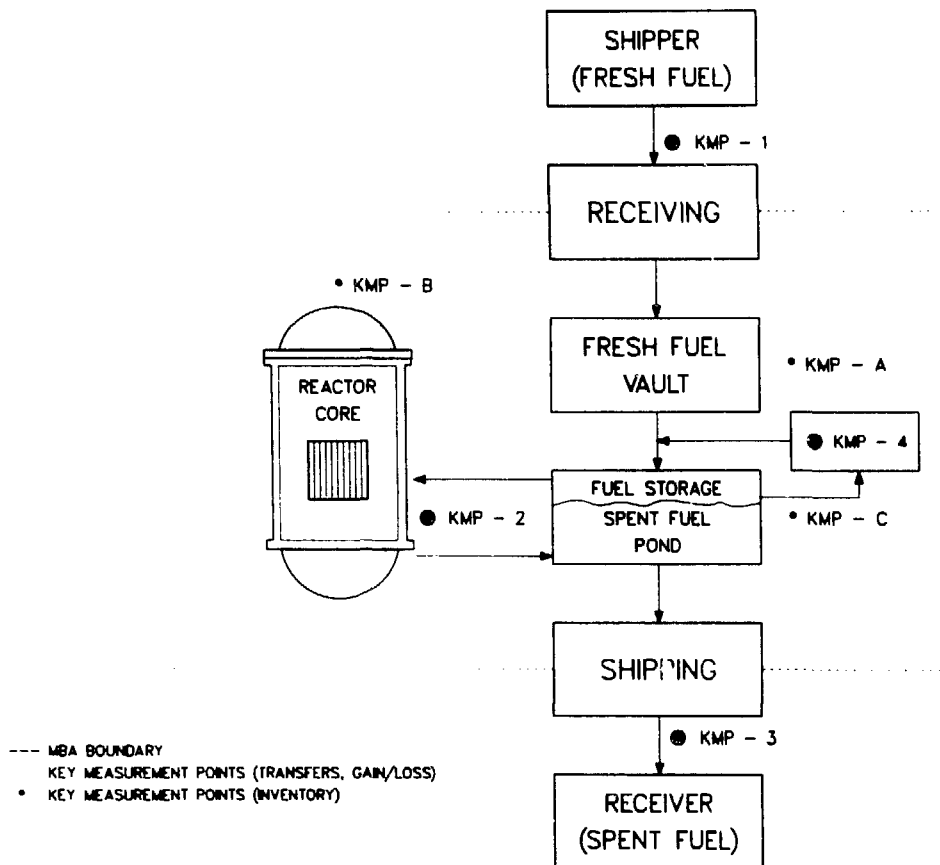
NUCLEAR MATERIAL FLOW

33. SCHEMATIC FLOW SHEET FOR NUCLEAR
MATERIAL
(identifying measurement points, accountability
areas, inventory locations, etc.)

DIAGRAM(S) ATTACHED UNDER REF. Nos.
See Attachment 33.1

Reference Attachments 34.1

See attached illustration (page 6-A)



Schematic flow sheet for nuclear material (Ref. Q33 of DIQ).

NUCLEAR MATERIAL FLOW	
34. INVENTORY STATE QUANTITY RANGE, NUMBER OF ITEMS, AND APPROXIMATE URANIUM ENRICHMENT AND PLUTONIUM CONTENT FOR (under normal operating conditions)	See Attachment 34.1 180 fuel assemblies plus 200 pins 217 fuel assemblies Up to 800 fuel assemblies (plus pins) <1 effective kg
35. LOAD FACTOR (power reactor only)	1982 (actual) - 0.66 1983 (actual) - 0.60 1984 (estimated) - 0.70 1985 (estimated) - 0.75
36. REACTOR CORE LOADING (number of elements/assemblies)	51,000 fuel rods 1600 poison rods 217 fuel assemblies

NUCLEAR MATERIAL FLOW										
37. REFUELLING REQUIREMENTS (quantity, time interval)	Interval between refuelings: 12-18 months Shutdown time: 3-5 weeks Quantity: 35 tons of fuel 70-90 spent fuel assemblies are unloaded and 70-90 fresh fuel assemblies are loaded									
38. BURN-UP (average, Maximum)	<table> <thead> <tr> <th></th> <th><u>First Cycle</u></th> <th><u>Equil. Cycle</u></th> </tr> </thead> <tbody> <tr> <td>Average:</td> <td>14,000 Mwd/t</td> <td>20,000 Mwd/t</td> </tr> <tr> <td>Maximum:</td> <td>21,000 Mwd/t</td> <td>30,000 Mwd/t</td> </tr> </tbody> </table>		<u>First Cycle</u>	<u>Equil. Cycle</u>	Average:	14,000 Mwd/t	20,000 Mwd/t	Maximum:	21,000 Mwd/t	30,000 Mwd/t
	<u>First Cycle</u>	<u>Equil. Cycle</u>								
Average:	14,000 Mwd/t	20,000 Mwd/t								
Maximum:	21,000 Mwd/t	30,000 Mwd/t								
39. IS THE IRRADIATED FUEL TO BE REPROCESSED OR STORED? (if stored, indicate site)	Disposition: • Stored on site • On-site storage time: until off-site storage is available									
NUCLEAR MATERIAL HANDLING										
40. FRESH FUEL ii) packaging (description)	See Attachment 40.1									
iii) layout, general arrangements and storage plan	DRAWING(S) ATTACHED UNDER REF. Nos. Reference Attachments 10.3-10.8, 13.1									

Date:

CONFIDENTIAL
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NUCLEAR MATERIAL HANDLING										
43. REACTOR VESSEL (showing core location, access to vessel, vessel openings, fuel handling in vessel)	DRAWING(S) ATTACHED UNDER REF. Nos. See Attachment 43.1 [Access from top only, after lid removal]									
44. REACTOR CORE DIAGRAM (showing general disposition, lattice, form, pitch, dimensions of core, reflector, blanket, location, shapes and dimensions of fuel elements/assemblies; control elements/assemblies; experimental elements/assemblies)	DRAWING(S) ATTACHED UNDER REF. Nos. See Attachment 44.1									
45. NUMBER AND SIZE OF CHANNELS FOR FUEL ELEMENTS OR ASSEMBLIES AND FOR CONTROL ELEMENTS IN THE CORE	<table> <thead> <tr> <th></th> <th><u>Number</u></th> <th><u>Size</u></th> </tr> </thead> <tbody> <tr> <td>Fuel assembly channels</td> <td>217</td> <td>20.441 cm x 20.441 cm</td> </tr> <tr> <td>Control assembly channels</td> <td>91</td> <td>2.10 cm O.D. (size of each rod in the assembly)</td> </tr> </tbody> </table> See Attachment 45.1 - Control Rod Assembly Locations		<u>Number</u>	<u>Size</u>	Fuel assembly channels	217	20.441 cm x 20.441 cm	Control assembly channels	91	2.10 cm O.D. (size of each rod in the assembly)
	<u>Number</u>	<u>Size</u>								
Fuel assembly channels	217	20.441 cm x 20.441 cm								
Control assembly channels	91	2.10 cm O.D. (size of each rod in the assembly)								
46. AVERAGE MEAN NEUTRON FLUX IN THE CORE thermal: fast:	Thermal: 5.3×10^{13} neutrons per $\text{cm}^2/\text{second}$ Fast: 2.2×10^{14} neutrons per $\text{cm}^2/\text{second}$									

NUCLEAR MATERIAL HANDLING	
47. INSTRUMENTATION FOR MEASURING NEUTRON AND GAMMA FLUX	See Attachments 47.1-47.3 • Gamma flux is not measured
48. IRRADIATED FUEL	i) layout, spent fuel storage plan and general arrangements (internal and external) DRAWING(S) ATTACHED UNDER REF. Nos. Reference Attachments 10.3-10.8
ii) method of storage	a) Fuel is stored as assemblies b) Fuel is stored in a pool cooled by water Reference Attachment 13.2 Reference Attachment 10.3-10.8
iii) design capacity of storage	290* assemblies in spent fuel pool (external building) No assemblies are stored in the reactor building (internal building) *According to the facility description, the maximum storage capacity is ~800 fuel assemblies
iv) minimum and normal cooling period prior to shipment	Minimum cooling period: 180 days Normal cooling period: 400 days

NUCLEAR MATERIAL HANDLING

IRRADIATED FUEL CONTINUED

vi. description of irradiated fuel transport
equipment and shipping cask
(if no information on site, where is it held?)

DRAWING(S) ATTACHED UNDER REF. Nos.

See Attachments 48.1-48.2

49. MAXIMUM ACTIVITY OF FUEL/BLANKET
AFTER REFUELLING
(at the surface and at a distance of 1 metre)

No blanket

50. METHODS AND EQUIPMENT FOR HANDLING
IRRADIATED FUEL
(except for that already given under Qs. 41, 48.v)

See Attachment 50.1

Reference Attachments 10.3-10.8, 13.2, 41.1,
41.5, 48.1, 48.2

NUCLEAR MATERIAL HANDLING

51. NUCLEAR MATERIAL TESTING AREAS

(except as already given under Q. 40)

For each such area briefly describe

i) Nature of Activities

None

ii) Major Equipment Available
(e.g., hot cell, fuel element decladding and
dissolution equipment)

See 1) above

iii) Shipping Containers Used
(main material, scrap and waste)

See 1) above

iv) Storage areas for both unirradiated and
irradiated materials

See 1) above

v) Layout and General Arrangement

See 1) above

DRAWING(S) ATTACHED UNDER REF. Nos.

COOLANT DATA	
52. FLOW DIAGRAM (indicating mass flow, temperature and pressure at major points, etc.)	DRAWING(S) ATTACHED UNDER REF. Nos. See Attachment 52.1
PROTECTION AND SAFETY MEASURES	
53. BASIC MEASURES FOR PHYSICAL PROTECTION OF NUCLEAR MATERIAL	The inspector will receive an entrance briefing on physical security

PROTECTION AND SAFETY MEASURES

54. SPECIFIC HEALTH AND SAFETY RULES
FOR INSPECTOR COMPLIANCE
(if extensive, attach separately)

The inspector will receive an entrance briefing on health and safety rules. A briefing is given rather than published guidelines because conditions at the facility change over time. To assist the inspector and to assure his health and safety, a plant official will be assigned to escort him.

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL**55. SYSTEM DESCRIPTION**

Give a description of the nuclear material accounting system, of the method of recording and reporting accountancy data, the procedures for account adjustment after inventory, and correction of mistakes, etc. under the following headings:

i) General

(This section should also state what general and subsidiary ledgers will be used, their form (hard copies, tapes, microfilms, etc.) as well as who has the responsibility and authority. Source data (e.g. shipping and receiving forms, the initial recording of measurements and measurement control sheets) should be identified. The procedures for making adjustments, the source data and records should be covered as well as how the adjustments are authorized and substantiated.)

SPECIMEN FORMS USED IN ALL PROCEDURES ATTACHED UNDER REF. Nos.

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL**SYSTEM DESCRIPTION CONTINUED**

ii) Receipts

iii) Shipments

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL**SYSTEM DESCRIPTION CONTINUED**

- iv) Physical inventory
Description of procedures, scheduled frequency, methods of operator's inventory taking (both for item and/or mass accountancy), including relevant assay methods and expected accuracy, access to nuclear material, possible verification method for irradiated nuclear material, methods of verification of nuclear material in the core

LIST OF MAJOR ITEMS OF EQUIPMENT REGARDED AS NUCLEAR MATERIAL CONTAINERS ATTACHED UNDER REF. Nos.

- v) Nuclear loss and production
(estimation of limits)

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL**SYSTEM DESCRIPTION CONTINUED**

- vii) Operational records and accounts
(including method of adjustment or
correction and place or preservation and
language)

**56. FEATURES RELATED TO CONTAINMENT
AND SURVEILLANCE MEASURES**
(general description)

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL

57. FOR EACH MEASUREMENT POINT OF ACCOUNTABILITY AREAS, IDENTIFIED IN PARTICULAR UNDER QS. 13, 33, 34 GIVE THE FOLLOWING (IF APPLICABLE)*

i) Description of location, type, identification

IF NECESSARY, ATTACH DRAWING(S)

ii) Anticipated types of inventory change and possibilities to use this measurement point for physical inventory taking

iii) Physical and chemical form of nuclear material
(with cladding material's description)

* For each measurement point fill in separate sheet.

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL

FOR EACH MEASUREMENT POINT OF
ACCOUNTABILITY AREAS, IDENTIFIED IN
PARTICULAR UNDER QS, 13, 33, 34
GIVE THE FOLLOWING (IF APPLICABLE)*

iv) Nuclear material containers, packaging

v) Sampling procedures and equipment used

vii) Measurement method(s) and equipment
used
(item counting, neutron flux, power level,
nuclear burn-up and production, etc.)

* For each measurement point fill in separate sheet.

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL

FOR EACH MEASUREMETN POINT OF
ACCOUNTABILITY AREAS, IDENTIFIED IN
PARTICULAR UNDER QS. 13, 33, 34
GIVE THE FOLLOWING (IF APPLICABLE):*

vii) Source and level of accuracy

viii) Technique and frequency of calibration of
equipment used

ix) Programme for the counting appraisal of the
accuracy of methods and techniques used

* For each measurement point fill in separate sheet.

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL

FOR EACH MEASUREMENT POINT OF
ACCOUNTABILITY AREAS, IDENTIFIED IN
PARTICULAR UNDER QS. 13, 33, 34
GIVE THE FOLLOWING (IF APPLICABLE):

x) Methods of converting source data to batch
data
(standard calculative procedures, constants
used, empirical relationships, etc.)

xii) Anticipated batch flow per year

xiii) Anticipated number of items per flow and
inventory batches

* For each measurement point fill in separate sheet.

NUCLEAR MATERIAL ACCOUNTANCY AND CONTROL

FOR EACH MEASUREMENT POINT OF ACCOUNTABILITY AREAS, IDENTIFIED IN PARTICULAR UNDER OS. 13, 33, 34 GIVE THE FOLLOWING (IF APPLICABLE):*

xiii) Type, composition and quantity of nuclear material per batch
(with indication of batch data, total weight of each element of nuclear material and, in the case of plutonium and uranium, the isotopic composition when appropriate, form of nuclear material.

xiv) Access to nuclear material and its location

xv) Features related to containment-surveillance measures

* For each measurement point fill in separate sheet.

OPTIONAL INFORMATION	
58. OPTIONAL INFORMATION (that the operator considers relevant to safeguarding the facility)	

Signature of Responsible Officer

.....

Date:



INTERNATIONAL TRAINING COURSE ON IMPLEMENTATION
OF STATE SYSTEMS OF ACCOUNTING FOR
AND CONTROL OF NUCLEAR MATERIALS



June 3 - 21, 1985

Session Objectives

SESSION 39: COURSE EVALUATION, DISCUSSION, AND WRAP-UP

In the final session, attendees will be asked to complete a detailed course evaluation form that provides feedback on the effectiveness and value of the overall course, evaluation of individual sessions, and suggestions for improving follow-on courses. Time will be allocated for general discussion of questions and comments from course attendees. Provision will also be made for informal discussions, individual consultation, and follow-up on special problems, as appropriate, between individual attendees, course staff, and technical specialists from Los Alamos and SONGS.

The course will conclude with a summary and closing remarks by course sponsors and participating organizations.