JNMM

Journal of Nuclear
Materials Management

CONTENTS

Volume XXVIII, Number 2 • Winter 2000

PAPERS

Strong Technical Program Planned for 41st Annual Meeting

Having just returned from the INMM Technical Program Committee meeting in Chicago last week, I am excited about the high quality of the 41st Annual Meeting program. Charles Pietri and the Technical Program Committee have laid out a strong program which, for the first time, will extend through Thursday afternoon. Thursday morning will have regular concurrent sessions, with Thursday afternoon devoted to the closing plenary session. We hope these changes will strengthen the quality of the annual meeting, and we welcome your comments. We have many first-time presenters, as well as some "regulars," who have become familiar to us over the years. We expect the meeting to provide another excellent forum for sharing information about the latest technical advances, policy changes, and developing issues in the field of nuclear materials management. In addition, you will find many other opportunities to network, including the informal hallway meetings. I hope you will take advantage of the opportunity to participate in the INMM annual business meeting and executive committee meeting as well. You will find information about dates and times in the soon-to-be-published preliminary program.

While the entire world has an interest in nonproliferation and arms control, the INMM's role is vital because we are an international professional society engaged in global nuclear materials management. The INMM's commitment to this field is evidenced by its involvement in upcoming activities. There are a number of INMM-sponsored activities on the calendar for 2000 which provide opportunities for members and friends to share their experience in all facets of safeguards and nuclear materials management. The following are only a few of the upcoming activities: the Vienna Chapter of INMM will hold its annual symposium May 15; the Obninsk, Russia Regional Chapter will sponsor a conference on "Materials Protection, Control and Accounting" May 22-26 and a tripartite conference on "Physical Inventory Taking Methodologies" in early October; and the Japan Chapter and INMM join with ESARDA to sponsor a workshop on modern technologies of nuclear materials management in November 2000. Please look carefully at the JNMM calendar of events, on Page 48 for other events of interest.

The coming year presents the INMM and its members with challenges that require all of us to be involved. I am pleased to announce that Obie Amacker has taken over leadership of the Fellows Committee, and has made a serious commitment to revitalizing the role of the committee. He has given the INMM many years of committed service, and we are fortunate to have his leadership and the strong experience base of our Fellows. We are facing rapid changes in our profession and in the world in general, and our Fellows have much value to offer to INMM's future.

There are many opportunities to become more involved with INMM. While not all of us have time to serve on committees, I challenge each one of you to encourage your professional colleagues to get involved with INMM in some way, as there is much to be done. We can accomplish so much more with a strong, dedicated membership.

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A Look Back on the Publishing History of INMM
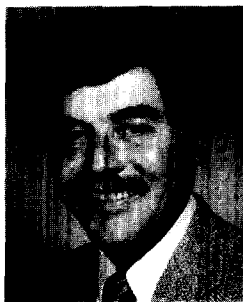
Vince DeVito, INMM secretary, sent me a summary of the INMM's publishing history. It's excerpted here:

Back in Time

The first publication of INMM was a newsletter published in August 1959 essentially one year after the founding of the Institute. The Newsletter was prepared by William B. Thomas, secretary of the INMM at Westinghouse Bettis Atomic Power Plant in Pittsburgh, Pennsylvania. In September 1961, Matthew N. Kuehn of Malinckrodt Chemical Works, St. Charles, Missouri, was named publisher and Ella Werner of the Atomic Energy Commission (AEC), Washington D.C., became editor of the Newsletter. Sheldon Kops, of AEC in Chicago, replaced Werner as editor in May of 1963 and the following year, Vince Donihee of General Electric, Richland, Washington, was appointed editor. In November 1965, Harley Toy of Battelle, Columbus, was named publisher. One year later, Raymond L. Jackson of Battelle, Columbus, assumed the role of publisher and editor remaining in that position until the final Newsletter was published in January 1972.

The first Journal of Nuclear Materials Management was published in April 1972 at Kansas State University in Manhattan, Kansas, with Dr. Curt Chezem as editor and Thomas Gerdis as managing editor. Chezem was head of the nuclear engineering department at the university and Gerdis was obtaining an advanced degree in communications. The initial issue was 16 pages and contained two technical papers. The annual subscription price was established at $15 for three issues and the annual meeting proceedings. Members, of course, received it as part of their membership benefits.

In July of 1974, Chezem left the university and Gerdis became editor and Willie Higginbotham became technical editor. Gerdis elected to leave INMM in summer 1980 to take a public affairs position with U. S. Ecology of Louisville, Kentucky. E. R. Johnson Associates was contracted to be the secretariat of the INMM (including publishing the Journal) until an association management firm could be selected. It should be noted, that before this time essentially all executive and administrative activities of the INMM were accomplished with volunteer help.

On October 1, 1981, the association management firm of Messervey and Co. took over the executive directorship of the INMM, including publishing of the Journal. The Sherwood Group has since succeeded Messervey and Co.

Thank you, Vince. To complete the picture, Willy Higginbotham remained technical editor until the fall of 1994. Darryl Smith took over the position for three years until fall 1997 when I assumed Darryl's place.

This issue of the Journal contains five technical articles of a mixed bag. The first is an extremely interesting article on monitoring the mass flow of highly enriched uranium. It's titled Fissile Mass Flow Monitor Implementation for Transparency in HEU Blenddown at the Ural Electrochemical Integrated Plant (UEIP) in Novourlsk, and is authored by Taner Uckan et. al. from Oak Ridge National Laboratory. The second paper, Sampling and Statistical Issues in Neptunium Safeguards, by Tom Burr and Bill Stanbro of Los Alamos National Laboratory, is a timely paper dealing with potential safeguards approach for neptunium, an issue the International Atomic Energy Agency recently has adopted. Stein Deron, David Donohue, Erin Kuhn, Kantika Sirisena, and Anatolii Tsarenko, in their paper, An Update of IAEA Analytical Capabilities for Safeguards: Goals, Results and Challenges, provide insights into the

efforts, capabilities, challenges, and activities associated with the IAEA's Safeguards Analytical Laboratory and Network of Analytical Laboratories needed to support the implementation of the Strengthened Safeguards System. The last paper, An Introduction to Focused Approach to Verification Under FMCT, by Victor Bragin and John Carlson of the Australian Safeguards and Non-Proliferation Office, discusses the Fissile Material Cut-off Treaty and possible approaches that might be considered for verification.

Other JNMM News

Our managing editor at INMM Headquarters, Renee McLean, decided to pursue a different path for a new career. We will miss Renee. It was certainly a delight to work with her, as she was an excellent team player as well as capable editor. We wish her the best in the future. She has been replaced by Patricia Sullivan (see story on Page 8). This change at INMM Headquarters has impacted slightly our schedules for publishing the issues of the JNMM. I trust you will understand and be patient with us.

Progress has been made with the peer review process with interesting results. We did a beta test and concluded that to be efficient and timely, we will need to use the electronic media. I also need to be clever and determine how to reconcile reviewers' comments on the same paper that range from "publish with minor revisions" to "reject". Stay tuned.

As always, I welcome any comments or suggestions you may have.

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Technical Division Reports

Physical Protection Technical Division

The Physical Protection Technical Division worked with INMM Headquarters to conduct a survey of active members to identify what types of workshops would be of interest in the area of physical protection. The survey sent to the membership asked the following questions:

- Which topic is of most interest to you or your colleagues?
- Within this topic, what would you like the focus to be?
- What would you like the format of the workshop to be?
- Who are the experts in the topic area you have chosen?
- Where would you like the workshop to be held?
- Would you be able to attend a workshop on this topic in the fall of 2000?
- What month in the fall is best for you?

The response was not overwhelming. The information we did receive from the survey will be used to plan future workshops.

The Physical Protection Technical Division conducted a workshop on Alarm Communications and Display Systems the week of February 21, 2000, focusing on functional requirements and technical and operational considerations for Alarm and Communications Display (AC&D) systems as they relate to power plants and other high-security facilities. The program provided participants the opportunity to present, discuss and exchange information on the design of AC&D systems. This workshop was intended to attract participants from the Department of Energy and its contractors, the Nuclear Regulatory Commission and its licensees, the DoD and high-security commercial industries.

The goal for the Division is to increase participation by the members in areas pertaining to physical protection. Despite our efforts, we may see a decline in participation this year at the annual meeting because of travel restrictions imposed by different branches of the federal government on their contractors.

Stephen Ortiz
Chairman, INMM Physical Protection
Division
Sandia National Labs
Albuquerque, New Mexico

International Safeguards Division

The INMM International Safeguards Division met at the Gakushi-kaikan in Tokyo, Japan, November 5, 1999. The meeting was attended by 19 members of the international safeguards community including participants from the IAEA, China, Japan, Pakistan and the United States.

Discussion centered on the impacts of the Integrated Safeguards System (ISS), including the new protocol, increased cooperation between individual states and the IAEA, and regional safeguards systems. As in past meetings of this division, it was recognized that many factors must be considered in the introduction of the variety of changes under the IAEA's new system. Clearly the meshing of the new and old systems and full implementation of the new system will be a challenge for all parties, requiring a great deal of cooperation.

The next meeting of the INMM International Safeguards Division will be held on Friday, May 12, from 9 a.m. to 12:30 p.m. at the Hotel Astron, Dresden, Germany, site of the 22nd ESARDA Annual Meeting. The suggested discussion topics for this meeting are the NPT Review Conference; the ESARDA meeting topics, including the IAEA's Integrated Safeguards System; and future R&D to support International Safeguards.

Note that the attendance in the ESARDA meeting is limited to 150 participants, first-come, first-served.

Planning continues for the Third Joint INMM/ESARDA Workshop on Science and Modern Technology for Safeguards, which will be held in Tokyo, Japan, during the week of November 13, 2000. This workshop will be co-hosted by the Japan Chapter of the INMM, the Korea Chapter of the INMM and the Australian Safeguards and Nonproliferation Office. It will be open to the memberships of the two organizations, as well as to others in the scientific and international safeguards community.

Cecil Sonnier
Chair, International Safeguards
Division
DOE Consultant
Jupiter Corp.
Albuquerque, New Mexico

Second Russian International Conference on Nuclear Material Protection, Control, and Accounting

The Second Russian International Conference on Nuclear Material Protection, Control, and Accounting is set for May 22-26 in Obninsk, Russia. The four-day event is cosponsored by Minatom, Russian Federation, the U.S. Department of Energy, the American Nuclear Society, the Russian Nuclear Society, and the Institute for Nuclear Materials Management.

To submit a paper or for more information on registering, contact the State Scientific Center of the Russian Federation-Institute for Physics and Power Engineering.
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Chapters

Japan

Guest speakers from the United States, the International Atomic Energy Agency, Pakistan, China, Korea, and Japan were among the 152 people who participated in the 20th Annual Meeting of the Japan Chapter of the INMM in Tokyo November 4-5, 1999. INMM President Deborah Dickman addressed the opening session and accepted a $1,000 donation to the INMM Education and Outreach Fund given to commemorate the chapter's 20th anniversary meeting. A plenary session on the first day featured a panel discussion, "Peaceful Use of Nuclear Energy and Nuclear Material Management in Asia." During the second day's technical sessions, 19 technical papers were presented.

The INMM Japan Chapter's International Safeguards Division met November 5 in conjunction with the chapter annual meeting. O.J. Heinonen, director, Division of Operations A, Department of Safeguards, IAEA, presented updated circumstances surrounding the Agency's integrated safeguards. Regional safeguards were also discussed.

Takeshi Osabe
Secretary, INMM Japan Chapter
Nuclear Material Control Center
Tokyo, Japan

Central Chapter

Efforts to reorganize and revitalize the Central Chapter of INMM are underway. In June 1999, the Central Chapter of INMM sponsored a technical session — "21st Center Safeguard Technology for Home and Abroad" — at the WaTTec Conference in Knoxville, Tennessee. Malinda Conger, of Lockheed Martin Energy Systems, chaired.

Chris A. Pickett
Chair, INMM Central Chapter
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Vienna Chapter

Last September, the Vienna Chapter selected officers for 1999-2000. The chapter executive committee members this year are:
- President: Jaime Vidaurre-Henry
- Vice President: Anita Nilsson
- Treasurer: Richard Hartzig
- Secretary: Diane Fischer
- Members-at-Large: Ira Goldman and Igor Tsvetkov
- Past President: Jill Cooley

Vice President Anita Nilsson delivered a presentation at the 20th anniversary of the Toyko Chapter of INMM, thus increasing and strengthening the relations among chapters. In this line, informal meetings were held with INMM representatives attending the 54th IAEA General Conference in September, and at a regional conference in Taejon, Republic of Korea. The Vienna Chapter chair met informally with the Korean Chapter chair and a member of the executive committee of the Japan Chapter. The chairs agreed to maintain informal communications to keep abreast of each other's chapter activities.

Jaime Vidaurre-Henry
President, INMM Vienna Chapter
IAEA
Vienna, Austria

Northeast Region Chapter

During the last quarter of 1999, the Northeast Region Chapter of INMM held a general membership dinner meeting in Middle Island, New York, near the Brookhaven National Laboratory (BNL). Earlier that day, chapter members toured BNL's new Relativistic Heavy Ion Collider. The speaker at the dinner was Nikolai Khlebnikov, director of the Division of Safeguards Technical Services of the IAEA. He addressed the objectives and process used by IAES to develop equipment for safeguards imple-

mentation. He noted that the primary objective is to be responsive to safeguards operations needs, and the process involves a mix of political considerations, sponsors for the projects, good ideas, and funding.

Kenneth Sanders
President, Northeast Region Chapter
U.S. Department of Energy
Washington, D.C.

Korea Chapter

INMM Korea Chapter held its third annual meeting on October 13, 1999, with 70 participants at the Korea Atomic Energy Research Institute in Taejon, Korea. The chapter meeting introduced newly elected Secretary Jang Soo Shin, and members-at-large Kun-Jai Lee and Young-Myung Choi. An international workshop on remote monitoring, which was co-hosted by INMM-KC, TCNC/KAERI and SNL, U.S.A. followed October 13-17. Lectures on the RMS Experiences and Perspective were presented by J. Whichello and W. Alston from IAEA, T. Niina from NMCC, Japan, and by some lecturers from Korean nuclear facilities. The RM technologies were covered by C. Harmon, D. Drayer, and R. Martinez from SNL. Many ideas on the RM benefits for the facility and the RM implication in national inspection, were discussed. Facility tours to the Yonggwang LWR site and the Wolsong HWR (CANDU) site followed the workshop.

Jang Soo Shin
Secretary, INMM Korea Chapter
TCNC/KAERI
Taejon, Korea

Spent Fuel Management Seminar XVII Decidedly Upbeat

The tone of the INMM's 17th Annual Spent Fuel Management Seminar was decidedly upbeat. Five topical sessions were addressed:

- Spent Fuel Management Programs and Policies
- Spent Fuel Storage Technologies
- Storage Projects and Regulatory Status
- Spent Fuel Transportation
- Status of Proposed Yucca Mountain Repository Disposal Project

The approximately 160 attendees included representatives of utilities, vendors, government and international agencies, regulators, national laboratories, consultants and the press. The representation included the United States, Canada, Japan, Korea, Spain, France, the United Kingdom, Germany, Austria, Hungary, and the Czech Republic.

The two private, independent spent fuel storage projects are progressing well. With successful resolution of prior QA difficulties, on-site storage has returned to good health; and the Yucca Mountain repository project is progressing on schedule.

The status of each of the two private spent fuel storage projects was described. The Private Fuel Storage project, located in Tooele County, Utah, on the Skull Valley Goshute Indian Reservation, has received the NRC's draft safety evaluation report based on its earlier license application and is proceeding to address remaining licensing issues and intervenor contentions. The current expectation is for receipt of the license in October 2001, in which case initial fuel storage could commence in 2003. The Owl Creek Project is located on privately-owned, rail-serviced land in Fremont County, Wyoming, a county in which there has been substantial uranium mining and hence a general understanding and accommodation of nuclear-related activities. This project intends to use DOE's existing Conceptual Interim Storage Facility design and its topical report, which has already been submitted and reviewed by NRC, and on which NRC's Safety Evaluation Report based on a generic site envelope is expected in early 2000. The Owl Creek application to NRC will thus consist of these existing documents plus its site-specific environmental report. The latter document will presumably demonstrate that the Owl Creek project and site fit within the environmental envelope that is expected to have received generic acceptance by NRC. The Owl Creek schedule anticipates initial fuel storage in 2004.

The current vendors of dry storage systems made presentations on the current status and future plans for their designs. One measure of the general health of the dry spent fuel storage business was provided by NRC, which described the growing pressures on its licensing activity that are imposed by the existing 14 dry storage licensees as well as the near-term planning for about 20 in-process or imminent license applications. Another encouraging trend is actual licensing or a stated intent to license most of the newer storage systems to be transportable after storage, without returning to the spent fuel pool. The other signs of a healthy dry storage business are the competition among vendors, including competition to increase the burn up and thermal capability of designs, and the growing numbers of orders. Some presentations were made on non-U.S. storage projects, including those that have adapted U.S. technology to the storage of spent fuel from Russian-designed reactors. A description of the project that will use NUHOMS technology to store spent fuel from the Chernobyl reactors was particularly interesting. This project is expected to be placed in operation within two years — including regulatory licensing.

The downside of the United States' growing demand for on-site dry storage is that little of this would have been needed if the U.S. Congress and the Administration had been able to resolve their differences on the matter so that the DOE could have begun accepting spent fuel on the statutory date of January 31, 1998. Several presentations summarized the history and current status of the utilities-vs-DOE legal proceedings and court rulings that have resulted from this situation.

The ultimate solution to spent fuel management is, of course, safe geologic disposal, and the seminar attendees were given a complete status report on DOE's Yucca Mountain repository project. The new director of DOE's Office of Civilian Radioactive Waste Management provided a project summary.

Additional presentations by staff members from DOE and DOE's management and operations contractor covered the development of the Site Recommendation Report and the Final Environmental Impact Statement; the waste characteristics and the waste package design; and the methodology for assessing the long-term isolation performance of the combined engineered and natural barriers against removal of radioactivity. An important conclusion is that DOE is on schedule to deliver the Site Recommendation Report in late-2000. This is expected to support the subsequent public review process and the development of the site recommendation by the secretary of energy to the president in mid-2001.

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New Managing Editor Joins INMM Staff
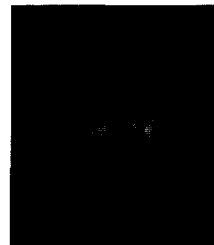
Patricia Sullivan has joined the staff of the Institute of Nuclear Materials Management as communications manager. Her duties include editing the *Journal of Nuclear Materials Management.*

Sullivan comes to INMM from the National PTA where she was associate editor of *Our Children* magazine, writing and editing news features on children's health, education and parenting issues, PTA-specific stories on management issues and copy editing all stories, departments and features. Sullivan is comfortable working with and for boards of directors and looks forward to the challenges of a new association's publishing goals and needs.

She also has written news, business and feature stories for the *Chicago Tribune, American Medical News* and the *Baton Rouge Business Report.*

Sullivan can be reached at INMM headquarters at 847/480-9573 or by e-mail at psullivan@inmm.org.

Agreement Expands Two Waste Disposal Firms Services

Two U.S. waste disposal companies have signed a memorandum of understanding that will allow both companies to expand the services they offer to the mixed waste (radioactive and hazardous) market. Commodore Applied Technologies, Inc., and Envirocare of Utah, Inc., a company providing a full range of low-level radioactive and mixed waste management and disposal services, announced the agreement in December 1999. Commodore brings its patented SET™ process and its nationwide EPA PCB permit, while Envirocare brings its Clive, Utah, treatment and disposal facility and more than 20 years of experience in the mixed waste market.

Envirocare began in the mid-'70s, when the U.S. Department of Energy and the state of Utah began the cleanup of an abandoned uranium mill site. Today, Envirocare's Clive facility offers a permanent disposal solution for low level radioactive and mixed wastes, as well as mixed waste treatment. Envirocare plays a key role in major federal and private cleanup efforts nationwide. Its customers include federal agencies such as the DOE, the U.S. Environmental Protection Agency, and the Defense Department, and large private firms including several Fortune 500 companies. The SET™ process will bring an added dimension to the site in the reduction of certain mixed waste to radioactive waste by converting the Resource Conservation and Recovery Act organic content of the waste into inert hydrocarbons.

International Flavor, Cooperation Focus of Decommissioning Symposium

The 4th U.S. Department of Energy International Decommissioning Symposium, also called IDS 2000, will be held June 12-16 at the Knoxville Convention Center and neighboring World's Fair Park in Knoxville,

Tennessee.

The four-day symposium will provide panel discussions, lectures, training sessions, and hands-on demonstrations on topics such as:

- Decommissioning of U.S. and international nuclear facilities;
- Treatment and disposal options for radioactive, chemical, low-level, and transuranic wastes;
- Technologies for metal and concrete decontamination; and,
- Reuse and reindustrialization of facilities, material, and equipment.

In cooperation with the International Atomic Energy Agency, the DOE is emphasizing the international aspects of the symposium. IDS 2000 is expected to draw more than 1,000 attendees from at least 20 countries. IDS 2000 will also focus on transforming former nuclear facilities into commercial industrial facilities.

Market needs including decontamination, decommissioning, and recycle and reuse will be complemented with demonstrations of available technology. Key components of the symposium include:

- Technical program emphasizing decontamination, decommissioning, and reindustrialization;
- Poster sessions and technical presentations;
- Business/technology exchange forums;
- Vendor exhibits and a trade show;
- Technology demonstrations and simulations.

U.S. to Dispose of up to 50 Metric Tons of Plutonium in Hybrid Approach

The U.S. Department of Energy announced in January that it will securely dispose of up to 50 metric tons of surplus plutonium from the United States in a hybrid approach and it will construct and operate three new facilities at its Savannah River Site (SRS) in South

Carolina. These actions are needed to ensure that surplus plutonium is never again used in nuclear weapons.

The hybrid approach allows for the immobilization of approximately 17 metric tons of surplus plutonium and the use of up to 33 metric tons as mixed oxide fuel. The new plutonium disposition facilities will provide pit disassembly, plutonium conversion, immobilization, and MOX fuel fabrication services.

These facilities will be located in F Area at the SRS site with construction scheduled to peak in 2003. Savannah River was selected because it has extensive experience with plutonium processing and the plutonium disposition facilities would complement the existing missions of SRS and take advantage of the existing infrastructure.

The pit disassembly and conversion facility will disassemble nuclear weapons pits and convert the resulting plutonium metal to a declassified oxide form suitable for disposition, using either immobilization or MOX fuel fabrication. The immobilization facility will use ceramic can-in-canister technology.

The MOX fuel fabrication facility will produce MOX fuel for irradiation in three existing domestic commercial reactors: the Catawba Nuclear Station near York, South Carolina; the McGuire Station near Huntersville, North Carolina; and the North Anna Power Station near Mineral, Virginia. Ultimate disposal of both the immobilized plutonium and the MOX fuel (as spent fuel) would take place in a geologic repository pursuant to the Nuclear Waste Policy Act. Because adequate reactor capacity is available in the United States, the DOE is no longer actively pursuing the option of using Canadian Deuterium Uranium reactors for the disposition of U.S. surplus plutonium. To assist the United States, Russia, and Canada in considering this option of the disposition of Russian surplus plutonium, the three

countries are conducting an experiment which will involve irradiating U.S. and Russian MOX fuel pins in a Canadian research reactor. This effort involves making a one-time shipment of a small quantity of fuel from the U.S. to Canada.

The selection of Savannah River to provide immobilization services for surplus plutonium clears the way for shipment of the surplus plutonium at Rocky Flats Environmental Technology Site near Denver to the Savannah River Site.

DOE Moves Forward with Nuclear Security Administration Plans

The U.S. Department of Energy took additional steps toward the formation of a new semi-autonomous agency, the National Nuclear Security Administration in January when Energy Secretary Bill Richardson met with the panel charged with conducting a search for a qualified undersecretary for nuclear security and the department's implementation plan for the NNSA was delivered to Congress. The new agency is established as of March 1, 2000.

The undersecretary for nuclear security will also serve as the administrator of NNSA under the statute that creates the new agency. The search committee is chaired by former Deputy Secretary of Energy Charles Curtis and its members are former Energy Secretary Admiral James D. Watkins, Admiral Henry G. Chiles, and Andrew Athy, chairman of the Secretary of Energy Advisory Board. Its work will aid Richardson in identifying a qualified candidate who will then be nominated to the post by President Clinton.

The NNSA will include the current offices of Defense Programs, Nonproliferation and National Security, and Naval Reactors. A deputy administrator will head each of these offices. In addition, the current Office of Fissile Materials Disposition will be incorporated into the new Office of Defense Nuclear Nonproliferation and will be

headed by the assistant deputy administrator, who will also serve as the special secretarial negotiator for plutonium disposition. The Albuquerque and Nevada Field Operations Offices will also be transferred to the NNSA. Employees of these offices, with the exception of those who are accountable to non-NNSA program offices, will become employees of the NNSA. At other department field operations offices with responsibilities for NNSA activities, employees who directly oversee those activities will become employees of NNSA.

The plan also provides support functions within the NNSA encompassing legal, security, counterintelligence, legislative affairs, public affairs, intergovernmental liaison, budget and procurement. Because of the short time allowed for implementation of the new agency and the importance of program continuity, among other factors, the plan calls for some DOE officers to serve concurrently in a few of these support function positions until January 2001. The field managers at selected field operations offices will also serve concurrently in dual positions during that time.

DOE Issues Second Penalty Issued to Rocky Flats Operator

The U.S. Department of Energy has issued a second civil penalty to Kaiser-Hill Co., LLC, operator of the Department's Rocky Flats Environmental Technology site in Colorado, for failure to adequately implement aggressive corrective actions it had committed to last year.

This fine is $55,000 and follows an earlier fine of $82,500 issued in August 1999 to Kaiser-Hill for problems with its procurement process in 1997 and 1998. In that case, DOE found that the contractor purchased 69 defective nuclear waste containers without adequately evaluating the quality controls used by the supplier, citing several failures to properly follow administrative controls and procedures.

DOE Establishes New Central Internet Database

A Central Internet Database which provides easier access to information on radioactive waste, hazardous materials, and U.S. Department of Energy facilities was established in January. The site can be accessed at http://cid.em.doe.gov.

The new database provides several ways to obtain information on the department, including ready-to-read options and more sophisticated search functions. Specifically, the CID provides detailed information on department inventories and management activities for low-level, transuranic and high level waste; contaminated media; spent nuclear fuel; facilities; non-radioactive hazardous waste; toxic chemicals; materials in inventory; and buried transuranic waste.

UC to Continue in Los Alamos Management Role

Energy Secretary Bill Richardson has approved the continuation of the Department of Energy's contract with the University of California for management of the Department's Los Alamos National Laboratory in New Mexico.

While the current contract is not scheduled to expire until September 30, 2002, the department could have exercised an "off-ramp" provision of the contract which would terminate the contract before its scheduled expiration. Exercise of the off-ramp was dependent on improved UC performance in specified areas of concern including performance of environmental management and safety and health. Richardson made the decision to continue with the contract after receiving a report from the Acting Assistant Secretary for Defense Programs General Thomas F. Gioconda who indicated that UC's performance had improved.

Fissile Mass Flow Monitor Implementation for Transparency in HEU Blenddown at the Ural Electrochemical Integrated Plant (UEIP) in Novouralsk

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■

Abstract

The Oak Ridge National Laboratory (ORNL) Fissile Mass Flow Monitor (FMFM) was deployed at the Ural Electrochemical Integrated Plant (UEIP) highly enriched uranium (HEU) blending facility in January and February 1999 at Novouralsk in Russia for the DOE HEU Transparency Program. The FMFM provides unattended monitoring of the fissile mass flow of the uranium hexafluoride (UF_6) gas in the process lines of HEU, the low enriched uranium (LEU) blend stock, and the product LEU (P-LEU) of the blending tee non-intrusively. To do this, uranium-235 (U-235) fissions are induced in the UF_6 by a thermalized and modulated californium-252 (Cf-252) neutron source placed on each process line. A set of detectors, located downstream of source, measure delayed gamma rays emitted by the resulting fission fragments. The observed delay in the time correlated measurement between the source and the detector signal provides the velocity of UF_6 and its amplitude is related to the U-235 content in UF_6. An on-line computer controls the source modulator, processes the collected detector data, and displays the results. The UEIP Main and the Reserve process lines were implemented with minor modifications. The FMFM monitors the HEU blending operation by measuring UF_6 flows in the process blending lines, and the traceability of the HEU flow from the blend point to the P-LEU. The detail operational characteristics of the FMFM software (FM2) and the measurement methodology used are presented.

Introduction

The Fissile Mass Flow Monitor, which was installed to the UEIP process lines in January and February 1999, determines the fissile mass flow rate by relying on two independent meas-urements: (1) the time required for the fission fragment to travel along a given length of pipe, which is inversely proportional to the fissile material flow velocity, and (2) an amplitude measurement, which is proportional to the fissile concentration (e.g., grams of U-235 per length of pipe).
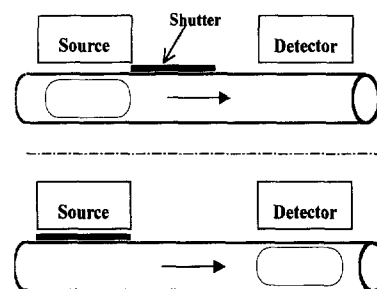
(a) When the shutter opens, fissions are induced and the fissile stream carries the fragments downstream

(b) Gamma rays from fission fragments are detected with a time delay. Velocity $\propto$ 1/delay Fissile concentrate $\propto$ amplitude

Figure 1. Fissile mass flow rate measurement concept

This paper describes the methodology used to interpret the data measured by the FMFM, the models used to simulate the transport of fission fragments from the source location to the detectors, and the implementation of these algorithms in the FMFM software FM2. The basic FMFM measurement concept is illustrated in Figure 1 and can be described as follows: (1) Fast neutrons from a Cf-252 source are moderated by a polyethylene block. (2) A neutron-absorbing shutter modulates the source strength, superimposing a time-dependent signature is in the fissile stream. (3) The moderated neutrons induce fissions

inside the process stream. (4) The resulting fission fragments are slowed down by the gas, and some are carried by the stream. (5) A downstream sensor detects delayed gamma rays emitted by the fission fragments. (6) A time-delay measurement is performed by detecting the signature caused by the shutter. (7) The fissile concentration is obtained from the measured detector response and a calculated calibration that is confirmed by measurements. (8) The fissile mass flow rate is determined by multiplying the average fissile velocity and the fissile concentration of step (7). This measurement methodology is insensitive to buildup on the pipe walls, and it can be applied to any flow stream capable of producing particles that emit delayed radiation that can be detected downstream.

In addition to measuring fissile mass flow, the FMFM traces the HEU through the blending tee by detecting in the P-LEU line detectors delayed gamma rays emitted by fission products generated in the HEU line. This traceability gives U.S. Monitors significant confidence that the HEU is indeed being blended into P-LEU.

Flow Monitor Algorithm

The FM2 software measures the time-dependent profile at the detector location following a shutter-induced pulse and compares it with all of the model-predicted profiles at different flow velocities. The time average flow velocity is the one that results in a minimum residual error, $\varepsilon(\mathbf{u})$,

$$\varepsilon\left(u\right)= \int_0^T \left[N_\gamma(t)- C\, N_{model}\left(t,u\right)\right]^2 dt \quad,$$

where T is the time period of the shutter motion, and C is the amplitude parameter, which is proportional to the detector-response. The detector response, N_γ, is proportional to the number of fissions induced, N_{fis}, which is proportional to the concentration of U-235 in the pipe. Thus the amplitude parameter C is directly proportional to the fissile density, and the product of the amplitude multiplied by the flow velocity is proportional to the fissile mass flow rate, ω:

$$\omega \frac{-g}{s} \sqrt{} \quad = \quad u \frac{-m}{s}\sqrt{\leftarrow}C \frac{-g}{m}\sqrt{\leftarrow}N_{model}\left(t,u\right).$$

A calibration factor is required to scale the model profiles, $N_{model}(t,u)$, so that the units of the amplitude parameter C are mass of fissile material per unit length (e.g., grams per meter). The calibration factor is calculated using a Monte Carlo computer code that simulates the flow-meter geometry and the detector efficiency (including the energy discrimination). The calibration factor is also confirmed by off-line benchmark tests. The basic steps performed by FM2 to evaluate the fissile mass flow rate are as follows.

A. Data Collection and Averaging

A new block of raw data is collected from the detector network in blocks of 60 seconds. These data consist of the detector counts per seconds measured as a function of time while the shutter is opening and closing. These new data are averaged with the old data using a running-average method. Two time constants are used for this running average. This results in a short- and a long-time-constant average block of data. Each of these average blocks is 60 seconds long but contains the average data over several hours. In the following steps, M(t) represents the average block. The formula used to compute M(t) is:

$$M(t) = \frac{M_0(t) \leftarrow (\tau - 1) + N(t)}{\tau},$$

where $N(t)$ represents each new 60-second block of data, $M_o(t)$ represents the old value of $M(t)$, and (represents the time constant (expressed in minutes).

B. Flow Velocity Determination

To determine the mass flow, the model described in Section A is fitted to the average block, $M(t)$, using a weight function, $W(t)$. To fit this model, FM2 first obtains the uncorrelated background, $bckgU$, correlated background, $bckgC$, and a fissile concentration, $C(u)$, that minimizes the residual error, $\varepsilon(u)$, for each trial velocity u,

$$\varepsilon(u)= \int_0^{T=20s} dt\, W(t)\, [M(t)-(bckgU+bckgC(t)+C(u)\, N_{model}(t,u))]^2.$$

Then, FM2 calculates $\varepsilon(u)$ for each trial velocity, and it finally selects the velocity that results in minimum error. This fitting process is performed every minute (after sampling each new 60-second block of data) using the short- and long-time constant averages. This process results in the best-estimate gas velocity for the data. The weight function, $W(t)$, in the above equation is set to 1 at all times when the shutter is not moving. During shutter motion, $W(t)$ is set to 0.

C. Mass Flow Determination

The fissile concentration is estimated for the optimal velocity using the above equation. Note that in the above equation, $N_{model}(t,u)$ is defined in the FM2 profile database, which is scaled so that the fissile concentration, $C(u)$, units are in grams of U-235 per meter of pipe. Finally, the fissile mass flow, $\omega(t)$, is determined by multiplying the gas velocity times the fissile concentration,

$$\omega(t) = u \leftrightarrow C(u).$$

A fissile gas velocity, a fissile concentration, and a mass flow rate are determined every 60 seconds, when a new block of data becomes available. These are based on the short- and long-time constant running averages.

D. Statistical Test for Flow of Fissile Material

Once every 60 seconds, a statistical test is performed on the average data to determine a confidence level of the algorithm fit described in the above sections. For this purpose, a statistical F-test is performed between the residual error calculated in Section B for the optimal velocity, and the residual error obtained by setting $C(u)$ equal to zero (i.e., forcing a mass concentration to zero). The result of this F-test is a confidence level on nonzero flow of U-235 and represents the quality of the flow measurement. As with the velocity, concentration, and flow measurements, FM2 computes a flow confidence using the short- and the long-time constants.
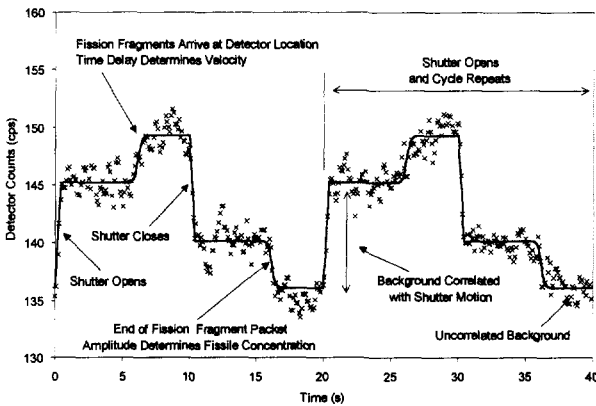
Figure 2. Illustration of the FMFM algorithm performance

Figure 2 shows an example application of the FM2 algorithm to data for a case of turbulent flow with a gas velocity of ~0.5 m/s and a source-detector separation of ~3-m. The crosses in this figure represent the average data [i.e., $M(t)$] and the solid line represents the optimal model selected by the FM2 algorithm as described in Section B. The uncorrelated and correlated backgrounds are evident in this figure. The fission-fragment-induced pulse is also evident at time 6 seconds, which is the expected time delay for a velocity of ~0.5 m/s and a distance of ~3-m. The amplitude of this pulse is proportional to the fissile concentration in the pipe.

Fissile Tracing Algorithm

The fission fragments that result from the Cf-252 induced fissions are relatively long-lived; thus their decay gamma rays can be detected at long distances from the source. This technique is used by FM2 to monitor flow continuity through a possibly complex series of pipes and volumes such as pumps.

The time constant for the "tagging signal" must be optimized based on the source-detector time delay and the number of mixing volumes. For a typical configuration, FM2 cycles the HEU-leg shutter open and closed every 5 to 10 seconds for a 10-minute period and then is closed for the next 10-minute period. This results in a 20-minute cycle of buildup and decay of fission products that allows for continuity monitoring by comparing the difference in the P-LEU detector counts with and without induced fissions. This concept is illustrated in Figure 3.

Disabling the HEU-leg shutter periodically (every other 10 minutes) affects the correlated background level at the P-LEU leg, because the P-LEU detectors may be located close to the HEU shutter and are affected by its motion. For this reason, FMFM traceability only uses the data when all shutters are closed. The FM2 tracing algorithm averages the shutter-closed data over the complete 60-second block. The data are then averaged into a tracing data block with the appropriate time delay so that the data from minute 1 are averaged with the data from minutes 21, 41, and so on. The data for minute 2 are averaged with the data from minutes 22, 42, and so on. The tracing data block thus contains 20 data points, one per minute, and it is synchronized with the cycle time of the HEU-leg shutter.
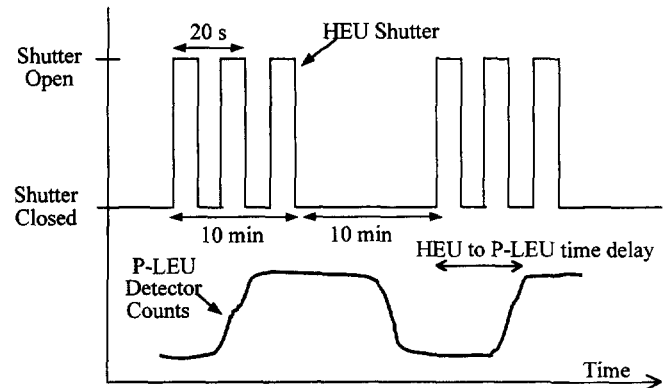
Figure 3. Illustration of shutter motion pattern to generate the low-frequency modulation required for HEU to P-LEU tracing
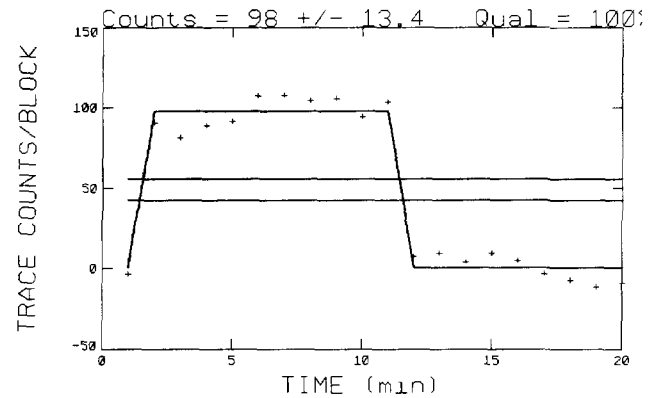
Figure 4. Sample tracing block showing high confidence of traceability

Figure 4 shows an example of a converged tracing block, which was measured using the procedure described above. Using the averaged tracing data block, two statistical tests are performed on the data. The first test compares the variance of the 20 data points with the theoretical variance if the measured data were perfectly random, which would result in a variance equal to the inverse of the number of counts averaged. The results of these tests define a confidence level that the data has "structure." A

second statistical test is performed to detect the 10-minute on, 10-minute off signature of the HEU-leg shutter in the data. For this test, a square wave with a variable time delay is fitted to the tracing block. The residual noise variance after removing the square wave fit is compared to the original variance using an F-test confidence level. The result of this last test defines the confidence level of the fitted tracing model.

After the above calculations are performed, FM2 reports two numbers:

1. The product of the two statistical confidence levels (the structure and the fit confidence levels).
2. The tracing counts per block, which correspond to the amplitude of the step in Figure 4, along with its calculated standard deviation.

Models and Correlations

To predict the detector response downstream of the source, it is necessary to model (a) the percentage of delayed gamma ray fission products that remain in the gas following an induced fission, (b) the flow of fissile material and fission products down the pipe, and (c) the decay of the fission products. Models and the resulting correlations are described in the following sections.

A. Fission Fragment Decay Model

The delayed emission data have been obtained by fitting a five-group model to measured data using the actual FMFM hardware. This model includes 300-keV energy-discrimination filters that are accounted for in the overall detector efficiency. The parameters of the five-group model are summarized in Table 1, and a sample measurement is shown in Figure 5. The parameters in Table 1 correspond to a best fit to the decay gamma-ray data following a fission event, so that

$$n_\gamma(\tau) = \sum_1^5 \alpha_i e^{-\lambda_i \tau},$$

where $n_\gamma(\tau)$ represents the average number of photons per second following a fission event, λ_i is the group yield constant, which is related to the group precursor fraction, β_i, as $\alpha_i = \lambda i \_ \beta_i$.

Table 1. Delayed Gamma-Ray Data

Group #	(αi (γ/s per fission)	λi (s^{-1})
1	0.35	0.4
2	0.06	0.04
3	0.015	0.008
4	0.0015	0.0008
5	0.0002	0.00005

Figure 5 shows the results of applying our delayed gamma-ray emission model to measured data obtained by irradiating a U-235 fission chamber for 60 and 600 seconds and measuring the decay gamma rays with the actual flow-monitor hardware. As seen in Figure 5, the delayed gamma-ray emis-
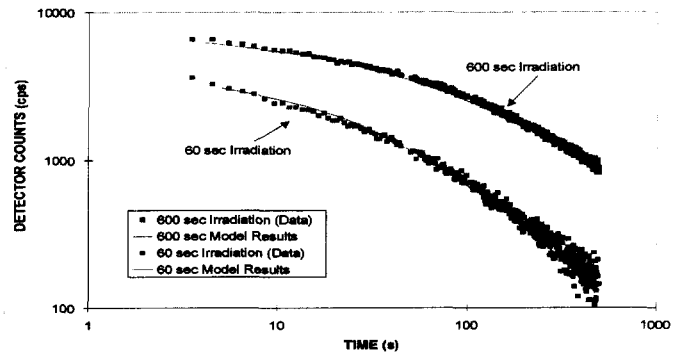
Figure 5. Comparison between ORNL irradiation measurements and decay model predictions

sion model predicts the measured data accurately up to 500 seconds following the fission event. This decay model also benchmarks well against the impulse-response data published in the literature.

B. Fission Fragment Range in Low-Pressure UF$_6$ Gas

Fission fragment ranges can be very large in low-density materials. For this reason, a methodology was developed to estimate the ranges and distribution of fission fragments with the goal of determining the fraction of fission products that remain entrapped in the UF$_6$ gas.

The basic range data are derived from the measurements documented in the *Nuclear Data Tables*[3]. These ranges are integrated path lengths of heavy charged particles traversing various media. Based on these data, the fission fragment ranges in UF$_6$ were computed as functions of gas pressure and fission fragment energy. The distribution of fragment energies can be approximated by two Gaussian distributions (one for the light fragments and one for the heavy fragments[4]). The parameters of

$$R_l + \frac{4174}{p}; \quad R_h = \frac{3154}{p}; \quad \sigma_l = \frac{183}{p}; \quad \sigma_h = \frac{236}{p},$$

this distribution are as follows:

where R_l and R_h are the average light and heavy fragments range expressed in millimeters, and σ_l and σ_h are their standard deviations, and the pressure p is expressed in Torr.

The above values represent the nominal ranges and their standard deviations. The range, however, represents the integrated path length, not the radial distance from the point of fission. To estimate the effect of nuclear scattering, the tabulated values were compared with measurements by Niday[5]. A comparison of the tabulated values with Niday's measurements indicates that the tabulated values (and the average ranges given in the Gaussian distributions) should be reduced by approximately 15%. Straggling, the statistical fluctuation in the ranges of charged particles traveling in a material, is accounted for by applying a 10% uncertainty to the ranges and to the standard

deviations[6].

To determine the number of fission-fragment absorptions in the pipe wall, a Monte Carlo-type calculation with special tallies was performed. For this run, a homogeneous source of gamma rays was placed inside an empty pipe of diameter, D; then the gamma-ray currents were tallied at different radii as a function of the age of the photon. These ages were directly proportional to the range that the photon had traveled before reaching the inner wall of the pipe and allowed for the development of a correlation for the fraction of fission fragments that were absorbed by the pipe as function of fragment range (i.e., UF_6 pressure) and pipe inner diameter. The fraction, ε_s, of fragments that remained in the UF_6 flow and that contributed to delayed gammas at the detector location was computed from this correlation and the probability distribution function for fragment ranges. Based on these data, two correlations for the source effective-

$$\varepsilon_s = \frac{1}{1 + e^{-0.025(pD-65)}} - 0.271 \; for \; (pD < 200 \; psi \; mm), \text{and}$$

$$\varepsilon_s = 0.943 - e^{-0.006(pD-10.1)} \quad for \; (pD > 20 \; psi \; mm),$$

ness have been developed:

where D is the pipe diameter in millimeters, and p is the gas pressure in pounds per square inch. The first correlation is more accurate but it can only be used for low pressures. The second correlation, while not as accurate, can be used at high pressures.

C. Fission Fragment Transport and Decay Model

The basic equation that describes the flow and decay of delayed gamma-ray fission fragments is the combined convection and

$$\frac{dc_i(r,z,t)}{dt} + u(r)\frac{dc_i(r,z,t)}{dz} = \beta_i N_{fis} S(r,z,t) - \lambda_i c_i(r,z,t),$$

decay equation:

where $c_i(r, z, t)$ is the concentration of group-i fission fragments at time t and location (r, z), $u(r)$ is the gas velocity at radial position r, β_i is the fraction of group-i precursors generated per fission, λi is the decay constant, N_{fis} is the number of induced fissions, $S(r, z, t)$ is the normalized shutter efficiency, which combines the source field of view and the shutter motion as function of time.

The total concentration of delayed gamma-ray fission fragments is the sum over all delayed groups. The number of

$$N_\gamma(t) = \frac{1}{\pi R^2} \int_0^R dr \; 2\pi r \int_{-x}^x dz \; \varepsilon_d D(r,z) \sum_1 \lambda_i c_i(r,z,t)$$

gamma rays per second counted at the detector, $N_\gamma(t)$, is determined by

where $D(r, z)$ is the normalized detector field of view of all the detectors together, (d is the overall detector efficiency.

The above model for fission fragment transport and decay has been implemented in a computer code. This code solves the above equations numerically and computes the precursor concentration at a number of axial and radial nodes inside the fissile stream.

The computer code solves the time and space equations converted to discrete form and determines the detector response for a particular flow regime, velocity, and shutter pattern. Figure 6 and Figure 7 show the calculated response profiles for turbulent and laminar flow, respectively. Once calculated, these profiles
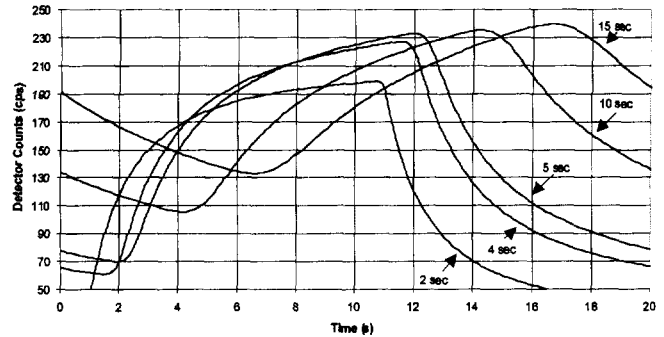
Figure 6. Calculated profile database for laminar flow and 1-m source-detector separation

are stored in the FM2 profile database and are used to determine the mass flow rate from the detector count measurements.

These profiles are calculated as functions of the time delay between the source and the detector center lines. For laminar flow, the time delay is defined as the distance divided by the average velocity. For both figures, the assumed detector efficiency is 22%, which includes an energy discrimination filter for gamma rays with less than 300-keV.

For the laminar flow case (Figure 6), the calculations assume a source-detector distance of 1-m and a fissile concentration of 7 g/m of the four-inch ID pipe (equivalent to ~1 psia pressure and 90% enrichment). For the turbulent flow case (Figure 7), the calculations assume a source-detector distance of 3-m and a fissile concentration of 0.1 g/m of 4-inch pipe (equiv-
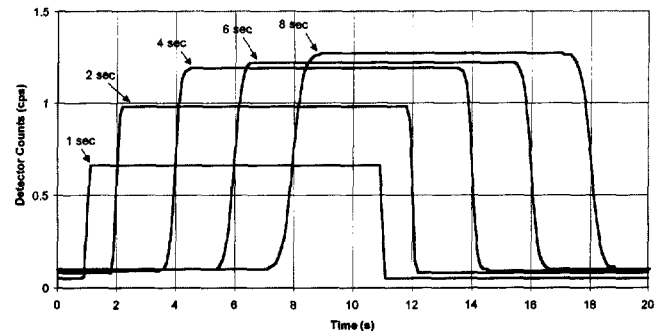
Figure 7. Calculated profile database for turbulent flow and 3-m source-detector separation

alent to ~1 psia pressure and 1.5% enrichment). Both cases assume that equilibrium conditions have been reached in the pipe and that the shutter efficiency is 95%. For these calculations, the shutter is opened and closed in 20-second cycles (10 seconds open and 10 seconds closed).

D. Correlated Background Model

Motion of the shutter inside the source modulator results in a change of background counts at the detector location. This background change is due to a change in the number of capture gamma rays emitted by the moderator and by a change in the number of capture gamma rays emitted in the pipe. This background change is correlated with the shutter motion and therefore is not reduced by increasing the measurement time. The correlated background magnitude is very substantial; it can be as high as 25% of the total background if the detector is located close to the source modulator. Thus, the correlated background must be taken into account in the model. Because of the possibly large amplitude of the correlated background, shutter synchronization is mandatory to allow for its removal during the data analysis process.
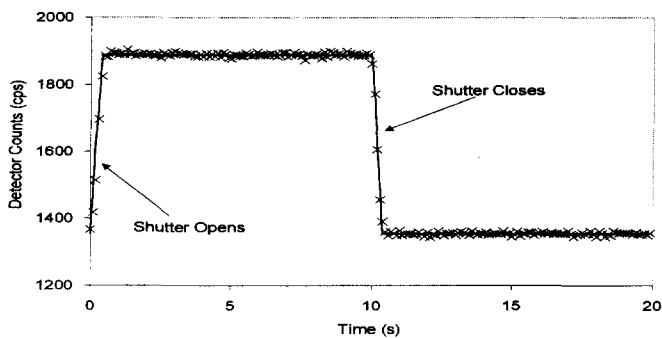
Figure 8. Correlated background (no UF₆ flow)

An example of a measured correlated background (with no gas flow) is shown in Figure 8. This figure corresponds to a detector assembly located 1-m downstream of a source modulator.

As shown in Figure 8, the FMFM correlated background model is a constant background between the times of 0 and 10

seconds, and a constant background of different magnitude between the times of 10 and 20 seconds. The two constant-background sections are connected by a linear interpolation at times 0 and 10, which represent the shutter motion. The duration of shutter motion is a field-selectable parameter that can be adjusted if different shutter speeds are used. For the nominal shutter speed, the shutter motion time is 450 ms, and the shutter settling time following this motion is typically 150 ms.

Conclusion

The FMFM was successfully implemented on the UEIP Main and the Reserve process lines at UEIP. The independent measurements of the FMFM measures the UF6 mass flow rate continuously in the process blending lines, and monitors the traceability of the HEU flow from the blend point to the P-LEU. These measurements give U.S. Monitors significant confidence that the HEU material is indeed being blended into a lower assay P-LEU material.

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Sampling and Statistical Issues in Neptunium Safeguards

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Abstract

The IAEA plans to initiate international controls on ^{237}Np. These controls will be based on Flow Sheet Verification at facilities capable of separating quantities of ^{237}Np. This paper describes a verification approach using random sampling based of the statistical properties of ^{237}Np flows to provide the necessary assurances while minimizing the sampling and analysis burden.

1. Introduction

Special fissionable material is defined in the IAEA Statute[1] to include ^{239}Pu, ^{233}U, and uranium enriched in ^{233}U or ^{235}U. These materials have been the subject of international safeguards since the 1960s. Recently, the IAEA has also recognized the possible proliferation potential of ^{237}Np. While not raising ^{237}Np to the status of a special fissionable material, the IAEA has decided to initiate inspection procedures at facilities that have the potential to separate this material. These measures, called Flow Sheet Verification (FSV), are designed only to ensure that undeclared separation does not occur. This novel approach is considered to be sufficient until and if large-scale separation begins to occur. At that time the IAEA would consider a possible change in status for ^{237}Np. This paper examines the statistical properties of ^{237}Np flows in reprocessing plants and suggests how those properties may be exploited to reduce the inspection burden on both the IAEA and the facility operator while not compromising Agency information requirements.

In the PUREX process commonly employed in commercial reprocessing plants, ^{237}Np follows a complex flow sheet. In the first extraction step, a variable amount (usually about half) follows fission products in the aqueous phase to the high-activity waste (HAW) stream. The balance of the ^{237}Np can end up either in one of the product streams or be separated and disposed of in the HAW. The goal of the IAEA control regime is to ensure that there is no indication that ^{237}Np is removed from the process at any other point. This assurance can be obtained without necessarily using the traditional technique of materials balance (MB) accounting.[2] Rather, the approach favored by the Agency would use the ratio of ^{237}Np to other materials in spent fuel as an indicator that no significant ^{237}Np separation was taking place. The absence of deviations in the ratios beyond that expected from experimental error would indicate that no significant separation had taken place. This paper shows that because of the statistical properties of the flows of ^{237}Np, it is possible to confirm non-separation through the analysis of a relatively few randomly taken samples in the input and output streams. Implementation of such an approach will significantly reduce the costs associated with control of ^{237}Np.

We will consider one typical reprocessing facility with one input stream for ^{237}Np (the input accountability tank) and two output streams for ^{237}Np: the HAW and Pu product streams. Denote the ^{237}Np concentration (g/L) as X and assume that we use some fission product (FP (g/L)), which is any fission product in spent fuel that goes to the HAW in the first extraction step, as one "tracer" and Pu (g/L) as the second tracer.

Also, we assume that no volume measurement is needed, or at least that the volume measurement error essentially does not contribute to the uncertainty of interest. This introduces an issue that must be addressed, particularly if some form of random sampling (measure the ratio of ^{237}Np to tracer only in randomly selected transfer batches) is adopted. Possibly, because a volume measurement will be made in the key transfer tanks to account for Pu and U, traditional MB accounting will eventually be required for ^{237}Np. But in the case we consider here (FSV that includes random samples of some transfer batches), we assume that we can extrapolate from the sample to the population. That is, we assume there is no loss of ^{237}Np and estimate the total inputs and outputs by extrapolating from the sample. This has never been accepted in full MB accounting. For FSV, it will obligate us to define acceptance/rejection criteria for each sampled batch, which will require both reactor calculations for the input batches and process knowledge for the output batches, in addition to a good understanding of the measurement capability. The MB equations for the ^{237}Np, FP, and Pu are

$$\sum V_{0i} X_{0i} = \sum V_{1i} X_{1i} + \sum V_{2i} X_{2i} , \qquad (1a)$$
$$\sum V_{0i} FP_{0i} = \sum V_{1i} FP_{1i} , \qquad (1b)$$
$$\sum V_{0i} Pu_{0i} = \sum V_{2i} Pu_{2i} , \qquad (1c)$$

where V is volume, 0 denotes the input, 1 denotes the HAW output stream 1, and 2 denotes the Pu product stream. Let $R_{X0 \cdot FP0}$

denote the ratio $\Sigma V_{0i}X_{0i} / \Sigma V_{0i}FP_{0i}$ and similarly for the other ratios. Upon division of the left side of Eq. (1a) by $\Sigma V_{0i}FP_{0i}$ and the right side of Eq. (1a) by $\Sigma V_{0i}FP_{1i} = \Sigma V_{0i}FP_{0i} \ \Sigma V_{2i}Pu_{2i} / \Sigma V_{0i}Pu_{0i}$ (assumes only one output stream for Pu), we have

$$R = R_{X_1,FP_1}/R_{X_0,FP_0} + R_{X_2,FP_2} / R_{X_0,FP_0} = 1 + FL + e , \qquad (2)$$

where FL is the fractional loss of ^{237}Pu and e is measurement error. If FL = 0 then the expected value of R, E(R), equals 1. For notational convenience we introduce the terms R1, R2, R3, R4, and write Eq. (2) as $R = R_3/R_1 + R_4/R_2 = 1 + FL + e.$*

We anticipate that the root-mean-square error (RMSE, σ_R) of the measured ratio will be an effective performance measure to compare candidate options. Because the expected value of a ratio is not necessarily equal to the ratio of the expected values, it is important to use the MSE (or RMSE), which includes both bias and variance of the estimator (MSE = bias^2 + variance).

Any candidate estimator of R can be judged on the basis of its RMSE, σ_R. Also, σ_R can be used to quantify what size FL can be detected with high (> 0.95) probability.

This paper is organized as follows. Section 2 introduces the two ratio methods. Section 3 describes our assumptions about the measurement performance and process variation. Section 4 compares their performances for a reasonable range of expected measurement and process variation in simulated data. Section 5 addresses the issue of acceptance/rejection limits for each sampled batch to address the issue that our ratio estimators extrapolate from the sample to the population (and hence implicitly assume that there is no material loss). Section 6 contains a summary and directions for future research.

2. Two Ratio Methods

Reactor variation leads to substantial variation in the input ^{237}Np, tracers, and Pu concentrations. Some reprocessing facilities use spent fuel having similar burnup in a given campaign, in which case the reactor variation is reduced. Nevertheless, reactor variation could easily be as large of 50% relative standard deviation (RSD). Also, all reprocessing facilities will have substantial processing variation, which leads to highly variable ratios of ^{237}Np to tracer or Pu (up to approximately 50% RSD).

Generally, ratio methods are attractive because the ratio of the true ^{237}Np to FP or Pu varies much less than the absolute ^{237}Np, particularly in the input stream, so the effect of reactor variation and process variation is minimized. From Eq. (2), note that we could use

$$\hat{R}_{a1} = \left(\sum_{i=1}^{n_0} V_{0u}X_{0i} / FP_{0i} \right) / \sum_{i=1}^{n_0} V_{0i} \approx 1/n_0 \sum_{i=1}^{n_0} X_{0i}/FP_{0i} \qquad (3)$$

or

$$\hat{R}_{b1} = \sum_{i=1}^{n_0} V_{0i}X_{0i} / \sum_{i=1}^{n_0} V_{0i}FP_{0i} \approx \sum_{i=1}^{n_0} X_{0u} / \sum_{i=1}^{n_0} FP_{0i} \qquad (4)$$

to estimate R1 $= R_{X_0,FP_0}$ or any of the other ratios in Eq. (2). Equation (3) is a volume-weighted average ratio and can well be approximated without using a volume measurement. Equation (4) is the ratio of sums of absolute measurements, and it also can be approximated without using volume measurements.

It is often claimed that measured ratios are preferred because absolute measurements involve an additional calibration step. However, the errors introduced by the additional calibration tend to cancel. This paper does not address performance claims regarding the measurement of ratios of absolutes being better or worse than direct ratio measurements. Instead, we consider a reasonable performance range for both methods and report our findings. However, we do point out that the expected value of estimator $\hat{R}_{1a}$, E($\hat{R}_{1a}$) $\neq \hat{R}_1$ and E($\hat{R}_{1b}$) $\neq \hat{R}_1$(because the expected value of a ratio is not in general equal to the ratio of expected values). Therefore, $\hat{R}_{1a}$ and $\hat{R}_{1b}$ are biased. However, the bias is expected to be negligible (thus, most of the RMSE is due to random error variance in the estimators rather than due to bias), as we demonstrate in the performance section.

Analytical approximations for the variances of $\hat{R}_{1a}$ and $\hat{R}_{1b}$ are available using the delta method[3] (consider the variance of the linear terms in a Taylor series). We use two approximations (one as an approximate lower bound, the other as an approximate upper bound) for the variance of a ratio of random variables x and y

$$\sigma(x/y) \approx (x/y)^2 \ (\sigma^2(x)/x^2 + \sigma^2(y)/y^2) \ \text{(upper bound)} , \qquad (5)$$

$$\sigma(x/y) \approx (x/y)^2 \ (\sigma^2(x)/x^2 + \sigma^2(y)/y^2 - 2\text{cov}(x,y)/xy) \ \text{(lower bound)} \qquad (6)$$

Both approximations are problematic if y can approach 0. For our case, we assume that the ratios have large variation but are positive and bounded away from 0. In our simulations, we achieve this by simulating lognormal random errors. We use only the upper bound for the final application to Eq. (2), which involves ratios of ratios. Also, Eq. (2) involves a sum of ratios of essentially independent ratios (there is a modest nonzero covariance between ratios R_1 and R_2 only). Therefore, approximations (5) and (6) applied to both $\hat{R}_{3a}/\hat{R}_{1a}$ and $\hat{R}_{4a}/\hat{R}_{2a}$ for estimator $\hat{R}_a$ (same for $\hat{R}_b$) can be added (the variance of a sum of independent rv's is the sum of their variances) to give a good approximation to the $\sigma^2_{R_a}$ or for $\sigma^2_{R_b}$). To estimate the variance of the individual ratios $\hat{R}_{1a}$ and $\hat{R}_{1b}$,we must specify a measurement-error model in order to apply Eq. (5) or (6), which we do in Section 3.

3. Measurement Errors and Process Variation

Consider the term $R_1 = R_{X_0,FP_0} = \Sigma V_{0i}X_{0i}/ \Sigma V_{0i}FP_{0i}$. The numerator is the total input mass of X per period of study ("campaign"). The input mass of X varies with each batch due to both process- and measurement-error variation. Following one typical multiplicative measurement error model,[4] we assume for example that

$$V_i = V_{Ti} (1 + S_V + R_{Vi}) , \qquad (7)$$

where V_{Ti} is the true volume in batch i, S_V is the systematic error (assumed fixed throughout the campaign), and R_{Vi} is i^{th} the random error (unique for each batch). We assume that E(S_V) = 0 and denote var(S_V) as $\sigma^2_{S_V}$, and similarly for R_{Vi}. We refer to the batch-to-batch variation in V_{Ti} as process variation and denote it

as $\sigma^2_{R_{Vp}}$ (random variance of the volume for the process).To summarize, we have V_{Ti} distributed as lognormal($V_T,\sigma^2_{R_{Vp}}$) ($V_{Ti} \sim$ lognormal($V_T,\sigma^2_{R_{Vp}}$)), and then Eq. (7) models the measured volume, with both V_S and V_{Ri} also following mean-centered lognormal distributions with the appropriate variance.

For the concentration of each species, we must consider all sources of correlation. For example, when the X concentration is high in a given batch, there is a strong tendency for FP and Pu to also be high in that batch. Therefore, given the true concentration of X, we assume the FP distribution is FP|X $\sim \mu_{FP}$ + (X - μ_X)ρ(process) $\sigma_{R_{FPp}}\sigma_{R_{Xp}}$+ e, where e follows a mean-centered lognormal distribution with variance (1-ρ^2(process)) $\sigma^2_{R_{FPp}}$ and ρ(process) is the correlation between the true FP and X concentrations. We do the same for all pairs (X and Pu, and FP and Pu) in each flow stream. Then, we do a similar procedure (using systematic and random errors) to generate the measured value of FP given a measured value of X, etc.

Because of the nonnegligible covariance between the numerator and denominator in each individual ratio (such as $\hat{R}_{1b}$), we include approximation (6) as one estimate of the variance of the individual ratios. For example, to apply approximation (6) to $\hat{R}_{1b}$, we must estimate (a) var($\sum V_{0i}X_{0i}$), (b) var($\sum V_{0i}FP_{0i}$), and (c) cov($\sum V_{0i}X_{0i}$, $\sum V_{0i}FP_{0i}$). For (a) and (b), we use the often-used stream average (assume all input volumes and concentrations are equal) approximation: var($\sum V_{0i}X_{0i}$) = ($\sum V_{0i}X_{0i}$)^2 ($\sigma^2_{S_V}$ + $\sigma^2_{S_X}$ + ($\sigma^2_{R_{VM}}$ + $\sigma^2_{R_{Vp}}$ + $\sigma^2_{R_{XM}}$ + $\sigma^2_{R_{Xp}}$)/n). This follows most simply by substituting the appropriate error models into the expression var($\sum V_{0i}X_{0i}$) and noting that the systematic errors factor outside the summation because they are assumed fixed throughout the campaign. For (c), we note that cov($V_iX_{0i},V_{0j}FP_{0j}$) = $X_{0i}FP_{0j}$cov(V_i,V_j) + $V_iV_j$cov(X_i,FP_j) + cov(V_i,V_j)cov(X_i,FP_j), with the third term being negligible. According to our error models, cov(V_i,V_j) = var(V) = V^2 ($\sigma^2_{S_V}$ +$\sigma^2_{R_{Vp}}$ + $\sigma^2_{R_{VM}}$) if i = j, and cov(V_i,V_j) = V_iV_j if i does not equal j. That is, the measured volume varies from batch to batch due to random process variation in the true volume plus both random and systematic measurement error. Next, cov(X_i,FP_j) = X_iFP_j (ρ(process)$_{X, FP}\sigma_{R_{Xp}}\sigma_{R_{FP}}$) + ρ(S$_{meas}$)$_{X,FP}$ σ_{S_X} $\sigma_{S_{FP}}$ if i = j and cov(Xi,FPj) = XiFPj (ρ(S$_{meas}$)$_{X,FP}\sigma_{R_{Xp}}$ $\sigma_{R_{FPp}}$) if i does not equal j. Note that the two sources of covariance between X_i and FP_i are due to the process () and due to measurement (ρ(S$_{meas}$)$_{X,FP}\sigma_{S_X}$ $\sigma_{S_{FP}}$). If i does not equal j, then only the systematic measurement errors contribute to cov(X_i,FP_j). Approximation (6) is applied to in a similar manner. The other three ratios have identical form for both version a and b of the estimators. The final expressions (upper and lower bounds) for var ($\hat{R}_a$) (or for var($\hat{R}_b$)) involves using approximations (5) and (6) applied to approximation (6) for each of the four ratios.

Simulation

To simulate a campaign with sampling, we first select the number of batches per campaign in each flow stream and specify all error models in percent RSD (assume all variation sources are multiplicative). We simulate the entire system (process variation, measurement error, and sampling effects) as follows (a

brief description only, intended to convey the concepts of simulating measurement variation (systematic and random variation) and process variation (random variation only)):

(A) For each flow stream;
 (1) generate S_V and S_X;
 (2) then generate S_{FP} according to S_{FP} |S_X= S_X ρS_{meas})$_{X,FP}$ $\sigma_{FP,S_{meas}}$ $\sigma_{X,S_{meas}}$ + e, where e is a shifted (mean 0) and scaled lognormal with variance σ^2_e = (1-ρ_2(S$_{meas}$)$_{X,FP}$)$\sigma^2_{FP,S_{meas}}$;
 (3) do the same for S_{Pu}|S_X.

(B) For each batch in each flow stream:
 (1) generate R_{Vm}, R_{Vp};
 (2) generate $R_{X,process}$ and generate $R_{FP,process}$|$R_{X,process}$ according to $R_{FP,process}$|$R_{X,process}$ = $R_{X,process}$ ρ(process)$_{X,FP}$ σ_{FP} σ_X + e, where e is a shifted (mean 0) and scaled lognormal with variance σ^2_e =(1-ρ^2(process)$_{X,FP}$)$\sigma^2_{FP,process}$;
 (3) do the same for and $R_{Pu,process}$| $R_{X,process}$.

(C) Compute $\hat{R}_b$ and $\hat{R}_c$.

The lognormal distribution allows us to include large coefficients of variation while ensuring positive measurement results. The nominal correlations will be very nearly exactly achieved (on average) using the appropriate conditional distributions, provided the number of simulations is large (100 or more).

4. Simulation Results

In this section we present simulation results for several sample sizes in which we assumed reasonable values for the numbers of inputs, outputs, and for the process and measurement variation. We assume 200 input batches per year (or campaign), 200 HAW (stream 1) batches, and 40 product (stream 2) batches. We parameterized our simulation code to allow rapid experimentation with the input parameters, ρ_X, ρ(process)$_{X,FP}$, ρ(S$_{meas}$)$_{X,FP}$, all the σ's, the true throughput of each species, and the batch sizes.

We present results here for two typical cases: all process σ's for concentrations are either (1) 50% or (2) 20%, and assume for both cases 200 input batches per year (or campaign), 200 HAW (stream 1) batches, and 40 product (stream 2) batches, process σ for input volume is 10%, process σ's for output volume are 20%, all systematic error σ's are 10% for concentration and 0.5% for volume, correlations in concentration due to the process are all 0.5 and due to measurement are all 0.2.

From the 200 input (or output 1) batches, we sample 1, 2, 3, 4, or 5 batches or from 10 to 200 in steps of 20, and from the 40 output-2 batches, we sample 1, 2, 3, 4, or 5 batches or from 10 to 40 in steps of 4. We plot $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$ versus total sample size (with the same number of samples per flow stream) for the 50% process variation case in Figures 1a and 1b, and for the 20% process variation case in Figures 2a and 2b. In all plots, the largest sample size plotted is 132 (rather than 440) because both $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$ are essentially constant for sample sizes larger than 132. We also plot the upper (Eq. 5) approximation and the lower (Eq. 6) approximation in the same plots. Note that the observed

values of $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$ fall either between the lower and upper approximations or below the lower approximation. We therefore believe that our upper and lower approximations are conservative, in that the true $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$ will be no more than the upper approximation and in some cases will be below the lower approximation. Also, note that "diminishing returns" set in at approximately 30 samples (10 per stream), because $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$ decrease very slowly after $n_{sample} = 30$. For 50% (20%) process variation, we can achieve $\sigma_{\hat{R}_a}$ (or $\sigma_{\hat{R}_b}$) of approximately 13% (10%) for our parameter values.

One convenient aspect of using ratios is that $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$ do
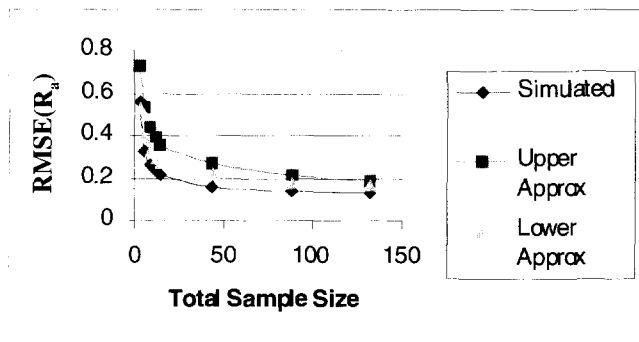
Figure 1a. 50% process variance, ratio estimator
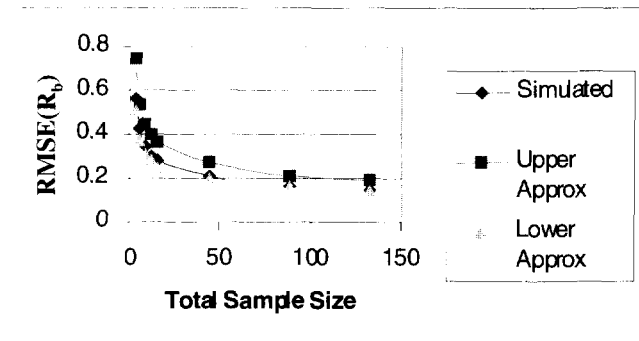
Figure 1b. 50% process variace, ratio of absolutes estimator

Figure 2a. 20% process variance, ratio estimator.

not depend on the amount of throughput per batch (although they do depend on the number of batches). Therefore, if we assume 200 inputs, 200 stream-1 outputs, and 40 stream-2 outputs, but vary the throughput by varying the batch size, we will
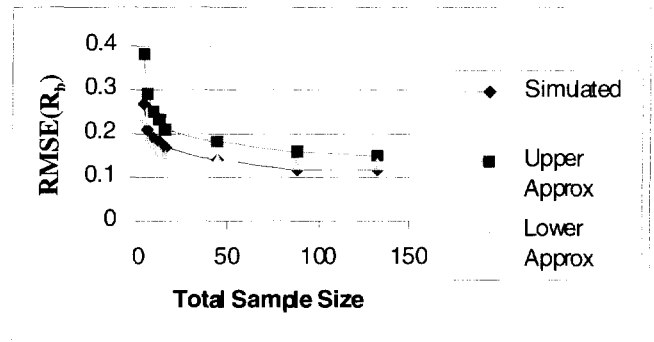
Figure 2b. 20% process variance, ratio of absolutes estimator.

not change $\sigma_{\hat{R}_a}$ and $\sigma_{\hat{R}_b}$. Therefore, we can assess loss-detection ability for any throughput (for the assumed number of batches) by converting the significant quantity (SQ) of interest to a fractional loss. For example, consider small, medium, and large plants having annual throughputs T = 2 SQ, 10 SQ, and 40 SQ, respectively, where SQ is the significant quantity (the amount that would represent a significant loss or diversion). These throughputs in terms of SQs are consistent with the following assumptions:

(1) the Pu throughput in kg/year is 500, 2500, and 10 000 for the 3 plant sizes;

(2) the [237]Np is 10% of the Pu (a reasonable upper limit for [237]Np); and

(3) the [237]Np SQ is 25 kg (not yet determined).

By expressing the annual throughput of [237]Np in multiples of SQ, we can easily calculate the detection probabilities for a given false alarm probability. Following typical convention,[5] let the false-alarm probability be 0.05 and perform one-sided testing (test for loss only, not for gain). Then the detection probability of a one-SQ loss for a throughput of k SQ is $P(MB/\sigma > 1.65) = P(Z > 1.65 - SQ/\sigma) = P(Z > 1.65 - SQ/(0.15 \, k \, SQ))$ where Z is a standard normal (mean 0, variance 1) random variable, $\sigma = \sigma_{\hat{R}}$, and we assumed 20% process variation so that $\sigma = .15$ (approximately, from Fig. 2). For $k = 2$, 10, and 40, we have a detection probability of >99%, 26%, and 8%, respectively. Therefore, we anticipate that the loss-detection probability will either be very large (for small plants), or will be quite small for large plants (as is the case for Pu). To compare to the Pu loss detection, we can expect the σ for a Pu material balance in a modern facility to be approximately 0.5% of throughput, which is 20 times better than the σ for the ratio-based [237]Np throughput under our assumptions. However, the Pu throughput is approximately 10 times the [237]Np throughput, and the Pu SQ is approximately 6 times the [237]Np throughput, so the net effect is that we can detect [237]Np losses better than we can detect Pu losses. A related issue is that it will probably be of interest to minimize the total sample size $n = n_0 + n_1 + n_2$. We have some flexibility in how to allocate the total sample size n among n_1, n_2, and n_3, and it is likely that the stream sample sizes will be allocated proportionally to the stream throughput. That can be shown to minimize $\sigma_{\hat{R}_a}$ or $\sigma_{\hat{R}_b}$ and it is the best choice when we consider sample size requirements for detecting large anomalies in a given batch (section 5).

5. Acceptance/Rejection Criteria for Sampled Batches

We must recognize that if we do not sample all batches from all streams, we could fail to detect an abrupt anomaly such as complete diversion of the ^{237}Np in an unsampled batch. This issue can be approached by using the "zero-defect" sampling scheme, in which the only acceptable number of "defective samples" is zero. So, if the ^{237}Np mass in a batch is much less than the stream average (for example, if the batch ^{237}Np mass is more than 5 σ_m less than the average m), then that sampled batch is "defective," and the facility would fail the inspection. Assuming we could define such a pass/fail criterion for each sampled batch, we could calculate the required sample sizes using an approximation to the exact hypergeometric probabilities that arise. The well-known approximation is $n = N(1 - \beta^{(1/D)})$, where D is the number of batches that would have to be diverted to acquire one SQ. To apply this "zero-defect" sampling scheme, we must select β and D. To do so, we will consider the loss-detection performance for Pu for a loss of 1 SQ. We find that the loss-detection probabilities are $1 - \beta = 0.93, 0.16$, and 0.09 for the 3 plant sizes for diversion of 1 SQ over 1 year (assuming each plant satisfies $\sigma_{MB} = 0.005T$). Note that, as we have seen, the T = 20 SQ and 40 SQ plants do not even approximately meet the strict goal of $1 - \beta = 0.95$. Using __ = 0.07, 0.84, and 0.91 and D = T/Ni, the approximation $n = N(1 - \beta^{(1/D)})$ gives the sample requirements in Table 1.

Table 1. Sample sizes for inputs (n_1) and outputs (n_2 and n_3) for the ^{237}Np loss-detection probability in 0-defect sampling to give the same detection probability for loss of one SQ or more of ^{237}Np as for loss of one SQ of Pu.

T = 2 SQ	T = 10 SQ	T = 40 SQ
$n_1 = 6, n_2 = 3, n_2 = 3$	$n_1 = 2, n_2 = 1, n_3 = 1$	$n_1 = 7, n_2 = 4, n_3 = 7$

Note that these sample requirements in Table 1 are less than those for which "diminishing returns" begin to apply in Figs. 1a, 1b, 2a, and 2b (in terms of slower rate of decrease in σ_R for larger sample sizes). Therefore, we can choose sample sizes on the basis of achieving adequate RMSE of σ_R and be confident that we also achieve adequate "abrupt loss" detection (to which the Table 1 entries refer).

6. Summary and Future Directions

We have considered the possibility of approximating full accountancy by selecting some of the input and output transfers to be measured for ^{237}Np mass. We have presented sample size calculations under idealized assumptions for 3 plant sizes for $\sigma_{process} = 0.20$ and $\sigma_{process} = 0.50$ (relative).

Although MB accounting is not anticipated, we believe that a useful objective performance criterion involves loss-detection and false-alarm probabilities. Certainly some criterion will be invoked because of the need to agree on required assay-perform-ance and sample-size requirements. It is logical to link the requirements to the RMSE, σ_R, of the estimated ratio because that

can be related to loss-detection and false-alarm probabilities. Our results suggest that diminishing returns (in terms of the reduction in σ_R) will begin at sample sizes of approximately 25-30% of the population (number of batches) sizes. Also, the Table 1 entries suggest that "abrupt loss" detection probability will be adequate if we choose the sample sizes motivated by consideration of diminishing returns in σ_R (because the commensurate sample sizes of 25-30% of the population are larger than the sample sizes needed for adequate abrupt-loss detection). However, we must develop a pass/fail criterion for each batch in order to implement a zero-defect sampling plan. That will obligate users to set thresholds based on historical variation, which is a potential difficulty if the facility does not operate consistently.

We currently do not have an explanation for why our "upper" and "lower" approximations do not completely bound the observed (simulated) σ_R for either of our two candidate estimators. However, we have indicated some possible explanations and believe that our current approximations are adequate, although they do not bound the simulated results. It would be useful to have approximations that do bound the simulated results. Also, work is in progress to acquire real data from a reprocessing facility to compare to our simulated results. Although we have used fairly detailed measurement-error models (based on results for Pu and other assays in reprocessing facilities) in our simulations, there is no substitute for real data to test our assumptions. Assay methods for ^{237}Np are currently under development in several laboratories worldwide.

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As an important aside, simple algebra shows that FSV for any facility having either (a) more than two output ^{237}Np streams, (b) more than one output Pu stream, or (c) more than one output Nd stream will require absolute ^{237}Np measurements for some species for some output streams. (Note that to derive Eq. (2) from Eqs. (1), we exploited the facts that Eq. (1b) and Eq. (1c) each had only one term on both sides.) Fortunately, most facilities are well approximated by our assumptions because minor output streams can be ignored.

The Use of Stable Xenon Isotope Monitoring in Strengthened Safeguards at Large Reprocessing Plants

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Abstract
An environmental monitoring technique using the on-stack capture and subsequent analysis of stable xenon isotopic ratios at a reprocessing facility is presented. This technique integrates existing technologies to strengthen safeguards at large-scale reprocessing facilities. A description of the monitoring technique and its potential application is described. The potential impact of such a technique for both the International Atomic Energy Agency and the reprocessing facility operator is also considered. Cost estimates and implementation analyses suggest that the use of this technique at a large-scale plant would provide a cost-effective and non-intrusive method for increasing confidence in facility declarations.

Introduction
International safeguards at large reprocessing plants present a challenge because of the amount of nuclear material that passes through such a plant each year. Optimistic estimates based on accepted values of measurement errors[1] suggest a standard deviation in material unaccounted for (MUF) of (0.3% of throughput.[2] For an 800-metric-ton-per-year plant, reprocessing fuel with a nominal 10 kg of plutonium per metric ton of heavy metal, this would mean an uncertainty in MUF of >24 kg of plutonium. Conventional INFCIRC/153[3] safeguards address this problem through additional measures such as containment and surveillance. The recent introduction of the Strengthened Safeguards System through the adoption by States of Additional Protocols based on INF-CIRC/540[4] provides new measures to further improve international safeguards at large reprocessing plants. One of the most significant of these measures is the right of the International Atomic Energy Agency to obtain environmental samples at points it accesses.

This paper will address an environmental-monitoring technique developed at Los Alamos National Laboratory that can be used to increase confidence that a reprocessing plant is operating in a manner consistent with its declarations. This technique focuses on the collection of samples of dissolver off-gases in the exhaust stack of a plant.[5] These samples are then analyzed to obtain the ratios of the various stable xenon isotopes in the off-gases that were produced by fission processes in the fuel while it was in the reactor. The basic concept of this approach has been considered in the past;[6-8] however, LANL has developed the tools and demonstrated the feasibility of this approach. One new component is a statistical analysis tool that compares the measured xenon isotopic data with an extensive database generated from rigorous modeling of xenon production in nuclear reactors as a function of burnup and reactor type. A key element in the use of this technique is that it provides these benefits without significantly increasing costs for either the IAEA or the operator.

The following section provides a description of the monitoring technology, including the plant processes resulting in the xenon signature, on-stack sample collection, xenon isotopic ratio measurements, reactor modeling to develop xenon isotopic ratio functions, and a Bayesian data-analysis technique. This is followed by a section describing a concept for the use of stable xenon gas analysis in reprocessing plant safeguards. A major objective in the development of this concept is to minimize disruption and costs both for the facility operator and the IAEA.

Description of Proposed Xenon Monitoring System
During the irradiation of fuel in a nuclear reactor, various fission products are produced, including the higher-mass xenon isotopes. The relative concentrations of the fissiogenically produced stable xenon isotopes depend on the neutron energy spectrum in the reactor, the fissioning isotopes, the power level, and the total number of fissions. Therefore, the xenon in the fuel provides a unique signature of the fuel's characteristics. Some previous attempts have been made to make use of these gaseous emissions; however, these studies proved inconclusive.[6-8]

The fissiogenic xenon gases are released during the chop and dissolve phases of reprocessing. Because of its chemically inert nature, the xenon generally travels directly to the stack and is relatively unaffected by chemical separations and porous filters.

The ^{131}Xe/^{134}Xe and ^{132}Xe/^{134}Xe isotopic ratios contain information about the fuel being reprocessed and have been shown to be a valuable monitor of reprocessing activities.[9-10] Also, since the gas is emitted through the facility's stack, the most likely collection point for taking samples (i.e., on-stack) is relatively far from the primary reprocessing activities which could promote both simplicity of collection and acceptance of the procedure by facility operators.

The proposed monitoring system consists of several distinct steps (Fig. 1). First, samples are collected on-stack at a reprocessing facility. For instance, samples can be collected through a probe in the stack. A portion of the stack gas is diverted to an appropriate sampling container (an evacuated bottle) and then saved for future analysis. It is likely that sample collection can be accomplished through the existing stack monitoring systems with slight modifications. The collected samples are then analyzed using a mass spectrometry system to determine the relative abundance of the stable xenon isotopes in the gas. Next, the xenon background air contamination is subtracted and the resulting purely fissiogenic xenon isotopic ratios are compared to a calculated database of reactor-physics parameters to determine the most likely fuel type and burnup that matches the measured ratios.
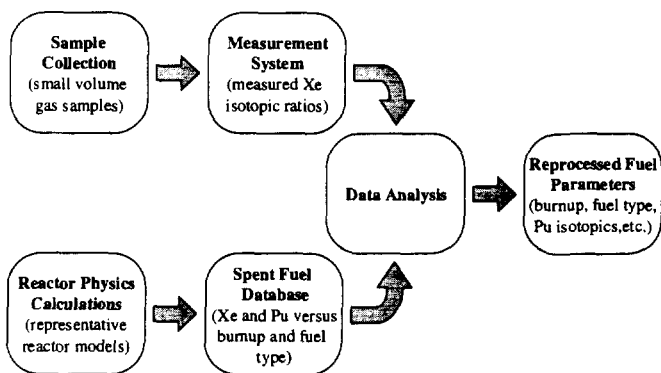

Figure 1. Overview of proposed monitoring technique.

Mass Spectrometry

One of the most important parts of the verification methodology is the noble gas measurement system. Determining spent fuel parameters from noble gas emissions requires precise and accurate measurements of the isotopic ratios of the stable fission gases. The fissiogenic xenon component from stack gases may contain up to 50% air xenon for some isotopes. This necessitates the use of a precise instrument that will allow for the extraction of the fissiogenic component with a high degree of accuracy to allow for discrimination between reactor types at various burnups. Because the isotopes of interest in this study are primarily stable noble gases, mass spectrometry is the ideal choice for a measurement system. Also note that all measurements are for isotopic ratios instead of absolute values. This eliminates the need for incorporating certain instru-

ment characteristics and enables greater generalization in the fissiogenic gas database. For a fully developed monitoring system, it is expected that commercially available mass spectrometry systems will be sufficient for the implementation of this technique.

Reactor-Physics Database

A reactor-physics database was developed for the statistical analysis portion of this technique. The reactor physics database contains xenon fissiogenic isotopic ratios and plutonium concentrations as a function of burnup for an extensive set of fuel types including

1. pressurized water reactor fuel,
2. boiling water reactor fuel,
3. Canada Deuterium-Uranium reactor fuel,
4. liquid metal fast breeder reactor blanket and driver fuel,
5. Calder Hall reactor fuel, and
6. RBMK fuel.

These ratios and plutonium concentrations were calculated using a series of state-of-the-art reactor analysis codes. The codes used in developing this database enabled accurate calculations of plutonium and xenon concentrations in spent fuel for a variety of reactor types. To properly couple the database to the measured isotopic ratios, the reactor-analysis codes were benchmarked for the production of xenon and plutonium in 12 different reactors.[9]

Data Analysis

The measured isotopic ratios can be coupled to the calculated fissiogenic-gas database through a Bayesian analysis technique that allows for the determination of the most likely fuel type and burnup from a set of measured isotopic ratios. A brief description of Bayesian analysis theory and development, along with the procedure used in these analyses, is given in this section. Also, the methodology used for subtracting the background air contaminant is given.

Because any realistically acquired samples contain both a fissiogenic component and a natural-air component, the sample's measured isotopic ratios will consist of a combination of the fissiogenic and atmospheric-air noble gases. Because some noble gas isotopes (e.g., 129Xe) are not produced in significant quantities via fission, these measured nonfissiogenic isotopes can be used to remove the background-air contaminant. This requires using known natural abundances of the xenon isotopes in air (either assumed or measured). For xenon, with normalizing isotope 134Xe and non-fissiogenic isotope 129Xe, the isotopic ratio of interest is given by

$$\frac{N_{f,u}^x}{N_{f,u}^{134}} = \frac{\left(\dfrac{N_{m,u}^x}{N_{m,u}^{129}} - \dfrac{N_{m,air}^x}{N_{m,air}^{129}} \right)}{\left(\dfrac{N_{m,u}^{134}}{N_{m,u}^{129}} - \dfrac{N_{m,air}^{134}}{N_{m,air}^{129}} \right)} \tag{1}$$

Thus, given a measurement of the isotope of interest and the

normalizing isotope (^{134}Xe) relative to ^{129}Xe in the unknown sample and in atmospheric air, the background-air contaminant can be removed directly.

Given a set of I measured isotopic ratios $[R^m = (R_1^m, R_2^m, ...R_I^m)]$, their associated standard deviations (σ_i^m), and a mutually exclusive, exhaustive set of J reactor models $[M = (M_1, M_2, ...M_J)]$, we can determine the most likely model (M_j) at a particular burnup (B_j) using a Bayesian analysis methodology. The reactor models are described by a database of calculated isotopic ratios $[R_{ij}^c = (R_{ij1}^c, R_{ij2}^c, ...R_{ijK}^c)]$ and their associated standard deviations (σ_{ijk}^c) for each model (M_j) at a series of K burnup points $[B_j^c = (B_{j1}^c, B_{j2}^c, ...B_{jK}^c)]$. The model based probabilities for each isotopic ratio [i.e., the probability that one would measure the isotopic ratio (R_i^m) given spent fuel from the reactor model (M_j) at burnup (B_j) and any background information (E)] can be calculated using the maximum entropy formulation:

$$p(R_i^m \mid M_j, B_j, E) = \frac{1}{\sigma_{ij}\sqrt{2\pi}} \exp\left[-\frac{(R_i^m - R_{ij}^\mu)^2}{2(\sigma_{ij})^2} \right]. \tag{2}$$

Using these model-based probabilities and a version of Bayes' theorem, the probability that the spent fuel is from reactor model (M_j) at burnup (B_j) given a set of measured isotopic ratios (R^m) can be determined from

$$p(M_j, B_j \mid R^m, E) = \frac{p(M_j, B_j \mid E)\prod_{i=1}^{I} p(R_i^m \mid M_j, B_j, E)}{\sum_{j=1}^{J}\prod_{i=1}^{I} p(R_i^m \mid M_j, B_j, E)p(M_j, B_j \mid E)}. \tag{3}$$

The quantity $p(M_j, B_j|E)$, called the prior, represents the probability that the fuel is of type M_j at a burnup of B_j given any background evidence alone. This background evidence could be information such as an inspector's observation that the fuel is not of a specific type, knowledge that a country does not possess a certain type of reactor, or any other data to which an inspector might have access. If there is no reason to prefer one model over any of the others, then the priors can all be set to unity.

It should be noted that the analysis described above assumes that our database includes a mutually exclusive and exhaustive set of reactor models. Because the database used in these studies contains fourteen different fuel types each evaluated over an extensive range of burnups, it will be assumed that this assumption is valid. However, one should note that if a reactor fuel of some type markedly different from those included here is reprocessed, this assumption will no longer be appropriate and other methods should be used (or the database should be modified). The database and analysis methodologies are contained in a software package produced by LANL. In the case of reprocessing a fuel not in the available database, this software package will respond to the input data by informing the user that the data not match any of the existing models at any burnup value. This is accomplished through the use of a confidence factor that conveys the probability that the given ratios match the available models [expressed above as $p(M_j, B_j|R^m, E)$]. Details of the soft-

ware and the derivation of the analysis methodology described above are contained in Reference 9.

The Application of a Stable Xenon Isotope Monitoring System to Safeguards

The proposed monitoring system described above can be applied to support a strengthened safeguards system at large reprocessing plants. The elements of a safeguards measure based on xenon monitoring are
1. examination of facility records to determine burnup and reactor type for each batch,
2. design information verification (DIV) to ensure that the stack being sampled contains the dissolver off-gas stream,
3. acquisition of samples,
4. analysis using mass spectrometry, and
5. reconciliation of results and records.
Each of these elements is discussed in more detail below.

Facility Records and DIV
The first step in a safeguards implementation is that for each sample taken facility records should be examined to obtain information on burnup and reactor type for each batch. This type of data is normally available as part of plutonium safeguards. This information will be needed for comparisons to the inferred results from the fission gas. In the normal course of the annual DIV, inspectors would have to take measures to ensure that dissolver off-gases go to the stack being sampled. This should be a straightforward extension of the current DIV process.

Sampling Issues
As indicated above, once a probe is in place, sampling of the stack for stable xenon isotopes is straightforward process. An estimate of the cost per sample is $120. This would include the collection of both a sample from the stack and a background sample (for subtraction of the background air contaminant). These samples can be stored for at least 10 years. This suggests that off-gases from each batch of fuel processed could be sampled for an annual cost of about $24,000 for a plant dissolving 200 batches of spent fuel per year. If this were to prove too burdensome, an alternative would be to sample batches on a random basis. Even in this case, it should be possible to sample a large fraction of the batches.

Analysis Issues
In contrast to the ease and low cost of sampling, mass spectral analysis for xenon, with the required precision, may be more complicated and expensive depending on the accuracy required. An estimate of the cost for a single analysis is $500. Thus, for one stack sample and one background sample the cost would be $1000. This suggests that the best approach is not to analyze every sample taken, but rather to analyze a few randomly chosen samples to provide an extra degree of transparency in facility operations. However, it is important that all of the other samples taken be retained for future analysis in the event that an

anomaly is found. The ability to analyze the other samples later would provide both a powerful proliferation deterrent and a means of increasing confidence in resolving anomalies.

In the case of random sampling, the number of samples that must be analyzed is dependent on the number of batches per time period, the required probability of detection of inconsistent batches, and the goal number of inconsistent batches that must be detected. For a zero-defect sampling scheme (in which the only acceptable number of defects is zero), the required number of samples (n) is then given by

$$n = N(1 - \beta^{1/M}) \quad (4)$$

where N is the total number of batches and β is one minus the required probability of detection of M inconsistent batches.[11] For example, the number of analyses required to detect at least one inconsistent batch out of 200 total batches (where 5 out of the 200 batches are inconsistent) with a 50% probability is 26. This would give an annual cost for analysis of $26,000. Therefore, the total annual cost for sampling and analysis would be on the order of $50,000. The annual cost of a regime where all batches are sampled and analyzed would be about $224,000.

Reconciliation

Once the required number of samples are analyzed, determination of burnup and reactor type are straightforward using the LANL-developed Bayesian analysis software. The user is prompted to enter the xenon isotopic ratios for each analysis, and the software produces its estimate of burnup and reactor type. This software has been benchmarked extensively for PWR and BWR fuels, and some comparisons for LMFBR fuels have been performed. However, data has not been available to verify its accuracy for graphite-moderated reactor fuels (such as Calder Hall or RBMK reactors). The LANL software would provide an estimate of the most likely reactor type and burnup based on the measured values, and these estimates could be used to verify the declarations made by a particular facility.

Summary and Conclusions

Reprocessing plants, because of the presence of large amounts of direct-use material, have always provided a challenge to international safeguards. The implementation of INF-CIRC/540 measures offers the prospect of significantly increasing the effectiveness of IAEA safeguards at little additional cost; however, attaining this goal is subject to the availability and implementation of tools that provide significant information to IAEA inspectors and can be deployed in a cost-effective manner.

Information to date indicates that xenon monitoring can be a cost-effective tool to increase the transparency of reprocessing plant operations. A key to this is an understanding of the cost structure, in which collection of samples is quite inexpensive while analysis is more expensive. One should note that the cost structure used in these studies was based on R&D analysis costs; as a facility moves toward a commercial (or

production mode) application of these analyses, the costs would drop significantly (perhaps 50% or more). A careful optimization of this process will yield increased confidence at a reasonable cost.

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3. MUF standard deviation calculated using

$$\sigma = \sqrt{\sigma_{in}^2(sys) + \sigma_{out}^2(sys) + \sigma_{in}^2(rand)\Big/n_{in} + \sigma_{out}^2(rand)\Big/n_{out}}$$

where σ's are errors expressed as absolute standard deviations and the n's are number of batches. The calculation assumes dip-tube volume measurements and isotopic-dilution mass spectrometry to determine the amount of plutonium in the input accountability tank and the use of electrobalances and gravimetric analysis to determine the amount of product in PuO_2 product. It is also assumed that there is no loss of material to other plant streams or to holdup.
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An Update of IAEA Analytical Capabilities for Safeguards Goals, Results and Challenges

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Abstract

Until 1991, the principal function of the IAEA's Safeguards Analytical Laboratory and Network of Analytical Laboratories for Safeguards analyses was to analyze samples of nuclear materials and heavy water requested by IAEA inspectors on the basis of random sampling plans in order to verify the accountability data declared by the states. Independence, accuracy and satisfactory response times are the paramount characteristics expected for such analyses. The major progresses in this respect will be reviewed.

After the Gulf War of 1991 and the conclusions of the 93+2 Programme, environmental sampling became a major tool of the IAEA for gaining confidence that illicit nuclear weapon capabilities were not developed clandestinely in contravention to safeguards agreements. The inspections first focused on the detection of potential undeclared uranium enrichment and plutonium separation. The NWAL was therefore expanded to include laboratories having the required expertise. A clean laboratory facility was added to SAL and equipped with Class 100 chemical laboratory rooms and highly sensitive instruments. It provides sampling kits, screens the samples when they are received in Seibersdorf, performs confirmatory measurements and manages a measurement quality control program.

The appropriate balance of resources for nuclear material analysis vs. environmental sample analysis is one of the new challenges, as budget restrictions threaten already with the progressive closing of several NWALs. We otherwise face five analytical challenges: (1) regularly reaching 0.1% or better accuracy in the accountability of large amounts of plutonium and high-enriched uranium, (2) equipping and operating an on-site laboratory at the Rokkasho-Mura plant, (3) clarifying the precision, accuracy and limits of detection required for environmental sample analyses, (4) developing reference and control materials suitable for checking the analyses of solid particulates in environmental samples, and (5) identifying the signatures and procedures appropriate for wide-range detection.

Introduction

An overview of the analytical capabilities and systems available for the safeguards activities of the International Atomic Energy Agency in 1994 was presented at a symposium in Vienna.[1] Several changes have taken place since then and have long-ranging consequences on the daily safeguards operations of the IAEA. The Non-Proliferation Treaty was extended indefinitely at an international conference in New York in 1995. In September 1996, the General Conference of the IAEA approved a plan of a strengthened safeguards regime in two parts following the conclusions of Programme 93+2. Part I measures are already in force and include the taking of swipes or other environmental samples at facilities where traditional safeguards agreements grant access to IAEA inspectors. Under Part II measures, model protocols have been worked out and will allow access to other locations, so that environmental sampling can be used also outside nuclear facilities and even in a wide-area mode in the future. The IAEA Action Team, in collaboration with the United Nations Special Commission (UNSCOM), is wrapping up its report on the dismantling of Iraq's nuclear weapon program and concentrates its efforts now on an ongoing monitoring and verification plan to confirm that Iraq abides by U.N. Resolution 1051. There is a new impetus with regional safeguards agreements. The New Partnership Approach between Euratom and IAEA brings sizable savings of inspection time in western European countries. The Argentinian-Brazilian Accountability and Control Agency applies a regime of mutual inspections. At the same time, Argentina and Brazil entered a comprehensive safeguards agreement with IAEA that enforces it in cooperation with ABACC. The states parties to the treaty banning the use and production of toxic chemical weapon grants to the enforcing organization (Organization for the Prevention of Chemical Weapons) the possibility to undertake unannounced inspections anywhere in their territory. These clauses opened the way for the introduction of similar tools in Part II measures of IAEA strengthened safeguards regime. The Comprehensive Test Ban Treaty was signed in New York in 1997 and will enter into force as soon as it has been ratified by

40 countries. The IAEA has offered to collaborate with the CTBT organization in the areas of worldwide radioisotope monitoring in air and on-site inspections. The Russian Federation and the U.S.A. proceed with stepwise reduction of their strategic weapons in the frame of the Strategic Arms Limitation Treaties. Encouraged by these progresses, the United States decided to reduce unilaterally its stock of nuclear weapons and placed the so-called "excess nuclear weapon materials" under IAEA safeguards. These include some two tons of separated plutonium and about 10 tons of high-enriched uranium. The U.N. Commission on Disarmament in Geneva pursues its discussion of a cutoff of the production of special nuclear materials for nuclear weapons.

However, continuing local conflicts and serious economic problems temper the optimism induced by the developments mentioned above. Implementation of NPT Safeguards Agreement is still suspended in the Democratic People's Republic of Korea. Tensions between Pakistan and India led to the series of nuclear weapon tests in April and May 1998. Public concern over the safety of nuclear energy industry operations and nuclear waste disposal, economic competitions and restrictions still block the expansion of commercial nuclear activities.

The present paper provides an updated information on IAEA analytical capabilities for safeguards today. It describes in particular the current goals, the results achieved and the challenges that lay ahead, in light of the developments that have taken place since 1994. It reflects the activities of the IAEA Safeguards Analytical Laboratory in Seibersdorf and of the IAEA Network of Analytical Laboratories (Table I) and the contributions of member state support and cooperation programs for the development of international safeguards.

Methods of Analysis of Nuclear Materials

Material accountability verification remains the basic concept applied by IAEA to obtain evidence that significant quantities of materials placed under its safeguards are not diverted from declared activities. It involves random but independent measurements of the amounts of materials, element and isotopes, combined with containment and surveillance measures. Destructive analyses (DA) are done in complement of on-site measurements and analyses, in order to:

- verify that protracted diversion of safeguarded nuclear materials has not occurred;
- certify working standards used for the calibration of NDA and installed verification instruments;
- provide assurance of the quality and independence of on-site measurements (e.g. validation of facility-specific procedures);
- carry out periodic verifications of operators' measurement systems.

Off-site DA are requested primarily for the determination of the chemical concentration and isotopic composition of safeguarded materials such as uranium, plutonium, thorium and heavy water. These analyses are made by SAL and NWALs on all types of solid and liquid materials encountered in bulk-han-

dling nuclear plants. DA verification measurements involve the following steps:

1. The taking of independent samples;
2. Their conditioning at the facility to ensure that they are in a chemical form adequate for maintaining their integrity during transport;
3. Their packaging, sealing and shipment to the SAL in Seibersdorf, near Vienna;
4. Their analysis by SAL or NWAL;
5. The statistical evaluation of the results of their analysis.

The critical review of IAEA safeguards regime in the 1990s pressed the IAEA to develop smarter or more comprehensive ways of safeguarding nuclear materials and activities. Consequently growing efforts are made to develop an analytical capability to analyze DA inspection samples of nuclear materials also for other actinides than uranium or plutonium.

The following sections give an overview of the analytical facilities, measurements and services that the IAEA currently uses for the analysis of nuclear materials.

IAEA Safeguards Analytical Laboratory Controlled Area

The controlled area, where samples of nuclear materials are received and analyzed, occupies now the entire 700 m^2 of space available in the original building erected in 1975-1976. The area equipped to receive and handle samples of plutonium and spent fuel materials and devoted to their chemical treatment and analysis was increased in steps to accommodate new instrumentation and analytical procedures. It reaches now about 275 m^2.

The main techniques applied for nuclear material analysis at SAL[2] are listed in Table II. The measurement precisions and accuracies reflected in the table by the random and systematic uncertainties, respectively, are values achieved in the analysis of materials of nuclear grade or similar chemical purity. They include the contributions of all uncertainties occurring after sampling. The effects of sampling, impurities and foreign components can increase the uncertainties and will vary with the type of material, to the extent that sampling uncertainties can become the dominant factor in the total measurement error. The analytical techniques and procedures installed at SAL were selected with the following objectives in mind:

- SAL must be equipped with a minimum capability to perform independent measurements of the characteristics most sensitive in safeguards.
- Its procedures must be suitably selective and achieve precisions and accuracies consistent with IAEA detection goals[3] and 1993 IAEA International Target Values ~
- The analyses must be completed and the results reported within 60 days of the inspection so that they can be reflected in the inspection report.[8]
- SAL must be able to provide evidence of the traceability of its measurements to the International System of Units.
- When necessary, SAL must be in the position to verify that actual samples do not introduce hidden interferences or biases in the standard method of analysis, by performing replicate analyses with an independent method

Table I: IAEA Network of Analytical Laboratories

LABORATORY	Nuclear Material Analyses	Heavy Water Analyses	Refer-enceMa-terial Provision	Environmental Particle Analyses	Environmental Bulk Samples Analyses	
					Radioactive	Not Radioactive
AEA Technology Harwell, UK	X		X	**)	X	X
Atomic Energy Commission Laboratory (AECL) Chalk River, Canada		X				
Air Force Technical Applications Center (AFTAC), USA				X		
Atomic Weapons Establishment (AWE), Aldermaston, UK				X		
Bundesanstalt fuer Materialforschung (DAM), Germany	X					
CEA Laboratories (Marcoule & Saclay), France			X			
CEA Laboratories (Bruyères la Chatel & Valduc), France	X			**)		X
***) DOE Network of Analytical Laboratories, (DOE SANES), USA					X	X
Institute for Reference Materials and Measurements (IRMM), EU			X			
KFKI Atomic Energy Institute, Budapest, Hungary		X				
**) V.G. Khlopin Radium Institute (KRI), St. Petersburg, Russian Federation	X		X		X	X
Laboratory for Microparticle Analysis, Moscow, Russian Federation				X		
DOE New Brunswick Laboratory (NBL), USA			X			
Netherlands Energy Research Foundation, Petten, Netherlands	X				**)	**)
Nuclear Material Control Center, Safeguards Analytical Laboratory, Tokai, Japan			X			
Nuclear Research Institute (NRI), Rez, Czech Republic	X					
Austrian Research Centre (OeFZS), Austria	X					
Safeguards Analytical Laboratory (SAL), IAEA	X				X	X
Transuranium Institute (ITU) Karlsruhe, EU				X	X	X
VTT Chemical Technology, Efpoo, Finland				**)		**)

**)   *Laboratories in the process of adding environmental sample analysis qualification*
***)   *SANES includes the following DOE Laboratories: Lawrence Livermore National Laboratory (LLNL); Los Alamos National Laboratory (LANL); Oak Ridge National Laboratory (ORNL); Pacific Northwest National Laboratory (PNL); Savannah River Technology (SRT)*

Table II: Main Analytical Techniques used by the Safeguards Analytical Laboratory

Analytical Technique	Analysed for:	Type of Material	Uncertainty (%rel.)	
			Random	Systematic
Elemental Analysis				
NBL Davies and Gray titration	U	U, U-Pu, U-Th[a]	0.05	0.05
MacDonald and Savage titration	Pu	Pu materials[a]	0.1	0.1
Controlled potential coulometry	Pu	Pure Pu materials	0.05	0.05
Ignition gravimetry	U, Pu	U oxides, Pu oxides	0.05	0.05
K-edge X ray densitometry	U, Th, Pu	U, Pu, U-Pu, U-Th[a]	0.2	0.2
K X ray fluorescence analysis	Pu	Pu materials[a]	0.2	0.2
Wavelength dispersive X ray fluorescence spectrometry	Pu, U	Pure U and Pu oxides, and MOX[a]	0.3	0.3
Isotopic Analysis				
Isotope dilution mass spectrometry	U, Pu	Spent fuel input solutions and Pu and U-Pu materials	0.1	0.1
Thermal ionization mass spectrometry	U and Pu isotopes	All Pu and U materials, and spent fuel input solutions	0.05[b]	0.05[b]
High resolution gamma ray spectrometry (Ge detector)	Pu isotopes, Am, Np	Pure U and Pu materials	0.5-2.0	0.5-2.0
Gamma ray spectrometry (NaI detector)	^{235}U	Low enriched U materials	0.2-0.5	0.2-0.5
Alpha spectrometry	^{238}Pu	Pu materials	0.2	0.3
Laser fluorimetry	Np	Pu materials	2.0	2.0
Plutonium (VI) spectrophotometry	Pu	Pu, U-Pu[a]	0.2	0.2

a   Except spent fuel.
b   For ratios of major isotopes.

of similar selectivity and accuracy.

Recent Developments in Nuclear Material Analytical Methods and Procedures at SAL

Significant developments took place in practically all fields of measurements. A few examples are described below. More comprehensive information is given in references 2 and 9.

Uranium by Potentiometric Titration

The New Brunswick Laboratory Davies and Gray titration is the basic method for the determination of uranium content in gram-sized samples of all types of nonirradiated materials. A third version of an automated titration system (Figure 1) was developed and assembled in SAL. It achieves relative precisions and accuracies of 0.05% or better, when aliquots containing 10 mg to 40 mg of uranium are titrated.

The success and reliability of the system stem in particular from the following original features:

- The temperature of the solution at the beginning of the Fe(II) reoxidation step is measured and used to determine the heating time for the next sample beaker;
- The titrant burette is calibrated gravimetrically and the temperature of the titrant is measured at the time of the titration and used to calculate the effective density and volume of the titrant delivered;

- The titration is calibrated against aliquots of a uranium standard solution prepared by weight from U metal NBL CRM 112A and titrated along with the samples.
- The titrant standard solution is prepared by weight from potassium dichromate NBS CRM 13 6e. This allows the measurement of the residual chemical bias of the titration.
- Aliquots of a control solution prepared by weight from UO2 pellets, CRM EC-1 10, are titrated along with the samples for an independent quality control.

The latter four measures ensure the traceability of the titration with a direct link to the International System of Units. In addition, they provide continuous monitoring of the validity of the traceability chain for the complete analytical procedure, including the weighing and dissolution of the subsample taken for the analysis. The first two features provide an effective control and correction of the maj or thermal effects in the absence of a strict and costly temperature control of the entire system or entire room.

Controlled Potential Coulometry of Plutonium

The instrument and procedure installed at SAL by a team from the Savannah River Laboratory[10] are used to verify the titer of plutonium standard solutions needed for the calibration of the MacDonald and Savage titration or for the preparation of large-size dried spikes. Four-mg aliquots of plutonium are measured

with a relative precision and accuracy of 0.1% or better. Typically the mean recovery of the coulometry of 47 aliquots of a number of plutonium standard solutions was equal to 100.00% with a standard deviation of 0.06%. A follow-up task of the U.S. Support Program is underway to develop components that will facilitate the maintenance of the instrument and its electrical calibration. Assistance is also requested to improve the precision and accuracy for the assay of 1-mg-size Pu aliquots. J. Tushingham at AEA Harwell was able to obtain relative repeatabilities of 0.1% for the coulometry of 1-mg aliquots of plutonium with relative biases of +1- 0.05% or less with the Harwell coulometer. This work was done in support of SAL under a task of the British Support Programme intended to develop a coulometer in line with a robotized anion exchange system for eliminating chemicals that may interfere in the coulometry of plutonium.[11,12]

Plasma Torch Inductively Coupled Mass Spectrometer
A PQ3 instrument of Vacuum Generators was purchased in 1996 and installed in SAL in December 1997 with the assistance of the British and French governments. Thirty-eight trace elements can presently be determined in dilute solutions of uranyl nitrate with detection limits of 0.5 fig/kg to 3 mg/kg of uranium. The plan is to test next the sensitivity of the system for potential analyses of Np/Pu, Am/Pu and Cm/Pu ratios in small samples of spent fuel and high-active waste solutions. This will be done before introducing the procedures of metal impurity analyses in samples of plutonium nitrate solutions.

Pu(VI) Absorption Spectrophotometry
In 1996, a Bentham spectrophotometer was installed with a dedicated personal computer and a custom-made software by a team from Dounreay in completion of a U.K. Support Programme task. Samples containing 0.05 g/L of plutonium can be analyzed with a repeatability of 0.015%. With samples con-
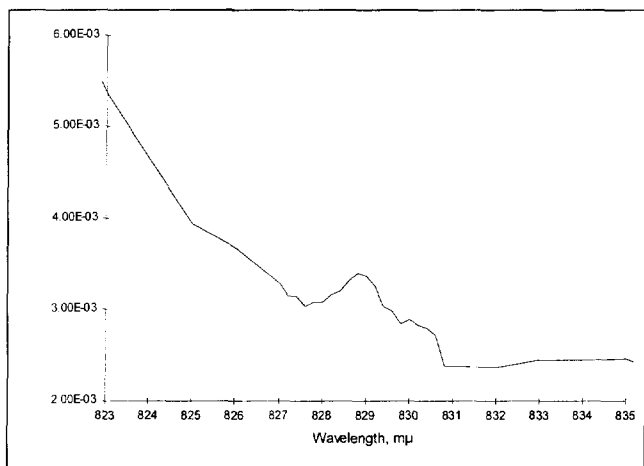
taining 0.02 mg/L to 0.2 mg/L of plutonium, the standard deviation of the measurement is typically 0.005 mg/L. It is therefore expected that a limit of detection of 0.015 g/L can be achieved in the absence of interfering elements like Nd (Figure 2). La Hague is sharing with SAL its experience in the application of this technique to analyze separated plutonium solutions, high-active liquid wastes and other process solutions. Activities in this area will continue in collaboration with the laboratories of the Tokai Reprocessing Plant and the Nuclear Material Control Center for potential applications at the on-site laboratory in Rokkasho, Japan.

Cm/Am/Np Separations from Samples of Spent Fuel and High-Active Liquid Waste Solutions
J. La Rosa[13] demonstrated the feasibility of separations using column extraction chromatography with TOPO to collect Pu and U, and recovering Cm, Am and Np from the effluents on a TRU column. In these tests the amounts of recovered Cm, Am and Np were measured by alpha spectrometry of fluoride coprecipitates with Nd fluoride.

Thermal Ionization Mass Spectrometry
The basic calibration parameters of both MAT 261 instruments were remeasured.[14,15] Measurable changes with time were observed for the relative cup efficiency factors used to collect the major isotopes. The revised parameters reduce the small biases that had appeared progressively on the quality control charts. They give in particular very consistent results of measurements of 233U/235U isotope ratios of the IRMM 072 series over a 100-fold change, using the 235U/238U ratio as an internal standard. The results of measurements of the same reference materials, the NBL 128 and NBL 144 plutonium isotopic standards using the procedure of total sample consumption are practically bias-free. This confirms that mass fractionation effects are very small, if present at all, in the conditions of current applications of this method at SAL. Discrepancies however

Figure 2: Absorption Spectrum of Pu(VI) in a nitric acid solution in the presence of Nd(III) [Concentrations: Pu(VI) 0.028 mg/L; Nd(III) 400 mg/L; NO_3- 3M]
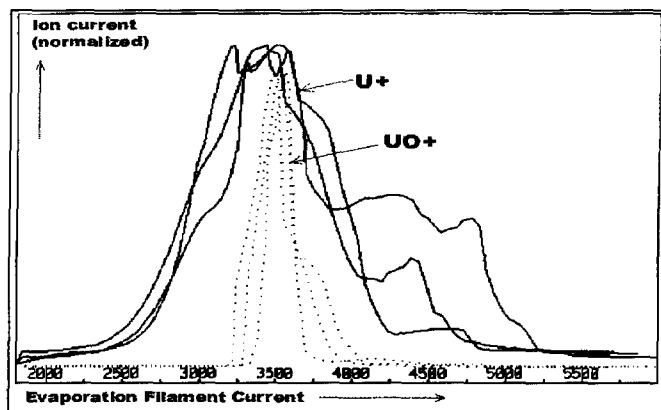

Figure 3: U+ and UO+ signal distribution of a uranium sample measured in total evaporation mode.

were found with the IRMM 290 and IRMM 047a isotopic standards of plutonium. These are being discussed with IRMM.

A Finnigan MAT 262 RPQ+ instrument was installed recently in SAL controlled area and is being tested. A software program was developed which allows the simultaneous recording of the emission of actinide metal ions, M$^+$, and actinide oxide ions, MO$^+$ (Figure 3). This feature will be extremely useful to optimize the preparation of the sample and its measurement so that the emission of metal ions is strongly favored.

Radiochemical Measurements

The characteristics of the low-level gamma spectrometer were measured. The Compton suppression system reduces the baseline due to residual "natural" radiation by a factor of 4 to 10. This has the potential to reduce the limits of detection by a factor of 2 to 3 compared to the limits reachable with the passive shielding alone.[16] However the "annulus" scintillator used for Compton suppression was found to contain small but unacceptable amounts of uranium. Its gamma ray emission is minute and would be negligible in all other applications than trace uranium detection. This was therefore considered as a serious defect and the annulus was returned to the supplier for a replacement free of charge. Alpha spectrometry and high-resolution gamma spectrometry were used to determine compliance of uranium oxide with nuclear fuel specifications. The limits of detection of direct measurements without chemical separations are given in Table III.

Table III: Detection Limits Of Some Radionuclides In Uranyl Nitrate Solutions

Gamma spectrometry	
Radionuclide	**Detection limit, Bq/g U total**
Zr-95	2.0
Ru-103	1.2
Ru-106	6.8
Sb-125	2.5
Cs-134	0.9
Cs-137	0.9
Tl-208	0.8
Th-234	72
U-235	3.5
Am-241	8.2
‡-spectrometry	
Radionuclide	**Detection limit, µg/kg**
U-232	0.3

Design of an On-site Laboratory at the JNFL Plant at Rokkasho-Mura, Japan

Within the framework of the IAEA project and with the cooperation of the Japanese government, a joint working group has been established to develop the technical specifications for the

equipment of an on-site laboratory at Rokkasho-Mura, Japan, and to prepare the procedures and arrangements for joint operation by IAEA and the Japanese National Safeguards Authority. The counterpart in this task is the Japanese Atomic Energy Bureau of the Science and Technology Agency, assisted by the Nuclear Material Control Center, the NMCC's Safeguards Analytical Laboratory and representatives of the plant owner, Japanese Nuclear Fuel Ltd. Representatives of all parties meet regularly. The conceptual design of the OSL was agreed upon in July 1997. JNFL will lease about 600 m2 of total space and heavy equipment in its own laboratory building for housing the OSL. Studies of the detailed design and costing of the fixed installations, heavily shielded hot cells, sample pneumatic transfer system, tong and glove operated alpha active boxes are progressing. A list of the main analytical equipment and instruments with an estimate of their expected cost was agreed upon in November 1997 (Table IV). Milestones for the definition of the specifications of analytical equipment and instrumentation, their procurement, installation and acceptance have been proposed. The same was done for the development of joint-use procedures. The general requirements of the IAEA for a joint use of the OSL are being revised as the bilateral discussions progress and as the IAEA develops its concepts for the safeguarding of nuclear materials in the separation, MOX conversion and waste process areas.

Table IV: Agreed List of Equipment for OSL/RRP

Category of Equipment	Equipment	Number of Units
Analytical Instruments	Automated Sample Treatment Systems	3
	HKED	2
	Pu(VI) Spectrophotometer	1
	Mass Spectrometers	2
	Density-meters	2
	Analytical Balances	6
	Potentiometric Titrator	1
	‡ - Spectrometer	1
	High Resolution Gamma Spectrometer	1
Ancillary Equipment	Computerized Data System	1
	HF Plastic Bag Sealer	2
Laboratory Furniture/ Installations	Hot Cells	5
	Glove Boxes	16
	Laboratory Benches (1.5 m long units)	11
	Fume Hoods	4
	Small Laboratory Equipment Set	1
	Reagent and Supplies Shelves and Cupboards	10
	Nuclear Material Storage Safes	2-3

Methods of Analysis of Swipes and Environmental Samples

Environmental sampling was introduced in 1996 as one of the new IAEA safeguards measures that contribute to the confirmation of the absence of undeclared nuclear material or nuclear activities. Collection of environmental samples at or near a nuclear site, combined with ultrasensitive analytical techniques such as mass spectrometry methods, particle analysis and low-level radio-metric techniques, can reveal signatures of past and current activities in locations where nuclear material is handled.

Initial implementation of environmental sampling for safeguards is focused on the collection of swipe samples inside enrichment plants and installations with hot cells. It is anticipated that implementation will be extended to other types of nuclear facilities and may include the collection of other types of environmental samples (e.g. vegetation, soil and water) exterior to facilities and sites. Samples are analyzed either in bulk or particle mode, depending on the sampling objectives and the activity levels of the swipes. Bulk analysis involves the analysis of an entire sample, usually by gamma ray spectrometry or isotope dilution thermal ionization mass spectrometry; the analytical measurements represent average results for the material contained. Particle analysis relies on the detection and analysis of individual particles in the micrometer-size range and provides as a measure of the isotope ratios of uranium and/or plutonium in these particles.

SAL Clean Laboratory

The SAL Clean Laboratory was inaugurated in December 1995 with the goal of providing a Class 100 clean-room capability for the provision and certification of sampling kits and for the receipt, screening and distribution of environmental samples from safeguards inspections. This facility significantly reduces the risk of crosscontamination that might lead to incorrect safeguards conclusions. The Clean Laboratory consists of over 200 m^2 of laboratory space, with approximately 100 m^2 at Class 100 cleanliness level (Figure 4). The laboratory is equipped with a suite of analytical techniques, including ‡, ·, gamma and X-ray fluorescence spectrometry. scanning electron microscopy with electron probe analysis and high-sensitivity thermal ionization mass spectrometry.

Environmental swipe samples received at the Clean Laboratory are given a code number to maintain confidentiality about their origin. The samples are then measured by low-background gamma-ray spectrometry to detect the presence of actinide elements (primarily U and Pu) and fission or activation products (such as ^{60}Co, ^{137}Cs and ^{106}Ru); radioisotope excited X-ray fluorescence spectrometry is used next to find U, Pu or other important elements. Alpha/beta counting is also applied to the samples to detect actinides or ·-emitting isotopes such as ^3H, ^{90}Sr or $^{99}Tc^m$. Scanning electron microscopy is used to examine small particles removed from environmental samples. The size and morphology of these particles can be examined at high magnification and their elemental composition measured by X-ray fluorescence spectrometry using an electron probe attachment.

Following the screening measurements, subsamples are distributed to laboratories of the NWAL for more detailed analyses. Selected samples are chosen for measurement in SAL Clean Laboratory by isotope dilution thermal ionization mass spectrometry, using a highly sensitive instrument equipped with pulse-counting detection. The ultimate limit of detection of this method is in the 10^{15}-g range for U and Pu.

One of the main activities of the Clean Laboratory is the preparation of clean sampling kits for collecting environmental samples. A kit consists of all the supplies needed by an IAEA

inspector in the field: clean swipe cloths, plastic minigrip bags, clean room gloves, a sample data form, a pen and labels. A roll of aluminum foil is provided to establish a clean working surface. A different type of swipe-sampling kit is required for sampling inside hot cells, where the subsamples must be taken with remote manipulators and shipped back to the IAEA in a special lead-lined container because of their higher radiation level.

Isotopic and Elemental Analysis at SAL Clean Laboratory

Pulse-Counting Thermal Ionization Mass Spectrometry
Screening measurements are used to decide which samples should be sent for more detailed analysis. Thermal ionization mass spectrometry is used to measure U and Pu concentrations and isotopic compositions in environmental samples. Measurements of environmental samples require a very high sensitivity, extending into the 10^9-g and 10^2-g ranges. This is achieved by the use of special sample treatment procedures, electrodeposition of the sample elements on the mass spectrometer filament and use of a pulse-counting detection system with high detection efficiency. A Finnigan MAT 262 RPQ mass spectrometer is used for these measurements. Isotopic spikes (^{233}U, ^{242}Pu or ^{244}Pu) are added to the samples during chemical processing to ensure efficient recovery of U or Pu. The accuracy and precision of this technique are about 1-10% for a U or Pu concentration in the $10^{-9}g$ range and for the ratios of the major isotopes in the sample.

Scanning Electron Microscopy with Electron Probe Analysis
The Clean Laboratory is equipped with a Philips scanning electron microscope with wavelength and energy-dispersive X-ray fluorescence detectors. Particles of interest are removed from the sample by ashing or ultrasoneration and deposited on a conducting substrate for introduction into the electron microscope. Under high magnification (1,000-5,000x), the particles are examined and the backscattered electron signal is used to search for particles containing heavy elements. Heavy particles can then be measured by energy-dispersive X-ray fluorescence spectrometry to give a semiquantitative elemental analysis. Particles containing U or Pu can be identified in this way; their size and morphology, as well as other elements present, will give information about the process that created them. This type of analysis forms a part of the classical "particle analysis" approach which is applied in certain NWAL laboratories in conjunction with thermal ionization mass spectrometry.

Secondary Ion Mass Spectrometry for Particle Analysis
A secondary ion mass spectrometer is under procurement to equip SAL Clean Laboratory with a capability to determine the isotopic and chemical composition of single solid particles of mi-~ crometer size. Depending on the size of the particle, the relative precision and accuracy of this approach is expected to be 1% for isotopes at the 1% to 9% abundance level and up to 10% for minor isotopes.

Services Provided by NWAL and the Member States

Analyses of Nuclear Materials and Heavy Water Samples

Up to now, seven laboratories of the network (Table I) shared with SAL the analyses of the samples of nuclear materials and heavy water taken during inspection by the inspectors of the IAEA or the UNSCOM/IAEA Action Team. Harwell, BAM, KRI, Petten, Rez and OeFZS used to analyze together about 40% of the spent fuel samples. In total, 5% to 10% of the uranium samples were sent to Harwell, KRI, Petten, Rez or OeFZS for a determination of the concentration and isotopic composition of uranium. All impurity analyses in uranium materials were done at Bruyéres le Chatel or lately at Valduc. Harwell analyzed impurities in plutonium samples. Occasionally a few samples of plutonium or plutonium/uranium oxides were sent to Harwell or Rez for the determination of the concentration and isotopic composition of plutonium and uranium. All heavy water samples are analyzed in Budapest, with a few samples being sent occasionally to Chalk River.

Representatives of these laboratories are periodically invited as consultants to review with the IAEA the quality of past services, to identify desirable improvements and to examine how the needs of the near future can be met in the most cost-effective way.[17,18]
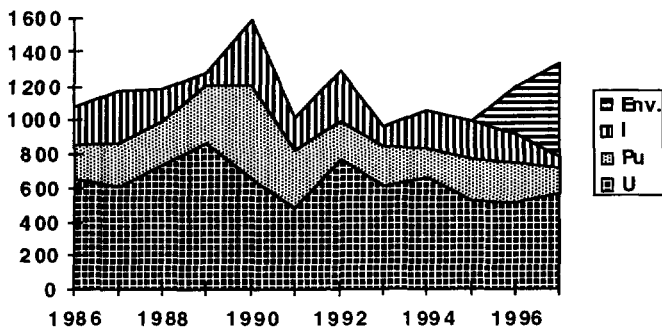

Figure 5: Number of samples received annually by SAL

Analyses of Swipes and Environmental Samples

All swipes and environmental samples are measured at SAL by the screening methods described above. All samples requiring more exhaustive analyses are sent to NWALs, with only a few replicate analyses being done at SAL for the time being. Swipes taken inside hot cells are sent presently to Harwell, SANES or ITU. KRI will be shortly in the position to receive this type of sample. Other swipe samples preferably are sent to AFTAC, Aldermaston, Moscow and ITU for single-particle analyses. ITU uses a CAMECA-1F6 secondary ionization mass spectrometer as described above. The other three laboratories apply the fission track method.

Fission Track Method

The technique involves first a step in which particles of interest, containing fissile isotopes such as ^{235}U or ^{239}Pu, are selected by the fission track method. These particles are then placed on a substrate for scanning electron microscopy and electron probe examination to measure the elemental composition. Particles of special interest are then mounted on the filament of a thermal ionization mass spectrometer for measurement of the isotopic composition of the U and Pu present, using a pulse-counting detection system. The abundance of the various isotopes can be estimated from the collected ion counts with a precision and accuracy of better than 1% relative for isotopes of 1% to 90% abundance in particles with a diameter of 1 μm to 5 μm. Particles with diameters down to 0.1 μm can be measured, but with less precision and accuracy.

Supply of Reference Materials

NBL, NIST (U.S.A.), IRMM (EU), CETAMA/CEA (France), ABA Harwell (U.K.), Khlopin Radium Institute in St. Petersburg (Russian Federation) and NMCC-SAL (Japan) provided lately the reference materials needed for IAEA Safeguards Analytical Services. Close to 30 different reference materials were delivered in 1996-1997 in response to purchase orders or to requests to member state support programs placed during this period or earlier. The present stocks at SAL and the identified needs were reviewed with representatives of suppliers members of NWAL during the Consultants' Meeting of April 1998.[19]

The most urgent needs for reference and quality control materials for the analysis of environmental samples concern materials carrying trace amounts of uranium and plutonium with certified isotopic compositions. These needs are being addressed with the assistance of the EURATOM/IAEA Cooperation Program and the British and U.S. technical support programs to IAEA safeguards. The Institute for Reference Materials and Measurements in Geel, Belgium, prepared and certified basic reference solutions containing low concentrations of pure ^{233}U, ^{242}Pu tracers, or uranium and plutonium with isotopic compositions close to the compositions found in the environment. The IRMM certified values were confirmed by replicate isotopic analyses done by Los Alamos National Laboratory, New Brunswick Laboratory and SAL. Aliquots of these standard solutions were diluted further by SAL to produce control solutions and swipes suitable for verifying the accuracy of isotopic bulk analyses. The need for reference and control materials for particle analysis is indeed even more urgent, but such materials are considerably more difficult to prepare and certify. IRMM prepared glass beads containing controlled amounts of uranium of accurately certified isotopic composition. These have been ground to produce particles with diameters in the range of a micron. Their certification is in progress. IRMM will then prepare similar glass particles carrying plutonium of known isotopic composition. Harwell is producing micron-sized particles of uranium oxide dispersed on clean IAEA swipes. SAL and U.S. laboratories will participate in the characterization or validation of these materials as certified control materials. Some of these particle control materials should

become available for NWAL analyses in 1998.

Pending needs include more accurately certified materials for more accurate analyses of "direct access" nuclear materials at plants with large throughputs or inventories of HEU or Pu. In particular, there are needs for tracers of large size or of ^{244}Pu and ^{236}U isotopes for isotope dilution mass spectrometry with total sample consumption or internal calibration. SAL is concerned also about the need to produce new batches of reference materials in low stock such as the NBL 126 and the CETAMA MP2 plutonium metal chemical standards, the IRMM 072 series of high accuracy uranium isotopic standards and the NBL 131 certified ^{244}Pu spike. Technical possibilities of separation ^{244}Pu isotope exist at the Arzamas facilities in Russia. A technical proposal was adopted at the Consultants' Meeting in Vienna in May 1996.[20] The IAEA is ready to obligate $200,000 (U.S.) over a three-year period from its regular budget to contribute to the cost of the production of separated plutonium isotopes, at the condition that up-front costs amounting to about $600,000 are covered by other sponsors, for example, the International Center of Science and Technology in Moscow.

Other Assistance by Member States
Maintenance and development of procedures for safeguards analytical services are currently done with the assistance of MSSPs. Six MSSPs sponsor tasks related to IAEA safeguards DA measurements. Nineteen tasks came to completion in 1996-1997. In December 1997 there were 12 active tasks with the same MSSPs, two U.S. tasks were on standby, and a total of 22 requests for new tasks were prepared for examination by eight different support or cooperation programs.

Quality Assurance Measures
Effective safeguards require that strict quality assurance measures be applied throughout the process of implementation of destructive analysis. These begin with the planning of the inspections, the taking of the samples, their shipment and apply to their handling and analyses at SAL and NWAL, and to the reporting and evaluations of the analytical results. Considerable efforts were invested in 1996-1997 to establish and document the required procedures and measures for environmental sampling. A parallel effort was made to update the documentation relative to the procedures and measures applicable to the destructive analysis of nuclear materials and heavy water. Training of inspectors was also stepped up to familiarize them well with the new DA procedures.

Documentation of Sampling Procedures
A third edition of the Safeguards Technical Report STR-69, Destructive Analysis and Evaluation Services for Nuclear Material Accountability Verifications, was issued by the Department of Safeguards in October 1996.[21] It includes guidelines for the taking and handling of samples of feed, product, waste and process materials encountered in nuclear fuel production and reprocessing plants. It serves as a reference in Section 4 of the Safeguards Manual, Part SMS,[22] Safeguards

Analytical Services, describing the activities that the inspectors perform to select the items to be verified by DA, obtain samples of these items and request their transfer to SAL. It is the responsibility of the relevant sections of the Divisions of Operations of the Safeguards Department to define, in consultation with the SAS Coordination Group, facility specific DA procedures consistent with SMS 4 and STR 69 and to document them in the relevant facility attachments.

A draft of a new section of the Safeguards Manual describes the activities to be carried out in environmental sampling.[23] Three working papers provide standard operating procedures to inspectors taking environmental samples. Issued in 1997 after intense in-house consultations, they incorporate the recommendations of consultants[24,25,26] who reviewed with the IAEA the experience acquired during the field trials conducted in 1993-1995 and during early implementation in the period of 1992 to 1996. Detailed working instructions describe the distribution of responsibilities and actions regarding the delivery of sampling kits to the inspectors, the tracking of the samples and their analyses, their integrity and the confidentiality of their origin.

Training of Inspectors
Ten to 15 new inspectors are appointed every year and attend the annual Introductory Course on Accountancy and Safeguards. During this course, the newly appointed inspectors attend lectures on the requirements and the statistical procedures for planning inspection verification measurements, and for the evaluations of the results of these measurements. They are also informed of sample shipment requirements and procedures and are trained in the use of installed, transportable and portable measuring instruments. SAL analysts brief them on the precautions to be taken when sampling nuclear materials and heavy water, and on the characteristics of typical methods of destructive analyses of such materials.

An intensive program was started in 1995 to train experienced inspectors in the taking and handling of environmental swipe samples. This continued in 1996 and 1997, with eight training courses carried out in IAEA headquarters and at Seibersdorf. Euratom inspectors were also trained in these sessions, and IAEA experts served as instructors in Euratom training courses held at the Institute for Transuranium Elements in Karlsruhe, Germany, in 1997 and 1998. Overall, more than 120 IAEA or Euratom inspectors were trained in environmental sampling for safeguards from 1996 to 1998.

NWAL and SAL Quality Assurance Documentation
The network was expanded in 1995-1997 to include six organizations that had been nominated by their sponsoring authorities in response to requests of the IAEA for the analysis of environmental samples. IAEA acceptance required demonstration of the expertise of the nominated organization in this type of analysis and the provision of a suitable quality assurance manual regarding the handling of the IAEA samples. All NWALs are expected to accept audits of their handling procedures and to analyze control samples that the IAEA includes along with the inspection samples.

The quality assurance coordinator of SAL conducts an annual internal audit of the quality assurance measures applied to samples of nuclear materials, in compliance with the relevant manual.[27] A customer's audit of these measures was made in October 1997 by representatives of the IAEA Department of Safeguards. In the frame of the U.S. Support Program to IAEA Safeguards, Radian Corp. developed, in collaboration with SAL staff, a quality assurance manual for SAL Clean Laboratory activities related to safeguards environmental sampling.[28] In this context, SAL has issued and implemented some 59 standard operating procedures. SAL quality assurance documentation will be revised in the next two years to conform to ISO 9000's Standard and ISO Guide 25 as it is integrated in the Quality Assurance System in development for the IAEA Laboratories at Seibersdorf.

Quality Control of SAL and NWAL Measurements

A batch of control sample is regularly analyzed along with each series of four to 10 batches of inspection samples of nuclear materials and heavy water. A batch of sample consists of one sample bottle in the case of heavy waters and nonirradiated uranium materials, or three sample bottles per sampled item in the case of high-active liquid wastes, spent fuel solutions and milligram-sized samples of nonirradiated plutonium materials. For U and Pu elemental assays in spent fuel solutions, SAL prepares control samples by mixing weighed amounts of a solution of a large-size dried spike with weighed amounts of a standard solution of a natural uranium chemical reference material like NBL 112a and of a standardized solution of plutonium nitrate with a high abundance of ^{240}Pu. Forty-six batches of such control samples were analyzed by SAL and NWAL in 1996-1997. A statistical evaluation of the precision and accuracy of the analyses of these control samples was presented and discussed (Table V) at the Consultants' Meeting of April 1998 on NWAL Services and Quality Control Program. In view of these results, the consultants recommended that the ITV for the random component of the uncertainties in IDMS of uranium and plutonium be decreased to 0.1%.[18]

In compliance with its standard operating procedures on measurement control, SAL also measures regularly reference materials for the purpose of calibrating the final analytical measurement technique and/or verifying the validity of the calibration parameters. Close to 3,000 control measurements of this kind are done every year. Table VI summarizes the most important of these measurements, made, respectively, in 1996 and 1997. Whenever SAL measured the same parameters with different reference materials, Table VI also can be used to confirm the con-

sistency of the certificates of these reference materials or to identify questions that may deserve to be discussed with the suppliers. A statistical evaluation of these control measurements is performed periodically to update the warning and control limits to be used in the next year for the various analytical measurements on samples of nuclear materials.[29]

SAL takes part in the analytical measurement evaluation program EQRAIN organized by la Commission d'Etablissement des Méthodes d'Analyse (CETAMA) du Commissariat & l'Energie Atomique, France. The EQRAIN program involves the receipt and analysis of blind control samples of uranyl nitrate, plutonium nitrate and mixed uranium/plutonium nitrate solutions. The elemental concentration of a uranyl nitrate solution is measured every three months, the plutonium and uranium/plutonium nitrate solutions every four months, along with inspection samples of uranium or plutonium products. Analyses of the control samples of runs 6 and 7 were completed and reported in 1996 and 1997. CETAMA provided the reference values of the control samples by fax within two days after SAL reported the results of its analyses. The final report[30] for the uranyl nitrate run No. 6 was received in spring 1996. Run No. 8 is currently underway.

At the Consultants' Meetings in April 1998,[18,19] the IAEA reiterated its plan of participating also in the REIMEP program administered by IRMM and in the SMEP program run by NBL. SAL priority interest would be to receive control samples of spent fuel solutions, UF^6, plutonium and mixed plutonium/uranium oxides. The IAEA expects that the measurement evaluation programs fulfill at least two requirements: Reference values exist before the participating laboratory reports its results, and the participating laboratory is informed of the reference value as soon as it has reported its results.

Similar control measures are introduced in the analyses of environmental samples at SAL and NWAL. All NWALs doing bulk analyses for IAEA safeguards environmental sampling received a first set of uranium and plutonium control samples. The results reported will be evaluated and discussed with the laboratories in the course of 1998.

Tasks Ahead

The appropriate balance of resources for nuclear material analyses vs. environmental sample analyses is one of the new challenges, as budget restrictions threaten already with the progressive closing of several NWALs. The laboratories traditionally analyzing nuclear materials are the ones in greatest difficulties. If they cease to work in this field, the tremendous capital of expertise that they hold will rapidly vanish. This would be dis-

Table V: Precision and accuracy of elemental analysis of quality control samples of simulated spent fuel solutions by isotope dilution mass spectrometry using LSD spikes (Coefficients of variation and relative difference to expected value, in %)

Element	Number of Laboratories	LAB effect	Spike batch	Spiking	Repeatability	Reproducibility	Mean difference
Pu	6	0.02	0.06	0.06	0.10	0.05	0.004
U	1	n.a.	VAR<0	0.03	0.05	0.06	0.040

Table VI: Typical results of quality control measurements

Reference Material	Measurand	1996			1997		
		N	bias	CV	N	bias	CV
Elemental assay by automatic titration (U: Davies & Gray, Pu: McDonald & Savage)							
NBL112A (U metal)	U assay	147	cal	0.03	340	cal	0.04
EC-110 (UO2 pellet)	U assay	76	-0.01	0.04	191	0.00	0.05
MP2 (Pu metal)	Pu assay	17	-0.02	0.11	10	cal	0.07
UK-Pu-1 (PU/Ga metal alloy)	Pu assay	80	cal	0.06	107	cal	0.07
Thermal Ionization Mass Spectrometry (TIMS), total evaporation measurement							
NBS947	240:239 Pu	105	0.03	0.04	90	-0.01	0.04
NBS947	242:239 Pu	106	-0.10	0.08	90	-0.16	0.17
UK-Pu-3	240:239 Pu	17	-0.03	0.03	15	-0.08	0.02
UK-Pu-3	242:239 Pu	17	0.13	0.06	15	0.03	0.04
UK-Pu-4	240:239 Pu	13	-0.06	0.03	15	-0.08	0.05
UK-Pu-5	240:239 Pu	15	-0.04	0.03	16	-0.09	0.02
UK-Pu-5	242:239 Pu	15	0.19	0.07	16	0.11	0.03
UK-Pu-5	244:239 Pu	15	0.28	0.05	16	0.23	0.05
UK-Pu-6	240:239 Pu	11	-0.02	0.04	14	-0.07	0.01
NBL128	242:239 Pu	20	0.10	0.07	15	-0.03	0.03
CBNM-047a	244:239 Pu	22	0.53	0.05	14	0.29	0.05
NBS010	235:238 U	101	0.00	0.13	86	0.03	0.06
NBS030A	235:238 U	71	-0.05	0.07	59	0.03	0.04
NBS200	235:238 U	10	-0.09	0.05	9	-0.01	0.06
NBS500	235:238 U	59	-0.05	0.06	65	0.03	0.04
NBS930	235:238 U	17	-0.08	0.08	20	-0.01	0.04
GUS3568	233:238 U	34	-0.05	0.13	60	0.03	0.05
GUS3568	235:238 U	34	-0.04	0.05	60	0.04	0.04
GUS3568	236:238 U	34	-0.01	0.05	60	0.02	0.03
CBNM-072/2	233:238 U	47	-0.11	0.06	63	-0.04	0.05
CBNM-072/2	235:238 U	47	-0.07	0.06	63	0.02	0.03
Isotopic Dilution Mass Spectrometry (IDMS)							
Spent fuel control spiked with	Pu assay	both years:			66	0.03	0.11
Large Size Dry (LSD) spike	U assay	both years:			65	0.04	0.07
UK82522 (Pu-Oxide)	Pu assay	16	0.15	0.11	13	0.00	0.06
UK82522 mixed with U	Pu assay	9	-0.02	0.11	7	0.11	0.23
NBL112A (U metal)	U assay	8	-0.03	0.14	7	0.14	0.15
High Resolution Gamma Spectrometry (HRGS) with MGA computer code for spectrum evaluation							
NBS947	241Am / Pu	99	0.21	0.41	83	0.16	0.47
SAL9984	241Am / Pu	99	-0.38	0.46	84	-0.34	0.36
NBS947	238:239 Pu	99	0.46	0.83	83	0.59	0.86
SAL9984	238:239 Pu	99	-0.58	0.46	84	-0.53	0.45
NBS947	240:239 Pu	99	1.48	0.86	83	1.36	0.82
SAL9984	240:239 Pu	99	0.32	0.56	84	0.42	0.59
NBS947	241:239 Pu	99	0.23	0.61	83	0.27	0.56
SAL9984	241:239 Pu	99	-0.60	0.45	84	-0.54	0.40
Alpha Spectrometry (activity ratio with respect to 239+240 Pu)							
NBS947	238 Pu	107	-0.16	0.24	93	-0.13	0.27
AEA90099	238 Pu	107	-0.48	0.18	92	-0.44	0.19

astrous for IAEA accountability verifications.

We otherwise face five analytical challenges: (1) reaching regularly 0.1% or better accuracy in the accountability of large amounts of plutonium and high-enriched uranium, (2) equipping and operating an on-site laboratory at the Rokkasho-Mura plant, (3) clarifying the precision, accuracy and limits of detection required for environmental sample analyses, (4) developing reference and control materials suitable for checking the analyses of solid particulates in environmental samples, (5) identifying the signatures and procedures appropriate for wide-range detection.

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An Introduction to Focused Approach to Verification Under FMCT

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Abstract

The proposed Fissile Material Cut-off Treaty is one of the most important items on the multilateral disarmament and non-proliferation agenda. Successful achievement of an FMCT would be an important step towards the goal of eliminating nuclear weapons. Taking a focused approach to verification of an FMCT as the point of departure, this paper discuses the purpose and scope of verification activities at specific types of nuclear fuel cycle facilities. The discussion covers fissile-material-production facilities (enrichment and reprocessing plants), associated R&D facilities, and downstream nuclear fuel cycle facilities that may handle fissile material produced after entry-into-force of an FMCT. The paper also discusses the relationship between FMCT verification and IAEA safeguards.

This paper reflects the views of the authors and does not necessarily represent Australian government policy.

Introduction

Clearly the purpose and scope of verification activities under an FMCT will be determined by the purpose and scope of the FMCT itself. Since formal negotiation of an FMCT has not commenced as yet, it cannot be foreseen with any certainty how these may eventuate. This paper deals with a key objective of the FMCT, to proscribe the production of fissile materials for use in nuclear weapons or other nuclear explosive devices. Key terms here, the interpretation of which will directly affect the scope of the FMCT, are *production* and *fissile material*. These terms are discussed further below.

In considering issues of purpose and scope, we are mindful that the proposed FMCT has a different purpose to *comprehensive safeguards* applied by the IAEA, i.e. under INFCIRC/153-type safeguards agreements which in almost every case have been concluded by non-nuclear-weapon states party to the NPT. Comprehensive safeguards are designed to provide assurance against horizontal proliferation, i.e. the acquisition of one or more nuclear weapons by a state in violation of commitments not to acquire such weapons. The sensitivity of comprehensive safeguards, reflected in technical parameters such as goal quantities, detection probability, and inspection frequency, has been set accordingly. In the case of states with comprehensive safe-

guards agreements, together with additional protocols (INF-CIRC/540), the FMCT is not expected to involve any additional commitments, i.e. satisfaction of CSA requirements would meet FMCT requirements.

With states outside comprehensive safeguards, including the recognized nuclear-weapon states, the issues are quite different. Here, the concern is essentially vertical proliferation. The FMCT will address this by providing assurance that the stocks of fissile material held outside international verification will not increase. The FMCT will make a major contribution to establishing the conditions under which further nuclear disarmament, involving all relevant states, will be both possible and encouraged.

We have argued elsewhere[1] that a focused approach, covering only the most proliferation-sensitive facilities and materials, is both appropriate and technically adequate for the FMCT. Such an approach would involve application of verification measures to all production facilities (i.e. facilities with enrichment and reprocessing capabilities), coupled with verification of unirradiated direct-use material produced after entry-into-force (EIF)[2], and supported with measures to detect possible undeclared production. Taking the focused approach as the point of departure, this paper discusses the purpose and scope of verification activities at specific types of nuclear fuel cycle facilities.

Apart from questions of the purpose and scope of verification activities under the FMCT, an essential issue to be settled will be the method of implementation of these activities, particularly which entity would have this responsibility. Since many of the FMCT verification activities would be similar to IAEA safeguards, it is difficult to imagine there is a viable and credible alternative to the IAEA as the verification agency for the FMCT, but this is one of many matters subject to negotiation, and is not discussed further in this paper.

Focused Approach To Verification Of An FMCT

As mentioned earlier, key terms determining the scope of the FMCT are production and fissile material. In this regard we have taken as starting points the following assumptions:

- *fissile material* would be given a meaning specific to the FMCT, reflecting the purpose of the FMCT, i.e. proscription of fissile material production for nuclear

weapons. If existing IAEA concepts such as *source material* (e.g. natural uranium) or *special fissionable material* (e.g. low enriched uranium), were applied under the FMCT, the result would be a treaty of very wide scope, similar to comprehensive safeguards. Likewise, application of the FMCT to *irradiated direct-use material* (e.g. spent fuel) would result in comprehensive coverage (essentially covering all reactors). Under a focussed approach, we envisage the fissile material subject to the FMCT would correspond primarily to the IAEA definition of *unirradiated direct-use material*, i.e. separated plutonium, U-233 and highly enriched uranium;

• *production³* in our view would mean *future* production, i.e. production *after* the FMCT's EIF. While some argue for possible coverage of pre-existing fissile material, it is clear inclusion of such material would not be acceptable to the NWS and others whose support is essential to any effective FMCT regime. In light of the discussion of *fissile material*, production would primarily mean production of separated plutonium, U-233 and HEU, i.e. *reprocessing* or *enrichment*, which are the processes by which these materials are produced.

Under a focused approach, therefore, international verification would be applied to:

• *all unirradiated direct-use material produced after EIF*, in both peaceful and non-proscribed military activities, including material in production and downstream facilities used for storing, processing, utilisation, or disposal of subject material; and

• *all production facilities*, i.e. facilities that are, have been, or could be capable of producing unirradiated direct-use material, including all decommissioned, shut-down, and future facilities with enrichment and reprocessing capabilities.

The fissile materials defined above would become subject to verification under the FMCT when:

• plutonium, HEU, or U-233 contained in irradiated material is introduced into a reprocessing plant or any other facility capable of separating subject material from fission products;

• plutonium, HEU, or U-233 contained in active waste is introduced into any facility capable of recovering and partitioning of these materials from fission products;

• any uranium is introduced into a uranium enrichment plant or any other facility capable of uranium isotope separation; or

• any plutonium is introduced into any facility capable of plutonium isotope separation (NB this latter point is included for the sake of completeness, but plutonium isotope separation, especially when this could be used for conversion of reactor-grade into weapons-grade plutonium, would seem inimical to non-proliferation objectives and we question whether in principle this should be accepted as a legitimate civil nuclear activity).

We also propose that fissile materials produced after EIF should cease being subject to verification under FMCT:

• upon *irradiation* of the fissile material in a nuclear reactor or other intense neutron source to a level to be specified;

• upon *blending* of HEU or U-233 with depleted, natural, or low enriched uranium so that the resulting uranium no longer meets the definition of fissile material (e.g., contains less than 20 percent of the isotope U-235); or

• upon a determination by the verification agency that fissile material has become *practicably irrecoverable*.

Appropriate definitions would have to be developed for specific types of facilities and materials mentioned above.

The figure on the following page illustrates the application of our focused approach to a typical nuclear fuel cycle with monitored production facilities and downstream facilities that may use fissile material produced after EIF, and with facilities outside verification arrangements.

Notes:
1. The figure necessarily simplifies complex situations, and should be read in conjunction with relevant parts of the text of the paper.
2. Pre-existing material would not be subject to verification — this figure should not be interpreted otherwise. However, if pre-existing material is processed or co-located with subject material, the verification arrangements will need to incorporate a way of distinguishing between the two.

Possible Fissile Material Acquisition Paths
Use by a state party to FMCT of any fissile material produced after EIF for nuclear weapons or other nuclear explosive devices would constitute a violation of that state's obligations under the FMCT. Thus an effective verification system associated with FMCT should be capable of verifying:

• that no fissile material is produced outside international verification after EIF; and

• that fissile material produced after EIF is not diverted for use in nuclear weapons or other nuclear explosive devices.

In assessing possibilities for application of measures along the lines of those available under conventional and strengthened safeguards, we find it useful to think in terms of acquisition paths or combinations of actions that a proliferator might undertake in order to divert for weapons fissile material produced after EIF. The principal acquisition paths which might be exploited by a State contemplating violation of its obligations under the FMCT are as follows:

• diversion from declared production, i.e. diversion of fissile material produced and declared after EIF either from a monitored production facility or from a monitored downstream facility declared handling such material;

• undeclared production of fissile material after EIF, either through undeclared activities at a declared and monitored facility, or through operation of a clandestine production facility.
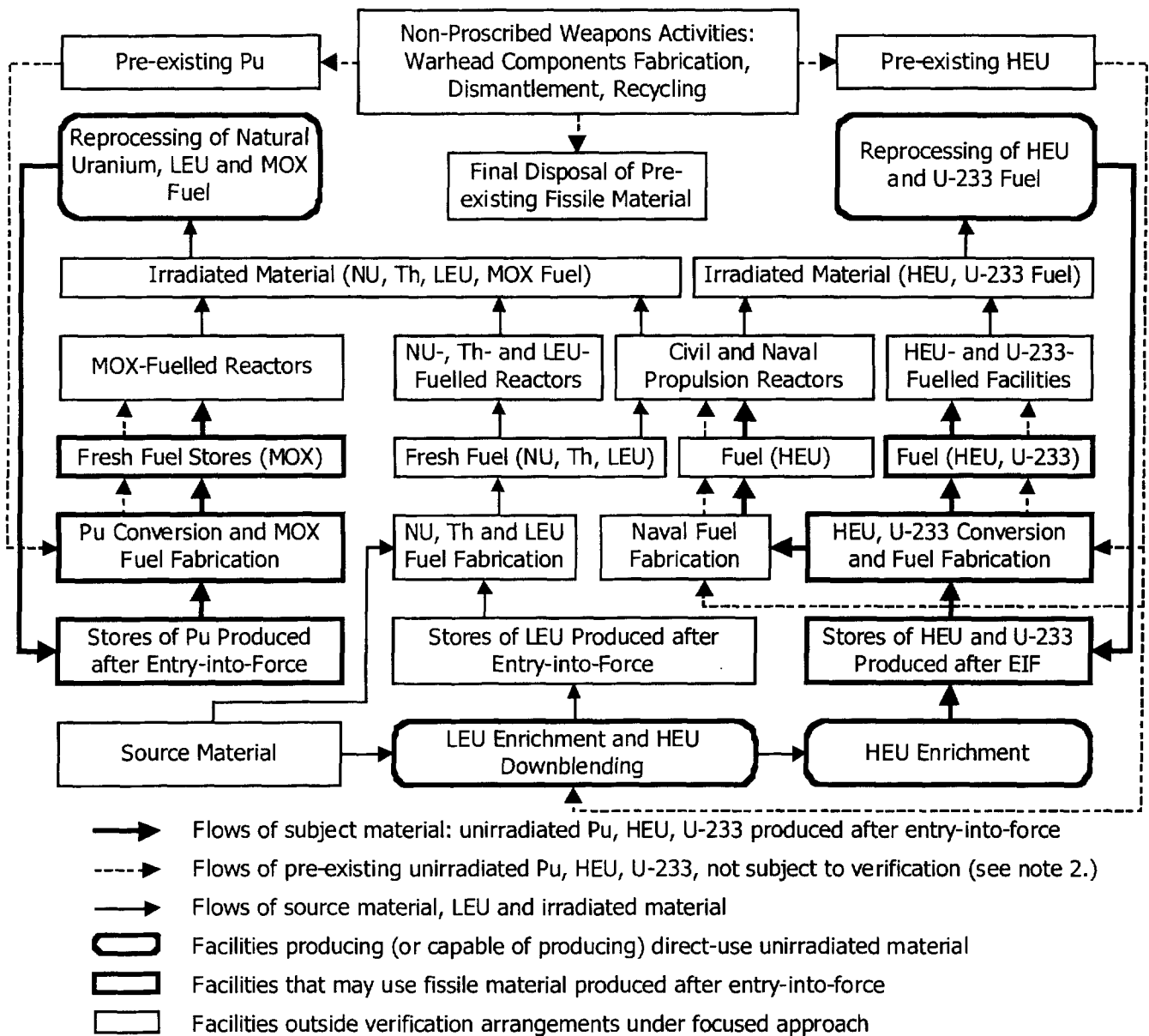
Figure 1. Representation of the Focused Approach to Verification of the FMCT

Monitoring Production Facilities

Possible verification measures to cover these potential fissile material acquisition paths are discussed in the following sections of this paper.

Monitoring Production Facilities

Separation of plutonium, HEU or U-233 from irradiated material and production of HEU for non-explosive purposes will not be proscribed by the FMCT. However, we believe, monitoring would have to be applied to *all enrichment* and *reprocessing activities* — probably even regardless of the presence of nuclear material — to ensure there is no undeclared production of separated fissile material after EIF. This would not only involve large-scale military, dual-use, commercial, pilot and test enrichment and reprocessing plants, but should probably also include hot cells above a certain size (to be defined) and other related R&D facilities, for which the verification agency established that nuclear material could be easily introduced and significant quantities of fissile material produced.

We expect that under the FMCT each state party would be required to provide a declaration covering all its reprocessing and enrichment facilities with detailed information on their operational status, current capacity, plans for possible future operations, extensions, modifications or decommissioning. It is not clear at this stage, however, what level of detail might be required
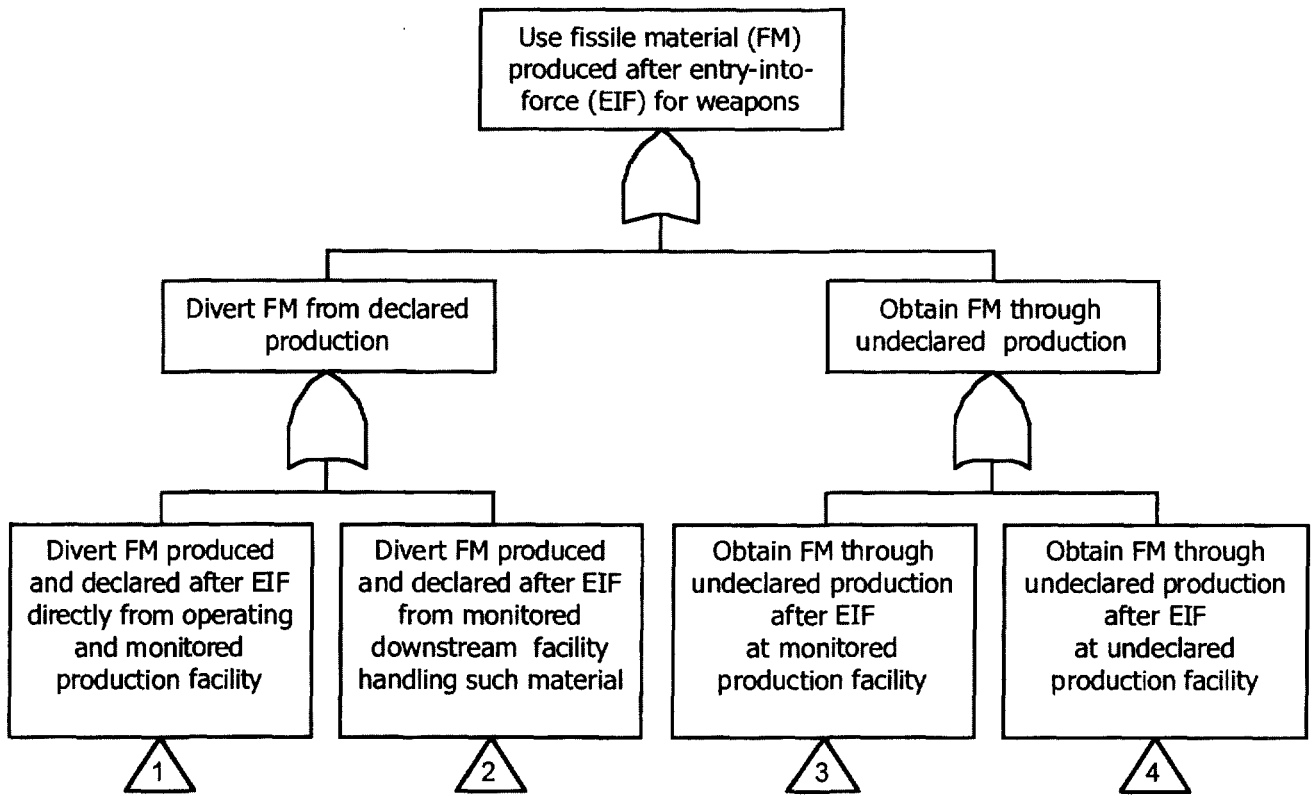
Figure 2. Logic Tree Describing Principal Fissile Material Acquisition Paths Relevant to the FMCT

for the information on the operating history of those facilities. The States' declarations would obviously have to include information on plans to construct and operate new production facilities. At some stage, of course, States party to the FMCT will have to provide regular design information needed for verification.

Operational Facilities
We consider that under FMCT verification arrangements for **operating production facilities,** modelled on NPT safeguards,[4] should be sufficient to detect possible:

- diversion of fissile material from declared flows and inventories of plutonium and uranium at an operating and monitored facility; and
- undeclared production of fissile material at a declared and monitored facility.

The verification arrangements for operating reprocessing plants would depend upon the scale of operations, but they would have to be intrusive and resource intensive in most cases. Former military reprocessing plants may pose a special challenge, as they are typically very poorly equipped with calibrated tanks and other instrumentation required for verification purposes.

At enrichment plants declared to be producing HEU for non-proscribed purposes, verification arrangements should be sufficient to ensure that all enrichment activities are verified and that all HEU and LEU produced is properly accounted for.

At enrichment plants declared to be producing only LEU, monitoring might be focused only on verifying the non-production of HEU. Environmental sampling could be instrumental for determining whether or not an enrichment plant produces HEU. One would need to develop different monitoring arrangements to verify that only LEU is produced in a facility that: has never produced HEU; was reconfigured from a former HEU production facility; or is co-located with an operating HEU production facility.

While LEU would generally be outside the verification arrangements under the focused approach, an issue to be examined is whether in some circumstances the ability to provide assurance of non-production of HEU might be enhanced if, in addition to measures aimed at detection of the presence of HEU within or around the LEU enrichment plants, the monitoring arrangements also included monitoring certain LEU-related operations.

While reactors would not be defined as production facilities under the focused approach, and generally would not be subject to monitoring (except when unirradiated fissile material subject to FMCT is present — see below), it may be necessary to consider some form of monitoring for reactors which are particularly capable of generating weapons-grade plutonium, such as some on-load refuelling reactors, and fast breeder reactors (FBRs) using blanket fuel for plutonium production.
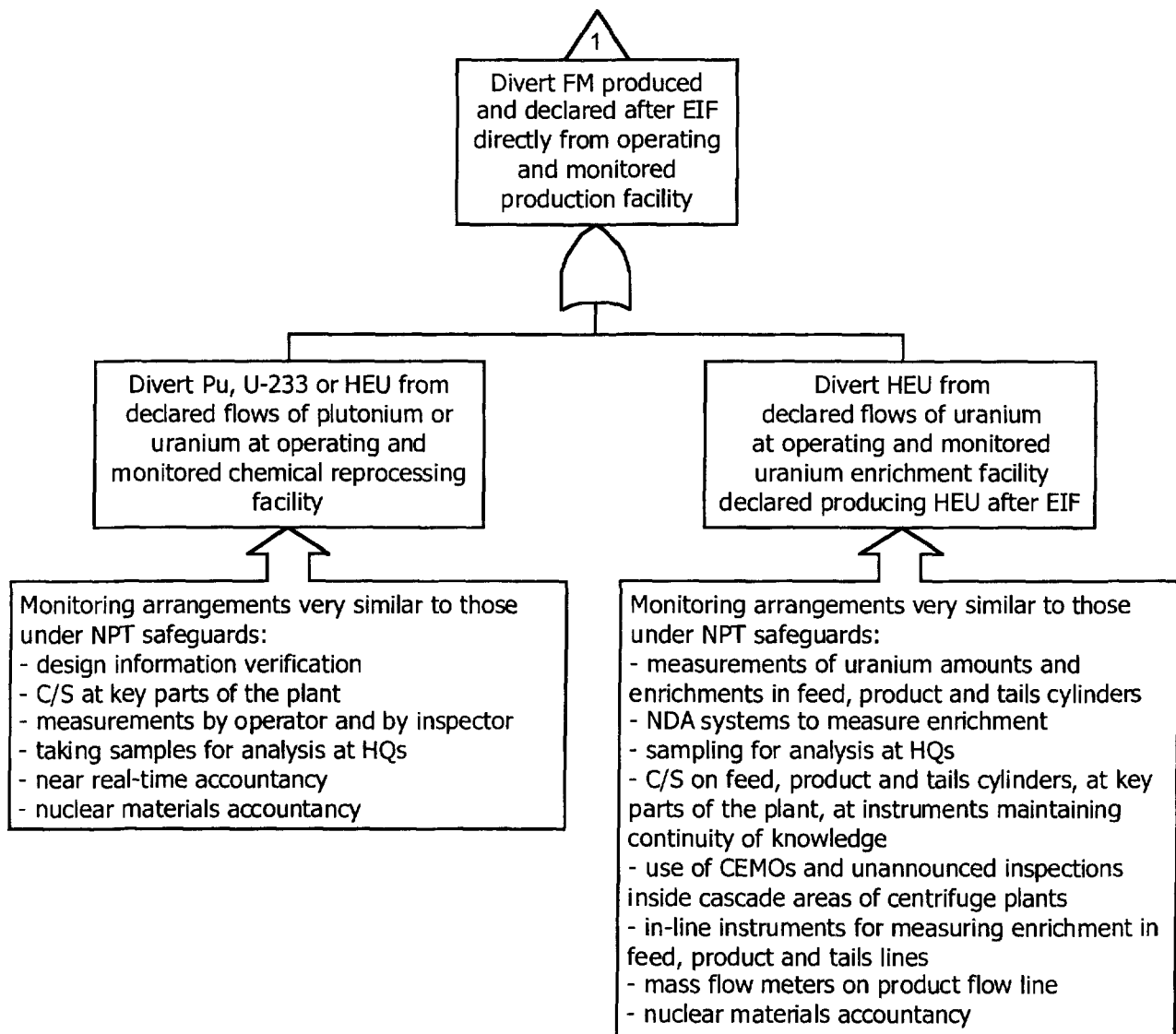
Figure 3. Fragment of the Logic Tree Related to Diversion from Operating and Monitored Facility

Shut Down and Decommissioned Facilities

When designing the monitoring approach for a facility declared inoperable, one will have to start with confirmation of its declared operational status. At closed-down and decommissioned production facilities the focus should be to confirm that no undeclared production occurs. A former reprocessing facility may be operated for non-reprocessing purposes. It can be held on operational stand-by. It can be decommissioned or abandoned.

Verifying that former production facilities remain shut down would be relatively simple. Appropriate monitoring approaches could be developed for each facility taking account of its status. If the verification agency concludes on the basis of available information that there is little chance a former production plant could resume operations without major renovations, very infrequent visits underpinned by periodical analysis of satellite imagery may provide sufficient assurance that this plant does not resume operations clandestinely. When a plant is maintained on operational stand-by it could probably be restarted with less effort and would obviously require more intrusive monitoring to provide assurance that it is not used for undeclared production. The cost and inspection effort required would vary from negligible cost for decommissioned or abandoned facilities up to more intensive inspections and equipment needed to verify that no undeclared production occurs at a former reprocessing facility being operated for non-reprocessing purposes.

Monitoring the Use of Subject Material in Downstream Facilities

As mentioned above, FMCT verification would have to follow all fissile material produced after EIF through downstream
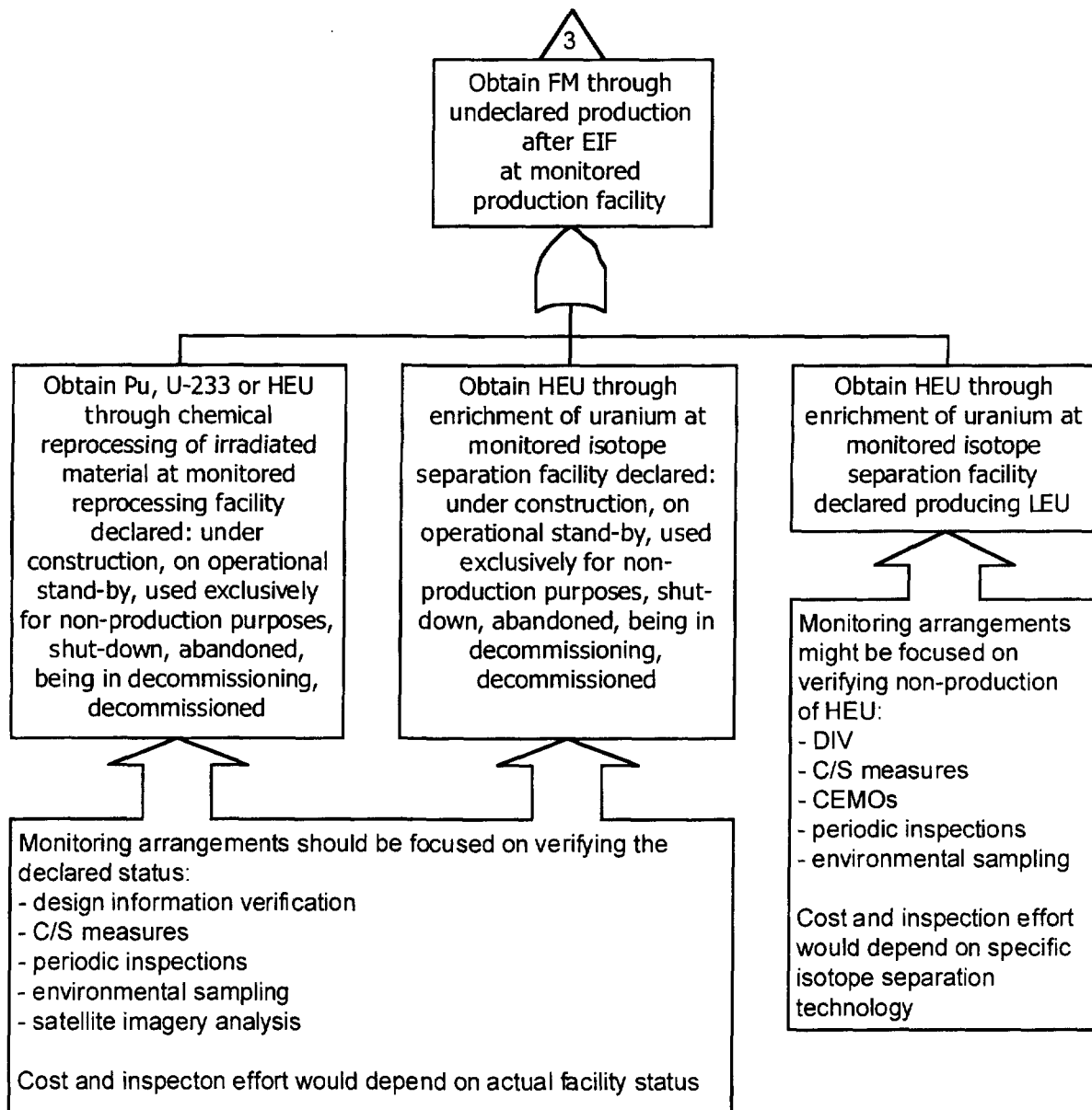
Figure 4. Fragment of the Logic Tree Related to Undeclared Production at Monitored Facility

facilities up to the point of termination of verification. Accordingly, we expect that the following types of facilities will come under monitoring, with material balance verification on the said subject material (subject to managed access requirements) at least for the period they handle fissile material produced after EIF:

* stores of separated fissile material;
* plutonium, HEU and U-233 conversion and fuel fabrication facilities;
* fresh fuel stores at research reactors, critical assemblies, and other reactors (e.g. FBRs and LWRs) using fuel containing plutonium, HEU or U-233.

States' declarations under the FMCT would have to provide

the verification agency well in advance with design information on all facilities, where fissile material produced after EIF is planned to be stored, processed, used or disposed of.

The case of HEU produced after EIF for use in naval propulsion raises difficult issues. The production of all HEU should be subject to appropriate verification arrangements at the enrichment plant. When new production is for naval propulsion, we believe monitoring should be maintained until the material enters naval fuel fabrication facilities, otherwise there could be a significant gap in the FMCT's coverage. However, the design of naval fuel is highly classified, and neither fuel fabrication nor the fuel itself could be verified using conventional nuclear materials accountancy, as that would require provision of sensi-

tive information. Thus, a considerable amount of work is needed to devise and implement verification using alternative principles, perhaps drawing on ideas which are being put forward for the verification of nuclear disarmament (e.g. the "Trilateral Initiative" involving the US, Russia and the IAEA.) Any eventual reprocessing of naval fuel would also necessarily be subject to monitoring at the reprocessing plant.

Measures Against Undeclared Production

In addition to the activities described above, the FMCT verification regime will obviously include measures aimed at detection of possible undeclared production facilities. Clearly the existence of an undeclared reprocessing or enrichment plant would be a violation of a State's obligations under the FMCT. We believe this should be the case even if the facility is not in operation, given the speed with which operations could be restarted. Thus under the FMCT the verification agency should be given sufficient legal authority for acquisition and analysis of all information related to possible undeclared production and for managed access to sites of concern. We expect that the access provisions included in the Additional Protocol and in the Chemical Weapons Convention might serve as an appropriate model for managed access under the FMCT.

A clandestine production facility may be detected through a combination of means including *inter alia*:

- analysis of satellite imagery that could reveal some characteristic structures of a production plant and trigger verification measures such as location-specific environmental sampling;
- wide-area environmental sampling, e.g. measures aimed at capturing characteristic gaseous effluents and particulates that may be deposited at significant distances from the facility;
- monitoring of international supply of reprocessing or isotope separation-related equipment; and
- acquisition and analysis of open source information, supported by information provided to the verification agency by States.

As under the provisions of the Additional Protocol, the verification agency would have the right to request managed access to locations to resolve inconsistencies in the information available to it.

Conclusion

The FMCT will have different impacts on individual states due to the enormous variance in their nuclear fuel cycles and the associated fissile material inventories. The challenge is how to negotiate a treaty that will achieve results favourable for all participants, given that interests and priorities vary so much. We consider that our focused approach, involving the monitoring of all production facilities, coupled with verification of unirradiated direct-use material produced after EIF, and supported by measures to detect possible undeclared production, is technically adequate for the FMCT. In our view the focused approach is the appropriate point of departure for the development of monitoring arrangements under the FMCT for specific types of nuclear fuel cycle facilities.

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2. Of course, states party to FMCT would retain their right to voluntarily offer the IAEA to place any of their unirradiated direct-use material under international safeguards, but, we believe, this is does not need to be part of the FMCT arrangements.

3. In accordance with the above definition of subject material, we refer to the following activities as "production of fissile material" under an FMCT: separation of plutonium, HEU or U-233 from irradiated nuclear material; recovery of plutonium, HEU or U-233 from active waste; increasing the abundance of the isotope U-235 in uranium through any isotope separation process; and increasing the abundance of the isotope Pu-239 in plutonium through any isotope separation process. Note that this definition excludes nuclear production, ie production of plutonium and U-233 in nuclear reactors or through the use of any other intense neutron sources. Although LEU would not be subject to the FMCT, all uranium enrichment activities would be subject to monitoring to provide assurance there is no undeclared HEU production.

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IEST 46th Annual Technical Meeting and Exposition, Rhode Island Convention Center and Westin Hotel, Providence, Rhode Island, U.S.A. Sponsor: Institute of Environmental Sciences and Technology. Contact: Joan Harpham; phone, 847/255-1561; fax, 847/255-1561; E-mail: iest@iest.org; Website, http://www.iest.org.

May 1–4
Light Water Reactor Fuel Performance International Topical Meeting, Park City, Utah, U.S.A. Sponsors: American Nuclear Society (ANS) Fuel Cycle and Waste Management and the ANS Materials Science and Technology Divisions in co-operation with the International Atomic Energy Agency, Atomic Energy Society of Japan, Korean Nuclear Society, and Atomic Energy Council of the Republic of China. Contact: P.E. MacDonald; phone, 208/526-9634; E-mail: pem@inel.gov.

May 3
ASTM Committee E-41 on Laboratory Apparatus, ASTM Headquarters, West Conshohocken, Pennsylvania, U.S.A. Contact: Dan Smith, 610/832-9727; E-mail: dsmith@astm.org.

May 3–5
Nuclear Energy Assembly, the Drake Hotel, Chicago. Sponsor: Nuclear Energy Institute. Contact: Tanya Caruso; phone, 202/739-8039 or E-mail: tmc@nei.org.

May 9
Nuclear Operations Seminar, London, U.K. Sponsor: British Nuclear Energy Society. Contact: BNES; phone, 0171 665 2241; E-mail: tillbrook_a@ice.org.uk.

May 22–24
ASTM Committee E-20 on Temperature Measurement, Sheraton Centre Toronto Hotel, Toronto, Canada. Contact: Felicia Quinzi, 610/832-9738; E-mail: fquinzi@astm.org.

May 22–26
Second Russian International Conference on Nuclear Material Protection, Control, and Accounting, Sponsors: Minatom, Russian Federation, U.S. Department of Energy, American Nuclear Society, INMM, and Russian Nuclear Society. SSC-RF—Academy A.I. Lepunski Institute for Physics and Power Engineering. E-mail: pshakin@ippe.obninsk.ru or ira@ippe.obinsk.ru.

May 23–25
Nuclear Technology 2000 Annual Meeting, Bonn, Germany. Sponsor: INFORUM GmbH. Contact: INFORUM GmbH; phone, 0049 228 507 223; E-mail: 100672,1424@compuserve.com.

June 4–8
ANS Annual Meeting, San Diego, California, U.S.A. Sponsor: American Nuclear Society. Contact: ANS at 708/352-6611, fax 708/352-0499, or Website: http:// www.ans.org.

June 12–16
4th U.S. Department of Energy International Decommissioning Symposium, Knoxville Convention Center, Knoxville, Tennessee, U.S.A. Sponsor: U.S. Department of Energy. Contact: Call 305/348-3752; E-mail: elaine@eng. fiu.edu, or write IDS 2000, Florida International University, Hemispheric Center for Environmental Technology, 10555 W. Flagler St., Suite 2100, Miami, Florida 33174.

June 19–21
World Engineers' Convention 2000. Hanover, Germany. Sponsors: VDI The

Association of Engineers and Expo 2000 Hanover. Contact: VDI The Association of Engineers, P.O. Box 10 11 39, D-40002 Duesseldorf, Germany; call +49 221 6214-440; fax +49 211 6214-167; E-mail: tagungen@vdi.de; Website: http://www.vdi.de/wec/.

July 16–20
41st INMM Annual Meeting, The Hilton Riverside New Orleans, New Orleans, Louisiana, U.S.A. Sponsor: Institute of Nuclear Materials Management. Contact: INMM; phone, 847/480-9573; fax, 847/480-9282; E-mail; inmm@ inmm.org; Website: http://www.inmm.org.

August 30–September 10
25th Annual Symposium of the Uranium Institute, London, U.K. Sponsor: Uranium Institute. Contact: UI; phone, 0171 225 0303; E-mail: ui@uilondon.org.

September 18–20
4th Conference on AeroSpace Materials, Processes, and Environmental Technology (Formerly the Aerospace Technology Conference),Von Braun Center, Huntsville, Alabama, U.S.A. Sponsors: Marshall Space Flight Center, NASA Operational Environment Team, NASA's Materials and Processes Working Group, Office of Space Flight, NASA Headquarters, National Center for Advanced Manufacturing, American Institute of Aeronautics and Astronautics, American Society of Metal International (R), Aerospace Industries Association, Environmental Protection Agency, National Center for Manufacturing Services, Sandia National Laboratories, Society for Advancement of Materials and Process Engineering, and the University of New Orleans. Contact: Jodi Weiner at 256/533-5923 (voice); 256/534-9899 (fax); E-mail: jweiner@aol.com, or http://ampet.msfc. nasa.gov.

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