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1958-1979

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EDITORIAL

Safeguards Lessons Learned From Three Mile Island

By Dr. William A. Higinbotham
Brookhaven National Laboratory
Upton, Long Island, New York

Three-Mile-Island was not a disaster, or even a near disaster. It **was** a very costly accident in terms of dollars and in terms of public confidence. What should be distressing to nuclear scientists and engineers is that the experts were not able to evaluate the severity of the risks for several days.

Every potential energy source has its advantages and disadvantages. People tend to accept the costs of using fossile fuels, not because the costs are small, but because they have been accepted in the past. It is too early to assess the risks of solar power, employed on a large scale. It is obvious that nuclear power implies the generation and control of large amounts of radioactivity, and that nuclear energy is associated with nuclear explosives. What can nuclear material managers learn from Three-Mile-Island?

There are some similarities between safety and safeguards, and some differences. The possible consequence of a reactor accident or of sabotage of a reactor are similar. Design of reactors in the interest of safety, makes them more resistent to sabotage attempts. There will be a lot of work on safety measures, now, some of which might improve sabotage resistance and some of which might not. It will be important to keep both safety and safeguards objectives in mind.

Three-Mile-Island emphasized the need to consider human factors, and the need for national plans to deal with emergencies, as well as the need for improved hardware. There will now be strenuous efforts to improve the ability of state and Federal agencies to respond to reactor incidents. What about response to theft or seizure of plutonium?

Human factors play a much larger role in safeguards analysis than they do in analysis of reactor safety. In the safeguards case, humans are the adversary and humans are the defense. Safeguards R&D has tended to concentrate on hardware solutions. T.M.I. should remind us how much safeguards depends on the interest, the understanding, and the motivation of our people.

The excellent record of reactor safety is due to the early recognition of potential problems and to the attention concentrated on the development of redundant and reliable safety features. The excellent record of safeguards, however, may be due to a paucity of adversaries, rather than to the robustness of our defenses. The lack of incidents means that assessment must be based on intuition, examination of scenarios, etc. or does it?

Many of the things that happened at T.M.I. had occurred at other reactors previously, though not all at once. Other critical values have been left turned off and other automatic valves have failed to close. In the future, such incidents will be studied more closely. Perhaps there are

(Continued on Page 106)

Dr. Higinbotham

Institute Continues Growth and Vitality

By G. Robert Keepin, Chairman
Institute of Nuclear Materials Management
Los Alamos, New Mexico

By the time this summer issue of the Journal rolls off the press, the Institute's 20th Anniversary meeting in Albuquerque will be history—and judging from the excellent technical program and the high level of interest being expressed both in the U.S. and abroad, I'm confident the record will attest to another landmark INMM meeting in 1979. Many topics of direct interest to INMM members—such as the formation of technical working groups, professional certification, standards, etc. are to be discussed at the Annual Business meeting in Albuquerque, and some of the results and conclusions therefrom will be summarized in the next [Fall 1979] Issue of the Journal.

As noted in my Annual Report to the Institute (presented at the Annual Business Meeting in Albuquerque) one good sign of a healthy, vital organization is member interest and participation in the election-of-officers process. In this year's election we had a record number of ballots cast (approximately 350) as well as a record percentage returned (nearly 60%). As announced at Albuquerque, our two newly-elected Executive Committee members are **Yvonne Ferris** of Rockwell International, Rocky Flats, and **Sam McDowell** of DOE Safeguards and Security, Washington, D.C. We are most fortunate to have the participation and input of these well-known safeguards and materials management experts in Institute planning, oversight and management, and we extend them both a hearty welcome to the INMM Executive Committee. At the same time we want to thank the two outgoing members of the Executive Committee, **Bill DeMerschman** of HEDL and **Dennis Wilson** of GE-San Jose for the excellent service and leadership they have provided during a period of very significant accomplishment and expansion of our Institute.

Keepin

Further evidence of the vitality of the INMM is the prospective formation of a Central Region Chapter of the Institute. This Chapter would encompass an area within a 300 mile radius of Cincinnati and would include cities as far distant as the Chicago, Pittsburgh, and Knoxville areas. Following an organizational meeting held recently in Cincinnati, a petition to form a Central Region Chapter was prepared for formal submission to the INMM Executive Committee at the Albuquerque meeting. With the Japan, Vienna, and Pacific Northwest Chapters already well underway, establishment of a Central Region Chapter will bring to four the total number of INMM Chapters around the world.

Let me turn now to quite a different topic that has lately drawn considerable attention and emphasis within the NRC, in Congressional Committees, the GAO, and elsewhere. In the wake of Three Mile Island, greater emphasis is to be placed on tightened regulations, upgrading of operator qualification and training requirements, better instrumentation and controls, etc. in the areas of reactor safety, emergency planning and preparedness. This general thrust toward tighter nuclear regulations and controls will undoubtedly also mean increased emphasis on upgrading of system performance criteria and operational capability in the areas of safeguards and materials management. Similarly, by analogy with current trends in reactor safety, concern for the more "conventional" safeguards areas of materials accountancy and control, physical security, containment and surveillance, may now be augmented by increased emphasis on contingency planning and response, emergency response capabilities, etc.

Common to both reactor safety and nuclear materials safeguards is the key role of plant operators in implementing effective in-plant safety and safeguards. In this connection I want to cite here the timely editorial by **Ralph Lumb,** President of NUSAC, Inc. (cf. p. 11, this issue) stressing the need for better safeguards education and training of plant operating personnel, greater awareness of safeguards and security requirements, alertness to unusual operating conditions, etc. Lumb indicates that too little is being done to stimulate and

(Continued on Page 106)

VICE CHAIRMAN'S COLUMN

20th Annual Meeting Program Efforts Successful

By G.F. Molen, Vice Chairman
Institute of Nuclear Materials Management
Barnwell, South Carolina

At this writing, our Annual Meeting in Albuquerque has not yet occurred; however, when you read this article the meeting will be fresh in our memories. As I reflect on all the preparations that have gone into the Albuquerque meeting and try to visualize the outcome of that meeting, I am struck by the tremendous talents that are contained within the Institute as a body. At this year's Annual Meeting, we have taken a new tact in several areas. We had a program with about as many Contributed Papers as we have ever had, and even with this, we still had to turn down a significant number of papers. We also had several invited sessions which covered a full range of topics and interests to those who are involved in the field of nuclear materials safeguards. As Chairman of the Annual Meeting Committee, it made me feel proud to see the way the several individuals of the Annual Meeting Committee gave unceasingly of their efforts. I would particularly like to commend **John Jaech**, as Chairman of the Program Committee and also **Bill DeMerschman,** who was Chairman of the Invited Papers Session and **Dick Chanda,** who was Chairman of the Contributed Papers Session. These gentlemen did an outstanding job.

Not only were the program efforts successful but also those many and varied meeting arrangements efforts which go toward making any meeting a real success have to be recognized for a superb job. Our Meeting Arrangements Chairman, **Joe Stiegler**, was ably assisted by **Roy Crouch**, as Local Arrangements Chairman, **Duane Dunn**, as Registration Chairman, **John Glancy**, as the Exhibits and Display Chairman, and last but certainly not least, **Tom Gerdis**, the Communications and Publicity Chairman. These gentlemen all deserve credit for the very fine job they did.

I think most anyone would agree with me that the Albuquerque meeting was certainly a success by most any measure of performance. The cooperation extended to us by Sandia and Los Alamos Scientific Laboratories in providing the very informative and interesting tours added a special bonus to the overall meeting. We want to thank these Laboratories and especially those staff members who gave so much of their time to make the tours a success. In particular, we would like to thank **Tom Sellers** of Sandia and **Darryl Smith** of LASL.

We are already beginning our planning for the next Annual Meeting which will be held at the Breakers' Hotel in Palm Beach, Florida. John Jaech will again serve as the Program Chairman, and Joe Stiegler will serve as the Meeting Arrangements Chairman. I am quite sure that these gentlemen will use their experience gained during the past year to improve the 1980 Annual Meeting. Therefore, we should look forward to that meeting with real anticipation. It certainly is not too early for you to express your interest and support in these meeting preparation activities. I would personally like to urge any of you who would like to become more active in the Annual Meeting planning activities too please contact me, John Jaech, or Joe Stiegler about what roll you might play. After all, the meeting is for you and you are the one who can make it the best meeting ever.


Chanda


Crouch


DeMerschman


Dunn


Gerdis


Glancy


Jaech


Molen


Stiegler

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INMM SECRETARY GETS SPECIAL PLATES—Vincent J. DeVito, Piketon, Ohio, is the proud owner of new Ohio license plates bearing the acronym, INMM. As far as the editors of this Journal know, Mr. DeVito is the first in this country to get INMM plates. If you know of others, please inform the editors. We congratulate Mr. DeVito for his zeal giving the Institute a little more visibility in his area of the country. Perhaps you may wish to consider getting "INMM" plates in your state or country.

SECRETARY'S CORNER

INMM Officers Re-elected

By V.J. DeVito
Secretary of INMM
Goodyear Atomic Corporation
Piketon, Ohio

According to Article III, Section 6, of the INMM Bylaws, "The Secretary shall notify each member in good standing of the results of the election by November 15 of each year." This notice in the Journal shall be construed as having fulfilled that obligation.

In accordance with Article III, Section 4, of the INMM Bylaws, the selection of candidates for the elected positions on the Executive Committee (officers and members) was properly received by the Secretary. The Nominating Committee selected the following slate of candidates:

For Chairman — **Robert Keepin**
For Vice-Chairman — **Gary Molen**
For Secretary — **Vincent DeVito**
For Treasurer — **Edward Owings**

For members of the Executive Committee:

Yvonne Ferris
Samuel McDowell
Tommy Sellers
Stanley Turel

In accordance with Article III, Section 5, a ballot was mailed to each of the Institute's 583 members of which 341 returned ballots.

There were no petitions for candidates to be added to the ballot; however, there were several write-ins.*

As a result of the balloting, the officers and the members of the Executive Committee for the terms of office beginning July 1, 1979, are as follows:

Chairman — **Robert Keepin**
Vice Chairman — **Gary Molen**
Secretary — **Vincent DeVito**
Treasurer — **Edward Owings**

Executive Committee Members at Large:

Dennis Bishop to June 30, 1980
Frank O'Hara to June 30, 1980
Yvonne Ferris to June 30, 1981
Sam McDowell to June 30, 1981
Roy Cardwell — Immediate Past Chairman

(Continued on Page 106)

DeVito

*For Chairman: **Edward Young, Shelly Kops, Herman Miller, Vincent DeVito, Ralph Lump,** and **Gary Molen.** For Vice-Chairman: **Larry Kull, Roger Moore, David Klein.** For Secretary: **Roy Cardwell.** For Treasurer: **Duanne Dunn** and **Lynn Vaught.** For Members at Large (Executive Committee): **Dave Zeff, Charles Vaughn.**

Special One-day Seminar In Late September

The following report was prepared by **Yoshio Kawashima** and **R. Hara,** officers of the Japan Chapter of INMM.

A working group has been organized under the leadership of Professor **Ryohei Kiyose** of the Department of Nuclear Engineering at the University of Tokyo. The group is discussing the proposed program for the first seminar on nuclear materials management in Japan.

The one-day seminar will be held during the last week in September with the date and place of the meeting to be announced later. INMM members in other nations who wish to participate in the seminar should notify the Japan Chapter, care of the Nuclear Material Control Center, Akasaka Park Building, 2-3-4 Akasaka Minato-ku, Tokyo, Japan.

The following topics will be tentatively covered in the program:

•Current Status of Nuclear Materials Management in Japan and its Problems.

•International Development of Nuclear Materials Management Programs.

Kiyose

•Safeguards Information Processing and Treatment.

•Review of Safeguards Equipment and Instruments for Fissile Materials Measurement.

•Reports of the Nuclear Materials Accounting Systems Currently Employed at Nuclear Installations-Power Reactor and Nuclear Fuel Utility Companies, and Foreign Installations.

Reports from overseas organizations are welcome. The program will include review papers on each topic followed by discussion. An emphasis will be placed on discussions and exchange of information among the participants. Abstracts of papers will be prepared.

Sinai Joins NUSAC

McLEAN, Va.—Dr. **Ralph F. Lumb,** President of NUSAC, Incorporated has announced the appointment of **Samuel B. Sinai** as a Senior Technical Associate in the Security Programs Division. Mr. Sinai's responsibilities will include the development of security programs in the computer, industrial and loss prevention fields.

Mr. Sinai comes to NUSAC from RMS International, Inc., a Virginia based consulting firm, where he was a principal involved in security services to the U.S. Government and companies throughout the world, offering expertise in security systems design, industrial security and systems management.

Mr. Sinai holds a B.A. degree from UCLA and master's degree from the University of Southern California.

NUSAC Incorporated provides staff and management consulting services for the nuclear power industry and security services to industry at large. Its services include physical security programs, security system development, security management audits, executive protection and loss prevention programs.

Sinai

Smith

James Smith Joins NUSAC

McLEAN, Va.—Dr. **Ralph F. Lumb,** President of NUSAC, has announced the appointment of **James L. Smith** as Senior Technical Associate in the Quality Programs Division of the firm.

Mr. Smith's responsibilities will include quality assurance audit and surveillance services during fabrication of fuel assemblies, ASME Code components, and Q-List safety-related items. In addition, Mr. Smith will be responsible for assisting NUSAC's clients in initially qualifying and auditing manufacturers and suppliers of ASME nuclear code materials to NCA-3800 requirements.

Prior to joining NUSAC, Mr. Smith was employed by Bechtel in Gaithersburg, Maryland and in San Francisco, California. At Bechtel, Mr. Smith developed and conducted ASME Code training programs, performed qualification and surveillance activities of manufacturers and suppliers of ASME nuclear code materials, and had overall supervisory responsibilities for the supplier quality related activities on four major nuclear projects.

NUSAC, Incorporated is a consulting firm providing services in the areas of Quality Assurance, Physical Security, Nuclear Materials Safeguards, and UF_6 confirmation.

NUSAC was acquired in December as a wholly-owned subsidiary of The Wackenhut Corporation, the Coral Gables, Florida security firm. NUSAC headquarters remain at 7926 Jones Branch Drive, McLean, Virginia, 22102.

Vienna Members Active In ESARDA, INFCE

By Iain Hutchinson
Vienna Chapter of INMM

Summer has at last come to Vienna. After the long gray overcast winter some warm sunshine is more than welcome. There are those who claim that Vienna also had a Spring this year. If so we failed to notice it.

Organization of the Vienna Chapter continues. A proposed Constitution and By-Laws has been distributed to the membership for comment, and unless there is a sudden wave of opposition it should be adopted within the next few months. No meetings have been held since the organizational meeting last January, but that activity should start on a regular schedule in the fall.

The ESARDA Symposium in Brussels last April undoubtedly is reported elsewhere in the INMM Journal, but a few lines must be taken to express congratulations to the organizers. The technical agenda, although crowded, was excellent, and the meeting was very well attended. It was also good to see a sizeable U.S. turnout. Plans for Edinburgh in 1980 are already underway; we hope to see you there.

It is hard to believe that it has been almost two years since the organization of the International Nuclear Fuel Cycle Evaluation (INFCE). The schedule called for each working group to submit its report to the Technical Coordinating Committee by 1 June. The universal laws governing the writing of committee reports being what they are, one can imagine the intensive writing and editing campaigns which have taken place in the last two months. The TCC is scheduled to meet in late July, and the closing General Conference is planned for February 1980. At least some of the working groups are operating on the assumption that in between there will be another round of working group meetings to edit and re-write reports, for example in the light of what other groups have concluded in related areas. Those of us who have been trying to attend two or even three meetings at the same time are thankful for even a summer to catch up on other work.

This letter will end with a word about what the majority of INMM members in Vienna are doing, namely performing international safeguards inspections. The glamour, as always, is in the development projects, the negotiations, the meetings, etc. When the system studies and the negotiations are over, however, it is the inspector who must spend half his life running from hotels to facilities to airports, implementing the grandiose plans others have developed. Anyone can (or should be able to) load film into a movie camera, but have you ever tried doing it while wearing a lab coat, hard hat, gloves, etc., and while standing at the top of a 15-20 foot ladder in a hot room? Have you ever arrived in country X, only to discover that the NDA instrument you checked out had been bent during transport and that the nearest source of spares and service was 3000 miles away?

So, what are INMM members doing in Vienna? Some of us are enjoying the prestige of developing instruments or systems, or attending important meetings. The rest are doing what has to be done, namely performing inspections!

Hutchinson

Improvements Can Be Made By Involvement

By Thomas A. Gerdis, Editor
Nuclear Materials Management
Manhattan, Kansas

During the past year, four issues of **Nuclear Materials Management**, journal of INMM, the 1978 INMM Membership Directory, and the Proceedings of the 1978 Annual Meeting held in Cincinnati, Ohio, were published by the INMM Publications Office at Kansas State University, Manhattan.

A fourth issue of the Journal was added this past fall. The issue was well accepted and is now a regular part of the Journal operation.

Two years ago, a goal of increasing the number of technical articles to 4-6 per issue was set by myself and Dr. **William A. Higinbotham** of Brookhaven National Laboratory. The Journal has reached the point where there is considerable technical material being made available for publication.

Perhaps the single most important achievement of the past year has been that it is apparent that more professionals working in the field want to have their work published in this journal. To us, that is quite an achievement in seven short years. The Journal first came out in April, 1972 with a 16-page issue. The winter 1978-1979 issue was 120 pages plus cover. It has been concluded that 120 pages is a bit large for us at this time. The current desirable issue size is in the range of 72-96 pages.

Much of the credit for the improving quality of the journal goes to Dr. Higinbotham who has done an excellent job as Technical and Editorial Editor of the publication. We all should be very grateful to "Willy" for his service to the safeguards profession. Without a doubt, he was very deserving of the INMM Distinguished Service Award for 1979 which he received at this annual meeting of the Institute for service in safeguards over the years.

Another very special person in the development of the Journal has been Dr. **Eugene V. Weinstock** of Brookhaven National Laboratory. His development of the book review section and summaries of seminars and other meetings has been very valuable. His counsel is deeply appreciated. Many others have written articles and reports for the Journal on a regular basis. Scores of names could be listed here.

We do want to stress, however, that the Journal is still striving to improve, particularly in editing, proofreading, and meeting the distribution deadlines. The editors are not sitting back and resting on past and present attainments. All of us understand that much work remains to be done. And with your help and suggestions, much more will be accomplished.

Gerdis

Higinbotham

Weinstock

GUEST EDITORIAL

The Need For Safeguards Education

By Dr. R.F. Lumb, President
NUSAC, Inc.
McLean, Virginia

Plant operators are the first line of defense of any security system. That is a truism and one on which I expect little disagreement. It is especially true in the prevention or diversion of special nuclear materials. Their alertness to unusual operating conditions or unusual activities of individuals can go a long way to deterring and preventing the diversion of materials.

It is my opinion, that too little is being done to stimulate and motivate plant operators and reinforce that line of defense. There needs to be an educational program to make the operators aware of basic security philosophy and to stimulate their alertness to such things as unlocked gates, doors left open, unauthorized personnel in material access areas, and the like. They need to be advised of what action to take when suspicions are aroused, who to contact, and what channels to go through. There is a need for the operators to be aware of the value of the material. (The more inexperienced operators do not have any appreciation of the dollar value of the materials which they are processing.) They must appreciate that these are precious materials which they are processing, both in terms of real dollars, as well as in terms of the potential damage to society from mishandling, theft, or diversion. At the same time, they must be apprised of the fact that there is essentially no market for the material as such, and that because of the licensing program, commerce in these materials is within very well defined channels. It needs to be emphasized that there is no black market for nuclear materials.

It is important that operators be made aware of the penalties for illegal possession or diversion of special nuclear material, including both substantial fines and lengthy imprisonment. Equally important, they should be made aware of the provisions for rewarding individuals who provide information relative to the diversion of special nuclear material. These penalties and rewards are discussed in more detail elsewhere in this issue of the Journal and a poster which should be useful in the education program is described.

Each facility should integrate into its operator training program subject matter on the safeguarding of special nuclear material. The facility should adapt the techniques used to maintain employee awareness of safety issues, and utilize periodic flyers and posters to maintain employee awareness at a high level. I submit that the proper education and training of operating personnel can be far more effective in deterring and preventing diversion than the arming of guards.

Lumb

Participating in the May 7-11 INMM advanced statistics course at Battelle Columbus Laboratories taught by John L. Jaech were (from left)—Jaech, Robert McBroom, Wayne Harbarger, Charles Roche, Jack Streightiff, Audeen Walters, Elizabeth Stasny, David Armstrong, Harley Toy, G. Anthony Adams, Lavella Adkins, Bob Eggers, Thomas Bishop, Harry Tovmassian and Brian Smith.

Institute Educational Program Showing Stability

By Harley L. Toy, Chairman

The past year witnessed growth and stability in the Institute's educational program. This was due in part to the formulation of a formalized scope of responsibilities for the Education Committee. The scope was drafted and approved following last year's annual meeting and paved the way for what has become an established and viable educational program. The stability of the program was due to the dedicated efforts of a few individuals. One individual was the mainstay of our program. That individual, *John Jaech*, has been the cornerstone of our program. During the past year John has conducted five formal statistics courses. Four were presented at Battelle-Columbus and the other at Heathrow, England. We certainly look forward to John's continued participation in our program. Our projected plans for the coming year definitely include the continuation of the formal statistics courses. This projection is based upon demand throughout the industry.

Toy

Jaech

Activities of the Education Committee during the past year centered on John Jaech's formal course offerings. The formal statistics courses presented at Battelle-Columbus was the major thrust of our educational efforts. Course enrollment was sixty-five. The courses were very well received as indicated by course evaluation questionnaires. Two courses of study for application of statistics in nuclear materials safeguards and control were offered: "Introductory Statistics," a three-day course designed for non-statisticians in managerial positions, and a follow-up five-day course in

INMM-Sponsored Safeguards Training Courses in Fall, 1979 Battelle Columbus Laboratories

"Introductory Statistics with Applications to Special Nuclear Material Control."	October 2-4, 1979 20 Attendees
"Selected Topics in Statistical Methods for Special Nuclear Material Control."	October 29-November 2, 1979 20 Attendees

Brochures on INMM-Sponsored Safeguards courses are sent to the INMM members. Brochures are available from the INMM Publications Office, 20 Seaton Hall, Kansas State University, Manhattan, Kansas 66506. Both courses will be offered at Battelle Columbus Laboratories and taught by John L. Jaech, Staff Consultant with Exxon Nuclear Co., Inc., Richland, Washington.

DOE SAFEGUARDS TECHNOLOGY TRAINING PROGRAM Los Alamos Scientific Laboratory Schedule of 1979 Courses

Listed below are the dates for the US DOE Safeguards Technology Training Program courses for 1979.

"Fundamentals of Nondestructive Assay of Fissionable Material Using Portable Instrumentation"	October 1-5, 1979 30 Attendees
"In-Plant Nondestructive Assay Instrumentation"	December 3-7, 1979 20 Attendees

Brochures on LASL Safeguards Courses are sent to members of the INMM as well as past attendees. The mailing list numbers about 800. This year the course announcements also were listed in *Physics Today* and *Nuclear News*.

"Selected Topics in Statistical Methods for SNM Control." Overall, the courses were presented well within assigned budgets.

Other activities of the Education Committee during the past year included:

•Liaison was established with NRC and DOE relative to their educational programs and instructional needs. This will continue to be an ongoing exchange of information and hopefully will result in co-sponsorship of instructional programs.

•Dissemination of course and seminar information to the membership through the Journal. Our efforts in this area will be increased. Our intent is to provide a timely compilation of all upcoming meetings, seminars, workshops, and course information to the membership. This program was initiated following last year's annual meeting and first appeared in the Summer 1978 issue of the Journal.

•Assisted in the presentation of the INMM Workshop on Implementation of IAEA Safeguards held in Washington, D.C. last December. Assistance included formulation of agenda and physical arrangements.

•Conducted survey of available educational and training programs in the area of nuclear materials safeguards and control. Results of this survey will form the basis of our program for the coming year.

The Institute's educational program has, in my opinion, turned the corner. We now have the administrative know-how and available physical facilities for providing a timely and comprehensive educational program in materials safeguards and control. This asset coupled with the vast experience of our membership to provide the necessary faculty will achieve our goals. Our goal will be one of meeting the educational needs of the '80s. Our immediate objective will be to expand our formal course offerings. Plans are currently underway to present formal courses in accounting and auditing techniques. At this stage, we have had discussions with several Institute members to serve as instructors. We have in-depth experience in the accounting area and we intend to utilize these talents in furthering the Institute's educational objectives. Finally, in the coming year, we will look more to the membership for their thoughts and requirements in the overall training area.

New Utility Member Encourages Involvement

By John Barry
INMM Membership Committee
Beaumont, Texas

Understating things, I welcome this opportunity to participate in INMM activities; I will learn much from members more actively involved in nuclear materials management than myself presently. Hopefully, I will contribute something in return through my efforts while on the membership committee to help increase nuclear utility representation in this organization. In beginning this responsibility **Jim Lee, Tom Gerdis** and **John Ladesich** have given me greatly appreciated direction and support.

Reflecting on the U.S. nuclear power utility industry's current concerns, I feel its representatives must become more involved in activities of the INMM. It is a significant, useful and necessary step in helping to provide an adequate, secure and economic mix of generating capacity for the domestic electricity customer in the 1980s and into the 1990s.

The "fallout" from the Three Mile Island incident definitely isn't radioactive in nature. In my opinion we are seeing some political implications for our country's peaceful nuclear power program that appear analogous to the Kristalnacht of Germany in the 1930s and its aftermath. More optimistically, this period of dislocative and divisive energy policies and politics will perhaps eventually stimulate a cohesive national survival instinct and a truly democratic remedy-ballot box directives. Amid the furor it is imperative that perspective reason and responsible actions in the meanwhile prevail in our industry and society; in essence we all still have our jobs to do. Professional organizations like the INMM can help us do just that.

In the areas of internal and external physical security and materials control problems are evident: attempted sabotage of fresh nuclear fuel at a nuclear generating plant site; demonstrators attempting to obstruct incoming equipment and to illegally penetrate sites; expeditious yet effective control of the inflow and outflow of technical personnel and equipment at plant sites. Today, not only industrial accidents but intentional threats on all types of hazardous materials in-place or in-transit may endanger the public safety and risk loss of controlled substances. Technological innovations can do much but the "technical fix" is no panacea. Institutional measures such as prosecutable laws, dedicated enforcement, integrated yet rational security procedures, and a supportive public information program which perspectively compares nuclear power risks to others considered both manageable and acceptable by our society appear as necessary as hardware developments. The INMM is uniquely able to contribute to understanding of the inter-relationship of these areas of effort and their betterment.

During my short time on the membership committee I have sent a series of two letters to selected electric utilities in order to locate potential INMM members within their organization, obtain input on INMM's role and how our segment of the industry may best profit from individual and corporate participation in fulfillment of that role. Fundamentally several respondee's asked "What's the INMM and how can it help us?" I believe this reaction will become less prevalent as we (the utilities) come to grips with the ramifications of the draft U.S.-IAEA Safeguards Treaty and contingent proposed NRC regulations and other national, state and local proposals for a general trend toward increasingly stringent requirements on power plant physical security and materials safeguards. Basically we need to determine what really is necessary and what is not (i.e. what is beneficial and what is not). Increasingly the utility industry needs better access to professional knowledge and experience of other fields which the INMM can facilitate. I look forward to meeting many of you at our annual meeting and attending the session on safeguards concerns of utilities. I sincerely invite any comments and information from the membership which will help me better carry out my committee duties.

New Addresses

The following changes of address have been received by the INMM Publications Office (Phone: 913-532-5837) at Kansas State University, 20 Seaton Hall, Manhattan, Kansas USA 66506, as of May 31, 1979.

Ralph G. Gutmacher, Los Alamos Scientific Laboratory, MS 541, Los Alamos, NM 87545.

Barry

R.J.S. Harry, Netherlands Energy Research Foundation ECN, 1755 ZG Petten, The Netherlands.

Alan M. Labowitz, Labowitz and Labowitz, 605 Prince Street, Alexandria, VA 22314.

Joseph E. Stiegler, Sandia Laboratories, Org. 1750, Albuquerque, NM 87185.

C.C. Thomas, Jr., Los Alamos Scientific Laboratory, MS/541, Los Alamos, NM 87545.

Kenneth E. Wilson, 4 Longfellow Place, Apt. 1111, Boston, MA 02114.

New Members

The following 36 individuals have been accepted for INMM Membership as of May 31, 1979. To each, the INMM Executive Committee extends its welcome and congratulations. New members not mentioned in this issue will be listed in the Fall 1979 (Volume VIII, No. 3) issue to be sent out November 1, 1979.

Dr. **Yumi Akimoto**, General Manager, Nuclear Energy Department, Mitsubishi Metal Corporation, 5-2, Ohtemachi-1, Chiyoda-ku, Tokyo 100, Japan.

Wayne R. Amos, Manager, Planning and Control, Nuclear Operations, Monsanto Research Corp., P.O. Box 32, Miamisburg, OH 45342.

Alfredo Caldwell, Section Head, Union Carbide Corporation, NMMSS, M.S. 7, ORGDP, Oak Ridge, TN 37830.

Dr. **Joseph M. Cameron,** Consultant in Statistics, 12502 Gould Road, Wheaton, MD 20906.

Howard E. Crowder, Engineering Supervisor, Union Carbide Corporation, Nuclear Division, Y-12 Plant, Oak Ridge, TN 37830.

Thomas S. D'Agostino, Acting Assistant Director for Plans and Policy, Office of Safeguards and Security, U.S. Department of Energy, Washington, DC 20545.

Dr. **R. Jack Dietz**, Group Leader, Q-4, Los Alamos Scientific Laboratory, MS 541, Los Alamos, NM 87545.

Dr. **Stanley L. Dolins**, Assistant Director, Energy Programs, Office of the Governor, State of Arizona, P.O. Box 25128, Phoenix, AZ 85002.

Dr. **Darryl J. Downing**, Statistician, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, TN 37830.

Dr. **Wolfgang Frenzel**, Section Head, International Atomic Energy Agency, P.O. Box 645, A-1011 Vienna, Austria.

Dr. **Kenneth D. Gerald**, Statistician, Rockwell International, Rocky Flats Plant, P.O. Box 464, Golden, CO 80401.

Wilhelm O.R. Gmelin, Section Head, International Atomic Energy Agency, P.O. Box 645, A-1011 Vienna, Austria.

Robby L. Hatcher, Manager, Inventory Control, General Electric, P.O. Box 11508, St. Petersburg, FL 33733.

James E. Haywood, Superintendent, Safety and Security, E.I. du Pont Nemours & Co., Savannah River Plant, Aiken, SC 29801.

Andrzej Janikowski, Safeguards Inspector P-5, International Atomic Energy Agency, P.O. Box 645, A-1011 Vienna, Austria.

Malcolm R. Johnson, Technical Records Manager, British Nuclear Fuels Limited, Windscale Works, Sellafield, Seascale, Cumbria, England.

Dr. **Ronald A. Knief**, Associate Professor, Chemical and Nuclear Engineering, University of New Mexico, Albuquerque, NM 87137.

Paul K. Makens, Senior Planner, United Nuclear Industries, Inc., P.O. Box 490, Richland, WA 99352.

Bernard Math, Adjoint Chef de Service, Commissariat a l'Energie Atomique, IPSE-DSMN, B.P. No. 6—Fontenay-aux-Roses, France.

Robert B. McCord, Manager, Process Development, Westinghouse Hanford Co., P.O. Box 1970, Richland, WA 99352.

Dr. **Dale A. Moul**, Manager, Security Programs Division, NUSAC, Inc., 7926 Jones Branch Drive, Suite No. 303, McLean, VA 22102.

Dr. **Ved P. Narang**, Inspector, International Atomic Energy Agency, P.O. Box 645, A-1011 Vienna, Austria.

Dwight J. Porter, Director, International Government Affairs, Westinghouse Electric Corporation, Suite 900, 1801 K Street N.W. Washington, DC 20006.

Michael J. Russell, Associate Scientist, Northeast Utilities, P.O. Box 270, Hartford, CT 06101.

Ernest A. Schnaible, Technical Programmer, Westinghouse Hanford Co., 325 Building 300 Area, Richland, WA 99352.

Donald F. Shepard, Senior Statistician, Rockwell Hanford Operations, P.O. Box 800, Richland, WA 99352.

James L. Smith, Senior Technical Associate, NUSAC, Inc., 7926 Jones Branch Drive, McLean, VA 22102.

Jack E. Streightiff, Senior Engineering Technologist, Virginia Electric & Power Company, P.O. Box 26666, Richmond, VA 23261.

Marta D. Tarko, Safeguards Statistical Assistant, International Atomic Energy Agency, P.O. Box 645, A-1011 Vienna, Austria.

Michel Ternault, 18, Avenue de Petit Chambord, 92340 Bourg-La-Reine, France.

Leslie William Thorne, Head, Far East Section, Department of Safeguards, International Atomic Energy Agency, P.O. Box 645, A-1011 Vienna, Austria.

Dr. **James L. Todd, Jr.**, Technical Staff, Sandia Laboratories, P.O. Box 5800, Albuquerque, NM 87185.

Edward Vejvoda, Director, Chemical Operations, Rockwell International, Rocky Flats Plant, P.O. Box 464, Golden, CO 80401.

George Weisz, Director, Office of Safeguards and Security, U.S. Department of Energy, Washington, DC 20545.

Dr. **James R. Whetstone**, Group Leader, Mass and Volume Measurement, National Bureau of Standards, Building 230, Washington, DC 20234.

Dr. **John C. Zink**, Manager, Nuclear Fuels, Public Service Company of Oklahoma, P.O. Box 201, Tulsa, OK 74102.

Examination Qualifying Safeguards Specialists Under Development

By Dr. Fredrick Forscher, Chairman
INMM Certification Committee
Pittsburgh, Pennsylvania

With the increasing importance of Safeguards and Non-Proliferation, the INMM is in the public spotlight as the only professional organization, worldwide, dedicated to these important issues. The requirements for professional recognition and public credibility includes a written test to evaluate the candidate's knowledge and understanding of the subject matter.

The most recent test for Certification as a Nuclear Materials Manger was administered by the INMM in June 1973. Neither the test, nor the certification procedure meets current requirements, for accreditation, completeness of subject matter, and professionalism. Much progress was made this year in developing a test of professional quality that could be fairly administered by a Certification Board. One lesson from the TMI-accident, among others, is the importance of assured professional quality and integrity of key personnel in the nuclear industry. Congress is calling for federally trained, and federally licensed professionals to operate facilities that impact on the health and safety of the public.

A hard working committee of Examination Question Formulators met at Rocky Flats (19 January 1979) and again in Pittsburgh (22 February 1979) for a full day each. The first order of business was to establish what is the pertinent subject matter and then went on to develop an appropriate number of test questions, about equally divided among the various specialties and subspecialties of our profession. The Outline of Pertinent Subject Matter appeared in the last issue of our Journal (see p. 27, Vol. VIII, No. 1). The pool of test questions stands now at about 480, equally divided among the three specialities: Material Accounting; Material Control; and Physical Protection and Secutiry.

It should be noted that the test will be open to all qualified applicants, irrespective of nationality, creed, race, sex, or age; even non-INMM members are eligible. In the future, we expect to find requirements for employment of safeguards specialists not only in industry and academia, but also in govenment—domestic, foreign and in international organizations. For these, and similar reasons, it is not just a matter of collecting test questions and letting the applicant struggle with them. It needs consistance in terminology, difficulty; grading ability and comparability to other professional test programs. In short, we need validation of our question pool. After validation we will proceed to administer the first test to all qualified applicants in the later part of 1979. No firm date can be set till validation is well on its way. The announcement will allow enough lead time for the applicants as well as for the administration of a fair test.

As indicated previously, we visualize a two-step process of certification. The first step would lead to a Qualified Safeguards Intern. After three years of applicable professional experience the candidate would be eligible to apply for Certified Safeguards Specialist in any one of the three specialties mentioned above. It is anticipated that presently practicing safeguards professionals could qualify for the latter without going through the internship.

Forscher

Kerr New LASL Director

SAN FRANCISCO, Calif.—**Donald M. Kerr, Jr.**, has been appointed Director of the Los Alamos Scientific Laboratory in New Mexico. The appointment was made May 30 by the Regents of the University of California upon the recommendation of UC President **David S. Saxon**, and was to be effective no later than August 1.

Kerr, 40, was Deputy Assistant Secretary for Energy Technology at the Department of Energy in Washington, D.C. He had responsibility for the office of nuclear waste management, fossil energy programs, field operations management, solar and geothermal energy programs, nuclear energy programs, and fusion energy.

Prior to joining the DOE staff in 1976, Kerr was on the LASL staff for 10 years, having served for the last year there as alternate energy division leader. As such, he was responsible for management of the hot dry rock geothermal energy program, basic geosciences research, solar energy, advanced heat transfer, low-temperature physics research, cryogenics engineering, and an energy systems analysis group.

"We are fortunate to have found for the Directorship at LASL a person who has had experience with the Department of Energy but who knows the Laboratory well. Donald Kerr is that person," said Saxon, "and I look forward to working with him to maintain the Laboratory's standards of excellence, and indeed, to increase them."

A native of Philadelphia, Kerr holds 3 degrees from Cornell University, including the Ph.D. in plasma physics.

Kerr

TABLE I
AN OUTLINE OF PERTINENT SUBJECT MATTER

A. Material Accounting	**B. Material Control**	**C. Physical Protection/ Security, at fixed sites and in transportation**
1. **Measurements**	1. **Process Control**	1. **Deterrence**
Bulk (mass, volume)	Process streams and flow	Laws and regulations
Chemical	Process measurement	Signs
NDA	Indicators	Personnel clearances
Treatment of data	Packaging	Procedures, operating
and uncertainty	Sampling	Physical characteristics
	Preparation for shipment	Seals
2. **Records**	2. **MBA System**	2. **Detection/Assessment**
Internal MBA records	Item identification	Access/Egress Control
Facility records	(serialization)	Sensors and alarms
Transfer documents	Physical inventory	Surveillance
Book inventory	Custodian/responsibility	Operating procedures
3. **Reports**	3. **Quality Control**	3. **Communication**
International requirements	Reference materials	Modes
National requirements	(physical standards)	Communications Security
Facility management	Standards	Redundancy
	Traceability	Network
	Sampling	
4. **Data Analysis**	4. **Laboratory Qualification**	4. **Delay**
Statistics	Sample exchange	Physical barriers,
Errors, bias treatment	Referee/verification	passive and active
Inventory difference		Remote response
Limit of Error		mechanism
Shipper/receiver dif.		
5. **Data Processing Technique**	5. **System Auditing**	5. **Response**
Licensee	Sampling	Reaction time
State System		Guard force
IAEA		Backup forces
6. **Audits**		6. **Audits**
System audits		
Sampling		7. **Transport**

*Compiled by the Institute of Nuclear Materials Management,
Certification Test Formulators
19 January 1979

Awards, Rewards — 1979

By S.C.T. McDowell
INMM Awards Committee
U.S. Department of Energy
Office of Safeguards and Security
Washington, D.C.

The Awards Committee (consisting of the Chairman, **S.C.T. McDowell, W.A. Higinbotham** and **Bernard Gessiness**) is pleased to announce another rewarding and successful year in helping to stimulate and promote activities and involvement in the areas of nuclear materials management and safeguards.

In addition to the second annual Student Award, the Institute presented its first "Distinguished Service Award" to an outstanding individual who has made significant contributions to the field of nuclear materials management, safeguards and nuclear energy.

The 1979 Student Award was presented to Mr. **Mark H. Killinger** of the University of Washington for his paper on "Optimal Use of Safeguards Expenditure for Verification Measurements." Three excellent student papers were placed in final competition and from them Mr. Killinger's paper was selected for presentation at the 20th Annual meeting held in Albuquerque, New Mexico. The selection made was influenced by the originality, technical correctness, and importance in the area of nuclear materials management and safeguards. A plaque acknowledging his achievement and a check for $500 was presented to him at the meeting.

The most prestigious of INMM awards, the "Distinguished Service Award," was presented to Dr. **W.A. Higinbotham** in recognition for 35 years of outstanding and continued contributions to the fields of nuclear materials management, safeguards and nuclear energy programs. The selection for this award is made through written recommendations and endorsements from Institute members or others based on contributions and outstanding accomplishments in the areas of nuclear materials management, safeguards and nuclear energy in the domestic and international communities.

Future activities of the Committee include more publicity for viable candidates for the Distinguished Service Award; early announcement for the Student Award Competition; promoting more interest in the Student Award from foreign colleges and universities; and thereby stimulating Institute membership growth and interest in helping achieve its goals in nuclear materials management, safeguards and nuclear energy programs.

Killinger

McDowell

Gessiness

Higinbotham

Future Trends in Safeguards

By Sylvester Suda, Chairman
INMM Safeguards Committee
Brookhaven National Laboratory
Upton, New York

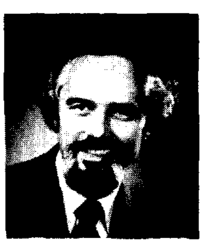
In the 1950s and the first half of the '60s, what we now call safeguards consisted largely of record keeping based on product measurements and transfer documents. Nuclear material managers were selected from knowledgable and proficient accounting or analytical chemistry personnel. Because of their aptitude for quality work and other characteristics, such as their understanding of product specifications and AEC reporting requirements, they became and were an important part of a somewhat informal but dedicated cadre of nuclear material managers.

As reporting requirements became more complex, it became more evident that safeguards must be a planned-for objective. As a result, safeguards concepts have advanced from mere record-of-transfer accounting to the present 100 percent inventory measurement and physical security concept of nuclear materials Safeguards. The evolution of the system progressed through the '70s until today the nuclear industry accepts the concept and practice of nuclear material safeguards as a management-inspired function of the total organization. With this evolution, new disciplines were developed. Non-destructive assay came into the limelight followed closely by related desciplines such as access monitoring and others.

To cope with the increase of requirements and disciplines, system effectiveness studies were performed to integrate all the functions into a total nuclear material safeguards system. The impact of these changes is still being felt.

What can we expect in the future? What are the trends in new technology and nuclear materials management? Which trends affect safeguards requirements? Will they result in evolution of new safeguards systems? New techniques? New organizational concepts?

Suda



Major advances have been made in automation and computer technology. Safeguards, which does not wish to be left behind by advanced technology, should examine and prepare for the problems that will result from automation of the measurement and monitoring functions. As the measurement techniques become more technically sophisticated and are applied to more and more in-process compounds, an individual still depending on old or manual methods is less likely to be able to judge their quality and their impact on safeguards effectiveness.

Because of the impact of process automation and other new technologies, safeguards tools also must be improved and developed. These include: automated information systems tied to measurement equipment; alternate or modified inventory and reporting requirements; tamper-indicating techniques and systems; universal utilization of standards; and training and re-education.

Major advances have occurred in development and automation of new measurement equipment, calibration, and data acquisition techniques. Most future facilities will be built with such systems already installed and will provide the plant operator with literally unlimited data on the quantity and location of the nuclear material.

Automated measuring systems make possible continuous dynamic monitoring of material in-process. This concept most likely will not free the safeguards organization from the statistical treatment of grossly errant or missing data.

The application of large computers and modern signal processing techniques to the safeguards problem will involve mathematical modeling, optimal estimation of process variables, and detection of abnormal changes and hypothesis testing of their significance. When it comes about, it will create a serious problem for the inspectors.

Convincing domestic and international inspectors that automated measurement equipment is accurate and that the safeguards program declaration based on those results are correct will be difficult. The total automated measurement and material accounting system must have an acceptance program of its own. That is, two conceptually distinct acceptances must be made: ac-

IAEA Safeguards Program Topic Of Meeting At BNL

The status of U.S. technical support of the International Atomic Energy Agency's (IAEA) safeguards program was discussed at a meeting at Brookhaven National Laboratory March 26-30. Among those present were Gerald F. Tape (left) President of AUI and former Ambassador to the IAEA; William Bartels, DOE; Leon Green, head of ISPO; and Dr. Johannes Gruemm, newly appointed Director for Safeguards of the IAEA. Participants included representatives of the IAEA, the State Department, Arms Control and Disarmament Agency, DOE and the Nuclear Regulatory Commission. They reviewed the 72 active projects now being managed by the International Safeguards Project Office (ISPO), in addition to the 60 new tasks that have been proposed by the IAEA. ISPO was established at BNL late in 1976 under the sponsorship of the above U.S. agencies. Its function is to consolidate and coordinate the U.S. effort aimed at strengthening the IAEA safeguards effectiveness.

ceptance of the measurement equipment and acceptance of the computerized facility (state) safeguards program. Acceptance of the measurement equipment requires the aid of universal standards and special test equipment which simulates and evaluates the measurement equipment. State acceptance of the automated safeguards program will immediately follow and will be tantamount to acceptance of the end product. This does not hold for international safeguards where the inspector must understand how the computer files are maintained and must be capable of independently testing and evaluating the logic and memory units and the computers' data manipulation routines and programs. New technologies are needed which assist the inspector to test the facility's automated safeguards program. The new technologies are in the areas of tamper-indicating, rapid data reduction and evaluation, and statistical structures based on conditional verification. Independent verification of key safeguards operation will be replaced with computer techniques operating on specification parameters and qualification procedures.

Alternately, the inspectorate might own and operate its own automated measurement equipment. This may in fact be the preferred option for key measurements in certain facilities. However, in general, the inspectors will continue to depend for much of their information on data obtained using facility owned and operated measurement systems.

In the era of computer-automated measurement systems, the program verification function will be one of the more significant areas of change in the inspection process. New technology in the form of tamper-indicating techniques and operations criteria will include provisions for international inspectors to use plant generated data with a high degree of confidence relative to the reliability of the data. Program verification will occur through the increasing use of more sophisticated data acquisition systems, use of electronic monitoring and the routine appraisal of these data using inspector supplied test values and validation criteria.

International inspectors must become acquainted with computer operations, techniques and language to be able to communicate so that their technical requirements can be used for formulation of input, for manipulation of input and for noting how results should be displayed. Inspector involvement in the development of the process is essential; otherwise, the program will fail.

It must also be recognized that man is the final source and sink for all information and no system can be better than its interface with the human user. It will be essential to consider the man-machine interface in judging safeguards effectiveness. It can be seen readily that the new technologies of automation and system engineering will have a profound effect upon our approach and practice of nuclear material management and safeguards.

Report on First Annual ESARDA Safeguards Symposium

By D.M. Bishop
INMM Executive Committee
General Electric Company
San Jose, California

The objective of this report is to provide a brief review of the recent Symposium on Safeguards and Nuclear Materials Management held in Brussels, Belgium on April 25 to 27, 1979. This highly successful meeting was sponsored by the European Safeguards Research and Development Association (ESARDA). Chairman of the Symposium was Dr. **Dipak Gupta** (KFK, Karlsruhe) a long time INMM member. Other members of the scientific secretariat who were responsible for organizing the meeting included **A.S. Adamson** (NMACT, AERE Harwell), **C. Beets** (CEN/SCK, MOL) and **L. Stanchi** (JRC, ISPRA).

The meeting was extremely well attended including over 230 safeguards professionals. Participants included representatives from national and international regulatory agencies, research and development laboratories and commercial processing facilities.

For those who may not be aware of ESARDA's overall scope and goals, a brief summary may be useful. ESARDA is an association of European organizations formed to advance and coordinate research and development activities in the safeguards area. It also provides a forum for the exchange of information and ideas between nuclear facility operators and safeguards authorities. Partners in the ESARDA organization currently include:

- The European Atomic Energy Community.
- The Kernforschungszentrum Karlsruhe (KFK)-Fed. Rep. of Germany.
- The Centre d-Etude de l'Energie Nucleaire (CEN/SCK)-Belgium.
- The Comitato Nazionale per l'Energia Nucleare (CNEN)-Italy.
- The Stitching Energie Onderzoek Centrum Nederland (ECN)-Netherlands.
- The United Kingdom Atomic Energy Authority (UKAEA)-Great Britain.
- Energistyreisen—Denmark.

The specific objective of this first annual ESARDA symposium was to provide a forum for the discussion of current nuclear materials methods of mutual interest. Particular emphasis was put on encouraging operators to publish the results of recent development programs. Topics relating to the development and implementation of international, national and subnational safeguards issues were reviewed.

There was notable INMM member participation in this ESARDA sponsored symposium. Over two dozen INMM members participated in various phases of the technical program. Although a complete list of INMM participants is too long to include, in addition to those members pictured, the INMM participants include **Syl Suda** (BNL and INMM Safeguards Committee), **George Huff** (AGNS), **Roy Nilson** (EXXON), **Tom Yolken** (NBS), **G. Cullington** (CCE), **A.G. Hamlin** (UKAEA), **M. Cuypers** (JRC, ISPRA) and others.

Papers presented at the meeting were organized into 11 sessions covering a broad range of current safeguards issues. These sessions included the following technical topics:

- Invited (Plenary) papers.
- Safeguards Concepts and Regulations.
- Containment and Surveillance.
- Destructive Analysis and Isotopic Correlations.
- Nondestructive Assay Methods and Instrumentation.
- Reference Materials and Interlaboratory Tests.
- Systems Analysis and Statistical Methods.
- Data Recording, Processing and Reporting Methods.

A total of approximately 100 individual papers were presented during the 11 sessions. This included 14 papers presented in four poster sessions. The non-destructive methods and instruments subject topped the list with 24

Gupta

Bishop

papers followed by those on containment and surveillance measures including 16 papers. The destructive methods and the systems analytical work had 14 presentations each. In 11 papers, different aspects of material accountancy were discussed. Ten papers dealt with reference materials, interlaboratory tests and the standardization of methods. In seven papers safeguards concepts for reprocessing and enrichment facilities, a fast critical assembly and a large research center were discussed. Four invited papers covered general aspects of international safeguards.

Highlights of individual papers are presented in the following summary. A more detailed summary of each session will be published in forthcoming ESARDA Bulletins. The complete text of each paper is also a available in the ESARDA meeting transactions.

Session 1 (Invited Papers)

The introductory session of the Symposium included comments on:

- the successful close cooperation between IAEA and EURATOM while executing safeguards activities in existing nuclear facilities within the European Community.
- the necessity of carrying on further research and development work for improving existing sealing devices
- the possible combination of some of the elements of international and domestic safeguards

Session 2 (Safeguards Concepts and Regulations)

These papers discussed interesting safeguards concepts for large scale or advanced reprocessing facilities. Further development work on conceptual designs for such safeguards systems was identified as necessary including both methodology and hardware.

The problem of adaptability of conceptual ideas in existing nuclear facilities was identified as a remaining significant issue.

The satisfactory functioning of current safeguards concepts in critical assembly and nuclear research centers was reviewed.

Sessions 3/4 (Containment and Surveillance)

Current work in the area of containment and surveillance was presented. Methodology for statements relevant to international safeguards were reviewed including the possibility of combining containment, surveillance and accountancy measures.

Conceptual and technical approaches to the remote verification of functioning containment and surveillance devices was discussed.

The list of containment/surveillance hardware was extended to include doorway monitors. Significant quantities of some of this equipment are now in routine use. Other hardware requires more development work before routine use will be possible.

Session 5 (Destructive Analysis and Isotopic Correlations)

These papers reviewed the continuing maturity of different destructive analysis methods. Work on destructive analysis appears to be progressing satisfactorily and will be adequate for international safeguards.

Dr. Dipak Gupta (KFK, Karlsruhe) chairman of the ESARDA Symposium, and long time INMM member, addressing the opening session of the symposium.

Great interest was expressed in the area of possible use of isotopic correlation techniques. However, in depth work on these correlations is still required before routine use in international safeguards will be possible.

Sessions 6/7 (Nondestructive Assay Methods)

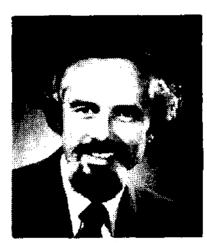
Non-destructive assay methods and instruments for the assay of plutonium and uranium in different forms and geometries were described based on both gamma and neutron techniques.

The most important conclusion is that a large number of these methods are currently in routine use. Some methods are even satisfactorily used by safeguards inspectors. However, the need for future improvements were identified including reducing the dependance on chemical methods and improved calibration techniques.

Session 8 (Nondestructive Assay Instruments)

In this session instruments were described which are currently in use. Experience indicates a high degree of maturity and shows that a large number of instruments

Beets



Suda

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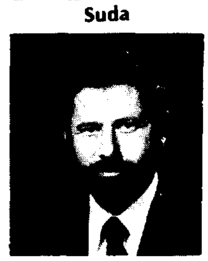


Yolken

Charles Beets (CEN/SCK), INMM Member and part of the Scientific Secretariate which organized the ESARDA Symposium.

Herman Miller (NNC) INMM Public Information Chairman, and his wife Jo Anne, standing in front of the National Nuclear Corporation exhibit at the ESARDA Symposium.

Dr. Anderson (AERE, Harwell), luncheon speaker during the symposium, stresses the need to put current nuclear risks in perspective with alternate energy sources.

Hans Buker (KFK-Germany), long time friend of the INMM, enjoys a few light moments with colleagues during a coffee break.

are readily adaptable to routine use by both plant operators and inspectors. Again, the need for further future improvement was identified.

Session 9 (Reference Material and Interlaboratory Tests)

Extensive international cooperation in the development and use of internationally standardized methods was recognized although some notes of caution were indicated.

Several papers showed progress in the development of different types of physical standards for both destructive and non-destructive purposes, including various forms of material, uranium concentrations and plutonium isotopic ratios. Such standards are available today for routine use.

Interlaboratory tests for different measurement methods starting as a small venture in 1970 were reviewed. This work has been extended into a dynamic international activity involving both destructive and non-destructive methods.

Session 10 (System Analysis and Statistical Methods)

A renewed interest in modelling and simulation techniques was identified. Further investigations on risk and diversion strategies was identified as a future need.

Computer techniques for establishing MUF and its uncertainties have become of great interest to both plant operators and safeguards authorities.

Session 11 (Data Recording, Processing and Reporting)

The whole question of 'real time' accounting received significant attention. Different approaches are being examined in different countries for various type facilities. However, it was stressed that although such accounting systems are being used in some of the facilities on a routine basis, the general use of such systems for international safeguards has yet to be demonstrated.

A second annual ESARDA symposium on Safeguards an Nuclear Materials Management is planned for March 26 to 28, 1980 in Edinburgh, Scotland. Interested parties should contact a member of the ESARDA scientific secretariat for the meeting including: **A.S. Adamson** (NMACT, AERE Harwell), **U. Ehrfeld** (KFK, Karlsruhe), and **L. Stanchi** (JRC, ISPRA).

Dave Rundquist (IAEA) and Dennis Bishop (GE) discuss possible areas of cooperation between the INMM N15 Standards Committee and the IAEA.

Joe Stiegler (Sandia Labs), and Mike Ryan (ANL) enjoy coffee and discussion of current international safeguards issues.

Joe Stiegler (Sandia Labs.), INMM Annual Meeting Arrangements Chairman, presenting a paper on Containment and Surveillance Systems.

Past INMM Chairman Jim Lovett (IAEA) and INMM member Nancy Trahey (NBL) discuss international safeguards measurement problems between sessions at the symposium.

Dennis Bishop (GE), INMM N15 Chairman, and member of the Executive Committee, answering questions after a paper on INMM Standards Activities.

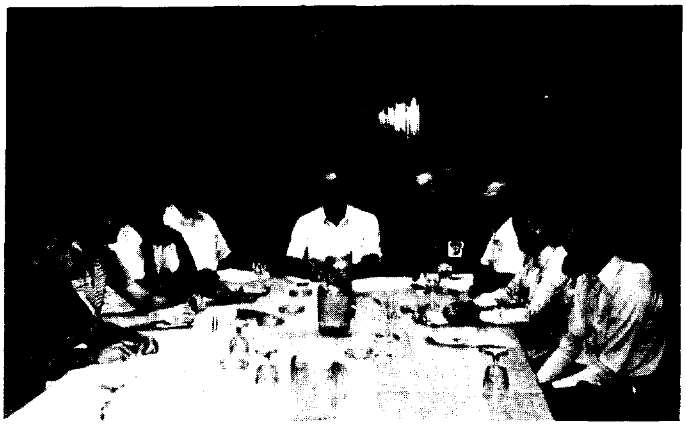
Dave Rundquist (IAEA), long-time INMM member, discusses the forthcoming implementation of IAEA safeguards requirements with Jim Lovett and A. Von Baeckmann (also IAEA).

Focus on Results

By D.M. Bishop, Chairman
N15 Standards Committee
(Nuclear Materials Control)
General Electric Company
San Jose, California

The opinion that the late 1970s and early 1980s are pivitol years in determining the future utility of the nuclear alternative has become a fact of life. With this in mind, the Institute of Nuclear Materials Management and its members have never worked more actively to bring a positive outcome to this technical-turned-social-turned-political issue.

Based on these efforts significant technical progress has been made on a variety of fronts to demonstrate both economic and technical feasibility. For example, throughout the past several years of hard winters and coal strikes, nuclear plants have quietly and effectively helped to meet the nation's energy needs. If we as an industry have made one mistake during these times it has been in not making the public more aware of this exceptional performance and safety record.



INMM-3—Statistics (from left): Merril W. Hume, Rockwell International—Rocky Flats; David Zeff, Babcock & Wilcox, NMD; Richard W. Mensing, Lawrence Livermore; Victor W. Lowe, Jr., Oak Ridge National Laboratory; J.L. Telford, USNRC, Chairman; Roger H. Moore, US NRC; Gary Tietjen, Los Alamos Scientific Laboratory; Dolores McCarthy, United Nuclear Corporation; and Frank Wimpey, Science Applications, Inc.

INMM-9.1—Material Characterization (from left): Dick Chanda, Fran Haas, Al Evans and Herb Smith.

This progress notwithstanding, much of the potential public advantage created by our recent years of toil has been seriously erroded during a single week in March, 1979, by the events in a sleepy little town near Harrisburg, Pennsylvania. The domestic and international ramifications of these events are just now being realized. However, at the minimum it would appear to have added yet another year or two delay to public acceptance of the nuclear alternative and achievement of our national energy goals.

The current problem may best be put in perspective by a brief story.

A friend recently related a situation involving his seventy-three year old mother. Six months ago she, as most of the general public had little understanding of the technical aspects of nuclear power—positive or negative. However, this dear lady recently called to question her professional sibling in considerable detail on the subject of power reactor cold-shut-down margin procedures, a term here-to-fore not normally cropping up in the Sunday afternoon family phone call home. One can only be convinced by such increasingly common place events that the vocal anti-nuclear minority, coupled with a sensationalistic press, has effectively capitialized on recent events and made major advances in the area of public information.

As a result of these unfortunate events the burden of proof is now even more squarely on us to repsond with redoubled vigor. The question remaining is how can we best respond to such often irrational pressures and yet remain accurate and productive. The answer must be in two areas: (1) systematically and realistically assessing current problems and (2) focussing on well conceived solutions to current nuclear issues. In a phrase this means "focussing on results" not the morass of rhetorical questions currently prevelent in some public circles.

In support of this needed focus the N15 Standards Committee has had a productive year to date. The following is a brief summary of key **results.**

A. **New Standard Issued**

A new INMM-American National Standards Institute Standard (ANSI) has recently been published. It is INMM-N15-23-1979, entitled: "Non-destructive Assay of the U-235 Content of Unpoisoned Low Enriched Uranium Fuel Rods." This standard was developed by the INMM-9 (Nondestructive Assay) Subcommittee chaired by **Darryl Smith** (LASL) under writing group INMM-9.5 (Techniques) chaired by **John Stewart** (GE). The standard has been under development for several years and represents a timely contribution to the field of safeguards measurement methods.

B. **Scope Assessment and Reorganization Complete**

A serious review of N15 Standards Committee past performance and future direction has been completed. This effort resulted in the identification of the following needs:

•Clearly defined and documented work scope down to the writing group level must be developed.

•A reassessment of previously issued standards to assure timeliness is needed.

•Writing groups must be established to address recent safeguards emphasis.

•A reassessment of the previous Sub-Committee organization and assignments is needed to assure maximum return from available volunteer resources.

•Expanded emphasis on external communication is needed to assure coordination and integration

•Expanded emphasis on international cooperation will be required to meet forth-coming needs.

The N15 officers and Subcommittee chairmen are actively pursuing the implementation of these goals. The current N15 organization is outlined in Table 1. Further suggestions or comments on N15 Standard Committee scope would be welcomed at any time.

C. **Measurement Control Sub-Committee Initiated**

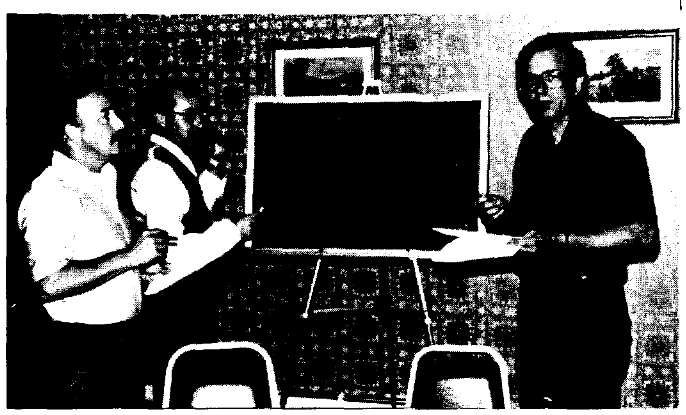
A new subcommittee level activity has been initiated to deal with safeguards related measurement control methods. This Subcommittee is chaired by **Yvonne Ferris** (RI-RFP) and is designated as INMM-5 (Measurement Controls). It currently includes the following members:

Willard B. Brown (NRC)
Donald D. Cobb (LASL)
Charles W. Emeigh (NRC)
William E. Gilbert, Jr. (DOE)
Rodney Hand (Allied Chemical)
Victor W. Lowe, Jr. (UCC)
Carson L. Nealy (Rockwell Inter.)
Munson M. Thorpe (LASL)

INMM-5 had its first meeting to define scope and goals in conjunction with the Albuquerque, 1979 annual meeting of the Institute. Interested parties are invited to contact Yvonne Ferris (303-497-4441) to get involved in this timely standards writing effort.

D. **INMM—8 Sponsors Uranium Hexafluoride Mass Measurement Seminar**

INMM-8 (Calibration Techniques) Subcommittee in conjunction with the National Bureau of Standards has sponsored a seminar and work shop on the industry implementation of ANSI Standard INMM N15.18, "Mass Calibration Techniques for Nuclear Materials Control." This event was conceived and planned by **Lou Doher** (RI-RFP), INMM—8 chairman, and **John Whetstone** (NBS). The purpose of the seminar/workshop was to acquaint Nuclear Regulatory Commission licensees and Department of Energy contractors with the techniques



INMM-9.4—Measurement Controls (from left): Robert B. McCord, Westinghouse Hanford Co.; Darryl B. Smith, Los Alamos Scientific Laboratory; and Richard H. Gramann, Nuclear Regulatory Commission.

Table 1.
INMM—N15 STANDARDS COMMITTEE ORGANIZATION

SUBCOMMITTEE	TITLE	CHAIRMAN	AFFILIATION	PHONE
---	N15 Chairman	Dennis Bishop	General Electric	(408) 925-6614
---	N15 Secretary	Robert Kramer	Northern Indiana Public Service	(219) 787-8531
INMM-1	Accountability and Control Systems	Howard Menke	Westinghouse	(412) 373-4511
INMM-3	Statistics	Frank Wimpey	Science Applications	(703) 821-4429
INMM-5	Measurement Controls	Yvonne Ferris	Rockwell International	(303) 497-4441
INMM-6	Inventory Techniques	Frank Roberts	Battelle—PNL	(509) 942-4767
INMM-7	Audit, Records and Reporting Techniques	Bob Sorensen	Battelle—PNL	(509) 942-4437
INMM-8	Calibration	Lou Doher	Rockwell - RFD	(303) 497-2575
INMM-9	Nondestructive Assay	Darryl Smith	LASL	(505) 667-6514
INMM-10	Physical Security	Tom Sellers	Sandia Labs	(505) 264-4472
INMM-11	Certification	Fred Forscher	Consultant	(412) 521-0515
INMM-12	International Safeguards (Proposed)	**		
INMM-13	Transportation (Proposed)	**		

**Currently under review to evaluate scope and feasibility.

proposed in ANSI N15.18 for measuring the mass of UF6 shipped and received in standard containers (eg., 30B, 48X, and 48Y). Discussions with Department of Energy and Nuclear Regulatory Commission representatives indicate that there is a strong possibility that this standard (ANSI N15.18) will meet both current and future regulatory requirements of each agency. This is an excellent testimonial to the usefullness of the standard.

This highly successful meeting was held on June 7, 1979, in Atlanta, Georgia. Results will be reported in subsequent issues of the Journal.

E. **Two Advisory Groups Initiated**

Positive steps have been taken to assess the need for standard writing activities in two timely safeguards areas:

(1) International Safeguards.

(2) Transportation and In-Transit Materials.

Advisory groups are being formed in both areas to make recommendations concerning the need and scope for future standards work. The chairman of the International Safeguards Advisory Goup is **Bob Sorenson** (Battelle—PNL). **Bob Wilde** (Sandia) will head the Transportation Advisory group. Membership inputs are being actively solicited in both areas. Recommendations will be reviewed and translated into a decision for possible action by early Fall 1979.

F. **ESARDA Contacts Initiated**

The INMM N15 Standards Committee program was reviewed for the European safeguards community during the first annual European Safeguards Research and Development Association (ESARDA) symposium on Safeguards and Nuclear Materials Control. The paper entitled: "USA—INMM Safeguards Consensus Standardization Program Status," was presented by **Dennis Bishop** (GE) N15 chairman. The paper was well received and resulted in contacts between INMM and ESARDA which will be developed to help assure the smooth implementation of forthcoming IAEA requirements and may result in future cooperative programs.

G. **Possible FTC Action**

For your information, in December 1978 the Federal Trade Commission (FTC) proposed regulations relating to voluntary standards and certification activities. The proposed rule is composed of 17 sections. These provisions can be divided into three general groups dealing with (1) procedures; (2) the substantive "duty to act;" and (3) certification. FTC hearings were held on the proposed rule on April 16, 1979 in San Francisco, Calif., and May 21, 1979 in Washington, D.C.

Although it is important that N15 members be aware of this possible regulation, ANSI evaluation of the

(Continued on Page 34)

INMM-9.6—Automation (from left): Larry East, Nick Roberts, Phil Ting, Walt Strohm, Norm Hall and Ron Brandenburg.

INMM-9.2—Container Standardization (from left): Fred Duff, Mound Laboratory; Tom Atwell, IRT Corporation; and John Birden, Mound Laboratory.

INMM-10—Physical Security (from left): Jim Prell, Sam McDowell, Al Winblad, Ed Kurtz, John Powers, Herb Dixon, Blythe Jones, E.L. Musselwhite and Don Moss.

INMM-9.3—Physical Standards (Standing from left): John Glancy, Bill Rodenberg, Tom McDaniel, Ron Harlan. Seated from left: Bill Reed, Nancy Trahey, and Steve Carpenter.

Implementation of ANSI N15.18 Uranium Hexafluoride Mass Measurements

By Lou Doher, Chairman INMM-8
Rockwell International
Energy Systems Group
Golden, Colorado

American National Standard ANSI N15.18-1975, "Mass Calibration Techniques for Nuclear Materials Control" was proposed by INMM 8.1 and published during the summer of 1975. A major portion of the standard is devoted to mass measurements of massive uranium hexafluoride (UF_6) cylinders.

The UF_6 portion of ANSI N15.18 has been supported by the Nuclear Regulatory Commission, Directorate of Standards Development, by funding the construction and subsequent calibration (by the National Bureau of Standards) of UF_6 cylinder Replica Mass Standards (RMS). The Replica Mass Standards are essentially stainless steel facsimilies (two each) of the Models 30B, 48X, and 48Y UF_6 cylinders as specified in ANSI N14.1-1973, "Packaging of UF_6 for Transport." One cylinder of each size has been filled to provide the RMS at two mass levels (full and empty) of each of the three types of UF_6 cylinders.

In order to evaluate the product mass measurement process of UF_6 cylinders, ANSI N15.18-1975 creates at each facility a process which produces known values of In-House Standards (IHS) using the RMS, the uncertainty of which is limited only by the precision of an ideal process. These IHS, when processed through the product mass measurement processes, provide quantiative estimates of the error bounds associated with the UF_6 product measurements.

Thus, ANSI N15.18-1975 minimizes two significant sources of bias by the use of RMS. Since the RMS are identical to the cylinders used in the transfer of UF_6, the variability associated with the buoyant forces is minimized, and the bias which is associated with the basis for the mass values assigned to all UF_6 cylinders in the system is eliminated.

With ANSI N15.18-1975 published and the artifacts calibrated, the question of future actions received considerable discussion at a meeting of INMM 8.1 at New Brunswick Laboratory (September 1975). Two decisions were made at this meeting: first, a proposal to institute a pilot measurement assurance program involving a select group of facilities who routinely measure UF_6, and second, an overview of how the concepts of ANSI N15.18-1975 could be incorporated and implemented in an industry-wide system.

The plan of the pilot measurement assurance program was published in 1976.[1] The results of pilot operations were reported first in 1977[2] and an expanded report, including results of product exchange measurements, in 1978.[3]

The implementation of the standards' concepts for UF_6 measurements in an industry-wide program has now been planned. The program has been made possible with the Nuclear Regulatory Commission providing funding to allow the National Bureau of Standards (NBS), Office of Measurements for Nuclear Safeguards, to actively provide the administration of the implementation of ANSI N15.18 UF_6 mass measurements throughout the industry. The current funding permits NBS activity in this area through September 30, 1980.

The NBS Office of Measurements for Nuclear Safeguards has appointed Mr. **E.G. Johnsen** as the NBS employee accountable for the administration of the program.

Mr. Johnsen, in conjunction with the chairman of INMM-8, called a meeting of the INMM 8.1 (Mass Calibration Techniques) Task Force at the NBS during

Doher

April 1979 to discuss the ANSI N15.18 UF_6 mass measurement program. The result of this meeting was a firm committment to begin implementation on or before October 1, 1979. In a subsequent meeting with Department of Energy and Nuclear Regulatory representatives, the participants indicated that ANSI N15.18 will likely meet both current and future requirements of each agency, and therefore, encourage participation by the industry.

The goal of this implementation is to **Provide an Efficient Means for Obtaining Uniform Mass Measurement of UF_6 Based on the National Measurement System**.

The implementation plans have been formalized into the following action:

1. The ANSI-INMM-8 Subcommittee on Calibration Techniques for Nuclear Material Control, in conjunction with the National Bureau of Standards, conducted a seminar and workshop on the industry implementation of ANSI N15.18, "Calibration Techniques for Nuclear Material Control," concerning uranium hexafluoride mass measurements.

The meeting was held in Atlanta, Georgia on June 7, 1979.

The purpose of the seminar/workshop was to acquaint Nuclear Regulatory Commission licensees and Department of Energy contractors with the techniques proposed in ANSI N15.18 for measuring the mass of UF_6 shipped and received in standard containers (30B, 48X, and 48Y). The techniques were of interest to the participants, as it has been demonstrated by selected portions of the nuclear industry that they comprise an efficient and economical method for weighing these items.

A questionnaire was distributed to the attendees of the meeting, inviting their response for participation in the program. Heavy participation is expected.

2. The questionnaires will be reviewed and summarized by selected membership of INMM 8.1 and a report prepared for a July 1979 meeting of INMM 8.1 and the interested participants at Albuquerque, New Mexico, in conjunction with the 20th Annual Meeting of the INMM.

3. The Administrator will prepare a proposed Standard Operating Procedure (SOP) for the program for presentation and review at the July Albuquerque meeting.

4. The results of the questionnaire and the proposed SOP will be discussed and the program modified accordingly at the INMM 8.1 meeting in July.

5. The NBS committee members will invite interested contractors and licensees for a "hardware" workshop at NBS during the month of August 1979.

6. The NBS administrator will make arrangements to ship the RMS to the first participant on or before October 1, 1979. (Goodyear Atomic Corporation—Portsmouth will store the RMS during the program.)

The membership of the INMM will continue to be informed of the progress of the program through articles in the INMM Journal and/or formal presentations at the annual meetings of the INMM.

References

1. Doher, Louis W. "INMM-8 Pilot Program," **Nuclear Materials Management**, Vol. V, No. 1, p. 47. Spring, 1976.

2. Pontius, Paul E. and Louis W. Doher. "The Joint ANSI—INMM 8.1—Nuclear Regulatory Commission Study of Uranium Hexafluoride Cylinder Material Accountability Bulk Measurements," **Proceedings, 18th Annual Meeting, Institute of Nuclear Materials Management**, June 29, July 1, 1977, Washington, D.C., p. 480.

3. Doher, Louis W., P.E. Pontius, and J.R. Whetstone. "A New Approach for Safeguarding Enriched Uranium Hexafluoride Bulk Transfers," **Proceedings, International Atomic Energy Agency Symposium on Nuclear Materials Safeguards**, October 2-6, 1978, Vienna, Austria.

N15-Focus on Results

(Continued from Page 31)

action indicates that only standards relating to specific "products" will be involved. According to the FTC a product is defined as follows: "A prescribed set of conditions or requirements, or portion thereof, applicable to any product in any market, established by agreement among buyers, sellers, professional groups, standards developers, certifiers, or others." According to this definition industry wide activities relating to nuclear safety and nuclear materials safeguards are excluded from FTC action. Therefore the proposed regulations have no expected impact on N15 Standards Committee activities.

These are but a few of the current activities in which the INMM N15 Standards Committee is currently involved. Please take the initiative to get involved in one or more areas of interest. Your continued contributions are vital to our future success.

A View From The Gas Queue

By Herman Miller, Chairman
INMM Public Information Committee
Redwood City, California

"Sail driven ships in the harbor, wood burning stoves in the White House! Revive the Pony Express, it might improve the mail service." (Al Hix)

While waiting in line at the gas station, I had the great fortune to find myself just ahead of a top Government official, B. Small. It was gas day for the odds. What a break! Since he is a man of the people, I decided to talk to him about a matter which was uppermost in my mind and had been troubling me. Particularly, since Bob and Tom at the INMM keep phoning me to do something.

After first checking to make sure nobody was trying to move in line ahead of me, I got out of my car and approached B. Small.

Mr. Small, I said, could I interrupt your meditation to discuss safeguarding of nuclear materials?

"What's that?"

Safeguarding of nuclear materials . . . you know! This is the program to protect strategic nuclear materials by the U.S. and other countries. This program has been in force for decades and considerable resources are being devoted to continually monitor and control nuclear material. As Chairman of the INMM Public Information Committee, I am charged with the responsibility of providing information on this program to the public. As a leading public official, you want to know what's being done and could tell us how we can improve this program.

"Well I don't really want to talk about Nuclear Safeguards, since I am against nuclear power. My Energy Committee has advised that nuclear power is not economic, it's unsafe, and it's not needed. They have looked at the energy sources that we now depend on, oil, coal, and nuclear; and concluded that we really have plenty of oil and gas, which our state is almost entirely dependent on, at reasonable prices. Even though we can't use more coal in our state, because of environmental controls; we can develop solar, wind and other undiscovered sources in a very few years, so nuclear power won't be required. That's why we killed the Desert nuclear power plant. We really think that should be a 1000 MW solar power plant."

Miller

Where will we get all the energy needed for home, industrial, and commercial use, including the increased amounts required to maintain our standard of living and bring the less well endowed up to the norm?

"Well my advisors and I have concluded we can get by with much less energy and we can replace some proposed large power plants with small coal and wood burning stoves, and solar and wind powered units that each family can install and operate. That will get us back to the small operations which are most beneficial and to the lifestyle everybody wants."

In other words, be more like American Motors than General Motors?

"Well I'd say even smaller, maybe more like Crosley Motors."

Do you think that individual units can be operated effectively by most families. They seem to have enough problems with their present equipment like automobiles, radios, TV, appliances, furnaces, air conditioners, and so on.

"Well the solar power people are working on simple systems that won't require much maintenance and adjustment. Once installed, they will produce trouble free power, at no cost, for as long as the sun is shining."

Where do you get your information and how do you make your decisions?

"Well, I select and appoint people from all walks of life to the key posts to handle the day by day business. For critical problems, I set up Committees to provide advice to me and my Administration. As an example, I have an Energy Committee to advise us on our energy programs.

The people I select for this Committee have had no previous experience in big government or big business and no connection with the industry or technology that they are studying. These people can therefore start from square one on the most complicated technologies, and be completely unbiased."

Could you give an example of your selection of people?

"Well, as an example, I have put D.K. in charge of my Energy Committee. He has had over ten years of experience in organizing anti-government and anti-business demonstrations. He knows how to get things moving. Oh yes, he's also married to that Broadway star that won the Tony Award. That sure increases his clout."

Yes, but how does that help in producing energy?

"You don't understand that our problem is not a shortage of energy, but that we are being ripped off by big government and big business. We have to get things back to manageable size. D.K. and I are planning to do this by having everyone install a wood burning stove in their living rooms for heat, solar panels for hot water and windmills to provide electricity."

But won't that produce more smog, visual pollution and home maintenance problems, in addition to being more costly than providing bulk energy? Things are cheaper at the supermarket than the corner grocer.

"Well I can see you really don't know anything about energy and the way our economic system works. You should do some meditating on that."

I will. What do you think about continuing to develop and use high technology like electric automobiles, jet airplanes, high speed commuters and so on.

"I think we should go back to a simpler life style. Life was less complicated and hectic then. After all earlier this century, with our soft technology, 90% of our families were living on farms, raising the food we eat. You know that now only 4% of our families can live on farms, and the rest of us must do other things, like you are doing. What do you do?"

I'm an engineer.

"Hmmm. Well now I didn't know that, but it really doesn't matter. You shouldn't rely so much on facts and information you get from books and technically trained people. It's more important to be in touch with your constituency, to talk and meditate. Then you can come up with consensus programs that the majority of people will support. You also should be more flexible; if public opinion changes, you must change, and quickly."

How about safety? I understand that more people have been killed or injured waiting in line for gas in the last month, than by civilian nuclear power in its twenty years of existence.

"Well that's just loose talk by the proponents of nuclear power. After all, we can't see radiation, and even though you can measure radiation at very very low levels of one part in one billion, we don't know what can happen; and besides, it's too complicated for the average person to understand. Besides, these gas lines are temporary, President Carter told me last week we were going to get increased allocations in our state next month."

Where is he going to get it?

"Well . . . What's your next question?"

I understand you may make a run for the Presidency, and took a recent fact finding trip to Africa with a friend. Looking at energy in the long range worldwide perspective, how do you feel about our present dependence on OPEC for almost half our oil and U.S. payments of over $50 billion per year for that oil. You know that even at present restrained levels of nuclear power, nuclear energy is replacing the equivalent of over 1,000,000 barrels of oil daily, at a savings in U.S. overseas payments of over $7 billion dollars per year. Equivalent oil savings from nuclear power now equals the present oil output from Alaska. That savings can double by 1985 even under our present program. Just think of that, we can save two million barrels of oil per day by 1985, using nuclear power plants that are operating or being built.

Well, I don't know that your facts are straight, but regardless, solar and wind look better to me."

You certainly qualify on the latter. How do you view the assured supply of energy for our children and grandchildren?

"Well, first and foremost, we have an unlimited supply from the sun, especially during the days when it doesn't rain or there's not fog; and in many places the wind blows hard most of the time. There's plenty of oil, if we could get the oil companies to release it. I suppose we would have to build more refineries and pipelines and ships. But we could do that. We're short of coal here in our State, but there's plenty in the U.S. We could build a big coal plant in another state and import electricity. The only fuel that's in short supply is nuclear."

The U.S. has more nuclear fuel in processed form than France, and France has more energy potential in its nuclear fuel stockpile than in all the proven oil reserves in Saudi-Arabia! Would you consider that a large supply?

"Well that doesn't sound right to me. Besides nuclear power costs much more. We can't afford it."

The National Economic Research Association reports nuclear electric power will cost 13% less than a coal-fired plant; and, of course, with OPEC skyrocketing prices, oil-fired plants cost much more than either coal or nuclear. And, of course, about 100% of the supplies and costs are U.S. controlled.

"Well my people don't believe such self serving calculations, we believe that nuclear power has to cost more because of all the government restrictions, delays, and added safety costs. We don't have to make technical and economic studies or use those made by others, it's just obvious to us from what we know."

How do you view the relationship of energy, jobs and standard-of-living?

"It is my policy to expand the economy and provide jobs for our ever increasing population. Those with lower incomes should be given opportunities to better themselves. We plan to do this by decreasing taxes and attracting new industry to our state."

I thought you were against proposition X to reduce taxes. As a matter of fact, didn't you say before the election, it would lead to disaster and chaos in the state?

"Well that was before the voters overwhelmingly approved it. Then it became obvious I was really for it all along."

Hmmmm.

"You must understand how these things are done, there is a logical development of ideas and policies which provide opportunities for all our citizens to get jobs."

Well aren't your governmental policies and red tape discouraging industry? I heard that a big chemical plant and an oil company gave up on big plants in our state after years of bureaucratic delay and are putting their money in other states.

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International Safeguards Discussions

By James P. Shipley
Safeguards Systems Studies Group
Los Alamos Scientific Laboratory

Dr. **James P. Shipley**, Alternate Group Leader of the Safeguards Systems Studies Group at Los Alamos Scientific Laboratory (LASL/Q-4), recently participated in safeguards discussions with the staffs of the International Atomic Energy Agency (IAEA) in Vienna, Austria, and the European Economic Communities Joint Research Center (EEC-JRC) at Ispra, Italy. The discussions, held during the last two weeks of January, were requested by the IAEA and the EEC-JRC as part of continuing efforts, sponsored by the U.S. Department of Energy—Office of Safeguards and Security, to maintain and improve the effectiveness of international nuclear safeguards.

Current concerns with increasingly stringent materials accounting requirements have led the IAEA and the Euration safeguards community to consider the possibility of near-real-time accounting. The Systems Studies Section at the IAEA, headed by Dr. **Tolchenkov**, is investigating the application of more timelý accounting techniques to fuel fabrication and reprocessing facilities. Similarly, the Fissile Material Control Project, under the leadership of Dr. **Marc Cuypers**, at the EEC-JRC is developing improved materials accounting methods for Euratom facilities, most notably fuel fabrication.

In the U.S., LASL has been a long-time proponent and major developer of near-real-time accounting concepts and systems. For example, the LASL Safeguards Systems Studies Group, led by Dr. **R.J. Dietz**, has defined and evaluated conceptual safeguards systems for several nuclear facilities in the backend of the LWR fuel cycle. Because of these mutual interests, particularly in the systems studies area, Dr. Shipley was invited to present aspects of the LASL safeguards work and to learn more about the special requirements of IAEA and Euratom safeguards.

The formal presentations began with a brief overview of the LASL safeguards program in instrumentation development, subsystem implementation, systems studies, and international safeguards. Following this survey, Dr. Shipley concentrated on the advanced materials accounting data analysis techniques being developed at LASL.

Dr. Shipley first gave a summary of the new methods available for detecting possible diversion of nuclear material. The methods have the common framework of decision analysis, and they operate on data from several balance periods to take advantage of relations among materials balances in discerning diversion patterns. The analyses are done sequentially in time for maximum efficiency, and all possible diversion scenarios are treated. An innovative display called an alarm-sequence chart provides the complete analysis results in easily readable and transparent form.

The application and usefulness of the techniques were illustrated with examples drawn from a study of a 3300 MT/yr plutonium nitrate-to-oxide conversion facility. The operations of the procedures and the appearance of the results to safeguards personnel in near-real-time were demonstrated by a movie showing the time evolution of the analyses.

Dr. Shipley concluded the presentations with several important points. First, the new methods work not just because of the availability of additional data with a near-real-time accounting system, but through more complete analysis of the same kinds of accounting data that traditionally have been gathered for the standard MUF/LEMUF treatment. Second, the techniques can predict the expected performance of proposed materials accounting systems, and can provide near-real-time analyses of data from operating facilities. Finally, the methods are structured for efficiency, comprehensiveness, and consistency, all essential characteristics for future safeguards systems.

The visit ended with discussions of the benefits and difficulties of applying the advanced data analysis techniques to the real problems of the IAEA and the EEC-JRC. Although the potential advantages are apparent, it is clear that further engineering development is required for the methods to be routinely practicable. Engineering development of the kind necessary to make the tools useful to safeguards practitioners must take place in facilities like those to be safeguarded. The Tokai Advanced Safeguards Technology Exercise (TASTEX) is an example of current efforts for small-scale reprocessing plants. Advancement of safeguards technology and methods for putting the enhanced capabilities into practice were identified as areas of common interest for continued collaboration.

Shipley

BOOK REVIEW

Nuclear Policies: Fuel Without the Bomb, Wohlstetter, A., Gilinsky, V.; Gillette, R.; and Wohlstetter, R., Ballinger, Cambridge, Mass., 1978

Reviewed by **Anthony Fainberg**
Brookhaven National Laboratory
Upton, Long Island, New York

A set of five essays on nuclear proliferation has been put together by the California Seminar on Arms Control and Foreign Policy as a result of several meetings held in the past few years. The prime force in organizing these conferences was, according to Robert Bacher's foreward, Professor Albert Wohlstetter, one of the most outspoken advocates of restraint in developing a plutonium cycle.

This is an opportune time for discussing proliferation hazards for a number of reasons. First, there is the very recent "Pakistan affair," wherein the U.S. has suspended all aid to that country because of its surrepititious efforts to collect enrichment equipment. A study of this matter as a model scenario for proliferation would be interesting, and comparison of the general points of view of the essayists with the specific example is most instructive. Second, virtually undisguised efforts by some nations to develop nuclear weapons may have an impact on nuclear energy, (let alone an impact on world peace), which is not less than that of Three-Mile Island, and thought must be given immediately in order to develop actions to thwart such efforts. Third, the Non-Proliferation Treaty (NPT) will come up for review in the near future, and the impact of the Carter policies on the attitudes of non-weapons states must be understood ahead of time if we do not wish the whole fabric of international accords on non-proliferation to dissolve.

This volume makes several contributions to the current dialogue. **Robert Gillette's** essay lays out logically and clearly the main technical features of current and possible future nuclear power reactors and cycles. Not too many conclusions are drawn, but the exposition is useful. One disagreement I have, however, is with his statement that "necessary chemical processes (for reprocessing) are within the capabilities of even modestly developing nations," as proved by India's success. Gillette does recognize that "India's nuclear establishment ranks high in the developing world" (as should be obvious), but he seems unaware that India is one of the biggest economic powers of the world, in spite of its extreme poverty. The fact that India can reprocess spent fuel does not prove that "modestly developing nations" are capable of the same thing.

Professor Wohlstetter's article, the centerpiece of the collection, is rather more contentious. It is essentially an updated version of his testimony before Justice Parker's inquiry, setting forth a position against the enlargement of the Windscale reprocessing facility on economic and proliferation based grounds. Wohlstetter's intervention was on behalf of Friends of the Earth, Ltd.

Wohlstetter first summarizes the arguments of British Nuclear Fuels, Ltd., (BNFL), in favor of reprocessing. The main argument touched upon deals with proliferation hazards, inherent in the widespread export of Pu and PuO_2 and of the technology to reprocess spent fuel. BNFL contended: a) that a nation intending to obtain nuclear weapons would not be significantly impeded by a prohibition on the export of such knowledge or materials and b) that a nation denied reprocessing assistance would thus be encouraged to develop such a capability by itself. Wohlstetter's responses are less than convincing. He acknowledges that critical facilities with their large Pu inventories, which are legitimized by international agreement, already provide an excellent opportunity for the rapid production of nuclear weapons. This seems to render irrelevant the concern about plutonium reprocessing, which would require a great deal of effort. Wohlstetter does not overtly suggest here that critical facilities should be banned from non-weapons states (although he suggests it), and unless he does, it makes little sense to be concerned about the much lesser dangers in reprocesing and Pu export, particularly if safeguards are improved and strengthened. I should add, that because of the "volatility" of large stocks of Pu (or highly-enriched U) in critical and other research facilities, it seems imperative to try to develop international agreements, if not to ban them from non-weapons states, then to increase greatly IAEA control in this area, perhaps restricting them to multi-national centers of research and/or fuel cycle operation.

Wohlstetter does make the excellent point that widespread diffusion of reprocessing technology could make "timely detection" of diversion by the IAEA impossible. A group of knowledgeable and experienced technicians and engineers could construct a small clandestine reprocessing facility which could, within days, or, at most, weeks of diversion produce bomb-grade materials. It is pointed out by Wohlstetter (and later by Gilinsky), that the production-reactor route to Pu is not as attractive to a weapons-seeking state. Being difficult, if not impossible, to hide from satellites or intelligence agencies, a production reactor would signal to the world the intention of constructing a bomb and thus bring down international opprobrium, sanctions, and a uranium embargo upon the guilty party. Thus, spread of reprocessing technology contains a clear incremental risk and does rely on safeguards to render it acceptable. The question is, is the incremental increase in risk inherent in reprocessing great enough to warrant its avoidance, at least for now? Wohlstetter thinks so, and I feel bound to agree with him (and with the current administration), particularly given the minute economic ad-

Fainberg

vantages which would accrue. These are detailed by Wohlstetter, relying on ERDA and GESMO reports. The argument that reprocessed fuel is necessary to start off breeders is true, but since breeders will not start on a large scale for twenty years, if ever, they cannot be used as a reason for current reprocessing.

I am inclined to consider the risk of proliferation through reprocessing to be rather small (if IAEA safeguards are radically improved, as one expects is being done). However, in the absence of compelling reasons to do so, I see no reason for taking the risk at all.

However, there is another argument which is not easily dismissable. That is, that states will seek to remove their dependence on the U.S. for the low-enriched uranium to provide a large fraction of their electrical energy; that they will be pushed to develop their own reprocessing facilities, which could be **outside** international safeguards, thus resulting in a worse situation than export of reprocessing **under** safeguards, as is now being proposed. Wohlstetter's answer is that, following BNFL's proposals, fuel dependence on the U.S. would be replaced by fuel dependence on the IAEA, and thus non-weapons states would not be any more independent. It appears to me that this argument ignores international realities—namely, that the U.S. would find it far easier to cut off supplies to a particular nation (an executive order could accomplish this at once) than the IAEA would. The latter entity makes decisions by international consensus (including input from the affected state) and, by virtue of the ponderous nature of international decision-making, which is strongly influenced by international politics, alliances of convenience, etc., it is by no means certain that the IAEA would be able to embargo a member state. Further, the U.S. could arbitrarily and capriciously cut off supplies whereas, in reality, the IAEA could not. Thus, there is no doubt that a state is better off if its supplies were assured by the IAEA rather than by the U.S.

A better response to this problem of encouraging independent reprocessing if one calls a halt to Pu export is to be found elsewhere (and the argument is admittedly weak). It lies in the existence of several sources of low-enriched uranium which would be available on the world market, thus increasing in real terms, the independence of customers for nuclear fuel. Combined with an international halt to the export of reprocessing technology among the few states with this ability, one could achieve a situation where there would be little incentive for a state to go the plutonium route. This, I suggest, is desirable.

Historically, regarding acquisition of nuclear weapons in the past, Wohlstetter is not quite as straightforward as he could be. It has been argued that nuclear power programs have never led to weapons in the past—the research reactor, dedicated production reactor, or enrichment have been the paths used until now, and thus, concern about the proliferation hazards of nuclear-generated electricity is misplaced. This is not really a good argument, because, irrespective of the past, the international availability of weapons-grade material resulting from the spread of nuclear power can well be a factor in the future. All the argument really can show is that banning nuclear power by no means assure non-proliferation. Nevertheless, Wohlstetter seems needlessly sensitive about this point, and he tries to show that in the cases of the United Kingdom, France, and India, the existence of plutonium production facilities, which could (but even now, after decades, have not yet done so) provide fuel for power reactors, were opted for in advance of the overt decision to make nuclear explosives. The weasel word here is "overt" (which Roberta Wohlstetter seems to have forgotten while repeating this line in her essay). Since no plutonium-fueled power reactors exist even today, 34 years after the British decision to acquire plutonium, and since the nuclear explosives in question have been in existence for quite a while, it is clear to a rational observer that the decision to produce plutonium was a decision to make explosives or at least to reserve the option. The fact that the decision was not "overt" but shrouded in secrecy is not at all puzzling. The British wanted a weapon because the U.S. had frozen them out, the French (particularly de Gaulle, who insisted upon a strong weapons program) wanted grandeur and the Indians wanted to have something which to oppose China and to gain prestige in the Third World. None was pushed along to explosives by a fortuitous existence of plutonium stockpiles.

In summary, I am able to agree with Wohlstetter's opposition to the spread of reprocessing, but I do find that several of his arguments are not well thought out. I should add that although Professor Wohlstetter is widely perceived in the nuclear business as being anti-nuclear, he clearly states "I believe that some forms of nuclear electric power are comparatively safe, and that they will play a useful role in the generation of electricity." I see no reason to doubt him.

Roberta Wohlstetter deals with the Indian case in her article. She draws several conclusions which bear mentioning. One is related to Wohlstetter's argument above, that India had somehow drifted into nuclear explosives because the plutonium happened to be there. I have already indicated that I find this unconvincing. The plutonium was there to provide India with the ability to choose. The lack of public statements by Indian leaders in the late 50's and early 60's regarding intentions relative to nuclear explosives is in no sense an indication of pure and peaceful intent.

Another conclusion is the hypocrisy of India's disarmament rhetoric while acquiring nuclear explosives. Of course, India is hardly alone in this regard. From this comes the corollary that "current pure intentions are not enough" when a state receives nuclear material. One can hardly disagree. Also, the conclusion that safeguards, while necessary, are not sufficient, especially if only partial safeguards apply in a given state, is quite true, and one which some of us in the safeguards business tend to forget at times. It is only human to miss forests because of trees.

However, the main and most controversial conclusion of R. Wohlstetter is that U.S. policy should end absolutely all nuclear assistance to countries which refuse to forgo nuclear explosives (e.g., India, Pakistan, Israel, Egypt, etc.). This may hurt some people's sensitivities, but it appears to me to be essential if we are serious about non-proliferation and if we wish to convince the world of our seriousness.

The final two articles are by **Victor Gilinsky,** one of the five Nuclear Regulatory Commissioners. In the first, he expresses support for the Carter Administration's attempt to halt the spread of plutonium internationally. He supports the effort to negotiate a ban on any and all (including "peaceful") nuclear explosives and urges the subjection of all nuclear activities of a state to IAEA safeguards, to close some current proliferation loopholes. Finally, he feels that controls on retransfers of nuclear material and reprocessing should be increased by agreement with (particularly) Euratom. This will certainly be the subject of a difficult negotiating process.

While his first essay is a reasoned general defense of current policies on proliferation, the second becomes somewhat more specific. Here, he is particularly concerned with the stockpiling of separated plutonium from spent fuel. Even if this material is meant for power production, its existence allows a state to construct a bomb(s) with it quite rapidly (in days, if the non-nuclear parts have already been built and tested), thus rendering "timely detection" by the IAEA virtually impossible. This provides a strong argument against allowing reprocessing facilities to proliferate throughout the world, an as Gilinsky points out, it is easier rapidly to divert material which you legally possess than to build a clandestine facility.

What are we to conclude from this volume? A strong case is made for restricting the flow of plutonium throughout the world. The point is made in several of the essays that reactor-grade plutonium has been used as a nuclear explosive, and one hopes that this argument has finally been put to rest. Reprocessed spent fuel is fine for a bomb.

Further, the existence of experimental facilities with large amounts of Pu or U-235 is a most serious danger and should be examined as to how the danger can be reduced. Also, to be effective in halting the spread of nuclear weapons, one must impose sanctions on those who refuse to abandon the attempt to acquire them. This is a good place to take the quantum leap from generalized policy discussions to the real world.

Pakistan has recently been discovered to be in the early stages of trying to construct an enrichment facility. This empirical case can be used to test a variety of theories regarding proliferation and safeguards. How was the attempt discovered? Not through IAEA safeguards. The project was clandestine and the IAEA didn't enter the picture, except in that it may have deterred Pakistan from taking an easier course. In this case, one could argue that Agency safeguards did their job.

The Pakistanis were, first of all, high on the list of states suspected of wanting a bomb, having a strong motive (India's explosion), technical ability and a suspicious interest in (guess what?) a reprocessing facility that the French almost provided them. Note that last December, former Prime Minister **Ali Bhutto** announced that his interest in reprocessing had been definitely for producing a bomb. Thus, one presumes that various Western intelligence units were wary to begin with. There seem to have been at least two clear actions which revealed what was up. One was an attempt by Pakistanis to buy high frequency inverters and high-grade steel, both of which are essential to separation centrifuges. The cover story was that the inverters were for a textile mill. Perhaps the lead of Israel (which originally did claim the Dimona reactor was a textile mill) was followed. The second action was the extreme interest shown by a Pakistani expert in the details of the URENCO separation facility. These facts, and perhaps others not yet announced, were put together, primarily by Dutch intelligence, and the cat was out of the bag. The U.S. reaction was not only to cut nuclear assistance (as R. Wohlstetter suggested in such a case), but to cut all aid to Pakistan, as well as to try to encourage India to renounce nuclear weapons (also suggested by R. Wohlstetter).

All this makes one think that safeguards have some use, but obviously must be supplemented by intelligence agencies and by an elementary knowledge of international events and relations. This brings me to a final point. There are other nations besides Pakistan who would like nuclear weapons. For example, Libya has openly announced through high officials that it is seeking nuclear weapons. In fact, there are stories that part of the Pakistani effort was financed by Libya to get a "Moslem bomb." However, Libya, as an NPT signatory, is receiving a Soviet-made power reactor as well as a research reactor. **Jeremy Stone,** of the Federation of American Scientists, has pointed this out, but, to my knowledge, the State Department has not made any representation to the IAEA or to the Soviet Union that Libya, which has, by stating its intentions, **ipso facto,** violated the NPT, has no right to reactors supplied by NPT signatories. This drastic "oversight," perhaps caused by our dependence on Libyan oil, cannot go uncorrected. We cannot rely forever on Qaddhafi's clumsiness and incompetence to prevent proliferation.

What about the future? Aerodynamic nozzle or laser techniques will certainly one day allow many other states to produce the bomb. I contend that, as much as one may try, the proliferation of knowledge may be slowed but not stopped. The nuclear nations will have a very hard time convincing other states to renounce voluntarily all lines of research that could lead to nuclear weapons. In fact, it is clearly impossible, in spite of the desires of naive commentators, to proscribe say, laser research, particle accelerators, and research reactors from have-not nations. This attempt will be met by derision and by changes of technological imperialism. Specific cases, such as critical facilities, may possibly be controllable, but even this will be difficult.

All one can hope for, is to slow proliferation enough so that, by the time non-weapons nations are able to produce the bomb, other technological advances and the international political climate will be such that the resulting destabilization will be negligible. This is probably already the case regarding the Indian explosion. In a looser sense, one could argue that this was also the case regarding the UK, France and China.

Therefore, we must realize that in fighting nuclear proliferation, we are fighting a holding action. The main object is to keep the bomb out of the hands of, particularly, irresponsible parties (including sub-national groups) by stringent safeguards, and to remove incentives to states to acquire weapons. The latter also requires safeguards, but, in addition, one needs to keep temptation at a minimum by severely restricting the availability of plutonium, and mainly by structuring the international situation so that no real advantage accrues to a new member of the nuclear weapons club.

SEVENTH IN A SERIES

Titles and Abstracts of Recent Safeguards R & D Publications and Reports

Editor's Note—This is the seventh in a series of listings of titles and abstracts of recent safeguards R&D publications and reports from agencies and R&D laboratories. It has been compiled by the professional staff of Battelle, Pacific Northwest Laboratory, Richland, Washington. A second series of listings from Canada is published elsewhere in this issue. Tentatively, the Fall Issue (Volume VIII, No. 3) will have a similar listing from Japan. If your agency or R&D laboratory is interested in being included in this series, please contact the editors, William A. Higinbotham (516-345-2908) at Brookhaven National Laboratory, or Thomas A. Gerdis (913-532-5837) at Kansas State University, Manhattan.

C.L. Timmerman, **Isotopic Safeguards Techniques**, International Safeguards Project Office, ISPO-25, PNL-SA-6761, June 1978. A generalized discussion of the application, demonstration and implementation of isotopic safeguards techniques to plutonium input measurements for chemical reprocessing facilities.

C.L. Timmerman and K.B. Stewart, **Isotopic Safeguards Statistics**, International Safeguards Project Office, ISPO-26, PNL-SA-6595, June 1978. The methods and results of statistical analysis of isotopic data using isotopic safeguards techniques are illustrated using example data from the Yankee Rowe reactor. The illustration provides greater insight into how statistics can be used to analyze and extract meaningful results from isotopic data. The statistical methods used are the paired comparison and regression analyses. A paired comparison results when a sample from a batch is analyzed by two different laboratories. Paired comparison techniques can be used with regression analysis to detect and identify outlier batches. The second analysis tool, linear regression, involves comparing various regression approaches. These approaches use two basic types of models: the intercept model ($y = a + Bx$) and the initial point model [$y - y_o + B(x - x_o)$]. The intercept model fits strictly the exposure or burnup values of isotopic functions while the initial point model utilizes the exposure values plus the initial or fabricator's data values in the regression analysis. Two fitting methods are applied to each of these models: (1) the usual least-squares fitting approach where x is measured without error, and (2) Deming's approach which uses the variance estimates obtained from the paired comparison results and considers x and y are both measured with error. Some statistical results using the Yankee Rowe data are presented. Review of these results indicates the attractiveness of Deming's regression model over the usual approach by simple comparison of the given regression variances with the random variance from the paired comparison results.

B.A. Napier and C.L. Timmerman, *Developing Isotopic Functions*, International Safeguards Project Office, ISPO-27, PNL-SA-6594, June 1978. The operation of nuclear reactors results in burning of uranium isotopes and production of plutonium isotopes. The burnup and transmutation of the uranium is a process yielding simple relationships between the amount of uranium remaining and the amount of plutonium produced. Certain simple relationships among isotopic concentrations have been observed to be remarkably consistent over various reactor types or burnup conditions. These simplified relationships are known as isotopic functions and generally consist of ratios of two isotopic variables. An isotopic variable can consist of sums or products of one or more isotopic concentration(s) or total elemental weight(s). The use of isotopic functions is a developed empirical method of regaining the simplicity of the transmutation relationships. Isotopic functions can be used in the verification of plutonium and uranium concentration measurements of spent fuel at the head end of the reprocessing plant for safeguards and/or nonproliferation purposes. They can also be used to verify or improve theoretical models. Knowledge of the existence and importance of isotopic functions has led to the development of a systematic method of forming and evalucating them. The method used at Battelle to form and evaluate isotopic functions is described in this paper, including definition of those properties considered to be important.

C.L. Timmerman, G.P. Selby and B.A. Napier, *Selected Isotopic Functions: A Description and Demonstration of Their Uses*, International Safeguards Project Office, ISPO-37, PNL-2761, October 1978. The report includes a description of eleven selected isotopic functions useful in the safeguards verification of input accountability measurements at a reprocessing facility. It provides a summary of how various factors affect the selected isotopic functions for pressurized water reactors. A similar summary table is provided for boiling water reactors. The two tables summarize the descriptive portion of the report. The eleven isotopic functions use pairings of various combination variables of uranium and plutonium isotopics and totals. A description and explanation of these variables and functions are provided in the report. Also included in a demonstration of the verification process utilizing isotopic safeguards techniques. The example used is the verification (or nonverification) of various Pu/U measurements. The technique demonstrates both the internal consistency check and the external data source verification which uses a similar reference data source.

K.B. Stewart, *Statistical Programs for Analyzing PAFEX Interlaboratory Test Data*, International Safeguards Project Office, ISPO-10, PNL-2571, December 1977. In November 1973 the IAEA initiated an experiment called PAFEX I (Process Analysis Field Experiment). The purpose of the experiment was twofold: (1) to study the administrative and logistical problems that occur when an international network of analytical laboratories is used to perform chemical analyses of samples taken during IAEA inspections, and (2) to obtain estimates of measurement error components of variance. The variance components determined on the basis of the experiment were expected to be typical of those that would arise in the course of IAEA inspections. PAFEX I was a cooperative effort involving analytic laboratories in nine countries. The IAEA arranged for preparation of the samples and coordinated the shipments to each laboratory. Three kinds of samples were analyzed in PAFEX I: plutonium nitrate solution, plutonium dioxide powder, and mixed oxide pellets. A second experiment, PAFEX II, was undertaken in December 1974. The objectives were similar to those of PAFEX I except that PAFEX II involved samples of dissolver solution from a reprocessing plant. This report describes several computer programs developed to analyze data generated by the PAFEX experiments. The programs apply the methods of analysis of variance to produce estimates of variance components and to perform statistical signifiance tests. The actual data analysis has been reported elsewhere. This report presents only the statistical tools and computer programs. Four computer programs were developed for analyzing the data generated during the PAFEX experiments. The computer program PAFEX is used to obtain variance compenent estimates. The computer programs NONORT I and II and NONINT do a nonorthogonal analysis of variance to test for the statistical significnace of effects, both main and first order interactions. Computation of nonorthogonal analyses of variance becomes very formidable on a desk calculator. Detailed descriptions of the program use are given in the report.

R.J. Sorenson, T.I. McSweeney, M.G. Hartman, R.J. Brouns, K.B. Stewart and D.P. Granquist, *Independent Verification of a Material Balance at a LEU Fuel Fabrication Plant*, International Safeguards Project Office, ISPO-7, PNL-2418, November 1977. This report describes the application of methodology for planning an inspection according to IAEA procedures, and an example evaluation of data representative of low-enriched uranium fuel fabrication facilities. Included are the inspection plan test criteria, inspection sampling plans, sample data collected during the inspection, acceptance testing of physical inventories with test equipment, material unaccounted for (MUF) evaluation, and quantitative statements of the results and conclusions that could be derived from the inspection. The analysis in this report demonstrates the application of inspection strategies that produce quantitative results. A facility model was used that is representative of large low-enriched uranium fuel fabrication plants with material flows, inventory sizes, and compositions of material representative of operating commercial facilities. The principal objective was to determine and illustrate the degree of assurance against a diversion of special nuclear materials (SNM) that can be achieved by an inspection and the verification of material flows and inventories. This work was performed as part of the U.S. program for technical assistance to the IAEA.

F.P. Brauer, W.A. Mitzlaff and J.E. Fager, *Uranium and Plutonium Analysis with Well-Type GE(Li) Detectors*, PNL-SA-6600, March 1978. Analysis of microgram and submicrogram quantities of ^{235}U and ^{239}Pu are required by the nuclear industry for process control, nuclear safeguards and effluent measurements. These analyses are of increasing importance in efforts to reduce inventory discrepancies and uncertainties. Current analytical laboratory methods used for measurement of small quantities of uranium and plutonium include X-ray fluorescence methods, spectrophotometric methods, fluorometric methods, radiometric methods and mass spectrometric methods. Many of these analytical laboratory methods measure only total plutonium and uranium while newer nondestructive analysis (NDA) methods, which have been developed primarily for in-plant use, can measure specific isotopes of uranium or plutonium. Adapation of some of the NDA techniques to the analytical laboratory would result in more rapid and more specific analyses. This paper discusses an NDA method for rapid laboratory analysis of ^{239}Pu and ^{235}U. Gamma-ray spectrometric methods can be used in the analytical laboratory for both direct measurement of sample aliquots (NDA) and for performing measurements on samples following laboratory processing. Samples can often be prepared for gamma-ray spectrometric measurements with considerably less effort than is required for measurement by other methods. Gamma-ray spectrometric methods can measure specific radionuclides, an important consideration in facilities processing enriched uranium. Gamma-ray spectrometric methods also differentiate between ^{241}Am and plutonium and can be used for plutonium isotopic analyses. A well-type Ge(Li) detector was used for measurements on standard uranium ore, uranium and plutonium samples. This paper discusses the results of these measurements and the application of X-ray and gamma-ray spectrometric measurements to laboratory uranium and plutonium determination.

F.P. Brauer, J.E. Fager, J.H. Kaye and R.J. Sorenson, *A Mobile Computerized Gamma Ray Spectrometric Analysis and Data Processing System,* PNL-SA-6571, March 1978. A mobile system was designed, assembled and evaluated. The system consists of a specially constructed vehicle, multichannel analyzer, and data processing equipment mounted in the vehicle, and GE detectors that can be moved to locations external to the vehicle for use. Applications of the system include nuclear material safeguards measurement, in-situ radionuclide analysis, activation analysis and research support.

F.P. Brauer, J.E. Fager, J.H. Kaye and R.J. Sorenson, *A Mobile Nondestructive Assay Verification and Measurement System,* INMM, Nuclear Materials Management, VI, No. III, Fall 1977, pp. 680-694. A mobile, real-time, nondestructive assay system was developed for both nuclear material safeguards inventory verification and measurements on the Hanford project. The system includes electronic and computer support equipment mounted in a specially constructed vehicle, and passive and active neutron and gamma-ray measurement equipment transported in the vehicle but operated external to the vehicle. The system significantly increases safeguards verification and measurement capabilities.

T.I. McSweeney and R.J. Sorenson, *The Role of Assurance in Material Safeguards,* INMM, Nuclear Materials Management, VI, No. III, Fall 1977, pp 265-276. The role of assurance in materials safeguards has not been as clearly defined or emphasized as much as other safeguards measures. An effective assurance program provides a safeguards element not found in other safeguards measures, namely, that the physical protection and material control systems have been effective. This paper describes a quantitative assessment plan that can demonstrate such effectiveness. The major difficulties with evaluating safeguards measures are (1) defining a realistic goal for the assessment activities, and (2) obtaining the required data to quantify the results. It is much easier to assess for compliance with requirements than it is to evaluate systems' effectiveness and to express the results in a quantitative assurance statement. Statistical techniques are available to quantify many of the assessment activities. They require the concept of protecting against the diversion of a prescribed quantity of material, i.e., of goal quantity. Because of the difficulties associated with assessment, a number of strategies are employed depending on the specific situation. This results in a structured approach to assessment that emphasizes evaluating all of the strata from which diversion by an adversary is possible. Both the flow components as well as the more traditional inventory components are included because diversion from both strata classifications can occur. This paper summarizes the methodology and use of various strategies in a structural approach to assessment, which allows for quantifying the results. It also describes a computer code that enables rapid development of an assessment plan based on both the operation status at the time of the assessment and the material transfers since the previous evaluation. The application and limitations of the methodology are also presented.

C.L. Timmerman, D.E. Christensen and K.B. Stewart, *Statistical Evaluation of Isotopic Safeguards Data,* INMM, Nuclear Materials Management, VI, No. III, Fall 1977, pp 559-566. Statistical methods are being applied to the data base used in evaluating isotopic safeguards techniques. The statistics are used strictly as a means to achieve confirmation of the verification of the desired isotopic content. Utilizing two basic statistical approaches, paired comparisons and regression analysis, three conclusions have been reached thus far based on these statistical evaluations: (1) the random variance estimates determined from paired comparison analyses (where both a reprocessing facility measurement and an independent measurement are compared) and from least-squares regression analyses, indicate that a regression model where errors in both the x and y variables are considered is desirable, (2) a regression model is needed to analyze the data from enrichment groupings that will give consistent relationships between y and x. Several models are being considered, all of which have some advantages and disadvantages. Based on a study of the methods applied to the data bank, a given model does not have a clear advantage over another model, and (3) anomalous results are identified on a more positive level using results from both paired comparison and regression analyses rather than from either one separately.

M.F. Mullen, *Comparative Study of Materials Control Practices in Industries Handling Valuable Materials,* PNL-2469, November 1977. At the request of NRC/I&E a brief survey was undertaken of the materials control practices of industries that handle valuable materials. The objective was to compare materials control practices in the nuclear industry with those in other industries handling precious materials. This information would be helpful in assessing the reasonableness of (a) present or proposed nuclear materials control systems, and (b) process losses experienced in various nuclear fuel processing facilities. The findings reported here are necessarily limited both in scope and in depth because of the limited time available for the study. The cases studied are not necessarily representative and the materials control systems may not, in fact, function as effectively as described. The first section of the report describes briefly some earlier studies. Later sections outline materials control practices at the U.S. Mint (gold and silver), a copper smelter (gold and silver), a gold and silver refinery, a chemical plant (platinum), an alcohol plant and a winery.

T.I. McSweeney, R.J. Sorenson, R.J. Brouns, D.W. Brite, F.P. Brauer, M.G. Hartman and R.E. Kleinknecht, *Summary of Findings Evaluating Material Accounting Losses at Four Licensee Facilities,* PNL-2467, April 1978. This report summarizes the findings of investigations of nuclear material accounting systems of four licensee facilities. These studies were performed at the request of the Nuclear Regulatory Commission. The purpose of the investigations was to identify from an engineering viewpoint the probable causes of the licensees' loss mechanisms and determine if they are reasonable for these facilities. The studies provide an outside and independent perspective of any problems, but no attempt was made to determine whether licensees were in compliance with regulations. The overall objective of the studies was to provide NRC/I&E with a technically sound basis for evaluating the significance of materials accounting differences reported by the licensees. This objective was broken into three tasks: (1) evaluate and identify process loss mechanisms from a process engineering viewpoint, (2) analyze characteristics of the material balance and measurement systems, and (3) acquire an overall insight into the licensees' material accounting concerns. This document summarizes (1) generic material control and accounting concerns, (2) recommended short- and long-term goals for upgrading licensee material control and accounting programs, and (3) prioritized plant-specific deficiencies. The generic material control and accounting concerns are listed in the recommendation section. The body of the report discusses plant-specific deficiencies.

R.G. Clark, R.J. Brouns, A.D. Chockie and L.C. Davenport, *Estimated Incremental Costs for NRC Licensees to Implement the US/IAEA Safeguards Agreement*, PNL-2884, January 1979. At the request of the NRC, PNL conducted a brief study to identify the incremental cost for implementing the US/IAEA safeguards treaty agreement. The purpose of the study was to develop an estimate of the cost impact to eligible NRC licensees for complying with the proposed Part 75 of Title 10, Code of Federal Regulations (10 CFR 75), the rule that will implement the treaty. The study was conducted using cost estimates from several eligible licensees who will be affected by the agreement and from cost analyses by PNL staff. A survey instrument was developed and sent to 25 NRC licensees, some of whom had more than one licensed facility. Their responses were obtained primarily by telephone after they had reviewed the survey instrument and a list of assumptions. The primary information received from the licensees was the incremental cost to their particular facility in the form of manpower, dollars or both. In summary, the one-time cost to all eligible NRC licensees to implement 10 CFR 75 is estimated by PNL to range from $1.9 to $7.2 millions. The annual cost to the industry for the required accounting and reporting activities is estimated by PNL at $0.5 to $1.4 millions. Annual inspection costs to the industry for the limited IAEA inspection being assumed is $480K to $160K.

R.J. Sorenson, F.P. Roberts, R.G. Clark, R.J. Kofoed, R.J. Brouns, R.F. Eggers, J.C. Gibson, F.L. Adelman, J. Ballantine, J.F. Fagan, Jr., C.R. Schuller, D. Lowenfeld, R.A. Morris and A.M. Hankardt, Jr., *Proliferation Resistance Design of a Plutonium Cycle*, PNL-2832, January 1979. This report describes the proliferation resistance engineering concepts developed to counter the threat of proliferation of nuclear weapons in an International Fuel Service Center (IFSC). These concepts include (1) facility design and process considerations that provide passive resistance to proliferation, or enable the application of active use-denial technology, (2) technical aspects of a command, control and communication system (C^3) necessary to initiate active use-denial penalties, and (3) description of active use-denial technology that is either currently available or under development in other DOE programs. In addition, descriptions of the basic elements of an IFSC, including fuel reprocessing, fuel refabrication, product storage, transportation systems, the reactor facility, waste management process, and an advanced safeguards system are presented. Possible methods for resisting proliferation such as processing alternatives, close-coupling of facilities, process equipment layout, maintenance philosophy, process control, and process monitoring are discussed. The political and institutional issues in providing proliferation resistance for an IFSC are analyzed in terms of three major issues: (1) political acceptability of introducing passive and active use-denial technologies into an IFSC located in a host country, (2) the value of multinational presence in enhancing or reducing proliferation resistance, and (3) issues of organization, management and operation of a proliferation resistant IFSC. The conclusions drawn from a study of the major issues are: (1) use-denial can provide time for international response in the event of a host nation takeover. Passive use-denial is more acceptable than active use-denial, and acceptability of active denial concepts is highly dependent on sovereignty, energy dependence and economic considerations, (2) multinational presence can enhance proliferation resistance, and (3) use-denial must be nonprejudicial with balanced interests for governments and/or private corporations being served. The incremental costs imposed on the design, construction and operation of an IFSC by including the PRE concepts have been estimated Comparisons between an IFSC as a national facility, an IFSC with minimum multinational effect, and an IFSC with maximum multinational effect show incremental design costs to be less than 2% of total cost of the baseline non-PRE concept facility. The total equipment acquisition cost increment is estimated to be less than 2% of total baseline facility costs. Personnel costs are estimated to increase by less than 10% due to maximum international presence. The work performed in the PRE program has shown that the concepts as viewed on an integrated basis have been developed to the stage where they could be considered as plausible. Further work must be performed to make a conceptual definition possible. The authors of this report represent the following contractors: Pacific Northwest Laboratory (PNL); Sandia Laboratories, Livermore (SLL); System Planning Corporation (SPC) and Battelle Human Affairs Research Center (HARC).

R.J. Brouns, F.P. Roberts and U.L. Upson, *Considerations for Sampling Nuclear Materials for SNM Accounting Measurements*, NUREG/CR-0087, PNL-2592, May 1978. This report presents principles and guidelines for sampling nuclear materials to measure chemical and isotopic content of the material. Development of sampling plans and procedures that maintain the random and systematic errors of sampling within acceptable limits for SNM accounting purposes are emphasized.

R.J. Brouns and F.P. Roberts, *Procedures for Rounding Measurement Results in Nuclear Materials Control and Accounting*, NUREG/CR-0033, PNL-2565, November 1977. This report defines procedures for rounding measurement results for nuclear material control and accounting. Considerations for the applications of these procedures are discussed.

K.B. Stewart, *Minimum Variance Linear Unbiased Estimators of Loss and Inventory*, INMM, Nuclear Materials Management, VI, No. 4, Winter 1977-78, pp 47-54. The article illustrates a number of approaches for estimating the material balance inventory and a constant loss amount from the accountability data from a sequence of accountability periods. The approaches all lead to linear estimates that have minimum variance. Techniques are shown whereby ordinary least-squares, weighted least-squares and generalized least-squares computer programs can be used. Two approaches are recursive in nature and lend themselves to small specialized computer programs. Another approach is developed that is easy to program, could be used with a desk calculator, and can be used in a recursive way from accountability period to accountability period. Some previous results are also reviewed that are very similar in approach to the present ones and vary only in the way net throughput measurements are statistically modeled.

R.J. Cole, C.A. Bennett, H. Edelhertz, M.T. Wood, R.J. Brouns and F.P. Roberts, *Structure and Drafting of Safeguards Regulatory Documents*, NUREG/CR-0377, BNWL-2408, September 1977. The objective of this study was to develop hypotheses about the relationship between the structure and drafting of safeguards regulatory documents and the ability of the document users to understand and implement them in a way that reflects the intent and requirements of the NRC. Licensing offices, licensees, inspectors, and the general public must understand the NRC's requirements if the regulatory system is to function effectively and in compliance with legal requirements. Unless the NRC's processes for setting standards and imposing license conditions can communicate to licensees and others what they are expected to do, and unless inspectors understand what to inspect, the NRC cannot achieve the objectives of its safeguards program. Improving communication will require a sequence of decisions. Certainly the first and most important decision is: (1) should improvement of safeguards regulatory documents as communication instruments be an explicit NRC program? If an explicit program is advisable, the next decision is: (2) what specific methods of communication should be the focus of improvement efforts? The third decision, and the primary focus of this study, is: (3) what actions to improve communications are feasible and desirable? The final decision required is: (4) how should the NRC divide its available effort and resources among desirable actions in order to provide the most effective communication through regulatory documents? The NRC is already making decisions similar to these four, implicitly if not explicitly, each time it prepares and issues a safeguards regulatory document. This study was primarily concerned with how to bring about better communication (decision 3 above), not how badly improvements are needed or what should be communicated. As a consequence, the study reflects only partially and indirectly on the first two decisions in the sequence above. However, insights gained during our study lead us to make some comments and recommendations in all these decision areas. The summary is organized in terms of these four decisions. For each decision the factors involved are discussed, possible alternatives described, and recommendations for improvement given.

H. Edelhertz and M. Walsh, *The White-Collar Challenge to Nuclear Safeguards*, Lexington Books, D.C. Heath and Company, Lexington, MA, 1978. The book assesses the white-collar threat to the commercial nuclear energy field. The study examines the concept of white-collar crime in a descriptive fashion to pinpoint potential safeguards vulnerabilities.

M.A. Wincek, K.B. Stewart and G.F. Piepel, *Statistical Methods for Evaluating Sequential Material Balance Data*, NUREG/CF-0683, PNL-2920, February 1979. Present material balance accounting methods focus primarily on the 'material unaccounted for' (MUF) statistic, which utilizes the data from only one material balance period as an indicator of a possible loss of nuclear material. Typically a cumulative MUF (CUMUF) statistic, which utilizes all the available flow data, is also calculated but there is not statutory requirement that it be reported or evaluated. Previous work has shown that CUMUF has greater power than MUF to detect small constant losses. Techniques that emphasize the sequential nature of MUF (i.e., MUF as a sequence of values related over time) are also expected to be more sensitive for detecting losses. The recursive estimation algorithm known as the Kalman filter has been proposed as a possible solution that uses the above idea. The purpose of this study was to evaluate the application of the Kalman filter to the MUF problem, to propose other approaches to the problem, and to reexamine the traditional MUF and CUMUF statistics in more general settings. The report considers the material balance model where the only modeled variability is that due to the measurements of the net throughput (inputs minus outputs) and the inventories. The problem discussed is how to extract more information from all the available data. Section 2 considers material balance models that assume no loss, while Section 3 considers the constant loss and all-at-once loss situations. Emphasis was placed on explaining state variable models and Kalman filtering in relation to the general linear statistical model to which least-squares is applied, yielding a minimum variance unbiased estimator. All errors affecting material balances were assumed to be random.

C.A. Bennett, E.W. Christopherson, R.G. Clark, F. Martin and J. Hodges, *DOE Assessment Guide for Safeguards and Security*, HCP/W 1830-01, May 1978. This guide describes the philosophy and mechanisms through which safeguards and security assessments are conducted. The assessment program described in this guide is concerned with all contractor, field office and Headquarters activities that are designed to assure that safeguards and security objectives are reached by contractors at DOE facilities and operations. Some clarifications of the scope are: (1) SS has assessment responsibility only for DOE facilities, but has responsibility for basic research and development on safeguards and security systems for all applications (e.g., contractor, licensee and international), (2) certain activities of SS serve some DOE functions in areas other than safeguards such as nuclear materials management; other agencies are served in these areas as well, for example NRC and DOD, and (3) relative to classified information the primary responsibility applies to restricted data and it extends to (a) protection of other classified information received and stored by DOE facilities, and (b) assuring that DOE-restricted data are not transferred to outside facilities unless adequate storage and handling facilities exist. Headquarters' Assessment Branch responsibility includes provision of technical support concerning the determination of the adequacy of physical protection measures in other countries as a condition for nuclear export and certain aspects of bilateral safeguards. This guide takes into account the interlocking relationship between many of the elements of an effective safeguards and security program. Personnel clearance programs are a part of protecting classified information as well as nuclear materials. Barriers that prevent or limit access may contribute to preventing theft of government property as well as protecting against sabotage. Procedures for control and surveillance need to be integrated with both information systems and procedures for mass balance accounting. Wherever possible, assessment procedures have been designed to perform integrated inspection, evaluation, and followup for the safeguards and security program.

K.B. Stewart, *The Loss Detection Powers of Four Loss Estimators*, INMM, Nuclear Materials Management, Vol. VII, No. 3, Fall 1978, pp 74-80. The power-to-detect loss curves are developed for four loss estimators under different loss conditions. The loss estimators studies are MUF, CUMUF, $L_{(n)}$ and $M_{(n)}$ where $L_{(n)}$ and $M_{(n)}$, respectively are designed to have maximum powers for the constant loss and the one-time loss situations.

M.A. Wincek and M.F. Mullen, *INSPECT-A Package of Computer Programs for Planning Safeguards Inspections*, International Safeguards Project Office, ISPO-58, PNL-2559 (First Draft), March 1979. The Pacific Northwest Laboratory has developed a package of computer programs for use in planning safeguards inspections of various types of nuclear facilities. The INSPECT software package is a set of five interactive FORTRAN programs which can be used to calculate: the variance components of the MUF (Material Unaccounted For) statistic; the variance components of the D (Difference) statistic; attribute and variables sampling plans; a measure of the effectiveness of the inspection; a measure of the cost of implementing the inspection plan. This report describes the programs and explains how to use them.

M.F. Mullen and M.A. Wincek, *Estimation of Inspection*, International Safeguards Project Office, (First Draft, in review process), April 1978. The Pacific Northwest Laboratory developed a method for estimating the manpower required to inspect various types of nuclear facilities. This report describes the method that was developed. Part I explains the method in general terms. An overview of IAEA inspection activities is presented and the problem of evaluating the effectiveness of an inspection is discussed. Two models are described: an effort model and an effectiveness model. The effort model breaks the IAEA's inspection effort into components; the amount of effort required for each component is estimated and the total effort is determined by summing the effort for each component. The effectiveness model quantifies the effectiveness of inspections in terms of probabilities of detection and quantities of nuclear material to be detected, if diverted over a specific period. In Part II the method is applied to a 200 MT per year low-enriched uranium fuel fabrication facility. A description of the model plant is presented, a safeguards approached is outlined, and sampling plans are calculated. The required inspection effort is estimated and the results are compared to estimates obtained by the IAEA. In Part III other applications of the method are discussed briefly. Examples are presented that demonstrate how the method might be useful in formulating guidelines for inspection planning and in establishing technical criteria for safeguards implementation.

C.L. Timmerman, *Isotopic Safeguards Data Bank (ISTLIB) and Control Program (MISTY)*, International Safeguards Project Office, ISPO-34, PNL-2726, September 1978. The Pacific Northwest Laboratory has developed a computer code and data bank to aid in the safeguards verification of spent fuel content at the head end of a reprocessing facility. A description and user instructions that use isotopic safeguards techniques are presented for MISTY, a computer program for analyzing an isotopic data base (ISTLIB). The input, operating procedures, and output from MISTY are explained in detail. An output listing of an example computer run is provided to illustrate the program's operation. The contents of the data bank are summarized and show the isotopic data sets that are available.

R.J. Sorenson, J.E. Fager and F.P. Brauer, *Recent Experience with a Mobile Safeguards Nondestructive Assay System*, IAEA-SM-231/82, PNL-SA-6826, September 1978. A mobile, real-time, nondestructive assay system for nuclear material safeguards applications has been designed, assembled and evaluated. The system is designed to be used by either an independent agency for verification of prior measurements or by plant personnel for various sample measurements. The system consists of electronic and computer-support equipment mounted in a specially constructed vehicle. This vehicle also carries passive and active neutron and gamma-ray measurement equipment that is operated outside the vehicle. Currently the analysis capabilities include gross sample weight, neutron counting, spontaneous fission neutron counting, gamma-ray spectrometry, and fissile material detection by fissions induced with a neutron source. The minicomputer mounted in the vehicle is used for measuremennt control, data acquisition and data analysis. Recent field experience with the system includes handling and measuring plutonium metal, plutonium oxide and plutonium nitrate. A variety of fuel research materials have also been measured, including ^{233}U, ^{235}U, plutonium, and thorium in various matrices. The system also has been used to measure amounts of material received, stored, or shipped. Field measurements are now underway on a variety of fuel cycle waste materials such as low-enriched ^{235}U, high-enriched ^{235}U, and plutonium in heterogeneous matrices. During field use a number of practical problems were encountered that are as important as the technical considerations in achieving results with the system. The question of calibration standards and our attempts to operate without such standards are also discussed.

R.J. Sorenson, K.B. Stewart and R.A. Schneider, *A Structured Approach to Inspection*, BNWL-SA-5731, INMM, Nuclear Materials Management Vol. V, No. III, Fall 1976. The report describes a structured approach to inspection, the purpose of inspection and its specific objectives, with the aim of providing a basis for an inspector to structure his activities in order that the inspection results may be expressed quantitatively. The various objectives of inspection are discussed as they relate to the origin of threat (adversary), the degree of assurance required, and the inspection body. The basic aim of inspection is discussed as it relates to the role of assessment. The degree of the safeguards assurance is described in increasing levels of inspection activity; and the various roles (responsibilities) in the inspection process are discussed as they relate to the threat they are designed to counter.

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Titles and Abstracts of Recent Safeguards R & D Publications and Reports

Editor's Note: This is the eighth in a series of listings of titles and abstracts of recent safeguards R&D publications and reports from agencies and R&D laboratories. It has been compiled by **R.M. Smith** and colleagues from the Safeguards Group at Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, Canada. Another series of listings from Battelle Pacific Northwest Laboratory is published elsewhere in this issue. Tentatively, the Fall Issue (Volume VIII, No. 3) will have a similar listing from Japan. If your agency or R&D laboratory is interested in being included in this series, please contact the editors, William A. Higinbotham (516-345-2908) at Brookhaven National Laboratory, or Thomas A. Gerdis (913-532-5837) at Kansas State University, Manhattan.

D.G. Boase and J.D. Chen, Non-Destructive Analysis of Irradiated CANDU Fuels, AECL-6316 (1979). A review and assessment of passive gamma-ray, passive neutron, and active neutron interrogation methods for measurement of the fissile content of CANDU fuel bundles. Only active interrogation is expected to provide reasonably accurate data.

D. Tolchenkov (IAEA), M. Honami (IAEA), D.W. Jung (IAEA), R.M. Smith (AECL), P. Vodrazka (AECL) and D.A. Head (AECB). A Safeguards Scheme for 600 MW CANDU Generating Stations, IAEA-SM-231/109 (1978). The scheme is based on item accounting and the use of containment/surveillance measures (to be published in the IAEA 1978 symposium proceedings).

A.J. Stirling and V.H. Allen, The Application of Safeguards Design Principles to the Spent Fuel Bundle Counters for 600 MW CANDU Reactors, IAEA-SM-231/38 (1978). The counters are designed with tamper indicating and self-checking features to record the number of spent fuel bundles released singly or in pairs from the reactor and sent to the spent fuel storage bay.

S.T. Crutzen (ISPRA) and R.G. Dennys (AECL), Use of Ultrasonically Identified Security Seals in the 600 MW CANDU Safeguards System, IAEA-SM-231/124 (1978). CANDU spent fuel bundles are to be stored in tamper resistant containers in the spent fuel bay. The ultrasonic seal designed at ISPRA was redesigned jointly with AECL for sealing these containers.

D.G. Boase, P. Campbell and E.M. Gardy, A Fresh Fuel Interrogator for 600 MW CANDU Reactors, Phase 1, WNRE-432 (1978). Report on joint Canadian-IAEA project on instrument to monitor fresh fuel being fed to a CANDU reactor. A laboratory model was demonstrated which measures the U-235 content of each bundle. A simple counter was also investigated.

D.A. Head, The Safeguards of the WNRE Dry Storage Canister Demonstration Program, Interim Report No. 1, WNRE-286-1 (1976). First report on joint Canadian-IAEA project to develop safeguards for the dry storage canister demonstration project at Whiteshell Nuclear Research Establishment.

D.A. Head, The Safeguards of the WNRE Dry Storage Canister Demonstration Program, Interim Report No. 2, WNRE-286-2 (1976). Similar to above, but for a different canister geometry.

D.L. Amundrud, The Safeguards of the WNRE Dry Storage Canister Demonstration Program, Report No. 3, WNRE-286-3 (1978). Reports tests of use of long fibre optic seals (6 metres) to verify integrity of canister containment.

A. Waligura, Y. Konnov (IAEA), R.M. Smith, D.A. Head (AECL), and J. Hodgkinson (AECB), Safeguarding On-Power Fuelled Reactors-Instrumentation and Techniques, IAEA-CN-36/185 (also AECL 5712). A review article presented at the IAEA International Conference on Nuclear Power and Its Fuel Cycle, Salzburg, 1977 May.

USA-INMM Safeguards Consensus Standardization Program Status

By **Dennis M. Bishop**
General Electric Company
San Jose, California

Editor's Note—This paper was presented by **Dennis M. Bishop,** Chairman, N15 Standards Committee at the First Annual Symposium on Safeguards and Nuclear Materials Management April 25-27, 1979 in Brussels, Belgium. The symposium was sponsored by the European Safeguards Research and Development Association (ESARDA). More information about the meeting appears elsewhere in this issue of Journal.

Abstract

This paper summarizes the status of nongovernment nuclear materials safeguards related consensus standardization programs led by the Institute of Nuclear Materials Management (INMM) in the United States of America (USA). A well-integrated INMM program addressing a broad range of technical disciplines and complex safeguards issues is reviewed. Increased international communications and cooperation is proposed as a vehicle for improving the effectiveness of current domestic safeguards systems, assuring coordination as *international requirements become effective*, and aiding in public acceptance of nuclear energy alternatives.

INMM Description

The Institute of Nuclear Materials Management is a nonprofit technical organization made up of over 600 professional engineers and scientists around the free world in government, industry and academic institutions working with nuclear energy technology. Its prime emphasis includes such timely technical issues as nuclear materials safeguards, worldwide nuclear nonproliferation and international nuclear trade. Specific INMM objectives include the advancement of Nuclear Materials Management topics including:

a. The application of technical and business principles for the safeguarding of nuclear materials and facilities.

b. Promotion of related research and development.

c. Promotion of professional cooperation and communication.

d. Members professional developments, education and training.

e. Development of consensus standards consistent with professional goals and regulatory requirements.

ANSI Description

The American National Standards Institute (ANSI) is an internationally recognized standards organization which works to establish consensus guides and codes promoting understanding and uniform practice within the industrial community. Areas in which ANSI has successfully developed standards include:

1. Definitions, terminology, symbols, and abbreviations.

2. Performance characteristics of materials, parts, equipment and designs.

3. Methods of testing, inspecting, analyzing, and rating.

4. Units of size, weight, volume, and rating.

5. Practices promoting the safety, health, and welfare of employees and the general public.

6. Procedures for operating, processing, handling, storing, and transporting materials, parts, and equipment.

7. Selecting, training, and evaluating operators of equipment and processes.

The ANSI does not, in itself, develop standards in any of these areas. Rather, ANSI serves a central review, communication, and approval function. Specific technical responsibilitilies for the development of standards are assigned to Technical Advisory Boards which make specific assignments to technical societies or related groups with specific knowledge and experience in the area where standardization is required.

Both by charter and emphasis, ANSI's primary goal is ensuring that its standards represent the national consensus in a particular area. This is accomplished through active participation at the writing group level by individuals from related sectors of industry, and by extensive review of proposed standards by the peer groups to ultimately use the standard.

Why Consensus Standardization

Standards developed by the INMM and issued by ANSI are intended to provide information in the form of recommendations for a particular operation, which are based on established practice. If properly developed and used, ANSI standards are beneficial because they

- Establish recognized levels of acceptable quality, performance, reliability, and safety.
- Reduce misunderstandings between producers and users.
- Provide a rational basis for contracts and increase opportunities for trade.
- Provide guidance for design, construction, operation, surveillance, maintenance, and inspection.
- Provide economy through uniformity and interchangeability.
- Form the bases for regulations, and provide guidance for the application and implementation of such regulations.
- Provide ease of communication through standardization of definitions, sizes, and symbols.
- Provide logical alternatives to slow and costly trial-and-error methods.

More specifically, the development of ANSI nuclear standards

- Assists in the standardization of nuclear facilities.
- Ensures a high level of public health and safety and environmental protection in the design, construction, and operation of nuclear facilities.
- Assists industry in complying with government regulations.
- Provides bases for more expeditious accomplishment of reviews for permits and licenses.
- Provides assurance that nuclear facilities will operate reliably.

Each of these benefits has a corollary in the international safeguards area.

Standardization Method

The actual mechanism for developing an ANSI standard is shown schematically in Figure 1. First, the need must be determined, based either on a request from an interested party or based on identification by a standing writing group. Next, the project needs must be defined and assigned to a particular writing group. Current N15 INMM-9 efforts on a standard for the nondestructive assay of low-enriched light water reactor UO_2 fuel rods are an example of one such outside request—in this case, made by the Nuclear Regulatory Commission (NRC) to Standards Committee N15. Next, initial and revised drafts of the needed standard must be developed, reviewed, and revised. Following the

FIGURE 1

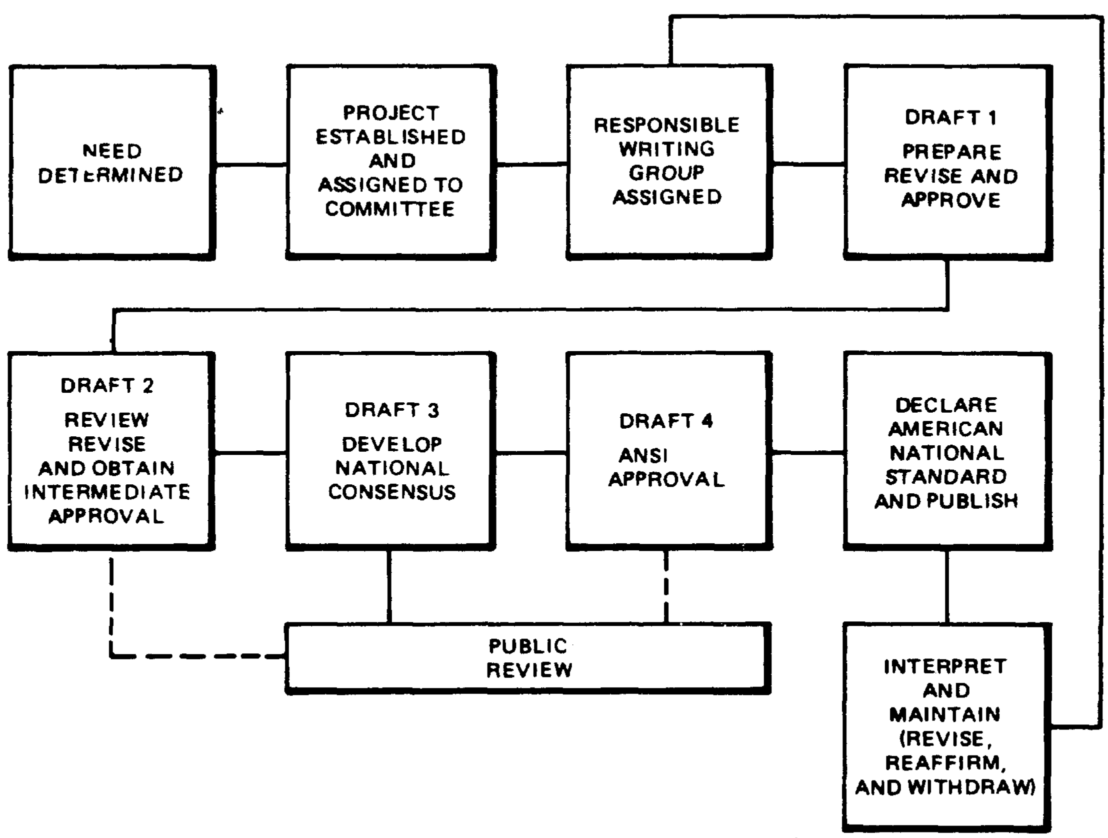

TABLE 1
SUMMARY - INMM N15 STANDARDS COMMITTEE ORGANIZATION AND STATUS

Subcommittee	Title/Writing Groups	Number of Standards: Issued	Under Development	Proposed
INMM-1	Accountability and Control Systems 1.1 Classification 1.2 Control Systems	6	1	2
INMM-3	Statistics	4	1	3
INMM-5	Measurement Controls			1
INMM-6	Inventory Techniques	1	1	
INMM-7	Audit, Records and Reporting Techniques	2	2	
INMM-8	Calibration 8.1 Mass 8.2 Volume 8.3 Nondestructive Assay 8.4 Calorimetry	4		
INMM-9	Nondestructive Assay 9.1 Material Categorization 9.2 Container Standardization 9.3 Physical Standards 9.4 Measurement Controls 9.5 Techniques 9.6 Automation	1	6	
INMM-10	Physical Security	1	1	1
INMM-11	Certification		1	
INMM-12	International Safeguards (Advisory Group)			1
INMM-13	Transportation (Advisory Group)			1
	Subtotal	19	13	9
	Grand Total			**41**

resolution of internal comments, ANSI-Board of Standards Review (BSR) and public comment reviews must be initiated. All negative comments resulting from these reviews must be reconciled in writing or incorporated in the standard prior to submittal of the final standard to ANSI for approval and issuance. Finally, after initial development and approval stages are complete, and throughout its life at a minimum of every 5 years, each ANSI standard is reviewed, reaffirmed, and if necessary, revised or withdrawn. The result is a dynamic set of guidelines or recommended practices for the industry, which are established and maintained by a panel of experts to assure timeliness and technical accuracy.

Life On An ANSI-INMM Standard

Within ANSI, the Nuclear Technical Advisory Board (NTAB) is assigned responsibility for developing standards relating to design, construction, and safe and reliable operation of nuclear facilities. Under this broad charter, NTAB invites various technical societies to coordinate standards development activities on specific "nuclear' topics within their principal area of expertise. Sixteen such standards committees currently exist under NTAB. The INMM is responsible for the N15 Standards Committee dealing with methods for nuclear materials control. Standards Committee N15 operates under the following charter:

> "Standards for the protection, control, and accounting of special materials in all phases of the nuclear fuel cycle, including analytical procedures where necessary and special to this purpose, except the physical protection of special nuclear material within nuclear power plants."

In order to effectively address this scope, the INMM N15 Standards Committee is divided into eleven (11) subcommittee activities. Each subcommittee addresses a high priority area of the current domestic safeguards program, including over 40 ANSI-INMM Standards in various stages of development, as shown in Table 1. These subcommittee activities are further subdivided into over 20 individual writing groups consisting of approximately 5 to 10 contributors. Thus, the INMM N15 Standards Committee represents a significant resource of nearly 200 dedicated engineers and scientists from all segments of the USA nuclear industry. This broad-based participation has been the key to the high rate of acceptance and implementation of INMM N15 standards.

International Cooperation

The current channel for international communication in the area of safeguards standardization is based on the International Standards Organization (ISO). The INMM is keenly interested in stimulating increased communication and cooperation under ISO. Such efforts can become a vehicle for improving the effectiveness of current safeguard systems, and assuring coordination as international requirements become effective. The exact mechanism for initiating this international cooperation is currently somewhat vague and has not been well exercised in the safeguards area. Possible future avenues which should be stimulated include:

1. Formulation of international integration advisory groups.

2. Formation of international standardization writing groups.

3. Cooperation at the draft and peer review stages as standards are developed on the national level.

4. International compatibility reviews of existing issued national standards.

5. Intercomparisons programs involving physical standards (round-robins).

Each of these areas should be evaluated to establish the most time and cost effective mechanism for satisfying today's rapidly changing international safeguards requirements.

As international safeguards needs grow, the need for internationally acceptable and comparable safeguards systems has become a vital issue. With careful planning, safeguards standardization can contribute significantly to overall public acceptance of the nuclear energy alternative. In its tenth year of existence, the INMM N15 Standards Committee has grown with the USA safeguards program to become the single most effective contributor of standards per capita member in the ANSI organization. The INMM is now extremely interested in expanding this effort, on either a formal or informal basis, to assure that forthcoming international requirements are adequately addressed. Interested parties are invited to get involved in the safeguards standardization process on both a domestic and international basis.

Radioactive Sludge

The removal of radioactive sludge from an old waste storage tank at DOE's Savannah River Plant (SRP) has been successfully demonstrated.

The sludge removal program, carried out by E.I. du Pont de Nemours and Company which operates the facility for DOE, is the first phase in a program to replace 23 early-design waste tanks with 18 tanks of an improved design.

DOE is currently storing some 21 million gallons of high-level liquid radioactive waste at SRP which resulted from the production of special nuclear materials for the nation's defense efforts.

In March 1972, some 768,000 gallons of liquid waste were transferred from Tank 16. What remained—about 78,000 gallons of sludge—was essentially all removed by placing three 35-foot-long sludge slurrying pumps into the tank. The material was then moved to a neighboring waste tank by a transfer pump. Any residual sludge in the tank will be removed by chemical cleaning in a program which was scheduled to begin in July.

Tank 16, chosen for this sludge-removal demonstration, experienced a series of stress corrosion cracks in 1960 which resulted in the release of a small quantity of waste materials outside the tank's primary containment. The tank was subsequently taken out of service.

The design of the new waste tanks is a direct result of improvement in engineering and metallurgical technology gained from operating experience in the more than 20 years of managing radioactive waste at the plant.

Some features of the new 1.3 million gallon tanks include a double-walled design tank with walls extending the full height of the tank. The walls of the primary tank are heat stress-relieved to prevent stress corrosion cracking, and leak detection grids are installed beneath the secondary tank.

The total tank replacement program at the plant will take eight years. The final phase of the program will provide for the ultimate disposition of the old tanks.

A View—Gas Queue

(Continued from Page 36)

Oh we didn't want those plants, they were too big and would have harmed our environment. Besides, I have been overseas, and with proper inducements, I can get them to put some plants in our state. It all boils down to the lifestyle we want and our priorities. These are things that are best decided by political rather than technical or economic considerations."

Well, we've been talking for ten minutes and I see the cars just moved up a notch so I better move my car up. But just to recap, your view is small is better, we don't need or want nuclear power, there will be plenty of oil and gas for automobiles and industry, and we can count on solar and wind to provide the energy we don't get from our wood and coal burning stoves in our homes. Many thanks!

As I got into my car, the attendant put a sign on the back of my car which read "Last Car—No more gas." I would have liked to ask one more question, but Mr. Small was engaged in an animated discussion with the attendant. As I moved my car ahead, I heard, "Do you know who I . . . "

Fate may put us next to Mr. Small in a future pilgrimage to the Gas Station and if so, I hope to continue this conversation. After all, we are all concerned about our children's future.

Sandia Laboratory's Plutonium Protection System Project

By H.E. Hansen
Facility Systems Development
Sandia Laboratories
Albuquerque, New Mexico

INTRODUCTION

The Plutonium Protection System (PPS) is an integrated safeguards system developed by Sandia Laboratories for the Department of Energy's Office of Safeguards and Security. This safeguards development project had as its overall goal the design and construction, for evaluation, of a prototype plutonium storage system that would combine the concepts of barriers, intrusion alarms, security, operational control, and material control and accountability. The combined utilization of these concepts would provide positive, uninterrupted control of plutonium items. This control would begin at the time the plutonium item is received at the storage facility, continue through its storage life, and terminate with the authorized release of the item from storage. While at the storage facility, the item would be protected in such a way that any attempt to steal, divert, or sabotage material would be readily detected and alarms would be generated.

Sandia Laboratories first reported a description and status of the PPS project to INMM during the 1977 Annual Meeting.[1] Both Sandia and Rockwell Hanford Operations (RHO) has completed test and evaluation of the prototype system. RHO, as their part of the test and evaluation, performed an operational demonstration at their facility in Hanford, Washington. Sandia performed additional evaluation using the PPS at Hanford as well as the development PPS installed in the Sandia Systems Integration Laboratory (SIL) in Albuquerque. This article provides a review of the operational features of the system, describes the overall test and evaluation activities, and summarizes the more significant results.

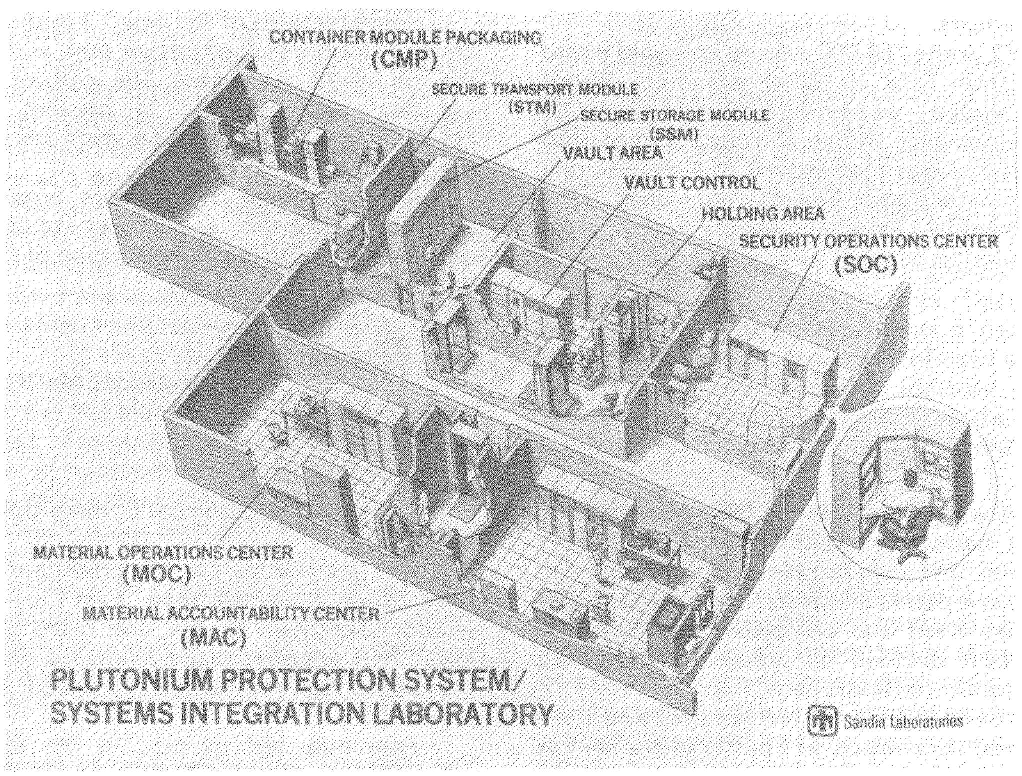

Figure 1. The development Plutonium Protection System as installed in the Systems Integration Laboratory at Sandia Laboratories. This prototype system, together with a similar system installed at Hanford, has undergone operational test and evaluation.

SYSTEM DESCRIPTION

Relative to existing systems, the PPS is designed to provide more positive control and accountability of packaged material and personnel access while reducing radiation exposure of personnel and complying with all safety requirements for handling of plutonium. These functions are implemented through three computer control centers: the Material Accountability Center, The Material Operations Center, and the Security Operations Center (Figure 1). Integrated with these centers are: (1) a hardened vault area, which includes a personnel corridor, the vault control room, and the vault storage area where the secure storage modules are located, and (2) a plutonium packaging area.

When operated as an integrated system, the three PPS centers provide stringent control and rapid accountability for each package of plutonium. To reduce vulnerability to insider threats, access to and movement of plutonium require active concurrence from the three physically separated control centers. These centers are individually programmed to separate the accountability, operations, and security functions. Overlapping of responsibilities, which provides an inherent set of checks and balances, is achieved by requiring independent and fail-safe approvals from the individual centers prior to implementation of PPS activities. To permit the requisite exchange of information, the control centers, the vault area, and the packaging area are interconnected by a data communications network.

Activities within the PPS are covered by four types of transactions: (1) deposit, (2) in-vault movement, (3) withdrawal, and (4) inspection/maintenance. When a particular transaction has been authorized, the appropriate data are entered into the system, thus establishing a transaction file. Data from this file are used by each computer center to set up the transaction and assure that it is conducted only as authorized.

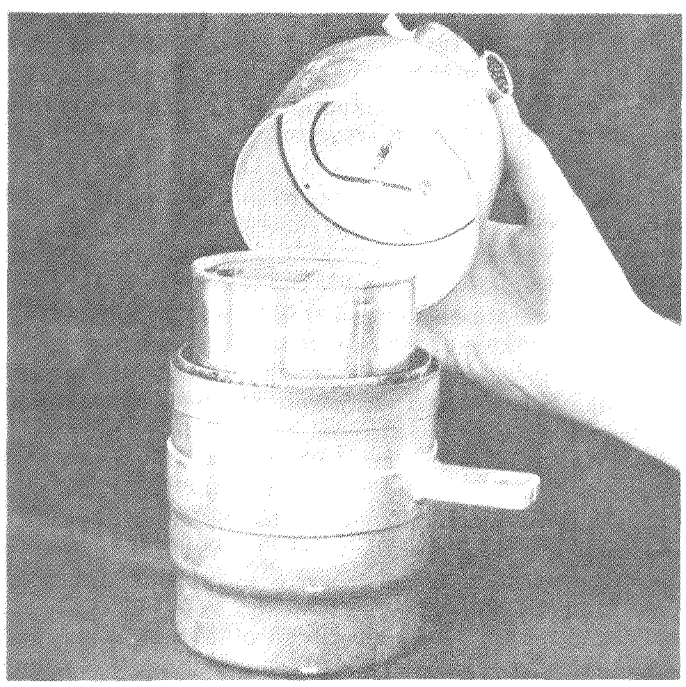

Figure 2. Container modules designed to hold packaged plutonium items provide security, safety, and rapid inventory. Electronics and sensors in each container provide item identification and monitor temperature, seal integrity, and can distortion.

At present RHO seals plutonium items into food-pack cans for storage. In the PPS packaging area, each of these plutonium-containing cans is further sealed into an overpack container (Figure 2). Logic circuits and sensors which provide unit identification, material security, safety, and rapid inventory are integrated into the upper half of this container. The lower half includes a tang into which a deadbolt is inserted to secure the container in its storage location.

After the container module is sealed, each unit is logged into the system data base and the presence of a threshold amount of radioactive material is verified. A secure transport module is then used to move the containers from the packaging area to the vault area for storage (Figure 3). In the vault area, the containers are placed into a secure storage module, which provides physical protection for each packaged item, controls access to the item, and provides the final link for maintaining accountability and inventory of special nuclear materials.

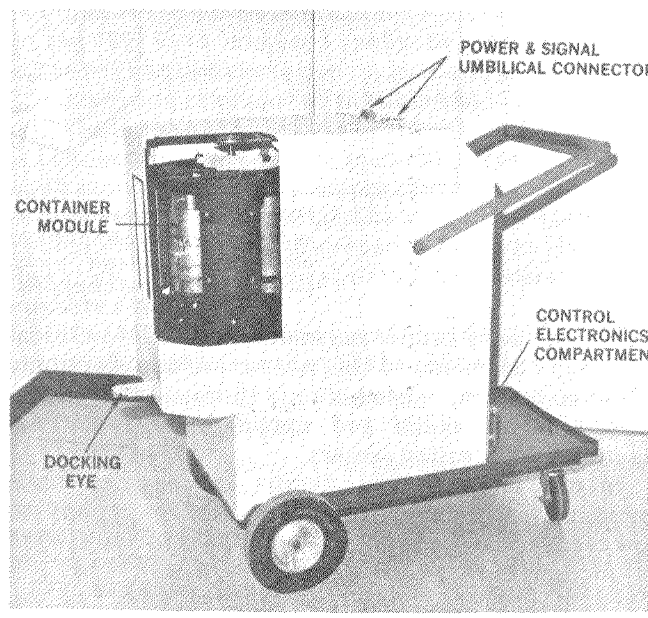

Figure 3. Plutonium protection for material during intraplant movement is provided by a secure transport module.

The secure storage modules (Figure 4) are designed to provide in-depth protection for the plutonium, i.e., a vault within a vault. Each module contains four storage carrousels within a massive structure that has steel doors and walls of precast, steel-reinforced concrete. The storage slots in the carrousels are arranged in a cylindrical configuration designed for single-container-only access and positive locking of each container. Rotation of each carrousel is computer-controlled to allow only the prescribed container to be released at the appropriate time. While the containers are in the storage carrousels (Figure 4), the status of each container is continuously monitored by a microprocessor in communication with the vault computer. Although only 280 containers can be secured and monitored in the two-module configuration tested at Hanford, the control system has the capability to accommodate thousands of containers in storage.

An integrated entry-control, intrusion-detection, and assessment system can detect unauthorized entry at-

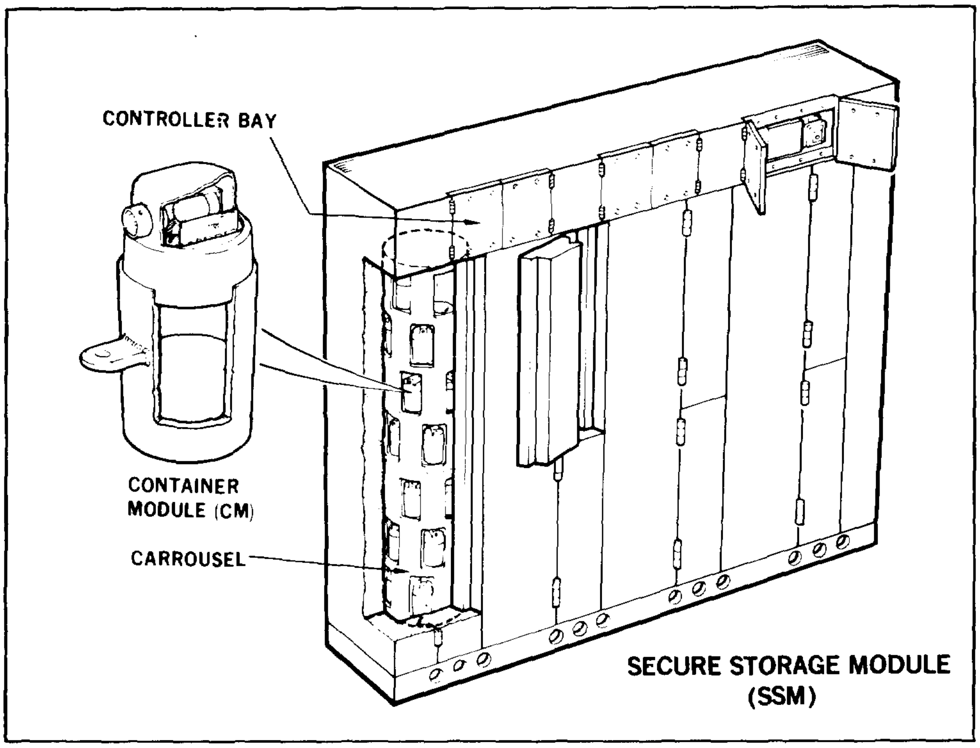

Figure 4. Each secure storage module is designed to provide protection in depth for up to 140 plutonium items. The status of each container module is continuously monitored while secured in one of the four computer-controlled carrousels.

tempts and verify proper personnel access into critical areas. Major elements of this system include electronic credential readers, closed circuit television, an identification booth, metal and special nuclear material detectors, and intrusion sensors.

In designing the PPS, Sandia Laboratories made provisions for supervised contingency operations to allow recovery in the event of personnel errors or system malfunctions during transactional activities.

TEST AND EVALUATION

Test and evaluation of the PPS was carried out in four coordinated activities:

1. A "hot" operational demonstration by Rockwell Hanford Operations
2. Analysis
3. Performance testing
4. Safeguards testing

OPERATIONAL DEMONSTRATION AND EVALUATION

The Rockwell "hot" operational demonstration involved 59 transactions in which 84 containers of plutonium were processed through the system. This demonstration realistically tested the system in an operating environment providing data for evaluation. The following nine specific design criteria were used by RHO to guide the evaluation:

1. Provide protection-in-depth
2. Release of one item at a time
3. Verification that only authorized material is being deposited or removed
4. Separation of material and personnel during entry to and exist from the vault and the container module packaging room
5. Control of personnel access to the vault storage area
6. Production of real-time inventories
7. Reduction of personnel exposure to radiation
8. Uniform packaging of material
9. Provision for protection of special nuclear materials in transit

During the demonstration, RHO concluded that the system met all of the design criteria.

No special criteria were used in selecting personnel to operate the PPS; however, they were required to participate in a training program which included considerable practice in operating the system.

The operational demonstration was conducted according to a three-phase operations plan which included loading, storage, and withdrawal operations. Inspection/maintenance-type transactions were conducted throughout the demonstration.

Phase I involved the packaging and deposit of 84 containers of special nuclear materials. Twenty deposit transactions involving from two to five containers were conducted over a three-week period. One storage carrousel was completely loaded with plutonium oxide and radiation levels were measured. Another carrousel was then completely loaded with plutonium metal and radiation levels were again determined. The remaining container modules, loaded with plutonium oxide, were then loaded into other carrousels.

Phase II exercised the system with twelve in-vault movement transactions involving 48 different containers. This phase, which was completed in one week, confirmed the ability of the PPS to provide inventory and accountability tracking of the items. As in the other phases of the operational demonstration, the ability of the system to monitor foodpack can bulge and temperature was checked.

Phase III utilized 21 withdrawal transactions to remove all stored items from the system. This phase required two weeks.

Some of the significant Rockwell conclusions from the operational demonstration are summarized in the following paragraphs.

Use of the PPS increases the time required to transfer plutonium items to and from storage vaults. The average time during the Rockwell demonstration was about 30 minutes per item for the PPS compared to an average 2.5 minutes per item for their existing operation. The additional safeguards obtained by sealing the items in instrumented containers, utilizing personnel access control, and holding the operating personnel while an inventory is taken were largely responsible for the additional time required.

Dose rates to operating personnel from hand-contact gamma radiation were reduced by 94 percent and full-body dose rates were reduced by 50 percent as compared to dose rates received in existing vaults. Use of the automated inventory techniques employed in the PPS as a replacement for manual periodic inventories would further reduce radiation to personnel.

Rockwell concluded that the rate of alarms registered during the "hot" demonstration was too high to be operationally acceptable. More stringent selection criteria together with more extensive training of operators would significantly improve operating efficiency and reduce alarm rates. Improved hardware reliability and better programming to evaluate and assess alarm conditions would also reduce the frequency of alarms.

Overall, Rockwell concluded from its demonstration and evaluation that the safeguards concepts employed in the PPS are compatible with an operational environment.

ANALYSIS

Analysis, supported with test data where appropriate, was employed to evaluate improved physical protection, separation of functions, and real-time accountability features of the PPS.

Improved Physical Protection

The effectiveness of the PPS in providing improved physical protection against threats involving (a) theft or sabotage by one insider with authorized access, (b) outsiders who do not normally have facility access, and (c) outsiders in collusion with an insider was evaluated using a path analysis technique.

A spectrum of 18 threats was considered in the analysis and the minimum detection probability associated with each family of paths was determined. Nuclear material stored in the secure storage modules was found to be well protected against outsider theft and sabotage; however, some correctable vulnerabilities to insider theft and system defeat were identified.

Separation of Functions

In the analysis of the PPS separation of functions feature, generic centralized and distributed systems were compared. The PPS distributed system for material and personnel control and accountability was then related to the generic system. The conclusion reached is that the PPS distributed system provides safeguards advantages over a centralized system.

Real-Time Accountability

The primary contribution made by the real-time accountability function to the protection of material is to reduce the time between a malevolent act to remove material and its detection to near zero. This is achieved through continuous material monitoring and the performance of inventories concurrent with material handling operations. These features are supplemented by a capability for periodic on-demand inventory.

The real-time accountability features implemented in the PPS were analyzed under two operational conditions: (1) system operation under static conditions, and (2) system operation during transactions.

During static operation of the PPS, a full inventory is conducted hourly by the material accountability and vault computers. The inventory results, as obtained, are compared with the material data base in the Material Accountability Center. A discrepancy between the current inventory and the material data base generates an immediate alarm.

The PPS full inventory capability differs from a conventional manual inventory in the following ways:

1. The length of time between inventories is short (hours vs. months)
2. The time required to conduct the inventory is short (minutes vs. days)
3. Inventories can be performed quickly on demand
4. Book inventories of material and personnel associations are updated concurrently with material transfer operations

The PPS also provides a continuous material monitoring capability under static conditions through electronic communications with each container module. Even if all other security features are defeated, the container continuity checks provide effective and immediate detection of any tampering or removal of material.

During transactions, access to critical areas and material items is precisely controlled through transaction authorizations which are subjected to computerized verification ensuring that all file data are legitimate. The PPS also updates the inventory as transaction activities proceed.

After transactions within the vault, the transaction party is detained in the holding area until the system carries out a "quick" inventory of stored material. This inventory provides a capability to detect unauthorized activities, which is another example of the protection-in-depth philosophy demonstrated in the PPS.

PERFORMANCE TESTING

Sandia conducted performance tests using both the Hanford and Sandia PPS installations. These tests verified system capabilities prior to start of the operational demonstration, provided data on operational characteristics, and identified conditions that might impact on safeguards.[2]

The verification tests identified several problem areas in which the transactions were not completely satisfactory. These problems were noted and corrected where necessary prior to the start of the operational demonstration. A detailed treatment of these problems together with a complete discussion of the test and evaluation can be found in the final report.[2]

Times required to conduct various activities were determined so that the operational impact of the improved safeguards features in the PPS could be evaluated. Figure 5 shows the times required for conduct of (a) packaging room operations, (b) processing of personnel through the vault personnel corridor, (c) vault operations, and (d) inventories.

In terms of overall time requirements, the Sandia performance test data indicate that the conduct of a five-container module, three-person deposit transaction takes approximately 1 hour, 10 minutes or about 14 minutes for each container module. This compares with an average 2.5 minutes per item for similar operations in the existing Hanford facilities and about 30 minutes for each container module by Rockwell during their demonstration. Fourteen minutes per container is believed to be the minimum handling time that can be achieved through increased experience in operating the PPS.

SAFEGUARDS TESTING

The objective of safeguards testing was to identify scenarios, involving one insider* and outsiders, which could lead to the undetected loss of material from the PPS. Information from the performance tests and the ideas of personnel familiar with the PPS were used to develop these scenarios. Using the PPS prototype at Sandia, we then attempted selected scenarios. Sixteen of the scenarios tested resulted in the defeat of at least one PPS safeguards feature. Only one of these scenarios resulted in conditions which could lead to undetected loss of material. In all other cases, adequate protective features remained to effectively prevent the undetected loss of material.

The scenario cited above, which could defeat the system, involved a complex series of planned events with an insider assisted by an outsider. We found that an item being moved within the vault could be passed through the portal door of the secure transport module, left ajar in a previous transaction, to an outsider. Creation of an alarm condition then causes loss of accountability for that item.

This possible vulnerability can be corrected with software changes but serves to illustrate the importance of extensive testing to identify potential problems in any system.

The safeguards testing further confirmed the need for a protected Security Operations Center and an alternate Security Operations Center, features proposed in the original PPS design and now required by DOE regulations. Other improvements identified, which would enhance the safeguards effectiveness of an operational system, include:

- Provision for added physical protection of sensitive devices and components within the system, e.g., computer switches, data communications equipment, Material Accountability Center/Material Operations Center security interface equipment, and software.
- Improved security for the communication system by implementation of message encryption and/or line supervision.
- More positive personnel identification.
- More positive accountability of unsecured items.
- Improved system activities log.
- Improved alarm criteria and clearing procedures.

CONCLUSIONS

While PPS test and evaluation identified areas in need of improvement, this prototype design successfully demonstrated the generic concepts of improved physical protection, real-time accountability, and separation of functions and demonstrated their operational acceptability in an integrated system.

Further conclusions and detailed recommendations can be found in the final report.[2]

ACKNOWLEDGMENT

The author gratefully acknowledges the contributions made to the Plutonium Protection System Project by personnel of DOE/RL and Rockwell Hanford operations. Special mention goes to the many individuals in Sandia Laboratories who contributed to the concepts, design, hardware, software, analysis, and test and evaluation.

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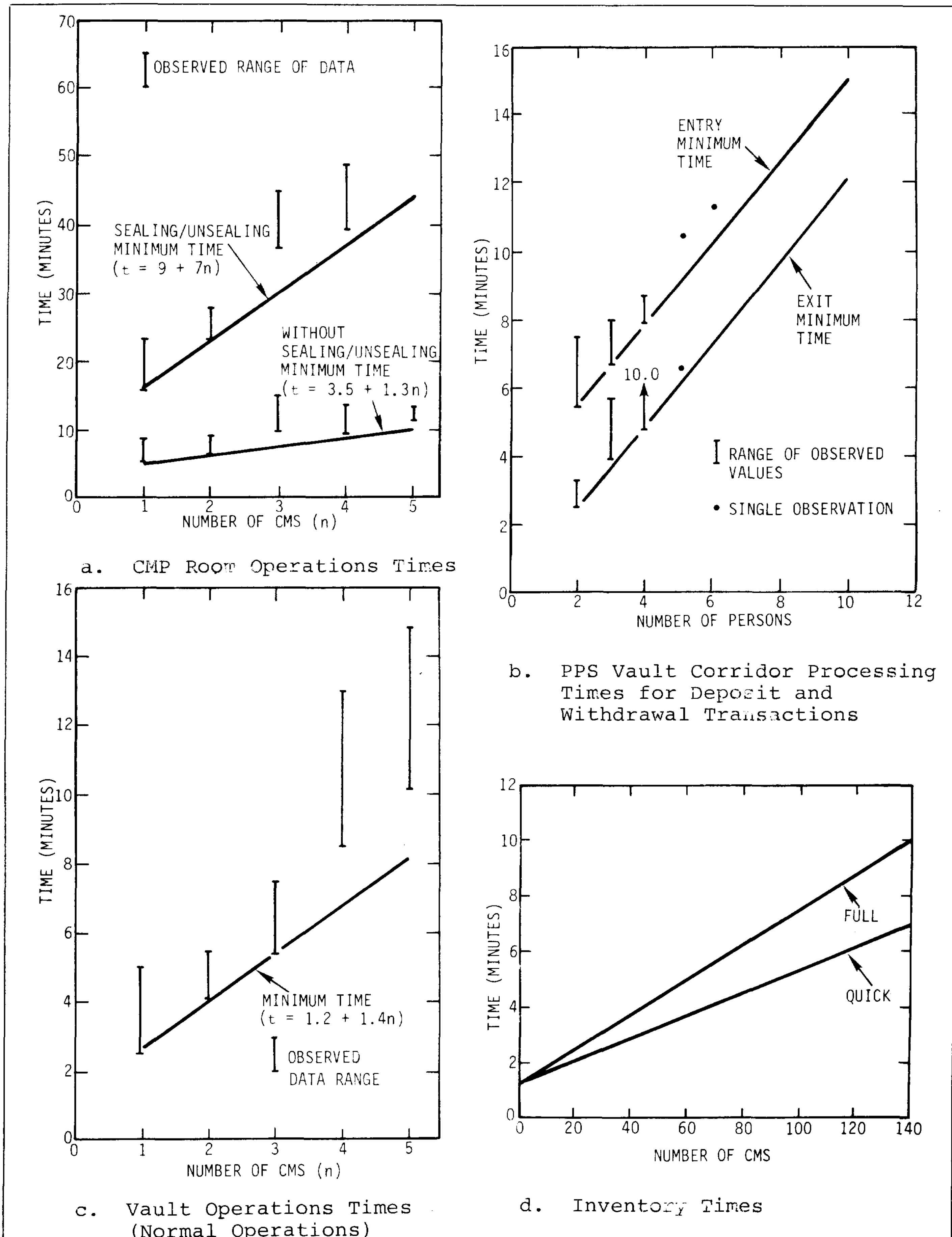

Figure 5. Operational times for PPS activities.

Self-Multiplication Correction Factors For Neutron Coincidence Counting

N. Ensslin, J. Stewart, and J. Sapir
Los Alamos Scientific Laboratory
Los Alamos, New Mexico

ABSTRACT

We have developed self-multiplication correction factors for the nondestructive assay of plutonium oxide or metal in thermal or fast neutron coincidence counters. One correction consists of a series of curves for oxide derived from a Monte Carlo simulation of coincidence counting. Another correction is an approximate, geometry-independent formula which is applicable to any well-characterized material. Both correction factors are compared with measurements.

I. INTRODUCTION

Neutron coincidence counters are often used to provide nondestructive assays of plutonium samples. If the isotopic composition of the sample is known, the total mass of plutonium can be obtained quickly. For plutonium samples larger than a few tens of grams, it is necessary to use fast electronic coincidence circuits and to apply count rate corrections to the data.[1] When this is done, it is often found that the assay accuracy is limited by self-multiplication within the sample.

An example of sample self-multiplication is given in Fig. 1. This figure is a plot of the coincidence response of a series of plutonium oxide standards as a function of ^{240}Pu mass (about 10% of the total). Errors due to counting statistics and count rate corrections are about equal to the size of the circles. The largest sample, about 800 g, shows a 35% increase in the coincidence response. (Fig. 1 also illustrates the application of self-multiplication corrections to the data, as will be described later.) For 1-2 kg metal buttons, the response may be increased by a factor of two or more, as illustrated in later figures. Because this increase depends on sample geometry, density, isotopic composition, etc., it usually cannot be corrected for by a nonlinear calibration curve.

The increased coincidence response may be ascribed to the following two effects:

a) Neutrons produced in the spontaneous fission of the even plutonium isotopes may induce other fissions before leaving the sample. This process will increase the number of neutrons emitted by a factor M, which is the net leakage multiplication (if reflection in the detector is included). These additional neutrons will increase the observed number of neutrons per fission above the actual spontaneous fission multiplicity ν_S, and this will increase the coincidence response.

b) α-particles produced in the decay of plutonium may react with other elements in the sample to produce neutrons. While the single neutrons produced in (α,n) reactions will not increase the coincidence response, they may induce fissions of multiplicity ν_I. These fissions are not correlated in time to spontaneous fission events, and will result in a separate average induced fission multiplicity, which will increase the coincidence response.

Section II of this paper describes a new technique for carrying out Monte Carlo calculations of self-multiplication effects in coincidence counting. Section III contains correction factors for plutonium oxide derived from these Monte Carlo calculations which should be useful for many typical oxide samples. Section IV contains a derivation of simple correction factors for self-multiplication based on the principle that the ratios of single, double, or triple events recorded by the coincidence circuitry can yield an effective fission multiplicity. This effective multiplicity is the key to an approximate, geometry-independent correction factor which can be used wherever Monte Carlo calculations are impractical or unavailable. In Section V this simple correction factor is compared with measurements and with the reference Monte Carlo calculations.

II. MONTE CARLO TECHNIQUES

The Monte Carlo method is ideally suited for computing coincidence response because it can yield information about events involving integral numbers of neutrons. For computing singles response, both Monte Carlo and discrete ordinates methods are useful.

The Monte Carlo technique described here does not require a detailed model of the detector or the coincidence circuitry. The detector is simulated by a hollow cylindrical volume of polyethylene, lined with cadmium to minimize the multiplying effect of reflected, thermalized neutrons on the sample. For n neutrons entering the detector over a time span shorter than the detector resolving time, the coincidence response is proportional to $\varepsilon^2 n(n-1)/2!$, where ε is the detector efficiency. A Monte Carlo calculation of neutron flight times showed that more than 99% of the neutrons from dry PuO_2, and more than 97% of the neutrons from very wet PuO_2, reach the wall of a 30 cm inner diameter detector within 40 nanoseconds. Thus the technique described here should be valid for both thermal (30-100 microseconds coincidence resolving time) and fast (40 nanoseconds resolving time) detectors. Krick[2] has shown this expression to be exact for the shift register circuitry used in this work. As will be shown later, use of this expression allows definition of a coincidence correction factor for multiplication which is independent of detector efficiency. Compared to the approach where detector geometry and circuitry are modeled explicitly, this Monte Carlo method yields a large increase in computational efficiency. It is of course possible, and sometimes appropriate, to compute the detector response directly, such as in detector design studies.

If more than one value of n is possible, the coincidence response will be proportional to

$$\varepsilon^2 \sum_{n=2}^{N} \frac{n(n-1)}{2!} P(n), \qquad (1)$$

where P(n) is the probability of n neutrons entering the detector per initial event, N is the maximum value of n, and $\sum_{n=0}^{N} P(n) = 1$.

If spontaneous fission in plutonium is the initial event, the multiplicity varies between 0 and essentially 6 with a Gaussian distribution.[3] For a very small sample with no multiplication, the coincidence response is given by

$$C_{SF} \propto \varepsilon^2_{SF} N_{SF} \sum_{\nu=2}^{6} \frac{\nu(\nu-1)}{2!} P(\nu) \ . \qquad (2)$$

Here ε_{SF} is the detection efficiency for neutrons with a spontaneous fission emission spectrum, and N_{SF} is the spontaneous fission rate in the sample.

For spontaneous fissions in a multiplying sample, the coincidence response is given by

$$C_{SFM} \propto \varepsilon^2_{SFM} N_{SF} \sum_{n=2}^{N} \frac{n(n-1)}{2!} P_{SF}(n) \ . \qquad (3)$$

Here $P_{SF}(n)$ is the probability per spontaneous fission of the net leakage of n neutrons into the detector, and ε_{SFM} is the detection efficiency for these multiplied neutrons. N is the maximum value of n observed in the process of multiplication of spontaneous fission neutrons. Note that $P_{SF}(n)$ is a net leakage probability, since the detector may reflect some neutrons back into the sample and these may then induce fissions. Both thermal and fast neutron detectors are customarily designed with an inner layer of cadmium or boral to minimize reflection of low energy neutrons. The present technique models these absorbing layers while following neutrons which have entered the detector until their energy falls below .004 electron volts. Thus for the entire range of important neutron energies, we account for reflection implicitly in computing the values of $P_{SF}(n)$.

When an (α,n) reaction is the initial event, there are no net coincidence counts produced by a nonmultiplying sample, because only a single neutron is produced in each reaction. In a multiplying sample, there will be a coincidence response due to induced fissions or (n,xn) reactions (<0.1%) given by

$$C_{\alpha nM} \propto \varepsilon^2_{\alpha nM} N_{\alpha n} \sum_{n=2}^{K} \frac{n(n-1)}{2!} P_{\alpha n}(n) \ . \qquad (4)$$

Here $P_{\alpha n}(n)$, $\varepsilon_{\alpha nM}$, and K are analogous to the quantities defined for Eq. (3), except that an (α,n) reaction is the initial event. $N_{\alpha n}$ is the (α,n) neutron production rate in the sample.

Combining Eqs. (2), (3), and (4) yields a coincidence correction factor CF_C =

$$\frac{N_{SF}\varepsilon^2_{SFM} \sum_{n=2}^{N} n(n-1)P_{SF}(n) + \varepsilon^2_{\alpha nM} N_{\alpha n} \sum_{n=2}^{K} n(n-1)P_{\alpha n}(n)}{N_{SF}\ \varepsilon^2_{SF} \sum_{n=2}^{6} \nu(\nu-1)P(\nu)}, \qquad (5)$$

or $CF_C = 1 + f_{SF} + f_{\alpha n}$, (6)

where $(1 + f_{SF})$ is the coincidence correction for net multiplication of spontaneous fission neutrons, and $f_{\alpha n}$ is the coincidence correction for net multiplication of (α,n) reaction neutrons. CF_C is the ratio of the coincidence response as calculated by Eqs. (3) and (4) to the response expected if the sample multiplication were unity. Therefore the measured coincidence response should be divided by CF_C to correct for multiplication effects.

For later reference, it is convenient to derive here an expression for the net leakage multiplication of the sample/ detector configuration, which is the net number of neutrons entering the detector per initial source neutron. The net leakage multiplication for spontaneous fission neutrons is

$$M_{SF} = \frac{1}{\bar{\nu}_s} \sum_{n=1}^{N} n\, P_{SF}(n) \ . \tag{7}$$

For (α,n) reaction neutrons the net leakage multiplication is

$$M_{\alpha n} = \sum_{n=1}^{K} n\, P_{\alpha n}(n) \ . \tag{8}$$

The average net leakage multiplication $\bar{M}$ is a weighted average of M_{SF} and $M_{\alpha n}$, namely

$$\bar{M} = \frac{N_{SF} \sum_{n=1}^{N} n\, P_{SF}(n) + N_{\alpha n} \sum_{n=1}^{K} n\, P_{\alpha n}(n)}{\bar{\nu}_s N_{SF} + N_{\alpha n}} \cdot \tag{9}$$

A slight modification of this formula gives the multiplication correction for singles counts,

$$CF_s = \frac{N_{SF}\, \epsilon_{SFM} \sum_{n=1}^{N} n\, P_{SF}(n) + N_{\alpha n}\, \epsilon_{\alpha nM} \sum_{n=1}^{K} n\, P_{\alpha n}(n)}{\bar{\nu}_s\, \epsilon_{SF} N_{SF} + \epsilon_{\alpha n} N_{\alpha n}} \tag{10}$$

where $\epsilon_{\alpha n}$ is the detector efficiency for the emission spectrum of (α,n) neutrons. CF_s reduces to $\bar{M}$ only if all four efficiencies are equal. $\bar{M}$ reduces to the M defined in the Introduction if the emission spectra of spontaneous fission neutrons and (α,n) neutrons are the same. Approximations for $\bar{M}$ are discussed in section IV.

The detector efficiencies ϵ_{SF}, ϵ_{SFM}, $\epsilon_{\alpha n}$, and $\epsilon_{\alpha nM}$ will generally differ due to differences in the spectra of the net neutron currents into the detector. Some detectors are designed to be insensitive to such differences. For these detectors, CF_s becomes $\bar{M}$ and Eq. (5) becomes

$$CF_c = \frac{N_{SF} \sum_{n=2}^{N} n(n-1)\, P_{SF}(n) + N_{\alpha n} \sum_{n=2}^{K} n(n-1)\, P_{\alpha n}(n)}{N_{SF} \sum_{n=2}^{6} \nu(\nu-1) P(\nu)} \ . \tag{11}$$

This expression was used for the Monte Carlo coincidence correction factor results reported in subsequent sections.

The Monte Carlo transport code MCNP[4] was adapted to tally CF_c (Eq. 11) and $\bar{M}$ (Eq. 9) directly. The code also yields estimates of the statistical uncertainties (one fractional standard deviation) in the tallies.

In adapting MCNP to the approach described above, a special source subroutine was written. MCNP uses this subroutine to initialize the energy, position and direction of flight of source particles. Initial events are assumed to be uniformly distributed in the sample volume and to emit neutrons isotropically. Source particles emanate from two initial events, spontaneous fission and (α,n) reactions. In the source subroutine, spontaneous fission is picked as the initial event with probability $N_{SF}/(N_{SF}+N_{\alpha n})$, and an (α,n) reaction is picked with probability $N_{\alpha n}/(N_{SF}+N_{\alpha n})$. If spontaneous fission is chosen, the prompt multiplicity ν_s is sampled from a Gaussian distribution about the mean multiplicity $\bar{\nu}_s$. (Table 7.4 of reference 5 lists mean multiplicities, half-lives, and spontaneous fission rates per gram of fissionable isotopes found in the fuel cycle.) The spontaneous fission neutron energy is picked from a Maxwellian distribution with temperature T. (Table 1 of reference 6 lists Maxwellian temperatures for plutonium isotopes.) Regardless of the number of neutrons selected, the same energy spectrum is used. This approach is supported by a study[7] of the limited evidence available, which suggests little correlation between prompt neutron multiplicity and average neutron energy from spontaneous fission.

If an (α,n) reaction is selected as the initial event, a single neutron is started with an energy selected from pure $^{238}PuO_2$, $^{239}PuO_2$, or $^{240}PuO_2$ spectra.[8] The ratio $N_{\alpha n}/N_{SF}$ is calculated for the samples using their known isotopic compositions and the normalized spontaneous fission and $PuO_2(\alpha,n)$ yields given in Tables 7.4 and 7.3, respectively, of reference 5. The value of $N_{\alpha n}$ is adjusted for H_2O content using the factor

$$f = 1.6(1 - 0.01\, P_{H_2O}) - 0.6\, e^{-0.042 P_{H_2O}} \ , \tag{12}$$

where P_{H_2O} is the weight percent of H_2O in an otherwise pure PuO_2 sample. This factor was calculated by J. Brandenberger using his α-particle stopping power cross section code.[9]

In adapting MCNP to this problem, modification of the neutron transport treatment was necessary in order to yield information on integral numbers of neutrons. Capture is treated as a terminal event instead of the more standard approach of reducing the captured neutron's weighting factor. With the standard approach, fractional numbers of particles result. The integral number ν_I of neutrons produced in an induced fission is sampled from a Gaussian distribution about a mean multiplicity $\bar{\nu}_I$ which is dependent on the energy of the incident neutron. For induced fission (as with spontaneous fission), it is assumed that there is no correlation between multiplicity and energy of the secondary neutrons. The source neutron and all its progeny are followed from birth to death (or escape) to complete a history. Over a large number of histories, the probabilities $P_{SF}(n)$ and $P_{\alpha n}(n)$ are scored. Concurrently, these are manipulated as prescribed by Eqs. (9) and (11) to obtain the statistical estimates for $\bar{M}$ and CF_c.

With the version of MCNP described above, a series of reference calculations were made. These are described in the next section.

III. RESULTS OF MONTE CARLO CALCULATIONS

The techniques of the preceding section were used to carry out a series of calculations of the correction factor CF_C, to be used as a reference in assaying PuO_2 samples in neutron coincidence counters. These results are also used to evaluate the simple corrections discussed in sections IV and V. For these calculations, the plutonium mass, ^{240}Pu content, sample density, sample diameter, and H_2O content were varied. All calculations represent variations about an arbitrary nominal sample of 800 g PuO_2, with a density of 1.3 g oxide/cc. This sample contains 706 g of Pu at 10% ^{240}Pu(effective) and 1% by weight water, in a container of 8.35 cm i.d. For each calculation, only one parameter was varied from the nominal values. For the PuO_2 mass, density, and sample diameter variations, the fill height was adjusted to conserve mass. For the H_2O content variation, the sample density was adjusted to conserve volume.

In these calculations, the number of neutron histories traced by the code was 60,000-100,000. For this number of histories a typical range for the maximum fission chain length N or K (cf Eq. 3) was 10-20. The mean energy of spontaneous fission neutrons was 1.96 MeV, and the mean energy of (α,n) neutrons was 2.03 MeV. The net leakage multiplication for (α,n) neutrons was typically within ½% of the net leakage multiplication for spontaneous fission neutrons. However, it is important to note that the (α,n) and s.f. neutron emission spectra have different shapes, and that the former is á function of sample composition. Thus in general it is best to model both spectra in the Monte Carlo calculations.

Figure 2 is the plot of CF_C as a function of PuO_2 density. The curve shows that the correction is appreciable even at relatively low densities. Figures 3a and 3b are plots of CF_C as a function of plutonium mass and diameter, respectively. These plots show that multiplication effects are significant even for small samples. The shapes of the curves in Figs. 2, 3a, and 3b are similar, showing decreasing slopes which appear to be approaching constant values. This behavior seems contrary to the upward curvature expected from self-multiplication effects. However, upward curvature would be observed in plots of response as a function of density, mass, or diameter, as in Fig. 1. For plots of corrections factors (proportional to response/g) downward curvature is apparently appropriate. The data in Figs. 4, 5, and 7 all exhibit downward curvature, and data published by Lees and Hooten[10] also show this. Of course, at very high values of density, mass, or diameter, the slopes of CF_C should increase again as the samples approach criticality.

Figures 3c and 3d are plots of CF_C as a function of weight percent of ^{240}Pu in Pu and weight percent of water in the sample, respectively. Figure 3c shows a large correction factor at low ^{240}Pu enrichment, and a minimum near 16%. This behavior is due almost solely to the variation in the ratio $N_{\alpha n}/N_{SF}$ (cf Eq. 11) as ^{240}Pu enrichment is varied. Figure 3d shows an increase in CF_C as water content increases, due primarily to neutron moderation in water.

The solid points in Figs. 2 and 3 represent the nominal calculation, and the error bars represent one standard deviation. In general, the values of CF_C do not exhibit the scatter which might be expected from the 1σ error bars. This is due to correlations between some of the Monte Carlo calculations. Starting seeds for the random number generators were sometimes identical, so some neutron histories could be the same in two calculations in which only one parameter was varied.

The range of applicability of these curves for CF_C is not completely known. For samples that are similar to the cases calculated here, it would be appropriate to evaluate the individual corrections for mass, isotopic composition, etc., and use the product of these corrections as the overall correction. A somewhat extreme example is the application of these corrections to a 500 g sample of high density, low ^{240}Pu oxide measured in a can lying on its side, as described more fully in section V. Here the large variation in density, isotopic composition, mass, and sample shape from the nominal case yielded an overall correction of 1.76. A special Monte Carlo calculation carried out for this case yielded an actual value of $1.87 \pm .02$, 6% higher. This is believed to be representative of the results of large extrapolations from Figs. 2 and 3.

A less extreme application of the correction factor curves is to the data given in Table I. The plutonium oxide samples listed here are the ones represented by Fig. 1. Column 3 is the measured coincidence response/second-gram of ^{240}Pu, corrected for (small) detector deadtime effects. The number in parentheses is the 1σ error due to counting statistics. Column 4 is the ratio of (α,n)-produced neutrons to spontaneous fission neutrons being emitted by the sample. Column 5 is the net leakage multiplication, as defined in Eq. (9) and as determined from the Monte Carlo calculations. Columns 6, 7, and 8 are the coincidence correction factors defined in Eqs. (6) and (11). For samples 1, 2, 3, 7, and 8 the correction factors have been calculated directly by Monte Carlo, and for samples 4, 5, and 6 they have been obtained by extrapolation from the correction curves in Figs. 2 and 3. The Monte Carlo calculations yield a corrected response/g (column 9) that is almost constant, as illustrated in Fig. 1.

$$S_M - B = m\,G\,\varepsilon\nu_S\,M(1+\alpha), \tag{19}$$

$$D_M = m\,G\,\varepsilon^2\,F(M\nu_S)(M\nu_S-1)(1+\beta)/2. \tag{20}$$

A ratio that is independent of sample mass, detector efficiency, and coincidence circuit time constants is

$$r = \frac{D_M/(S_M-B)}{D_o/(S_o-B)}\;\frac{1+\alpha}{1+\alpha_o} = \frac{M\nu_S - 1}{\nu_S - 1}\,(1+\beta). \tag{21}$$

This ratio is larger than 1 because sample self-multiplication ($M > 1$) increases the doubles more than the singles. $D_o/(S_o-B)$ must be determined in practice by carefully assaying a small sample with negligible self-multiplication. Then r can be determined from $D_M/(S_M-B)$ whenever a larger, multiplying sample is assayed. If the larger sample is of the same composition as the nonmultiplying sample, then $\alpha = \alpha_o$. Otherwise α should be determined from Eqs. (13) and (14) or from the assay of a sample of known multiplication, using the relation

$$1+\alpha = \frac{S_M - B}{m\,G\,\varepsilon\nu_S\,M}, \tag{22}$$

which follows from Eq. (19).

From previous equations, we can express β as a function of M and α:

$$\beta = \alpha\;\frac{M-1}{M}\;\frac{\nu_I}{\nu_I-1}\;\frac{M\nu_I-1}{M\nu_S-1}. \tag{23}$$

This relationship provides an estimate of the number of fissions induced by neutrons produced in (α,n) reactions. Direct experimental confirmation of this relationship is not available. However, for the oxide samples listed in Table I the comparison with the ratio $f_{\alpha n}/(1+f_{SF})$ agrees on the average to within 12%. Eq. (23) also satisfies the constraint that neutrons from (α,n) reactions will not induce any fissions if the sample geometry or composition is nonmultiplying ($M = 1$, $p_f = 0$).

Solving for the leakage multiplication M between Eqs. (21) and (23) yields

$$\left(\nu_S + \frac{\alpha\nu_I^2}{\nu_I-1}\right)M^2 - \left(1 + \frac{\nu_I^2+\nu_I}{\nu_I-1}\,\alpha + (\nu_S-1)\,r\right)M + \frac{\alpha\nu_I}{\nu_I-1} = 0. \tag{24}$$

In this simple derivation the double coincidence response/g of the multiplying sample relative to that of the nonmultiplying sample is given by Eqs. (18) and (20) as

$$\frac{D_M/m}{D_o/m} = \frac{M(M\nu_S-1)(1+\beta)}{\nu_S-1}. \tag{25}$$

By comparison with Eq. (21) we can define a correction factor for self-multiplication as Mr:

$$\text{Corrected Doubles} = \frac{\text{Observed Doubles}}{\text{Mr}}. \tag{26}$$

This relation involves no arbitrary constants. It requires that the background and α be known. The ratio of doubles/singles for a small, nonmultiplying sample must be measured. Then the doubles/singles ratio observed during the assay of the unknown sample is used to calculate r, M, and the correction factor.

The second derivation, also for thermal neutron counters, is somewhat more realistic in the treatment of the average neutron multiplicity. The distribution of spontaneous fission multiplicities is allowed for by assuming a Gaussian distribution of r.m.s. width $\sigma = 1.1$,[3] which alters the coincidence response from $\nu(\nu-1)$ to $\sigma^2 + \nu(\nu-1)$, as derived by integration. Induced multiplication is considered to all orders. For each order, it is assumed that all neutrons are detected simultaneously and that the total width of the distribution increases with each induced fission. It is assumed that each fission distribution has the same width σ, and that the total width is given by folding Gaussian distributions, which is equivalent to summing the squares of the individual widths. Then the coincidence response from ν neutrons in nth order is $\sigma^2(n+1) + \nu(\nu-1)$. Selfmultiplication of spontaneous fission neutrons is then given by

$$\frac{D_M}{N_{SF}} = \frac{1}{2\nu_S}\Big\{\big[(\sigma^2 + \nu_S(\nu_S-1)\big](1-p_f\nu_S)$$

$$+\; p_f\nu_S\Big\{\big[2\sigma^2 + (\nu_S+\nu_I-1)(\nu_S+\nu_I-2)\big]$$

$$\times\big[1 - p_f(\nu_I-1)\big] + p_f(\nu_I-1)\Big\{\;.\;.\;.\qquad.$$

Self-multiplication of (α,n) neutrons is given by a similar equation with ν_S replaced by 1. Both equations can be written as infinite series in powers of p_f and summed. Using Eq. (15) to replace p_f by M yields a complex equation for M, which can be written more simply in numerical form by substituting $\nu_S = 2.17$, $\nu_I = 3$,

$$0 = 2(1+\alpha)M^2 - (1.022+2.192\alpha+.869r)M + (.192\alpha-.109). \tag{27}$$

r is given by the first part of Eq. (21). The correction factor for coincidence counting is again given by Eq. (26). The results of this derivation are similar to those of the first derivation (Eq. 24). The calculated values of M are closer to those expected from Monte Carlo calculations, as shown in the next section. For this reason Eq. (27) is considered more realistic, and is the recommended equation for thermal coincidence counters.

The third derivation is for a fast neutron coincidence counter with three plastic scintillators which can supply triple and double coincidences. This makes it possible to base a multiplication correction on the ratio of triple to double events, which are virtually background-free. The derivation given here is for pure plutonium metal only (α=0). Using T for triples, the count rates are given by

$$T_0 = mG\varepsilon^3 F' \nu_s(\nu_s-1)(\nu_s-2)/6 \ , \tag{28}$$

$$T_M = mG\varepsilon^3 F' M\nu_s(M\nu_s-1)(M\nu_s-2)/6 \ . \tag{29}$$

With $r = (T_M/D_M)/(T_0/D_0)$, the leakage multiplication is given by

$$M = \frac{2}{\nu_s} + \frac{\nu_s-2}{\nu_s} r \ . \tag{30}$$

The correction factor is given by

$$CF = M\left[\frac{1}{\nu_s-1} + \frac{\nu_s-2}{\nu_s-1} r\right] \ . \tag{31}$$

This derivation has also been extended to cases with α not equal to zero, again leading to a quadratic equation for M. Application of this equation to oxide samples has led to some improvement over the results obtained with Eqs. (30) and (31). However, this equation is not as well-tested or as well-founded as the equations given above, and it seems premature to publish it at this time.

Several important assumptions were made in the above three derivations. One is that the detector efficiency be constant over the volume of the sample. This can usually be realized in practice. Another assumption is that all events in the fission chain occur on a time scale small compared to the detector response. This approximation has been described before, and has been called "super fission" by Boehnel.[11] This will not be true for some neutrons which are reflected back into the sample after thermalization in the detector and which then induce fissions in the sample. This effect has also been described by Boehnel. Our initial Monte Carlo calculations included reflection explicitly, and showed that neglect of reflection constitutes an error of typically 15% in CF_c for a thermal detector.[12] For a fast detector the effect has not been calculated, but is presumably smaller because neutron moderation is less. With the exception of reflection effects, the assumption of instantaneous multiplication should be valid, as discussed in section II. The most serious assumption is the use of an effective average multiplicity, rather than folding the actual multiplicity and discrete leakage multiplication distributions, as was done for the Monte Carlo calculations. This assumption introduces large errors, but they tend to cancel when ratios are used, as illustrated in the next section.

V. APPLICATION OF SIMPLE CORRECTION FACTORS

Figure 4 summarizes the results of an experiment which demonstrates that the simple self-multiplication correction factors derived in the preceding section are geometry-independent. In this experiment a drawer of fast critical assembly metal plates was assayed in the dual range coincidence counter described in reference 1. The plates, which contained a total of about 1 kg of plutonium, were initially separated into two batches about 12 cm apart. They were then moved closer together in 1 cm steps until they were side by side. At each step the coincidence response was measured, and was observed to increase. This response was corrected for self-multiplication using Eqs. (24) and (26), with α=0, ν_S = 2.17, and ν_I = 3. The ratio of double to single events for a nonmultiplying sample was estimated from the response of the smallest plate, which contained only 25 grams of Pu. The corrected response is constant within statistical errors.

Figure 5 summarizes the assay results for a variety of plutonium metal cylinders, buttons, and discs, the largest of which weighs about 4 kg. The measurements were again made with the dual range coincidence counter in the high efficiency (25%) mode. Deadtime corrections were as large as 25% at 100 000 counts/s data rates. It is clear that the deadtime-corrrected data can not be fit by a nonlinear calibration curve because of geometry variations from sample to sample. The data were corrected for self-multiplication using Eqs. (26) and (27), with α=0. A 10 g Pu standard was used to obtain the ratio of double to single events for a nonmultiplying sample. The standard exhibited count rates of D_O = 15.6 cnts/s and S_O = 139.7 cnts/s, with B = 12.1 cnts/s. Then $D_O/(S_O - B)$ = 0.122 $\pm$ .001. The heaviest sample, a flat disc containing 4 kg of plutonium, exhibited an uncorrected response/g of 284 coincidence counts/s-g. For this sample, D_M = 59 917 cnts/s and S_M = 101 386 cnts/s. Then D_M/S_M = 0.591, and r = 4.84. Solving Eq. (27) yields M = 2.63. From Eq. (26), the self-multiplication correction factor is Mr = 12.8, so the corrected response/g is 22.3 coincidence counts/s-g. For other samples the correction factors were as large as 18.0. For all of the samples the corrected response/g is constant to within $\pm$ 13% (1σ). This represents an improvement of more than an order of magnitude over the uncorrected data. Excluded from this comparison are the two outliers at 35 and 60 g ^{240}Pu. Radiographs of these two samples show that they consist of irregular pieces of fibrous material. It is likely that oxidation has occurred to the extent that setting $\alpha = 0$ for these samples is not valid. This is an example of the fact that the correction factors derived in section IV are dependent on a knowledge of α.

Application of the simpler correction formula given by Eqs. (24) and (26) corrects the above data to within $\pm 18\%$. The leakage multiplication predicted by Eq. (24) for the 4 kg sample is 3.07, whereas the more realistic Eq. (27) predicts 2.64. The actual observed leakage multiplication is 1.88, using Eq. (22) with $\alpha = 0$. This value is confirmed by a shape-dependent formula based on early Monte Carlo calculations by Atwell, Smith and Walker,[13] which yields 1.83. Both the calculated leakage multiplication and the observed scatter in the data show that the more realistic Eq. (27) is better, although there is still room for substantial improvement.

One way to improve the corrected data without improving the physical model is to calibrate the multiplication-sensitive variable r as a function of the coincidence response/g with a known set of standards. Then the mass of an unknown sample can be determined from r, the observed response, and the calibration parameters relating r to response/g.[14] The disadvantage of this technique is that calibration parameters without physical significance are introduced into the analysis, which so far has been parameter-free. The advantage is that the correction can be optimized. As an example, a quadratic fit was used with the data in Fig. 5 to relate r to response/g. The corrected response/g is then constant to $\pm 4\%$, rather than $\pm 13\%$.

Application of the doubles/singles ratio correction factors to plutonium oxide is shown in columns 10 and 11 of Table I. A 10 gram metal sample was again used as a nonmultiplying reference. The correction is based on Eqs. (26) and (27), with α calculated from Eqs. (14) or (22). Comparison with the Monte Carlo results of columns 5 and 8 shows that the simple correction factor is usually more realistic than the intermediate calculation of the leakage multiplication M. The simple correction factor based on doubles/singles deviates by about 5% from the Monte Carlo calculation. This is due in part to the fact that the simple correction factor shown here was calculated from a different data set than that presented in column 3. For the data from which it was derived, the simple correction factor yields a constant response/g to within 2%. In general, the simple correction is inherently less accurate than the Monte Carlo calculation, but does yield a nearly constant response/g.

Another application of the doubles/singles ratio correction factor to oxide is shown in Fig. 6. In this example a glass jar was filled with oxide in small, carefully measured increments of 10 to 50 g up to a total weight of 500 g. After each increment the jar was assayed in an in-line coincidence counter while lying on its side. The correction for multiplication was calculated from Eqs. (26) and (27). The observed doubles/singles ratio for the smallest samples was extrapolated to zero mass to obtain a nonmultiplying reference ratio. α was calculated from Eq. (22) to be 2.40. Figure 6 shows that the corrected response/g is flat to within $\pm 1\%$. Fluctuations in the uncorrected response/g due to slight variations in powder geometry have also been smoothed out. Because of the high density and low enrichment of this particular oxide, the correction factor is large. At 500 grams M = 1.087 and CF = 1.79. A special Monte Carlo calculation carried out for the 500 g sample yielded actual values of M = 1.088 $\pm$ .002, CF = 1.87 $\pm$ .02, showing that the value given by the doubles/singles ratio is realistic.

The above two examples show that the simple self-multiplication corrections derived in section IV can be applied to plutonium oxide. However, they also show that the ratio of (α,n) neutrons to spontaneous fission neutrons (α) can vary greatly (cf Table I, column 4). The doubles/singles ratio can be applied only if α is precisely known. A certain relative error in the assumed value of α will result in a relative error of the same magnitude in the corrected response. This is because the singles count rate is directly dependent on α. Because variations in isotopic composition or variations in sample impurities can cause large variations in α, the inherent advantage of coincidence counting - that it is independent of uncorrelated background neutrons - may be lost. Thus the simple correction equations based on the doubles/singles ratio will be useful only for well-characterized material.

The final application of the self-multiplication correction equations is illustrated in Fig. 7. Here very pure electrorefined metal was assayed in a three-scintillator fast neutron coincidence counter (random driver).[15] The metal consists of irregular curved pieces stacked randomly in a can. The correction is based on the triples/doubles ratio (Eqs. 30 and 31). The corrected response/gram is constant to within 2.5%, demonstrating that the triples/doubles ratio also provides a geometry-independent measure of multiplication. For active assays of impure metal in this detector, Eqs. (30) and (31) have also been used successfully with the substitution of ν_I for ν_s.

VI. CONCLUSIONS

Monte Carlo calculations and measurements show that neutron coincidence counting of both plutonium metal and oxide is subject to large self-multiplication effects. The set of correction factor curves derived from Monte Carlo calculations should be useful for correcting assays of many oxide samples. For clean metal or very well characterized oxide, the ratios of single, double, or triple events provide a geometry independent correction with no free parameters. These equations should be useful with any existing coincidence circuits capable of handling high count rates.

The general problem of assaying large samples of unknown mass, geometry, composition, and moisture content is not yet solved. Total neutron counts and coincidence counts together do not supply enough information. Work is in progress at LASL on several alternative techniques that may be useful for specific problems. For example, for samples with a well-defined geometry and unknown impurities, it may be feasible to assume a value for the leakage multiplication M. Then it may be possible to use Eqs. (22) and (23) to calculate α, β, and a correction factor for the coincidence response.

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TABLE I

Self-Multiplication Correction Factors for the Plutonium Oxide Samples in Fig. 1. Columns 5-9 are based on Monte Carlo calculations and Columns 10-11 are based on doubles/singles calculations

1	2	3	4	5	6	7	8	9	10	11
Sample Mass (grams)	^{240}Pu effective %	Coincidence Response / g-s	$\alpha = \frac{N_{\alpha n}}{\nu_S N_{SF}}$	Leakage mult. $\bar{M}$	f_{SF}	$f_{\alpha n}$	Correction Factor CF_c	Corrected Response / g-s	From doubles/singles calculation M	CF
20	6.0	2.35(2)	2.11				1.03(1)	2.27(3)		
60	6.4	2.42(2)	1.83	1.005	.024	.026	1.05(1)	2.30(3)		
120	6.4	2.53(2)	1.60	1.010	.049	.041	1.09(1)	2.31(3)		
480	7.8	2.99(3)	0.76				1.28(1)	2.34(4)		
459	9.5	2.98(3)	0.64	1.046	.192	.068	1.26(1)	2.36(4)	1.040	1.20
556	9.9	3.03(3)	0.62	1.049	.215	.075	1.29(1)	2.35(4)	1.047	1.23
615	10.6	3.08(3)	0.60	1.056	.260	.084	1.34(1)	2.30(4)	1.057	1.28
779	10.4	3.26(3)	0.61	1.061	.285	.095	1.38(1)	2.36(4)	1.079	1.39

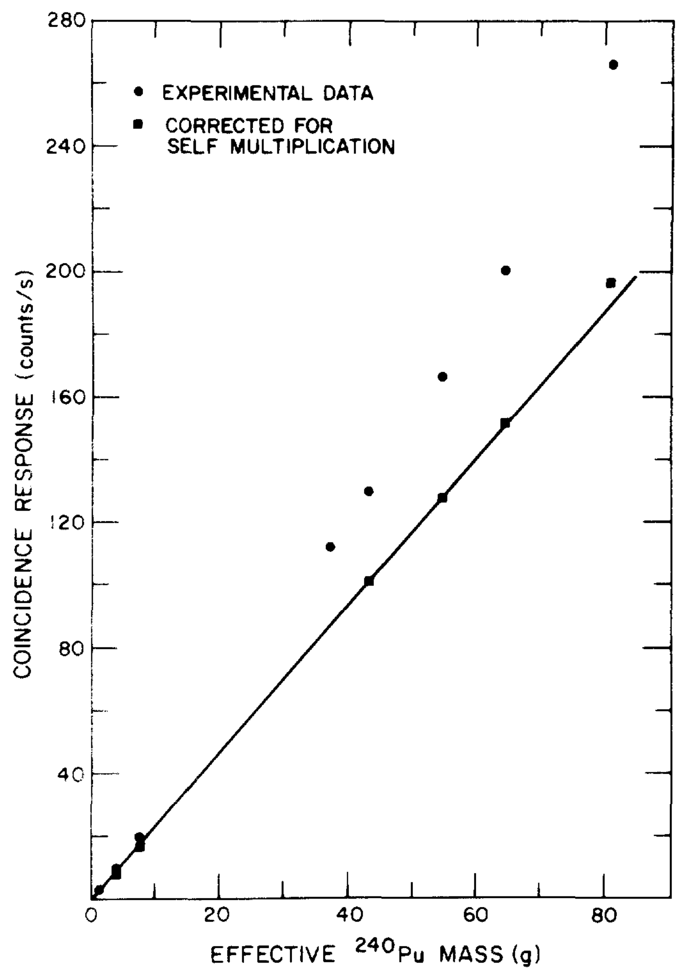

Fig. 1. Coincidence response of PuO_2 standards. The upward curvature in the data is due to self-multiplication in PuO_2. Monte Carlo calculations described in the text were carried out for all but the first and fourth samples in order to correct for self-multiplication, yielding a linear fit to the data.

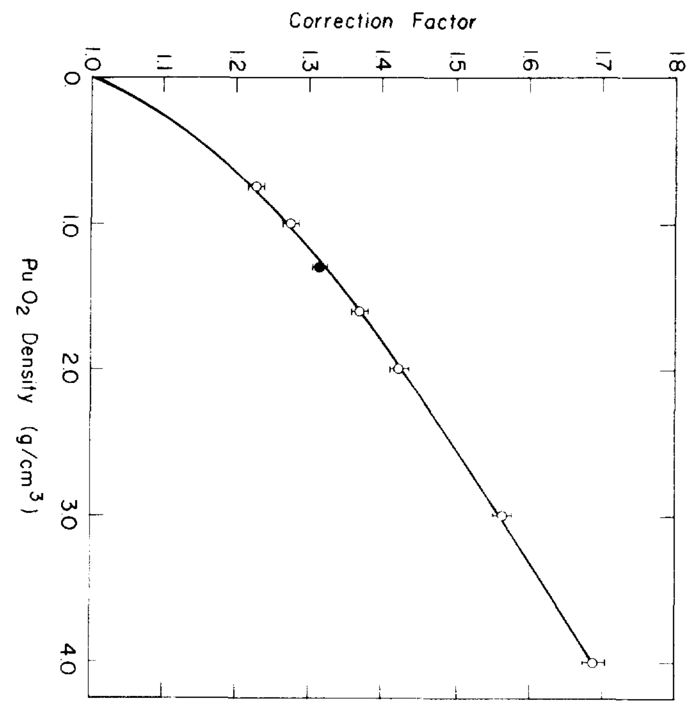

Fig. 2. Monte Carlo calculation of the effect of PuO_2 density on the self-multiplication correction factor required for neutron coincidence counting of PuO_2 samples. The solid data point denotes the nominal calculation.

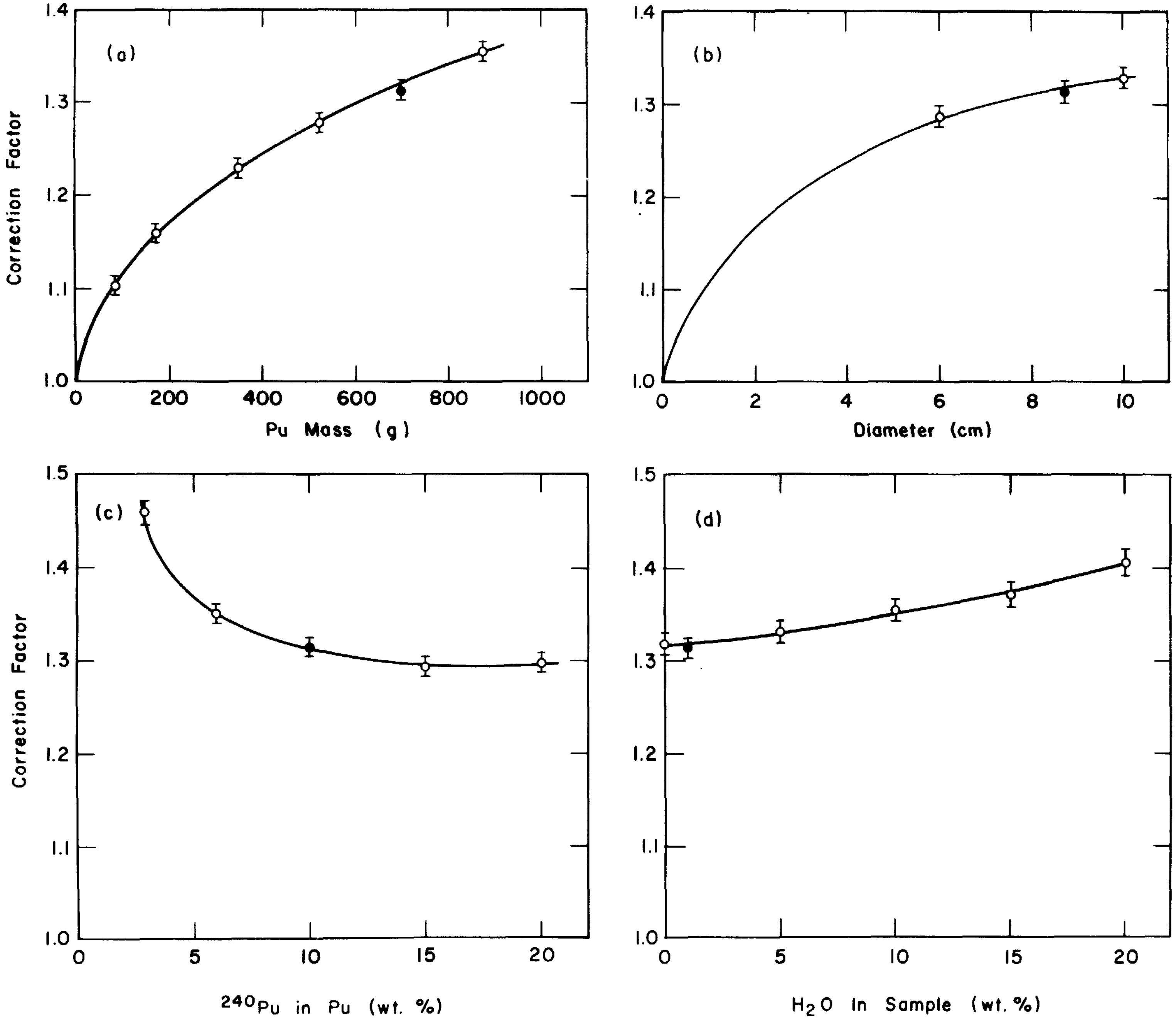

Fig. 3. Monte Carlo calculations of self-multiplication effects of various parameters on coincidence counting of PuO_2. The solid data points denote the nominal calculation.

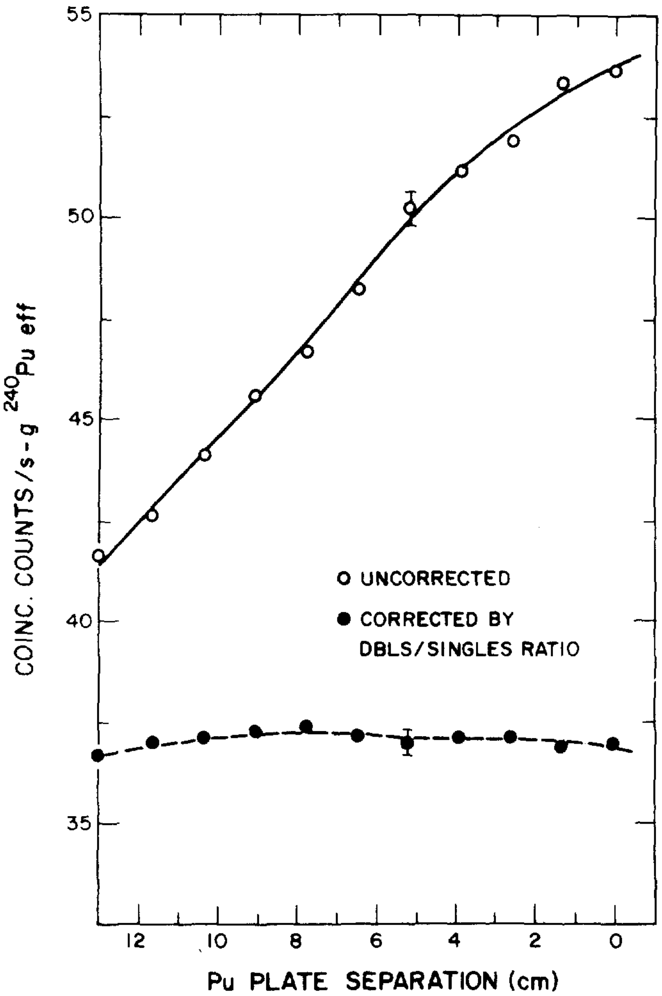

Fig. 4. Application of simple self-multiplication correction based on doubles/singles ratio to 1 kg of plutonium metal plates. As the plates are moved closer together, the uncorrected response per gram increases markedly.

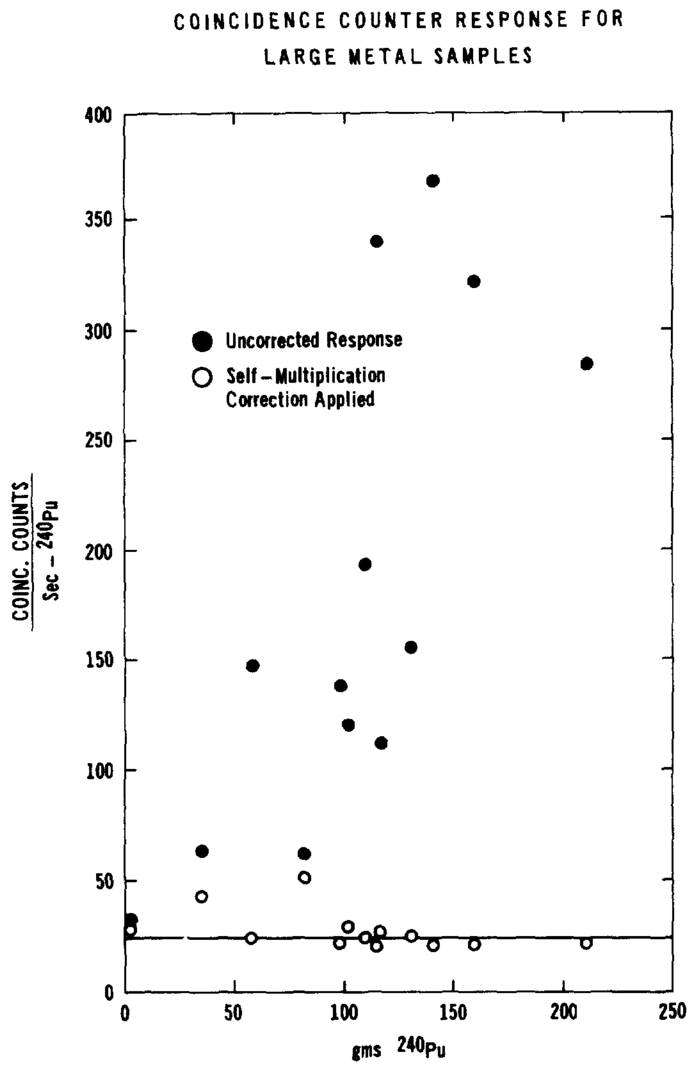

Fig. 5. Coincidence counter response for large metal samples. Statistical errors are about equal to the size of the circles.

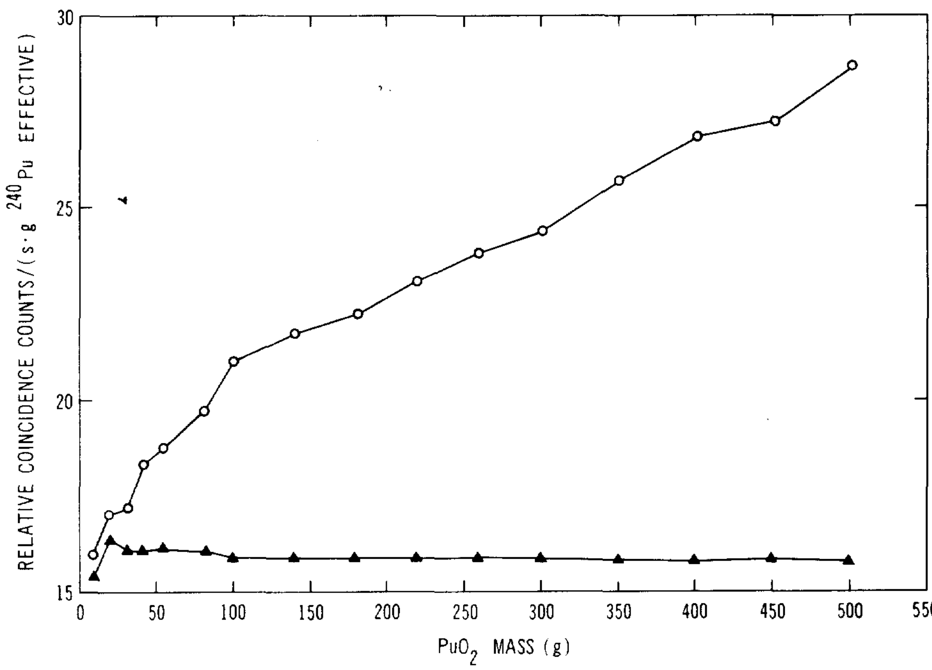

Fig. 6. Coincidence counter response for plutonium oxide which was ladled into a bottle in 10, 20, or 50 gram increments (open circles). Statistical errors are about equal to the size of the circles except for the smallest samples, where they are about 2%. The data was corrected for self-multiplication by using the doubles/singles ratio of section IV (solid triangles).

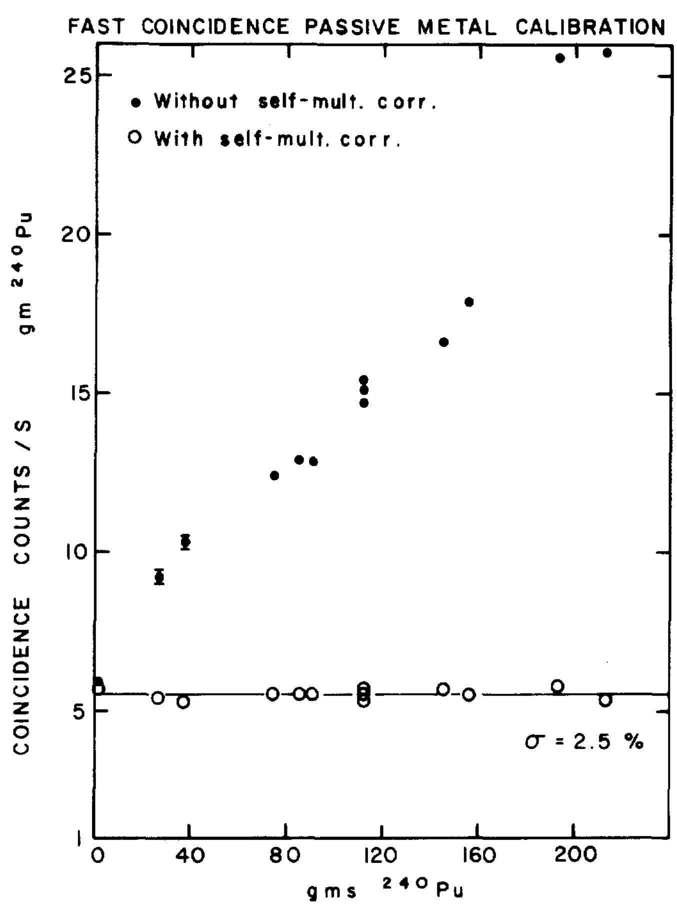

Fig. 7. Application of simple self-multiplication correction based on triples/doubles ratio to plutonium metal in a fast neutron coincidence counter. Statistical errors are about equal to the size of the circles except where shown. The corrected response/gram is constant to within 2.5% (one standard deviation).

Direct Determination of the Total Fissile Content in Irradiated Fuel Elements, Waste Containers and other Samples of the Nuclear Fuel Cycle

H. P. Filss*
Institute of Chemical Technology
Kernforschungsanlage Jülich GmbH
5170 Jülich, Federal Republic of Germany

Abstract

A non-destructive assay system for the direct determination of fissile material is presented which is not influenced by the presence of high level fission product radioactivity (~ 100 Ci). In order to explain the basic principles of this system, some neutron transport properties in hydrogeneous materials are discussed which lead to the specific attenuation of the source neutrons. The source neutron contribution is equivalent to 5 mg U-235 in the small sample system and 60 mg U-235 in the medium-size system. It is estimated to be 1 g U-235 in a conceptual NDA design for radioactive waste barrels. Some typical aspects from the measurement of medium-size waste boxes are reported. The system was originally developed for the assay of irradiated HTR pebble bed fuel materials. It was calibrated with U-235 and is well applicable to uranium from the thorium cycle too. In order to estimate the applicability to the U-Pu cycle, the spontaneous neutron emission of irradiated fuel-elements is compared with the neutron emission of fissile material in the source neutron field. According to the calculations, the system appears to be applicable to the U-Pu cycle and the medium-size asay system may be directly suited for the measurement of irradiated fuel rods. In general by the use of this type of NDA system, the solid radioactive material inputs and outputs of a reprocessing plant appear to be accessible to direct and fast NDA control. This could improve the overall accuracy of material balance and strengthen the confidence in an improved containment/surveillance control system at such facilities.

1. Introduction

Growing use of nuclear power needs reliable and fast methods of nuclear material control to be used both for unirradiated and irradiated fissile material in varying sample configurations. Destructive methods of analysis are at present well established, but require a considerable amount of work for sample taking and preparation.

At the KFA Jülich, Institute of Chemical Technology, a non-destructive assay (NDA) system has been developed and tested for the direct determination of irradiated fissile material from reprocessing experiments in small and medium size sample configurations. It could be verified that this NDA system is not disturbed by a fission product radioactivity of 100 Ci per sample. In case of an increased radioactivity content, it might be necessary to exchange the BF_3 neutron counters for boron lined tubes or for fission chambers. In addition to the undisturbed operation of the system in the presence of fission products the background count rate of this NDA system is very low, equivalent to 5 mg U-235 for small and 60 mg U-235 for medium-size containers. This permits small quantities of fissile material to be detected and measured per sample container.

2. The Small Sample System, Measuring Principles and Performance

NDA systems based on active neutron interrogation generally use external neutron sources to cause nuclear fission in fissile material containing samples and an appropriate measuring system to detect the fission events. In our assay systems which are shown in Fig. 1 and Fig. 3 the radioactive Sb-Be photoneutron source has been chosen [1]. The most important aspect of this neutron source is its low primary energy of E_n = 24 KeV. This kind of source neutrons will cause fission events in fissile material only, yet not in fertile material since the neutron energy is far below any fast fission threshold. In addition, the source neutrons are moderated in the surrounding material to a varying degree and thermal neutrons may play an important role in the interrogating neutron flux unless special precautions (like wrapping with Cd) are taken to eliminate them.

*Presently delegated to the Federal Ministry for Research and Technology, 5300 Bonn, Federal Republic of Germany.

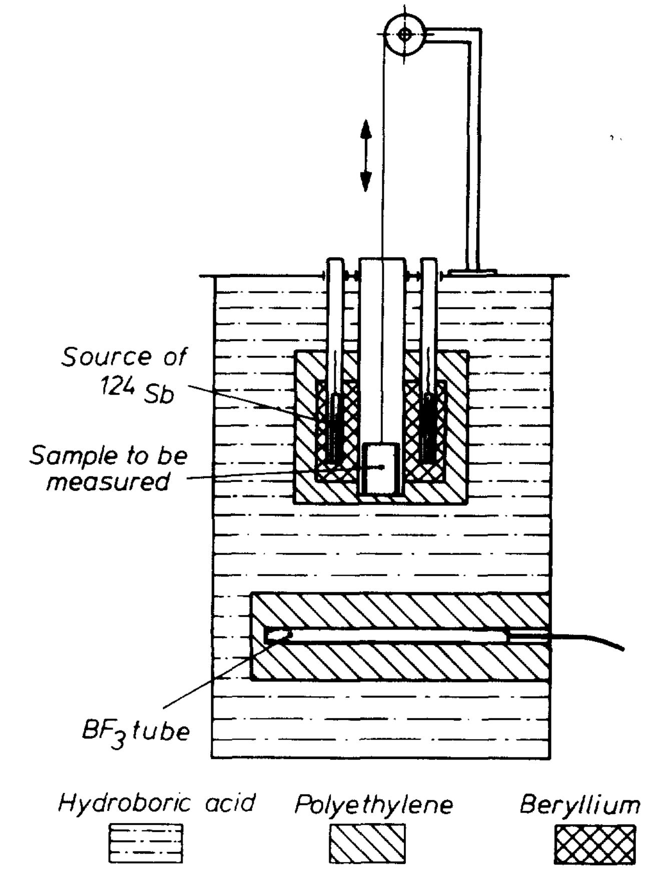

Fig. 1. Assay system (in 1977) for nondestructive measurement of irradiated fissile material in small samples. The total height of the barrel is approx. 1 m.

Samples to be assayed for their fissile content are taken to locations in the assay system where the source neutron flux is high [2]. In the case of Fig. 1 this is the central boring of the cylindrical Sb-Be source. The released fission neutrons from samples containing fissile isotopes have a higher neutron energy of MeV on average. Before registration these neutrons must be separated from the source neutrons in order to obtain low detection limits for fissile material. The separation is achieved in these systems by selective neutron transport (specific attenuation) in poisoned hydrogeneous material. In order to reach the integral neutron counter in the lower part, a polythylene moderated BF_3 tube, the neutrons have to penetrate a layer containing hydrogeneous material. In Fig. 1, this is the interstice between the sample and the polyethylene around the BF_3 tube, which was finally filled with boric acid. On this way, the source neutrons are attenuated to an unimportant background level, whereas the fission neutrons penetrate this layer to a high degree. Originally the length of the interstice and the composition of the liquid were variable. The curves shown in Fig. 2 demonstrate the obtained specific attenuation.

In Fig. 2 the count rate I of both source neutrons and fission neutrons decreases exponen-

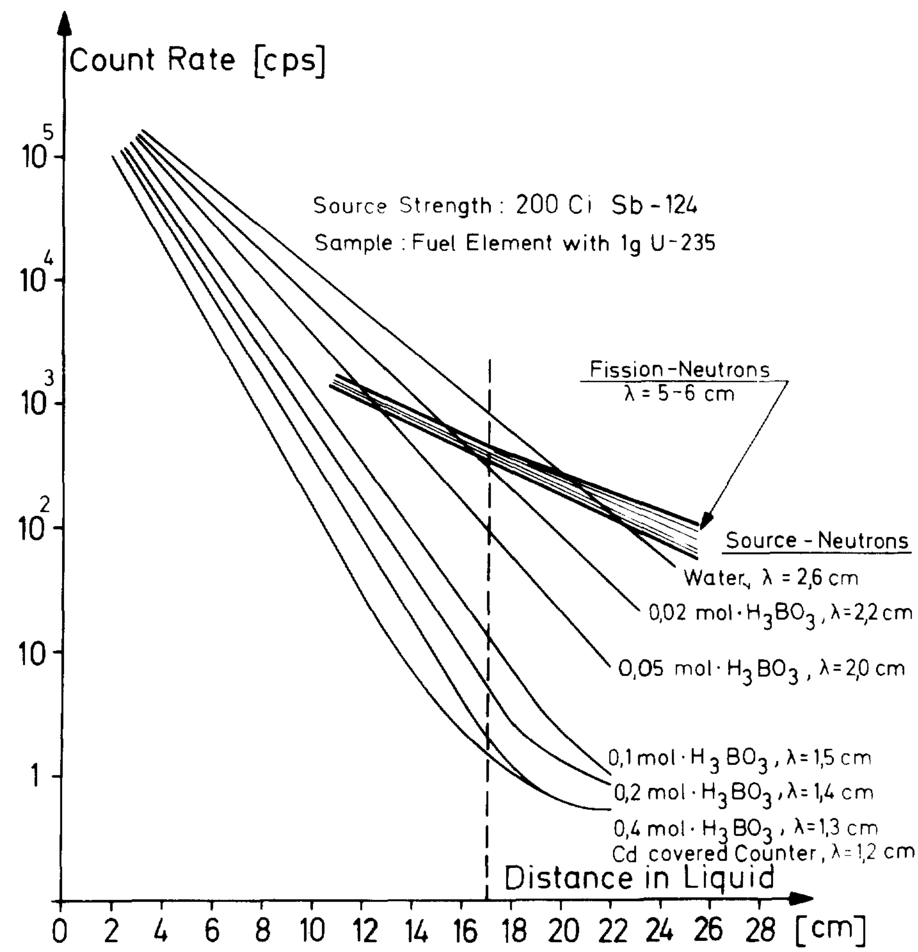

Fig. 2. Relaxation of source neutron count rate and of fission neutron count rate with the thickness of the liquid layer in Fig. 1. The indicated lambda values are obtained from the measured curves.

tially with the layer thickness x. The indicated relaxation constants λ according to

(1) $$I/I_o = e^{-x/\lambda}$$

are dependent on the boric acid content only for the source neutrons. In order to explain this result, it should be mentioned that an exponential relationship can be derived (for the one-dimensional case) from scattering removal of the primary neutrons and from thermal neutron diffusion theory as well. It should be mentioned that the scattering cross section of hydrogen is very much dependent on neutron energy. On the basis of these theories, the following values were calculated for pure water:

λ (2 MeV) = 5.18 cm (λ = mean free scattering path)

λ (24 KeV) = 0.86 cm (λ = mean free scattering path)

$\lambda = L_{thermal}$ = 2.88 cm (L - thermal diffusion length)

It is important to note, that in pure water

$$\lambda \text{ (24 KeV)} < \lambda = L_{thermal} < \lambda \text{ (2 MeV)}$$

The long distance transport process in water is considered [1,3] to be the one with the largest λ and this is

- thermal neutron diffusion for 24 KeV source neutrons
- fast neutron penetration for 2 MeV fission neutrons

Only thermal neutron diffusion can be effectively influenced by small amounts of thermal neutron absorbers such as boric acid or cadmium. Such admixtures have no or only minor influence on fast neutron penetration, which is indeed shown in Fig. 2. This peculiarity makes it possible to selectively attenuate the source neutrons with only a minor influence on the fission neutron intensity. Using concentrated boric acid instead of pure water, the source neutron intensity has been reduced by a factor of 1000 at the fixed distance of 17 cm whereas the fission neutron intensity has only been reduced by a factor of 2. The remaining source neutron count rate was thus finally equivalent to 5 mg U-235. The neutron counter was chosen under the aspect of good stability and dose rate capability. The moderated BF_3 tube used is not disturbed significantly by the radioactive Sb-124 and by fission products both in the radioactivity range of 100 Ci. Other neutron counters (e.g., fission chambers) might be required at a significantly increased radioactivity level.

The following performance was established with the system according to Fig. 1:

Φ (thermal in the sample position according to Fig. 1)

$= 10^6 \text{ sec}^{-1} \text{ cm}^{-2}$ * (source strength 200 Ci Sb-124)

$$r = \frac{\text{fission neutrons registered}}{\text{fission neutrons emitted from the sample}}$$

$= 4.10^{-5}$ **

s = fission events caused in 1 g U-235

$= 4.10^6 \text{ sec}^{-1}$

C_o = background equivalent = 5 mg U-235

Further details including some calibration curves are given in [4,5]. With small samples of 6 cm ø or below, this assay system showed optimum performance. The modifications given in the following were only introducted in order to apply the same measuring principles to samples of a larger size.

3. The Medium-Size Sample System+

The NDA system according to Fig. 3 was designed for the measurement of medium -size samples of unlimited length. This type of sample is supposed to be moved horizontally through the system. The open cross section for the movement is at present 21 cm. 28 cm. The height can be easily changed. The present size limitation would permit fuel rods, medium-size waste containers, and probably even whole fuel elements of some type of water reactors to be measured. For calibration and comparison with small samples, a separate "system I" was additionally provided which is similar to that shown in Fig. 1. In all cases, the measuring site is the center of the open horizontal tunnel between the neutron source in the upper part and the neutron counter including the attenuation layer in the lower part. It must be noted that the dimensions of this measuring site are by no means geometrical safe from a criticality viewpoint. Since the site is surrounded by well reflecting material, it is strictly necessary to avoid measurement of entirely unknown samples. Only if the possible range of fissile material is known to be limited, e.g., in the case of one spent fuel element no problems will arise from criticality. Otherwise critical accidents may occur by the introduction of slightly undercritical masses into the well reflecting geometry.

In order to obtain a good space average from waste boxes it appeared necessary to move and rotate them through the measuring site. A special mechanism "system II" was provided for the

* determined by gold foil activation.

**determined with a mock fission neutron source.

+ The work on this system was supported by the IAEA under research contract No. 1931/RB

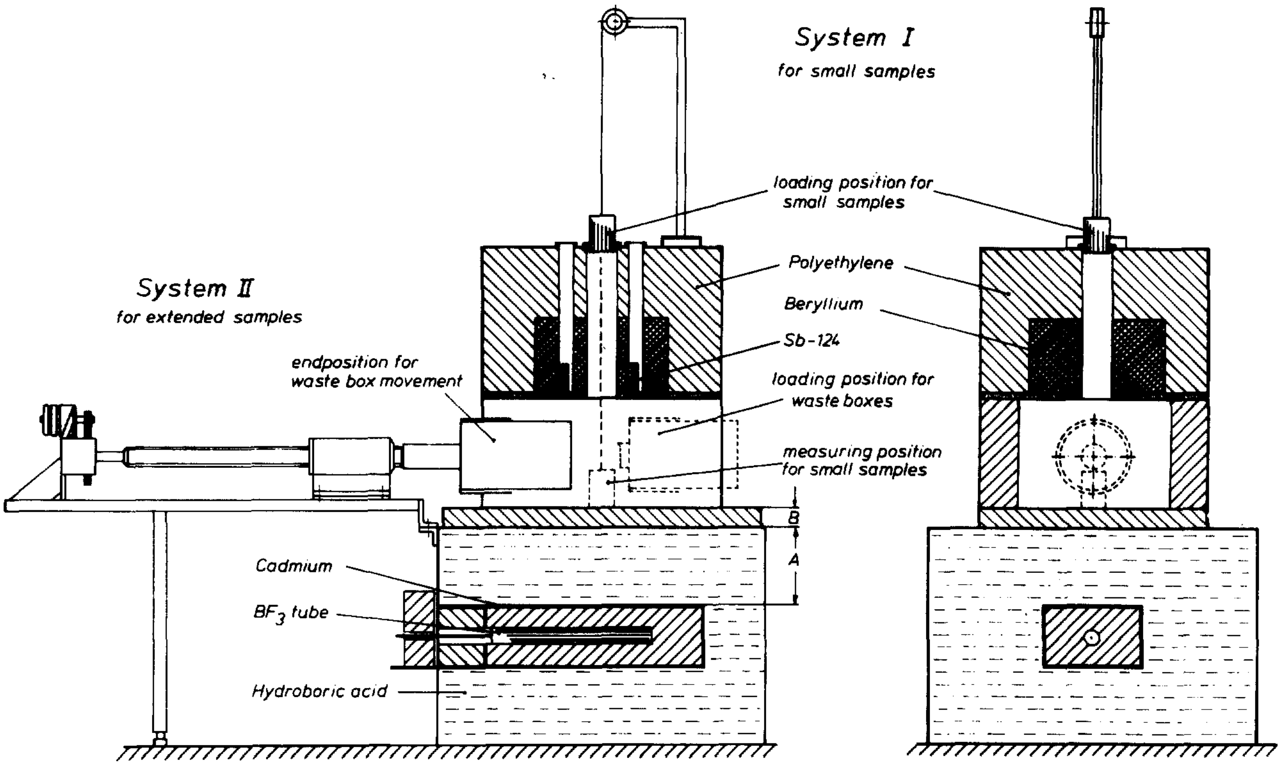

Fig. 3. Assay system for Medium-Size Samples Containing Mixtures of Fissile and Radioactive Material.

continuous helical movement of special waste boxes (16.5 cm diameter, 25 cm height), which are frequently used at the KFA Jülich. The measuring time of a waste box is 1000 sec, which is equal to the complete helical movement over a length of 35 cm. The mechanism for the horizontal movement had at first some problems, but in the end was operating successfully. Practical experience with the system was gained from the measurement of boxes. Compared to the small sample design (Fig. 1), the following differences were established:

- length of samples no longer limited
- size of samples increased from 6 cm Ø to 16.5 cm Ø
- source of neutron spectrum less moderated
- background equivalent to 30 mg U-235 in system I and to 60 mg when the sample was moved in system II
- same dose rate and radioactivity capabilities as in Fig. 1
- the averaging movement of a small sample in system II leads to 50% loss in count rate as compared to system I
- different positions of the fissile material in the waste box lead (typically) to 5% deviations from the mean count rate
- on average only slight changes (1%) by the addition of 500 g matrix material as metal wool or paper

4. Performance and Application

Both NDA systems were developed for process control and accounting purposes in reprocessing experiments with thorium high-temperature reactor (THTR) fuel. The systems were calibrated with U-235 and applied to fuel containing U-233 as well. No principal differences are to be expected with fuel from the U-Pu fuel cycle for which some special apsects are discussed below. The following facts should be mentioned in general:

- the system must be operated in a hot cell because of the radioactivity of the Sb-124 Be neutron source and the application to irradiated fuel
- the neutron souce strength varies with the half life of Sb-124 of 60.6 d. The changing strength is most easily compensated for by a comparison with a standard each day
- all fissile isotopes together contribute to the neutron count rate of the sample. The content of each single fissile nuclide is valued by the product of the fission cross section σ_f and $\bar{\nu}$ the average number of fission neutrons emitted. Since the product is strongly dependent on neutron energy, the above estimate should be verified experimentally with the isotopes under investigation.
- the fission product radioactivity and its associated dose rate leads to an increased base line fluctuation of the neutron pulses. The pulse height discrimination of the neutron counter was not affected by the highest available radioactivity of 100 Ci (fission products)
- isotopes emitting gamma rays with energies above 1.6 MeV produce additional neutrons by (gamma, n) reaction in beryllium. After

the decay of La-140 (Ba-140, $t_{1/2}$ = 12.8 d), the remaining contribution from Pr-144 (Ce-144, $t_{1/2}$ = 284.8 d) gamma rays can be neglected in typical fission product samples.

- compared to the just mentioned gamma-emitters, a contribution from U-232 decay products, especially Tl-208 with E_γ = 2.6 MeV could not be established. The measurement of uranium from the thorium cycle must therefore be considered to be possible in typical fuel elements.
- in most cases the fissile content of the measured samples was in the range of
  10 - 1000 mg U-235 for small samples
  0.1 - 4 g U-235 for medium-size samples
- typical count rates per g U-235 were
  100 counts per second (cps) in the small samples system
  30 cps in the medium-size sample system
- the exact relationship between count rate and fissile content of a special sample must be established with standards. At first approximation, the net count rate is proportional to the fissile content [5].
- samples containing highly enriched uranium in concentrated form, e.g., feed particles suffer from self shielding effects in the sample. In these cases epithermal neutron interrogation (wrapping of the sample with Cd) is recommended.
- samples of almost unknown composition such as graphite dusts or filter material can be compared with a spherical fuel element standard if the expected accuracy is 10% or below.
- The non fissile matrix composition of the sample, mainly contents of water change the neutron count rate.
  - a pebble bed fuel element immersed in water and measured in Fig. 1 gave a 15% higher count rate
  - the addition of 500 g paper to a waste box measured in Fig. 2 gave a 4% higher count rate.

The following considerations refer to the application of the system on materials of the U-Pu fuel cycle which is mainly utilized in water and fast breeder reactors. The fuel element is composed of fuel rods of typically 4 m length. A complete rod can be moved successively through Fig. 3 for non-destructive measurement. The amount of fissile material in the measuring section of Fig. 3 would be in the same range as with the HTR fuel element (1 g U-235). The measurement of unirradiated and irradiated fuel rods seems therefore to be possible without major problems. The measurement of complete water reactor fuel elements may in priciple be possible but the total fissile load self shielding problems and the fission product content would be much larger as experienced up to now and probably require a modified and enlarged layout of the whole assay system. A very important aspect in connection with irradiated U-Pu fuel is the presence of relatively intensive spontaneous neutron emitters, mainly the Cm isotopes. These may constitute a neutron count rate independent of the source strength and of the presence of fissile material. In order to eliminate this contribution the Sb-Be neutron source can be switched off by removing the Sb sources from the beryllium. In this case only the spontaneously emitted neutrons are left and registered. Additional fission neutrons caused by the source neutron flux in fissile material are present after the neutron source is switched on again. The obtained count rate is the sum of spontaneously emitted plus source induced neutrons. The pure fissile material contribution can be obtained by taking the difference. This will only be practicable with good accuracy if the difference is not small compared with the spontaneous neutron count rate. The following estimate is valid for an assay system according to Fig. 1. It was calculated in Section 2 that

$s = 4.10^6$ fission events take place per sec in a

1 g U-235 sample.

In [9], Fig. 6, a typical spontaneous neutron emission is reported to be

$$t = 10^9 \text{ neutrons/s MTU.}$$

With an assumed 1% fission uranium content t' = 10^5 neutrons/s g U-235 are calculated. This spontaneous fission neutron emission is only 1/40 of the expected neutrons from fissile material. This estimate may be further changed by geometry and self-shielding effects. In general, spontaneous neutron emitters are not to be expected to heavily impediment the direct determination of the fissile material in typical samples. On the other hand the simultaneous presence of three fissile nuclides (U-235, Pu-239, Pu-241) with different valuation factors has to be considered in calibration and evaluation of samples from the U-Pu cycle.

5. Concept of a NDA System for Direct Fissile Material Determination in Large-Size Samples

On the basis of the experience from the small and medium size NDA systems, a modified version is under consideration for the measurement of large waste barrels. The design principles of Fig. 3 are retained, but the axis for the movement and rotation of the sample is changed from the horizontal into the vertical direction. This leads to the general layout shown in Fig. 4. As already mentioned for the medium size system, care must be taken to avoid criticality accidents. Samples with completely unknown fissile content are not to be transferred to the well reflecting geometry of the measuring site. Considering the increase in the background count rate from 5 mg in Fig. 1 to 60 mg in Fig. 3, the background

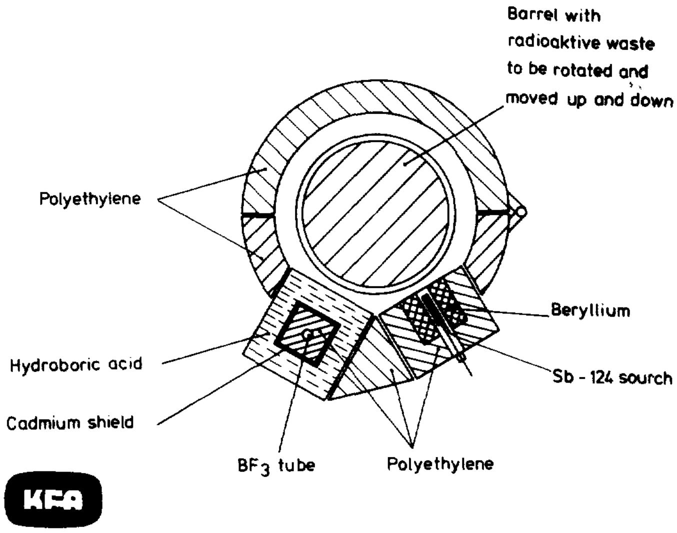

Fig. 4. Conceptual Design of a NDA System for Barrels Containing Radioactive Waste.

count rate of this new design is estimated to be in the range of 1 g U-235. Concrete walls of the hot cell instead of lead walls would be advantageous in reducing the background count rate according to the better neutron absorption of concrete.

Such an NDA system may be applied to the control of waste barrels which leave the reprocessing plant. The NDA measurement of their fissile content seems to be possible as long as the composition of the waste is similar to irradiated fuel. This limitation includes leached hulls and normal air-filters but probably not concentrates of alpha- and spontaneous-neutron-emitters separated in the course of reprocessing. The direct measurement of the fissile content in waste barrels at reprocessing facilities could establish that no fissile material can leave such a facility without direct control even not in waste barrels.

6. Conclusion

The aforementioned NDA systems based on selective neutron transport properties were originally developed for the assay of irradiated HTR pebble bed fuel materials but are generally applicable for fissile material determination without being disturbed by large quantities of fission products. This kind of radioactive material occurs in various stages of the reprocessing process. These NDA systems were calibrated with U-235 and applied to U-232 and U-233 containing material from the thorium-cycle. According to estimates of the spontaneously emitted neutron contribution the application on irradiated fuel of the U-Pu cycle seems also to be possible. The medium-size system which has been used for the measurement of irradiated HTR fuel elements seems also to be well suited for the measurement of complete LWR fuel rods. Some typical aspects of radioactive waste box measurement are shown and the conceptual design of a radioactive waste barrel system is given. The major solid material inputs and outputs of a reprocessing plant appear thus to be accessible to direct and fast control by this NDA system. The inclusion of radioactive waste containers into direct control would generally improve the overall accuracy of material balance in open fuel cycle stages such as reprocessing plants. It could further strengthen the confidence in an improved containment/surveillance control system at such facilities.

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Sandia Credential System

An electronic credential system which automatically detects passage of personnel into or out of controlled areas has been developed and successfully field-tested by Sandia Laboratories.

The system includes an entry/exit portal, which looks much like an airport metal detector, and a small credential badge worn by workers. The badge, activated by an alternating magnetic field produced by the portal, transmits a signal to a receiver/decoder.

"When the system is placed at strategic locations such as corridors or doorways leading to critical areas," says Project Engineer **Thurlow Caffey,** "it automatically monitors, identifies, and electronically logs the individual credential badge entering or leaving the area."

Major advantages of the system are: (1) the badges are long-lived because they use no batteries; (2) they are detected automatically when they pass through the portal; and (3) individual badges can be detected and accurately counted even though separated by as few as 60 cm when they pass through the portal.

The latter advantage means that "two Olympic sprinters with badges attached to their uniforms could run through a portal just an arm's length apart and still be detected by the system," Caffey says.

The basic system consists of a multi-turn coil or loop, called an exciter. This loop, which is wound within the portal, transmits a continuous tone of 112kHz. The badge, which weighs one ounce and is 2½" × 2¼" × 3/8", also contains a loop which transmits bursts of a 56kHz tone when passed through the portal. These signal bursts, transmitted back to the portal, contain a unique code which identifies the badge. When the signal is decoded, it is sent to a computer which notes that the badge has either just entered or exited the portal.

Depending upon the needs of the user, the badge's transmitted signal can trigger other actions. It can deny access to an area, for instance, if the allowed number of badges are already logged into an area.

Also, in the event of an emergency—for example, an accident within a facility—the system could be designed to identify any workers in the danger zone.

The credential system can be incorporated into a physical security system if measures are taken to insure that the credential badge is being worn by the person to whom it was issued.

"This can be done by several methods," Caffey says. "For instance, the badge's code could instruct a computer to call up a photograph of the person authorized to use a particular badge. TV monitors at a guard station would then show pictures of both the photograph and the badge wearer."

For its field test, the self-energized credential system was incorporated into a Plutonium Protection System developed by Sandia for demonstration at a government nuclear facility. The credential system was used at three places within the special storage complex: near the entrance of a corridor leading to the plutonium storage vault, within a personnel identification booth along the same corridor, and at the entrance of the protection system's operation center.

Sandia is a prime contractor to the Department of Energy.

Performance Analysis of Nuclear Materials Accounting Systems

D. D. Cobb and J. P. Shipley
Safeguards Systems Group Q-4
Los Alamos Scientific Laboratory
Los Alamos, NM 87545

ABSTRACT

Techniques for analyzing the level of performance of nuclear materials accounting systems in terms of the four performance measures, total amount of loss, loss-detection time, loss-detection probability, and false-alarm probability, are presented. These techniques are especially useful for analyzing the expected performance of near-real-time (dynamic) accounting systems. A conservative estimate of system performance is provided by the CUSUM (cumulative summation of materials balances) test. Graphical displays, called performance surfaces, are developed as convenient tools for representing systems performance, and examples from a recent safeguards study of a nuclear fuels reprocessing plant are given.

I. INTRODUCTION

One essential part of designing nuclear materials accounting systems is analyzing their expected performance in detecting losses of nuclear material. Systems performance analysis, in turn, implies the definition of suitable performance measures that can be easily related to externally established criteria. Thus, there are two aspects of the analysis problem: first, defining performance measures, and second, relating those measures to established, quantitative performance criteria.

Performance measures for any nuclear materials accounting system embody the concepts of loss-detection sensitivity and loss-detection time. Because of the statistical nature of materials accounting, loss-detection sensitivity can be described in terms of the probability of detecting some amount of loss while accepting some probability of a false alarm. Loss-detection time is the time required by the accounting system to reach some specified level of loss-detection sensitivity. Note that the loss scenario is not specified; that is, whether the loss occurs in an abrupt or in a protracted fashion, the total amount of loss is the measure of performance. Note also that loss-detection time only refers to the internal response time of the accounting system.

Detection is based on finding a statistically significant materials imbalance, although the definition of what constitutes a "detection" is debatable. It can be argued that detection must include the external functions of investigation and response following a positive indication of loss by the accounting system. However, performance measures for those activities external to the technical functions of any materials accounting system are difficult to quantify and may be subjective at best. The scope of this paper is restricted to analyzing the expected technical performance of accounting systems designs using quantifiable performance measures.

Performance criteria for materials accounting systems result from external judgements concerning acceptable, or at least desirable, performance goals. Criteria are established for, or are directly relatable to, four performance measures: total amount of loss, loss-detection time, loss-detection probability, and false-alarm probability (or level of significance). For any materials accounting system, the four performance measures are not independent but are related by a continuous function that depends on the uncertainties of

This work was performed as part of the US Department of Energy-Office of Safeguards and Security Research and Development Program.

the materials measurements and also, in general, on the particular loss scenario and the particular statistical test applied to the accounting data.

To be more specific, let Z be a set of N measured materials balances $\{z(i), i = 1,2...,N\}$ over N accountability periods, and let Θ be a set of parameters describing the uncertainties in the measurements. Symbolically, let $s(L,N)$ be a particular loss scenario in which a total amount L is lost during the N accountability periods, and let $\tau(Z,\Theta,\alpha)$ be a particular statistical test of the accounting data applied at the αth level of significance. Then, the loss-detection probability, or the power of test τ, for loss scenario s is

$$P[s(L,N);\ \tau(Z,\Theta,\alpha)]. \qquad (1)$$

The loss-detection probability P for a fixed level of significance α is a function of the two performance measures, total loss L, and loss-detection time (here denoted by N). One convenient way of displaying the loss-detection power of any test is a three-dimensional graph of the surface P versus L and N. We refer to such three-dimensional graphs as loss-detection power surfaces by analogy with the usual two-dimensional loss-detection power curves of P versus L.

Several tests are currently available that are tailored for specific loss scenarios,[1-3] such as uniform or random protracted losses and single or block abrupt losses, and each of these tests has its maximum detection power for a particular loss. In reality, the starting time, the duration, and the loss scenario are never known *a priori*. Ideally, every available test would be applied to every combination of the available materials accounting data to ensure that tests and loss scenarios are matched to maximize detection power. Furthermore, every test would be applied at several levels of significance to ensure that the most significant test result is obtained. The type of test producing the most significant result (greater than some specified minimum level of significance) and the subset of data for which the most significant result is obtained provide information for a subsequent investigation of possible losses. Other test results provide corroborating evidence.

II. PERFORMANCE SURFACES

Intuitively, the composite performance of a materials accounting system should be independent of particular choices of loss scenario or test. This composite performance is described by the function

$$P^*[L,N,\alpha], \qquad (2)$$

where P^* is the accounting system's probability of loss detection for specified values of L, N, and α. By analogy with the loss-detection power surfaces suggested previously for a single test, one convenient way of displaying the accounting system's expected level of performance is a three-dimensional graph of the surface P^* versus L and N for some specified value of α. We call such graphical displays performance surfaces.

The performance criteria determine the required level of significance α and the coordinates of a point in the three-dimensional space (N,L,P^*). If the volume bounded by the performance surface of some materials accounting system contains that point, the system is presumably judged to be acceptable. If not, the shape of the performance surface may suggest the necessary design modifications to make it more acceptable. Performance surfaces therefore could be useful systems design tools. They also portray (correctly) the expected performance of a proposed materials accounting system as a continuous function of the three performance measures, loss, time, and detection probability, rather than as a single point.

A. CUSUM Performance Surfaces

Clearly, the function in Eq. 2 can be mathematically complex, and Monte Carlo simulation techniques are generally required for its (approximate) evaluation because the results of applying several tests to many loss scenarios must be considered. Fortunately, there is one test, the CUSUM test, that does not depend on how the material was lost, but responds only to the total amount of loss L during the time interval N. Furthermore, the CUSUM test provides relatively good detection power for any loss scenario even though it is seldom the best test for any scenario.[2]

The CUSUM test statistic, $c(N)$, over some subset of N accountability periods is just the unweighted linear summation of the N materials balances,

$$c(N) = \sum_{i=1}^{N} z(i). \qquad (3)$$

The fixed-length (nonsequential) CUSUM test uses the standard hypothesis testing procedure[4] in which the test statistic, $c(N)$, divided by its standard deviation, $\sigma_c(N)$, is compared to a threshold that depends on the specified value of α. (It is assumed that $c(N)$ is normally distributed.)

If the CUSUM test is always one of the tests applied to the accounting data, the performance of an accounting system will always be at least as good as the loss-detection power of the CUSUM test, regardless of loss scenario. Thus, the CUSUM test provides a conservative, scenario-independent measure of systems performance. In other words, it is conservative to approximate the function in Eq. (2) by the function in Eq. (1) using the CUSUM test [$\tau = C$ in Eq. (1)] for two reasons: first, the detection power of the CUSUM test is independent of the loss scenario; that is, for any s,

$$P[s(L,N);C(Z,\Theta,\alpha)] = P[L,N;C(Z,\Theta,\alpha)];$$

and second, if the CUSUM test is always one of the tests used, the system's performance will always be at least as good as the CUSUM detection power; that is,

$$P^*[L,N,\alpha] \geq P[L,N;C(Z,\Theta,\alpha)]. \qquad (4)$$

We refer to a loss-detection power surface generated using the CUSUM test as a CUSUM performance surface because it is an approximation to the accounting system's true performance surface.

B. Two Examples

Figures 1 and 2 show two examples of CUSUM performance surfaces from a recent safeguards study of a 1500 tonne per year reprocessing plant.[5] The graphs were produced using a commercially available computer graphics program that generates the surfaces by plotting isometric contours of total loss L and materials balance number N. Note that contours of fixed loss-detection probability are also plotted on the CUSUM performance surfaces in probability increments of 0.1.

The two figures illustrate the use of CUSUM performance surfaces in analyzing the expected performance of proposed accounting systems designs. The expected performance of two possible near-real-time accounting systems, the "worst-case" and the "best-case" systems, are shown in Figs. 1 and 2, respectively, for simulated materials accounting in the solvent-extraction cycles of the plutonium purification process (PPP) of the reprocessing plant. Details of this process and its safeguards analysis are given in Ref. 5.

The design-basis throughput of the PPP is 50 kg of plutonium per day, and it is proposed that materials balances be closed every 8 hours during normal, continuous operation. This would be accomplished by using a variety of process-control and accountability instrumentation included in the reference process design and by adding on-line instrumentation to the input and output streams of the PPP.

The figures show the expected levels of performance during any four weeks of PPP operation (21 materials balances per week or a total of 84 balances at the end of four weeks). A one-sided, fixed-length CUSUM test at about the $\alpha = 0.001$ level of significance (3σ test threshold) was applied to the simulated materials balance data. Sequences of measured materials balances of lengths 1 to 84 were tested, and each sequence had simulated losses of 0 to 20 kg of plutonium. (The indicated choices of maximum N and L as well as α are at the discretion of the user.)

Note that values of detection probability for any sequence of length one, that is, for any single materials balance, are plotted at balance number N = 0, so that the contour at N = 0 is the loss-detection power curve of the single materials balance test. Note also that the plotted materials balance numbers are relative. In other words, the performance surfaces give the expected performance of any future sequence of materials balances starting with the current balance, or the past performance of any sequence of balances ending with the current balance.

In the worst case (Fig. 1), the in-process plutonium inventory in the solvent-extraction equipment is measured to 10% (1σ relative standard deviation), and the accountability instrumentation on the input-output streams is not recalibrated during the four-week period. At the end of any one-week period (21 balances), the total detectable loss at the 90% detection probability level, for example, is about 14 kg of plutonium. At the end of any four week period (84 balances), the corresponding loss-detection sensitivity is 50 kg. (Only losses up to 20 kg are shown in the figures.)

In the best case (Fig. 2), the in-process plutonium inventory is measured to 5%, and the input-output instrumentation is recalibrated every two days (every six balances) causing the scalloped patterns to appear in the fixed-loss contours. A dramatic improvement in loss-detection sensitivity is obvious by comparing the two CUSUM performance surfaces. For example, in the best case, the loss-detection sensitivity at the 90% level of detection probability is 7.6 kg at the end of any one-week period and 14 kg at the end of any four-week period, whereas, in the worst case, the corresponding loss-detection sensitivities are 14 and 50 kg, respectively.

C. Other Tests

The loss-detection powers of tests that are tailored for specific loss scenarios can be substantially better than the CUSUM test, if those tests are applied to accounting data obtained under the specific loss scenario.[1-3] For example, the application of a uniform diversion test (constant amount of loss per accounting period), based on the Kalman filter, to simulated data in the PPP yields the results in Table I. Sensitivities of the

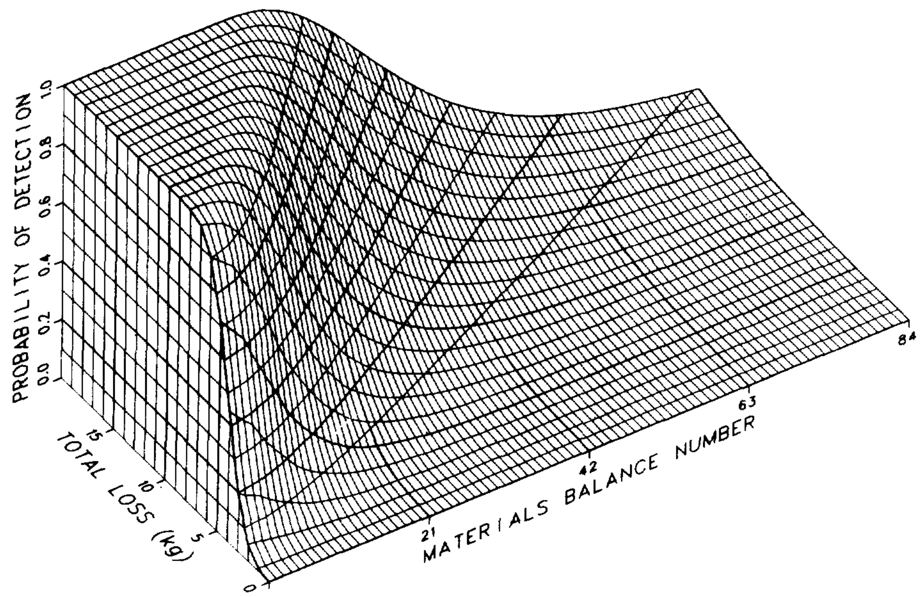

Fig. 1.
CUSUM performance surface for the worst case.

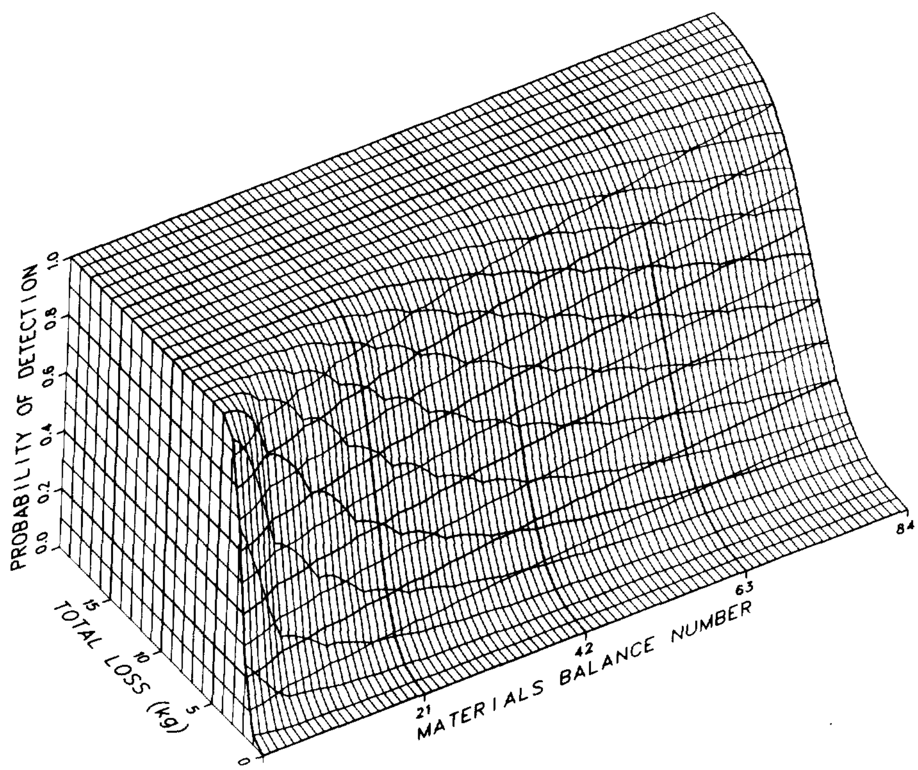

Fig. 2.
CUSUM performance surface for the best case.

TABLE I

DETECTION SENSITIVITIES OF CUSUM AND UNIFORM DIVERSION TESTS FOR UNIFORM LOSSES IN THE PPP

Detection Time[a]	Uniform Diversion Test Sensitivity[b] (kg)	CUSUM Test Sensitivity[b] (kg)
1 hour	3.68	3.68
8 h	2.76	3.73
16 h	2.41	3.90
1 day	2.52	4.13
2 d	3.80	5.19
3 d	4.12	5.52
5 d	5.44	6.46
7 d	6.51	7.44

[a]As defined in Sec. I.

[b]90% detection probability at $\alpha = 0.001$.

CUSUM and uniform diversion tests in detecting uniform losses are compared. The simulated measurements are the same as in the example of Fig. 2, but materials balances are drawn every hour rather than every 8h. Ratios of the loss-detection sensitivities of the two tests range from one to approximately 1.6.

In our experience, the sensitivities of other tests are seldom more than a factor of two better than the CUSUM test, indicating that the CUSUM performance surfaces are useful, conservative approximations to the expected performance of proposed accounting systems designs. The performance of the more powerful tests should always be compared with that of the CUSUM test for specific loss scenarios to ensure that the CUSUM approximation to the system's composite level of performance is not unduly pessimistic.

III. GENERALIZED CUSUM PERFORMANCE SURFACES

It is possible to develop CUSUM performance surfaces that describe the performance of a broad class of possible accounting systems designs. These so-called generalized CUSUM performance surfaces are derived in terms of the statistical parameters describing the accounting measurements. They may be used to obtain a quick estimate of the expected level of performance of a proposed accounting system or of the effect of a proposed system modification.

A. Models of the CUSUM and Its Variance

The measured materials balance for the ith accountability period is

$$z(i) = y(i-1) - y(i) + x(i),$$

where $y(i-1)$ and $y(i)$ are the beginning and ending measured inventories and $x(i)$ is the measured net transfer of nuclear material (inputs minus outputs) during the ith period. Note that the measured values, y and x, may be composites of many partial inventory and input-output measurements, respectively.

The statistical model assumed for the inventory measurements is[2,4,6]

$$y(i) = \mu_y(i) + \delta(i),$$

where $\mu_y(i)$ is the true inventory at the end of the ith period (beginning of the (i+1)th period), and $\delta(i)$ is a zero-mean random error with variance σ_y^2. The model assumed for the net transfer measurements is

$$x(i) = \mu_x(i) + \lambda(i) + \varepsilon(i) + \eta,$$

where $\mu_x(i)$ is the true net transfer of material not including any unmeasured losses or gains during the ith period; $\varepsilon(i)$ is a zero-mean random error with variance σ_ε^2, η is another zero-mean random variable with variance σ_η^2, and $\lambda(i)$ is the algebraic sum of any unmeasured losses or gains during the ith accounting period. The η measurement-error component may, for example, be caused by imprecise calibrations of the input-output measurements, and, thereby, the net transfer measurements having the same calibration are correlated. That is, the η error is a so-called "systematic error" in the terminology of Ref. 4.

Thus, under those measurement models, the accounting measurements are accurate but imprecise, the measurement errors are stationary, and the correlations between net transfer measurements caused by imprecise calibrations are included explicitly. The measurement models can be extended to incorporate additional correlations and biases, for example, those that would be unaffected by calibration. The extension to non-stationary errors is also possible, but the following analysis does not apply to that case.

The CUSUM over any sequence of N accountability periods is

$$c(N) = y(o) - y(N) + \sum_{i=1}^{N} x(i),$$

where the measured inventories $\{y(i), i = 1,2,..., N-1\}$ cancel in pairs in the summation of Eq. (3). The expected value of c(N) is

$$E[c(N)] = L, \quad L = \sum_{i=1}^{N} \lambda(i) ,$$

where L is the total amount of loss during the N accountability periods. The variance of c(N) is

$$\sigma_c^2(N) = 2\sigma_y^2 + N\,\sigma_\varepsilon^2 + N^2\,\sigma_\eta^2, \qquad (5)$$

where the N^2 term appears because of the correlated net transfer measurements.

B. Loss and Time Scaling

The CUSUM and its variance can be expressed in terms of scaled loss and time variables. Let $\sigma_I^2 = 2\sigma_y^2$ be the variance of the measured net inventory change, y(o) - y(N). If the CUSUM is scaled by σ_I, $\hat{c}(N) = c(N)/\sigma_I$, then the expected value of the scaled CUSUM is

$$E[\hat{c}(N)] = \hat{L} = \frac{L}{\sigma_I}. \qquad (6)$$

The variance of $\hat{c}(N)$ is

$$\hat{\sigma}_c^2(N) = 1 + N\,\hat{\sigma}_\varepsilon^2 + N^2\,\hat{\sigma}_\eta^2,$$

where

$$\hat{\sigma}_\varepsilon = \frac{\sigma_\varepsilon}{\sigma_I}, \quad \hat{\sigma}_\eta = \frac{\sigma_\eta}{\sigma_I}, \text{ and } \hat{\sigma}_c(N) = \frac{\sigma_c(N)}{\sigma_I}.$$

The materials balance number N (that is, the time) is scaled by a particular value of N, call it N_o, such that

$$\hat{N} = \frac{N}{N_o}, \quad N_o = \frac{\sigma_\varepsilon^2}{\sigma_\eta^2}. \qquad (7)$$

Note that N_o is the value of N for which the two net-transfer measurement-error variance terms in Eq. (5) are equal; that is,

$$N_o\,\sigma_\varepsilon^2 = N_o^2\,\sigma_\eta^2.$$

With the choices of loss and time scaling given in Eqs. (6) and (7), the scaled CUSUM variance reduces to the simple one-parameter form:

$$\hat{\sigma}_c^2(\hat{N},\gamma) = 1 + \gamma\,\hat{N}(1+\hat{N}), \quad \gamma = N_o\,\frac{\sigma_\varepsilon^2}{\sigma_I^2}. \qquad (8)$$

The loss and time scaling parameters, σ_I and N_o, and the system parameter γ are determined by the measurement-error standard deviations σ_y, σ_ε, and σ_η.

C. Generalized Performance Surfaces

Figures 3-8 are generalized CUSUM performance surfaces generated using the one-sided CUSUM test, the scaled CUSUM statistic, $\hat{c}(\hat{N})$, and the 3σ test threshold (α = 0.001). Generalized performance surfaces are shown for six values of γ in decade steps from 0.001 to 100. Contours of scaled total loss $\hat{L}$ are plotted in the range 0 to 50. Contours of scaled materials balance number $\hat{N}$ are given over different ranges depending on the value of γ.

To illustrate the use of generalized CUSUM performance surfaces, consider the example of simulated near-real-time accounting in the PPP area of a reprocessing plant using the "worst case" measurements (Fig. 1). The total in-process inventory in the PPP is 41.21 kg of plutonium distributed among seven items of process equipment. (The process equipment and their in-process inventories are described in Ref. 5; see especially App. G.) Thus, the total measured inventory is the sum of the measured inventory in each piece of equipment, seven measurements in all. The total inventory measurement-error variance σ_y^2 is the sum of the individual inventory measurement-error variances; thus σ_y^2 = 0.9775 kg^2. The variance of the measured net inventory change is $\sigma_I^2 = 2\sigma_y^2$ = 1.955 kg^2 or σ_I = 1.40 kg, which is the value of the loss-scaling parameter.

The process throughput is 2.089 kg of plutonium per hour or a total of 16.7 kg during each 8-h accounting period. For each input or output measurement, the ε-error relative standard deviation is 1.414% and the η-error relative standard deviation is 0.583%. If input-output measurements are made every hour, then, after 8 hours,

$$\sigma_\varepsilon^2 = (2)(8)(2.089)^2(0.01414)^2$$
$$= 0.0140\ kg^2,$$

and

$$\sigma_\eta^2 = (2)(8)^2(2.089)^2(0.00583)^2$$
$$= 0.0190\ kg^2.$$

Thus, in this example, the values of the parameters N_o and γ are

$$N_o = \frac{0.0140}{0.0190} = 0.737,$$

and

$$\gamma = (0.737)\left(\frac{0.0140}{1.955}\right) = 0.0053.$$

The value of γ in Figs. 3-8 closest to the actual system value is 0.01, and the generalized CUSUM performance surface in Fig. 4 is the closest available approximation to the actual performance surface.

The approximation to the actual system's level of performance can be improved

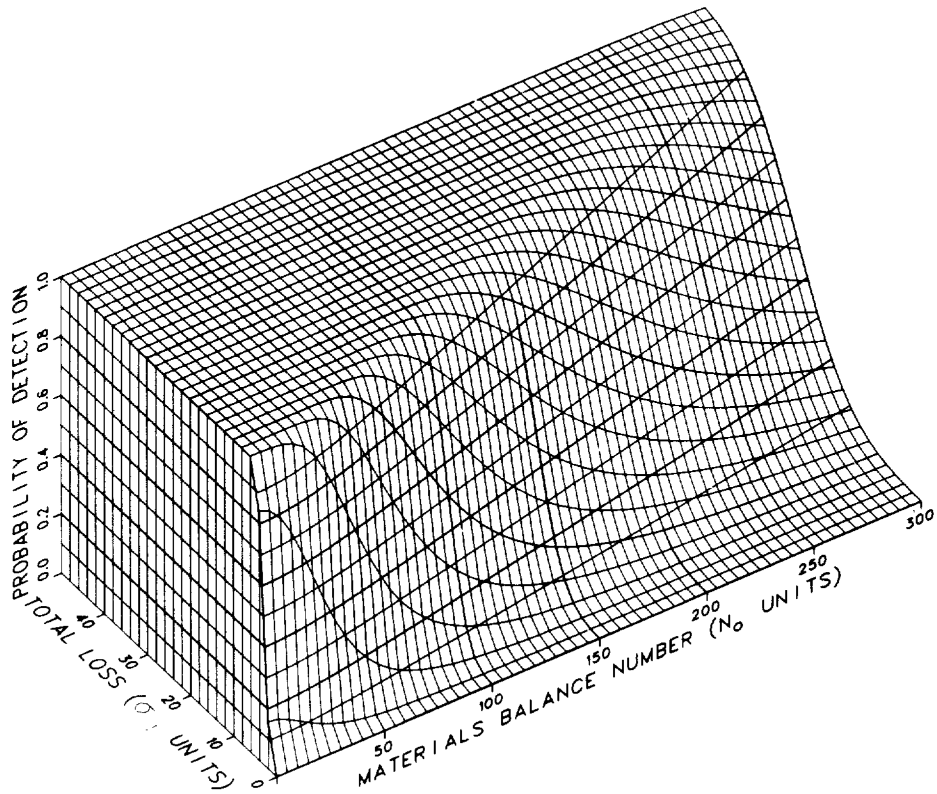

Fig. 3.
Generalized CUSUM performance surface for $\Upsilon = 0.001$, $\alpha = 0.001$.

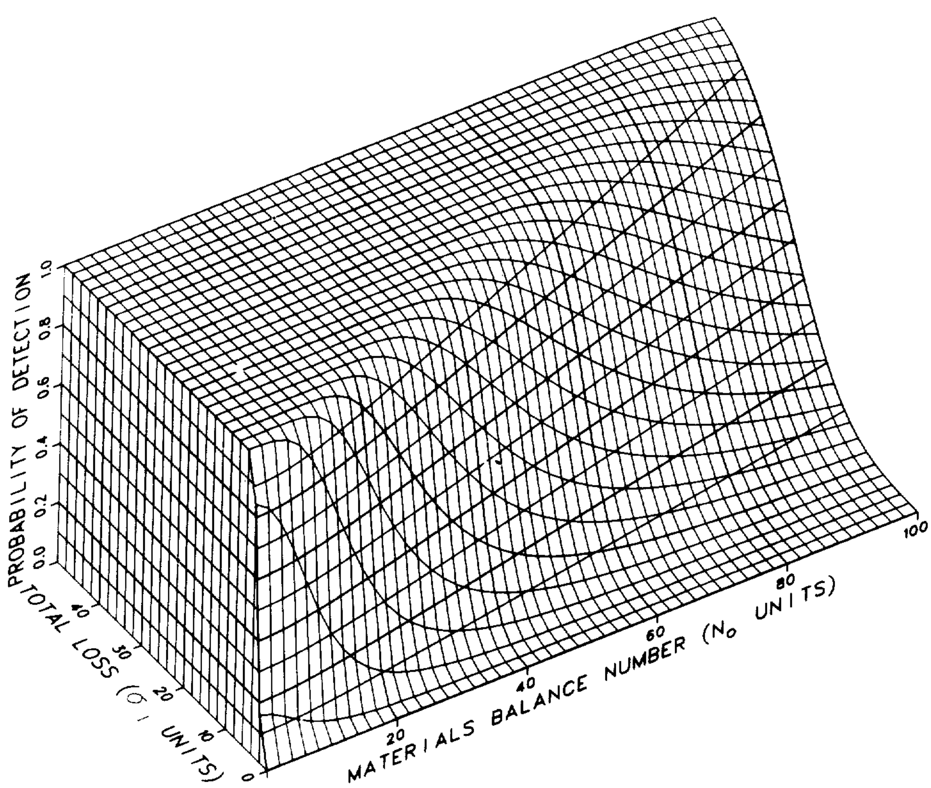

Fig. 4.
Generalized CUSUM performance surface for $\Upsilon = 0.01$, $\alpha = 0.001$.

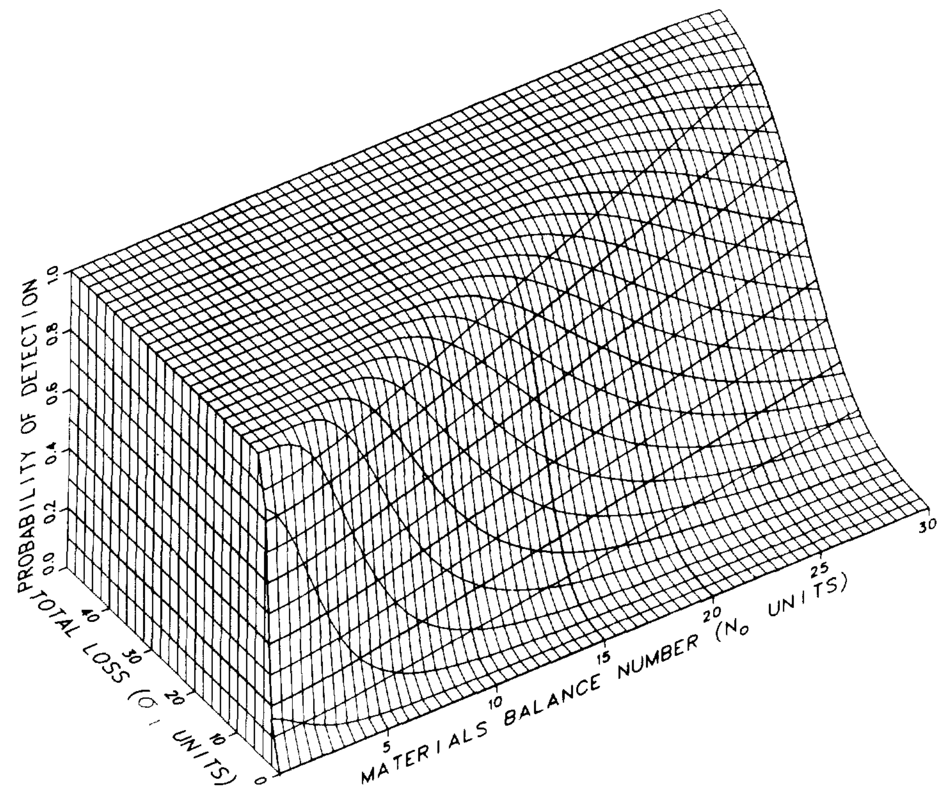

Fig. 5.
Generalized CUSUM performance surface for $\gamma = 0.1$, $\alpha = 0.001$.

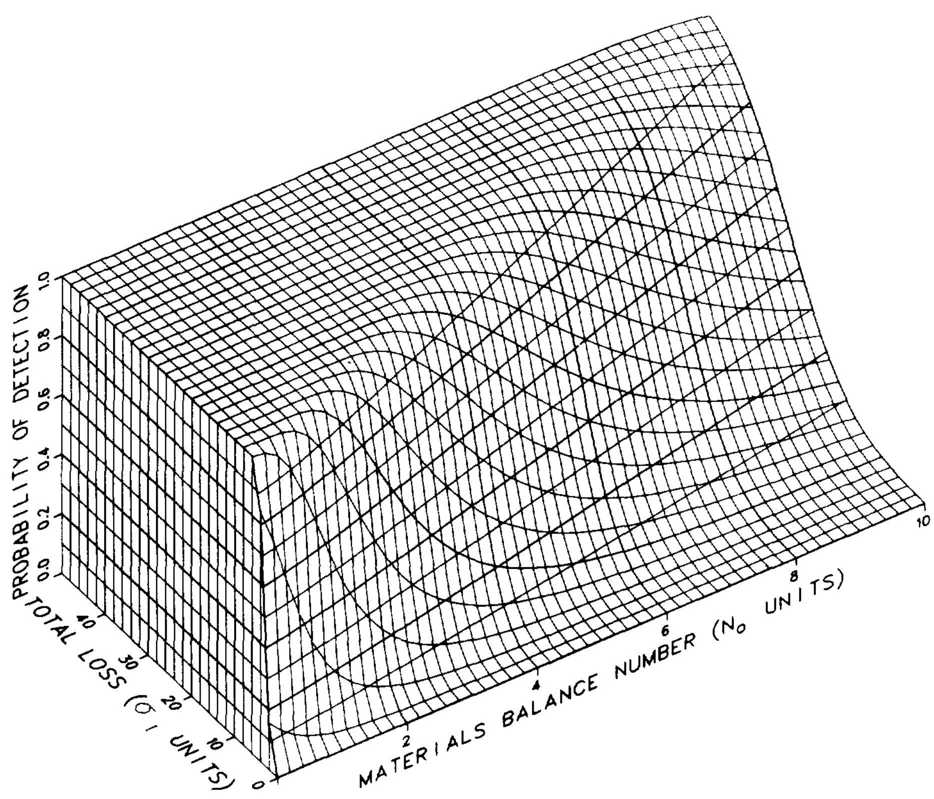

Fig. 6.
Generalized CUSUM performance surface for $\gamma = 1$, $\alpha = 0.001$.

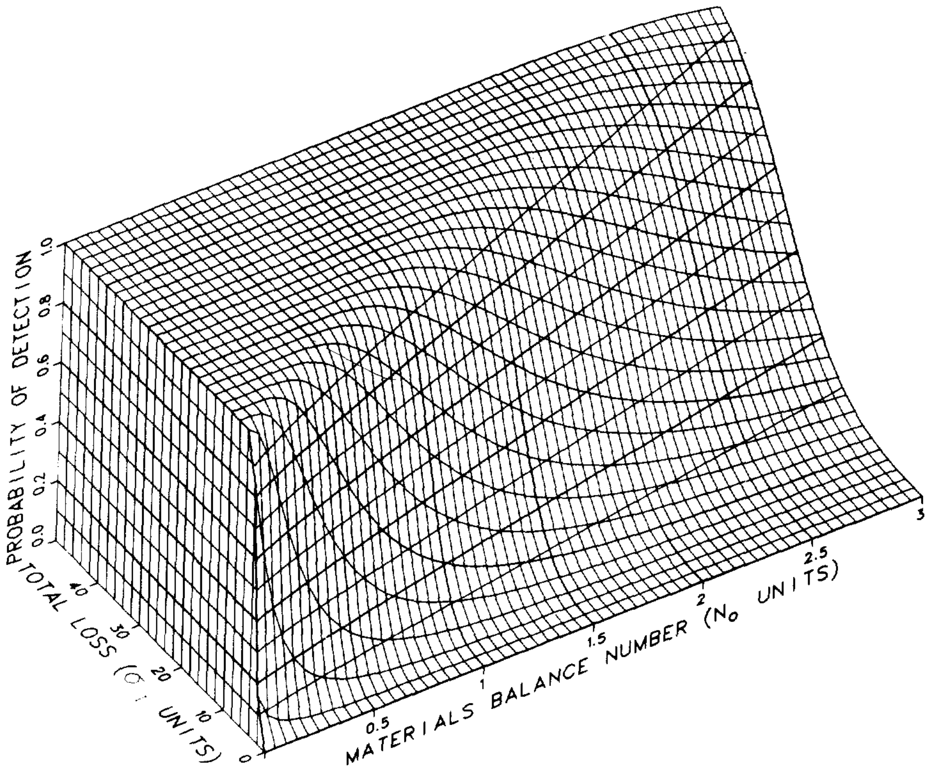

Fig. 7.
Generalized CUSUM performance surface for γ = 10, α = 0.001.

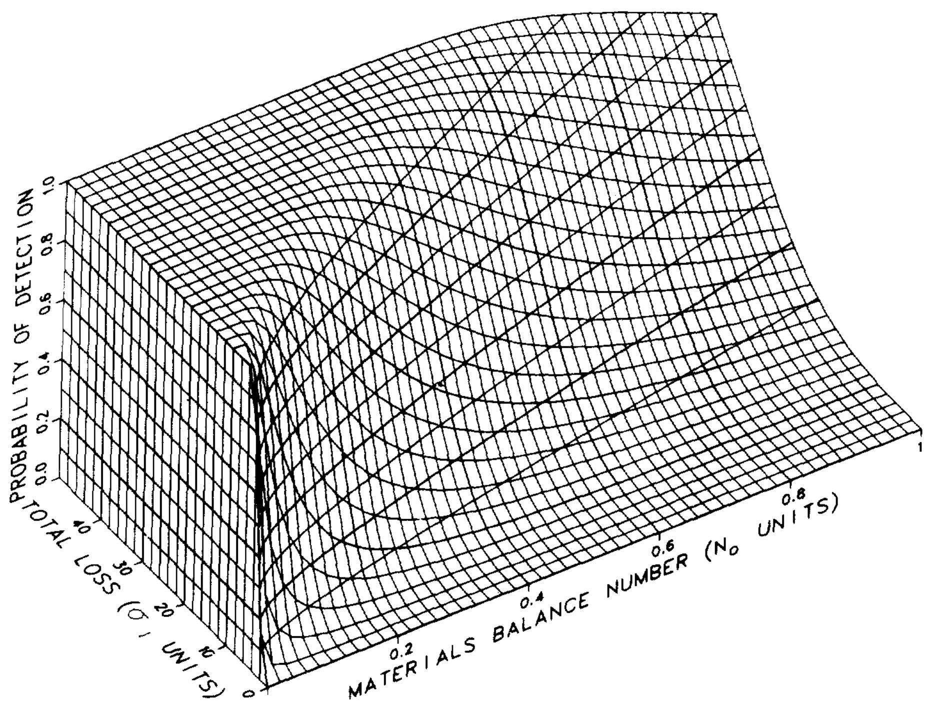

Fig. 8.
Generalized CUSUM performance surface for γ = 100, α = 0.001.

substantially by using the following relationship between the parameters N_o and γ,

$$N_o^* = N_o\left(\frac{\gamma^*}{\gamma}\right)^{1/2}.$$

This approximate relationship, which is obtained from Eq. (8), is useful when $\hat{N} \geq 5$. Over that range of $\hat{N}$, the effect of differences between the actual system value γ and the plot value γ^* can be compensated for by using a slightly different value of the time scaling parameter, N_o^*, rather than the actual system value N_o. The appropriate value of N_o^* is

$$N_o^* = (0.737)\left(\frac{0.01}{0.0053}\right)^{1/2} = 1.0.$$

Thus, the materials balance numbers in Fig. 4 are scaled by $N = \hat{N}\, N_o^*$ or $N = \hat{N}$ in this example. If $N = 21$ balances (one week), then the scaled loss value from Fig. 4 is $\hat{L} = 10$ at the 90% level of detection probability. The corresponding loss value is $L = \hat{L}\ \sigma_I$, or $L = (10)(1.4) = 14$ kg, which is a result obtained previously from Fig. 1. If $N = 84$ balances (four weeks), then $\hat{L}$ from Fig. 4 is 36 at the 90% level of detection probability, so that $L = 50$ kg, in agreement with another previous result.

D. A Special Case

One interesting subset of accounting systems arises in the limit of uncorrelated transfer measurements; that is, in the limit that the calibration error $\eta \to 0$. The CUSUM variance in that limit is

$$\sigma_c^2(N) = \sigma_I^2 + N\ \sigma_\varepsilon^2, \tag{9}$$

and the scaled CUSUM variance is

$$\hat{\sigma}_c^2(\hat{N}') = 1 + \hat{N}'.$$

As before, L is scaled in terms of σ_I, but N is scaled differently; thus,

$$\hat{N}' = \frac{N}{N_1},\quad N_1 = \frac{\sigma_I^2}{\sigma_\varepsilon^2} \tag{10}$$

The new time scaling parameter N_1 is the value of N for which the CUSUM variance terms, σ_I^2 and $N\sigma_\varepsilon^2$, are equal.

The generalized CUSUM performance surface for the special case of uncorrelated transfer measurements is shown in Fig. 9. Only one generalized performance surface is required because the parameter γ has been eliminated. Figure 9 can be used to estimate accounting systems performance in the manner illustrated by the previous examples.

E. The Effect of Recalibration

Figure 9 also can be used to estimate the effect on performance of periodically recalibrating the input-output instrumentation. We assume that each recalibration provides an independent estimate of the η-type errors in the input-output measurements. Other correlations or biases that are unaffected by recalibration are not included here. (A more thorough discussion of the effect of calibration on accounting systems performance is given in Ref. 6.)

Assume that the input-output instrumentation is calibrated every k accountability periods. Let K be the total number of recalibrations during N accountability periods; that is, $N = Kk$. Under the measurement models described previously, it can be shown that the CUSUM variance, $\sigma_c^2(K)$, over K recalibration periods is

$$\sigma_c^2(K) = \sigma_I^2 + K\ \sigma_k^2, \tag{11}$$

where

$$\sigma_k^2 = k\ \sigma_\varepsilon^2 + k^2\ \sigma_\eta^2.$$

Note that $\sigma_c^2(N)$ in Eq. (9) for the special case of uncorrelated transfer measurements is identical in form to $\sigma_c^2(K)$ in Eq. (11). Therefore, the generalized CUSUM performance surface in Fig. 9 also describes the recalibration case with the following identifications:

$$\sigma_\varepsilon \to \sigma_k,\quad \hat{N} \to \hat{K};\quad \hat{K} = \frac{K}{K_o},\quad K_o = \frac{\sigma_I^2}{\sigma_k^2}. \tag{12}$$

Thus, the scaled number of materials balances $\hat{N}$ in Fig. 9 becomes the scaled number of recalibrations $\hat{K}$ of the input-output instrumentation.

Consider the example of simulated near-real-time accounting in the PPP using the best-case accountability measurements (Fig. 2), that is, 5% inventory measurements and recalibration of input-output instrumentation every two days or six materials balances. Then, in that example, $k = 6$ and

$$\sigma_k^2 = (6)(0.0140) + (6)^2(0.0190)$$

$$= 0.7680\ \text{kg}^2.$$

Also, $\sigma_I = 0.8444$ kg, so that

$$K_o = \frac{(0.8444)^2}{(0.7680)} = 0.928.$$

During any 8-day period (24 balances), there are $K = 4$ recalibrations, and the scaled calibration number is

$$\hat{K} = \frac{(4)}{(0.928)} = 4.3.$$

The corresponding scaled loss from Fig. 9 is $\hat{L} = 10$ at the 90% level of detection probability; thus, $L = (10)(0.844) = 8.4$ kg, which is the same level of detection sensitivity found in Fig. 2 at $N = 24$ balances, as it should be.

During any four-week period (84 balances), there are $K = 14$ recalibrations, so that

$$\hat{K} = \frac{(14)}{(0.928)} = 15.1,$$

and $\hat{L} = 17$ from Fig. 9 at the 90% detection probability level; thus, $L = (17)(0.844) = 14.3$ kg, versus 14 kg found previously from Fig. 2.

IV. SUMMARY

The performance surfaces developed in this paper are a natural means of displaying the expected behavior of materials accounting systems in terms of commonly accepted performance measures. In particular, CUSUM performance surfaces are easily obtainable, conservative indicators of systems capability that should be particularly useful in designing and evaluating near-real-time materials accounting systems. The performance surfaces and their mathematical basis give insights into the most effective directions for systems improvement. Generalized CUSUM performance surfaces (Figs. 3-9) provide flexibility because they are applicable to many possible accounting systems designs.

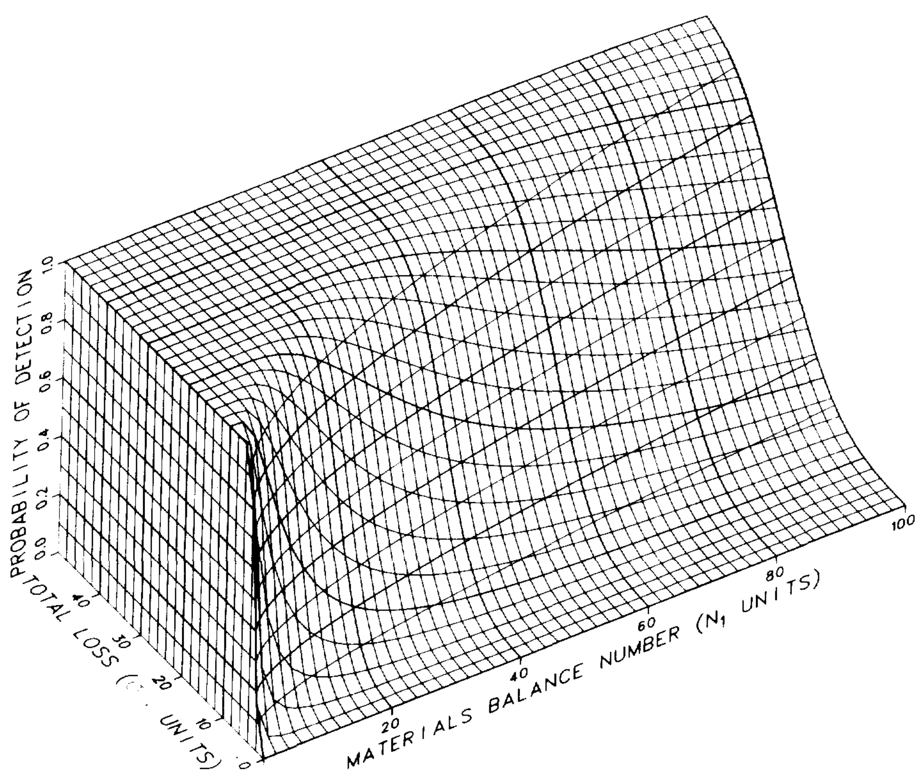

Fig. 9.
Generalized CUSUM performance surface for uncorrelated transfer measurements ($\alpha = 0.001$).

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Second LOFT Test in Idaho

The second in a series of nuclear tests in the LOFT (Loss of Fluid Test) reactor was conducted on May 12 in Idaho.

As in the first nuclear accident simulation, initial results indicate that the emergency core cooling system functioned as expected.

The 50 thermal megawatt LOFT reactor is the largest facility in the NRC's program of confirmatory research designed to study the effectiveness of systems intended to provide emergency core cooling (ECC) for light water-cooled reactors in the event of a pipe-break accident. Data from the many experiments in this research program are being used to help predict the performance of ECC systems in large reactors, and increase NRC's ability to confirm independently the margins of safety that have been estimated during licensing reviews.

The LOFT reactor, located at the Department of Energy's Idaho National Engineering Laboratory, is operated by EG&G, Idaho.

The May 12 test simulated the events which would follow the largest break considered possible in the reactor system piping—a complete rupture of a large pipe supplying cooling water to the reactor core. Instruments recorded pressures, fuel-rod temperatures, coolant flow rates, and the time required to again cover the core with emergency coolant to keep the nuclear fuel cooled.

The experiment began with the opening of two large blowdown valves in 20 thousandths of a second, simulating the sudden shearing of the coolant pipe. Steam and water were quickly discharged through the break to a suppression tank where the steam was condensed.

Although the experiment was conducted at a power level of about 1/90th that of a commercial power reactor, the power density in the LOFT fuel core was about equal to that of a commercial reactor. Initial results indicate that the emergency core cooling system functioned as expected and the behavior of the primary coolant system agreed in general with the predictions. In particular, the maximum fuel cladding temperature of nearly 1200 degrees fahrenheit was about equal to the value predicted for the blowdown phase (rapid depressurization). During the reflood stage, in which the fuel core is covered again with water, the peak fuel cladding temperature was about 250 degrees fahrenheit lower than the best estimate prediction.

In the first nuclear experiment, the peak temperature of the fuel cladding was 990 degrees fahrenheit, about 30 degrees F below the final best estimate. The experiment was conducted in December 1978 at a power level of about 1/120th that of a commercial power reactor, generating a power density in the LOFT core that was about two-thirds that of a commercial reactor.

Austrian, Dutch, Finnish, German and Japanese scientists, on assignment to INEL, observed the May 12 experiment and will assist in the detailed analysis of test data during the next several months.

Nuclear experiments in LOFT will continue at high power levels and will deal with a variety of pipe break sizes and locations and with alternate emergency cooling systems.

MACSY—A Minicomputer Accounting and Control System

V. Jarsch, S. Onnen

Kernforschungszentrum Karlsruhe GmbH
Institut für Datenverarbeitung in der Technik
Postfach 3640, D-7500 Karlsruhe
Federal Republic of Germany

ABSTRACT

A minicomputer-based system for nuclear materials control is presented. Significant properties of the system are interactive data entry by the plant operators themselves, user-oriented dialogues and user-independent internal data representation, and description of user processes by means of basic functions.

INTRODUCTION

The Karlsruhe Nuclear Research Centre is subject to the regulations of the non-proliferation treaty like all similar organisations. For this reason its research and development program includes the development of a data processing system for accounting and control of the nuclear materials stored and handled in the various laboratories and reactors in the centre.

Plant operators and plant management usually mistrust the introduction of new techniques and of computer systems in particular. However, they are ready to accept a data processing system which is able to fulfill not only its original task (to process data according to the Euratom regulations) but also to provide information they need to manage their facility. And so do the users of MACSY, the Minicomputer Accounting and Control System being implemented at Karlsruhe.

For a research centre with its variety of facilities and institutes, typically there is a great variety of processes and materials. So, evidently, a concept is needed, which is general enough to be applicable not only to a distinct facility, but also to all of the different institutes and facilities of the research centre and consequently to other plants subject to safeguards control.

NUCLEAR MATERIALS PROCESSES

The Centre's activities are mainly concentrated upon three problems in the field of nuclear energy:

- ensuring the continuity of fuel supply for the generation of nuclear power,
- reprocessing and ultimate storage of radioactive materials, and
- safety of nuclear installations.

These activities are done in various scientific and technical divisions, in which nuclear material is handled in a large number of processes. Examples: In the Fast Zero Power Reactor (SNEAK), different types of fuel elements are assembled to research fundamental physics of fast reactors and to determine physical and technical data for fast breeder reactors. At the plutonium laboratory of the Institute of Materials and Solid State Research, nuclear fuels with different chemical compositions are fabricated for experiments. In hot cells of the reactor technical division the fuel elements are cut into samples for examination and analysis of highly radioactive materials. Besides these divisions, there are other departments where the nuclear materials of different physical forms and chemical compositions are mounted, processed, and measured.

The nuclear materials control system MACSY maps a real-world process of nuclear materials handling onto a process running on the computer. A thorough analysis of all activities and processes involving nuclear materials in different nuclear facilities and a subsequent abstraction show that each real process can be defined as a sequence of three basic operations:

- transfer,
- composition, and
- decomposition of material.

Transfer is the name of a class of activities like receiving and shipping of materials.

Processes like cutting fuel elements, disassembling bundles, enrichment, etc., can be subsumed under decomposition of a discrete nuclear material into two or more distinguished materials with different data.

Assembling fuel elements, collecting waste, producing mixed powder, and other processes are considered as the composition of two or more discrete materials to one material.

To each basic operation corresponds a basic software function. Just as analysis of any real process leads to the basic operations (transfer, composition, and decomposition), the corresponding computer process is a synthesis of the basic software functions transfer, composition, and decomposition (Fig. 1).

THE 'SNAPSHOT METHOD' FOR MATERIALS DESCRIPTION

Pellets, powder, fuel elements, and liquid waste are examples of objects we find in our institutes and laboratories. At first glance, they seem to be uncomparable. So we leave the level of these real-world objects and look at them from the next higher level of abstraction. Here all these materials, regardless of how different they are in reality, are handled as logical objects of a certain type and of certain characteristics.

Any real object is described by a unique scheme consisting of a set of attributes. The attributes serve to express the properties of a real object on a logical object level.

We determined this set of attributes so that it constitutes a complete description of the material as needed for Euratom reports (ICR, MBR, PIL) and IAEA inspections, and - following the second aim - that it contains and furnishes further data which are needed for plant management and operation.

Figure 2 shows an extract of the set of attributes:

- the object's name,
- the type (e.g., fuel element, pellet, waste, ...),

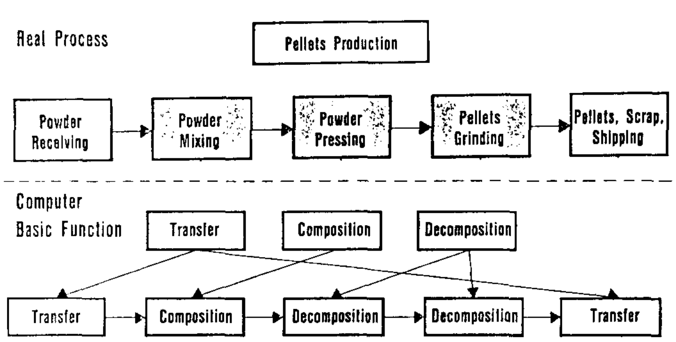

Fig. 1. Mapping of a real process into a computer process

```
LOGICAL OBJECT ::= <NAME>
                   <TYPE>
                   <LOCATION>
                   <DATE>
                   <OPERATOR>
                   <OPERATION>
                       :
                   <CHEMICAL COMPOSITION>
                       :
                   <WEIGHT U-234>
                       :
                   <WEIGHT Pu-241>
                       .
                   <SAFEGUARDS OBLIGATION>
                       .
```

Fig. 2: Attributes of a logical object

- the location (the institute, the room in the institute, the place in the room, where the object is to be found in the case of inspections),
- the operation which led to the object's present existence,
- the name of the operator who is responsible for this operation, and
- the day and time of the operation.

The second group of attributes describes the material specification, i.e., chemical composition, weights of elements and isotopes, as well as safeguards obligation, the materials' origin. In this way, any real object in the research centre is represented by one logical object.

As one logical object describes the state of a real object over a certain time interval, the totality of the logical objects describes the complete set of materials over a time interval. Informally, a logical object may be regarded as a snapshot of a nuclear material at a certain time, and consequently the complete set of logical objects is a snapshot of the plant's situation at a certain time. Just like in the history of photography, a snapshot is only the first step. As the objects in a facility are involved in actions (they are transported or manipulated), we need a kind of 'movie' as a description method.

A logical object describes a state of a real object. Whenever the real object is manipulated, it changes its state, i.e., it changes at least one attribute. For example in the case of a transportation, the value of the attribute 'location' is altered.

The real object's new state is described by a new logical object. The values of its unchanged attributes are the same as those of the old ones. The new values of the changed attributes depend upon the operation which has been done with the object.

Now there exist two logical objects for one real object. There is a relation 'predecessor <-> successor' between them. A logical object may have more than one successor or more than one predecessor, respectively.

If for example a fuel rod is cut into several pieces, a corresponding number of new logical objects is generated which are all successors of the old one. Besides the attributes 'length' and 'name', which have changed following the cutting operation, all new objects contain the same identical attribute values.

This results in a certain number of logical objects generated during the lifecycle of a real object. Containing date and type of any alteration, they constitute a complete history of the real object, which serves for reporting, inventory, and material follow-up purposes.

Up to this point we demonstrated the way we proceeded from the level of the real world to the logical object level. The description scheme of the logical level is complete and it is easy to handle. But in the form presented here it lacks efficiency.

Example (Fig. 3):

A fuel rod is transported from storage to the process line, there it is cut into two pieces, which then are transported back to storage. Fig. 3 shows the logical objects which are generated during this process. The first part of the process (transportation) changes the attribute 'location'. The second process-step (cutting) leads to new values of 'length' and 'name'. The transportation back to the storage alters the 'location' attribute again from 'process line' to 'storage'. Besides these attributes a number of others are changed: the name of the operator who executes the process, the kind of operation, the date. But the majority of attributes - the material specification - is not affected at all. This is the case for most of the manipulations which are done with the material.

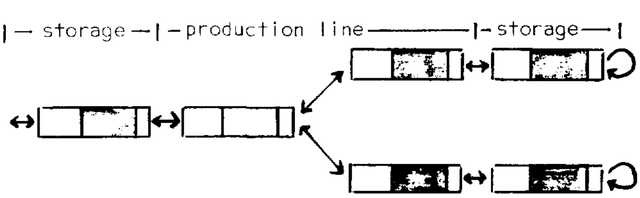

Fig. 3. Sequence of logical objects generated during a process

That is why we extracted the material specification attributes from the logical object and let more than one object share the same identical material description (Fig. 4).

A material specification describes the composition, the weights of elements and isotopes, and the safeguards obligation of a nuclear material.

ORGANISATION OF FACILITY LOCATION AND CRITICALITY CONTROL

For the handling and inventory verification of the objects a unique identification of the facility locations is necessary.

The subdivision of the centre into material balance areas and key measurement points as specified for safeguards purposes is not detailed enough for the identification of each location inside the facility.

A location can be identified by one or more attributes: box, room, building, area, etc. The location identification combined with the requirements of criticality safety demand a hierarchical organisation of the location data: a tree structure. Each location corresponds to one leaf of this tree. The nodes of the tree represent rooms, buildings or parts of the facility. All these attributes are stored in a relational database. Each leaf and node are represented by a record.

In addition to the location identification, this record contains information on the criticality limits, the inventory of nuclear materials, and the key which selects the inventory verification algorithm.

Searching the tree permits checking every amount of nuclear material for each location before the process is executed. The algorithm for calculating the quantities of materials for

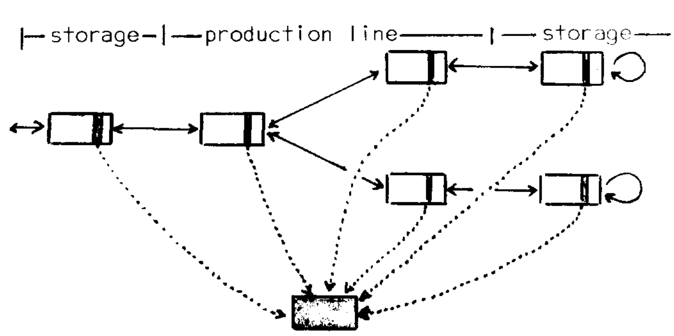

Fig. 4. Logical objects referencing the same material description

the criticality checks depends upon the location, as well as the actions which are undertaken in case limits are exceeded. Possible actions are message to the operator or his supervisor, or the system's refusal to execute a process.

Furthermore each leaf and node of this tree can be sealed. This means that no movement (shipping or receiving) of nuclear materials will be permitted from this location as long as the seal is set. Any local movement inside the sealed location is tolerated.

THE SYSTEM'S DESIGN

Briefly, the system's purpose is to acquire data given by an operator, to store these data in the database, and to retrieve, correlate and output data from the database.

In order to achieve a solution which is well-suited from the point of view of software construction, we divided the system into three independent functional modules as illustrated in Fig. 5.

The user interface transforms data objects from the individual representation familiar to the various users into a standardized internal representation, and vice-versa.

Storage and retrieval of data are performed by a database management system (FADABS) which has been developed to be used in MACSY and in other applications [1].

The consequent modularisation into a man-computer communication subsystem, a database management system, and special control functions facilitates the attempt to design a generalized system.

As the database management system has no direct concern with the problems discussed in this paper, we exclude it from our considerations.

The system is divided into two levels: the user level and the system level. On the user level, the user communicates with the system

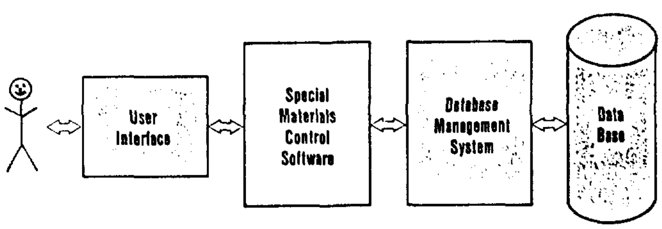

Fig. 5. Logical structure of MACSY

via alphanumeric terminals and describes the processes he executes with the nuclear material in his own everyday symbolism. This individual representation is projected onto the system level by functions supporting the dialogue and by transformations.

On the system level the result of a user's dialogue is a structured set of logical objects in a 'communication buffer'. The attributes of these objects have values, which have been set in the course of the user's dialogue. Thus they form a user-independent description of the user's real process.

An interpreter on the system level takes the logical objects stored in the communication buffer as input and triggers the execution of the basic functions described in the second chapter.

Any basic function called-up by the interpreter reads the attributes of the corresponding objects, completes them, and performs special functions like criticality checks, sum checks, and inventory verification. Then it inserts the object into the buffer again.

The communication buffer serves both as a dynamic extension of the computer memory and as an intermediate small database which assures the integrity of the main database. No user process may write directly into the main database. Each object which is referenced by a process, as well as a newly generated object, is copied into the communication buffer. A user process works on the objects stored in the communication buffer. It may read, alter, and insert them. Only at the end of a user process the system decides whether the objects are saved from the communication buffer into the main database or whether they are deleted. Saving is done only if the process has been terminated error-free.

CONCLUSION

The decision to implement a materials control system as a dedicated minicomputer system with its own terminals in the laboratories was made under safeguards aspects. A 'public' large computer open to anyone in the centre does not satisfy the desire for privacy. Besides we found it important to put the data entry stations as far as possible immediately at the locations where the data arise, and to make the laboratory operator himself responsible for the correct data input.

Preconditions to reach these aims are: to adapt the user-system-interface to the individual needs of the users, and to make the software general enough to avoid software adaptions.

MACSY is presently being implemented at the Nuclear Research Centre of Karlsruhe.

ACKNOWLEDGEMENTS

The authors thank J. Woit and Dr. A. Jaeschke for useful discussions concerning the nuclear materials description. We acknowledge the cooperation of F.-J. Polster and R. Kerpe, the implementers of FADABS, and of G. Tretter, who takes part in the implementation of the user interface.

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Victor Stello, Director of Inspection and Enforcement, John Davis, Deputy Director of NMSS

Victor Stello has been appointed Director of the Nuclear Regulatory Commission's Office of Inspection and Enforcement (I&E), Chairman **Joseph M. Hendrie** announced. Mr. Stello succeeds Dr. **Ernst Volgenau,** who resigned in July 1978 to enter private consultant work. **John G. Davis,** has been serving as acting director of I&E, has been named Deputy Director of the Commission's Office of Nuclear Material Safety and Safeguards (NMSS).

The Office of Inspection and Enforcement carries out inspections and investigations and initiates enforcement actions involving activities licensed by the NRC. In addition to its headquarters staff, I&E has five regional offices—at King of Prussia, PA; Atlanta, GA; Glen Ellyn, IL; Arlington, TX; and Walnut Creek, CA. The Office of Nuclear Material Safety and Safeguards, headed by William J. Dircks, is responsible for licensing and regulating the handling of nuclear materials, construction and operation of nuclear fuel cycle facilities, and the safeguarding of nuclear facilities against sabotage and nuclear materials against theft.

In announcing the appointments, Chairman Hendrie said:

"The Office of Inspection and Enforcement has a key role in helping to assure that public health and safety are protected in activities conducted by NRC licensees. Mr. Stello brings to his new position almost 20 years of experience in various aspects of the nation's nuclear program. Most recently, he was a senior NRC representative at the Three Mile Island site for 39 days. The Commission also wants to express its appreciation to John Davis, who has served as Acting Director of Inspection and Enforcement during a very demanding period. In his new position as Deputy Director of Nuclear Materials Safety and Safeguards, Mr. Davis will assist Mr. William Dircks in carrying out the important responsibilities of that office."

Mr. Stello has been Director of the Division of Operating Reactors in NRC's Office of Nuclear Reactor Regulation since 1975. He joined the regulatory organization of the former Atomic Energy Commission in 1966 as a member of the licensing technical staff. In 1971 he was promoted to branch chief of a boiling water reactors licensing branch and in 1972 he became chief of the reactor systems branch. He was named Assistant Director for Reactor Safety in March 1973, and Director of the NRC Operating Reactors Division in 1975.

Mr. Stello worked from 1960-1965 at the CANEL office of Pratt and Whitney Aircraft Company in Middletown, Connecticut, participating in analyses of the high temperature liquid metal reactors systems proposed for use in aircraft and space applications. From 1965-1966 he worked in the East Hartford, Connecticut, offices of Pratt and Whitney as part of the study of advanced jet engine concepts for military and commercial applications.

Mr. Stello received a bachelor of science and master of science degree in mechanical engineering from Bucknell University and has done graduate work at Rensselaer Polytechnic Institute.

Mr. Davis has been Deputy Director of Inspection and Enforcement since 1975. Before that he was Deputy Director for Field Operations of the AEC Directorate of Regulatory Operations, and also served as Assistant Director for Radiological, Environmental and Materials Protection. Before coming to Washington in 1973, he was director of the AEC's Region II office at Atlanta, and supervisory radiation specialist at the Region III office in Chicago.

From 1955 to 1957, Mr. Davis was with the E.I. duPont de Nemours and Company at the Savannah River Plant in the Works Technical Department-Health Physics Section. He was employed by the AEC with the Isotopes Extension at Oak Ridge and the Division of Licensing and Regulation at Germantown, MD, from February 1957 to August 1958, when he resigned to accept a position on the staff of Texas A&M College.

Mr. Davis received a B.S. degree in chemistry from the Virginia Military Institute in 1950, and M.B.A. degree from Georgia State College in 1968. He served in the U.S. Army in Korea and Japan from 1950 to 1954.

A Minimum Risk Trigger Index

F. H. Tingey

EG&G Idaho, Inc.
Idaho Falls, Idaho

Introduction

A fundamental problem in effective nuclear materials management is that of taking an appropriate action on the basis of the data at hand. To assist in this decision process one often computes from the data indices which, along with their associated error estimates, provide criteria for action. Such commonly used indices as the inventory difference, cumulative sum of inventory difference, minimum variance loss estimators, and autoregressive inventory difference estimates are of this form. The classical approach, assuming an index and its associated error is one of constructing statistical tests which under certain distribution assumptions make the likelihood of a false alarm small. The test's failure to indicate a loss/diversion when in fact one has occurred is quantified by the test's operating characteristic curve which can be used to select the "best" test among a number of candidates, assuming a given loss scenario.

This classical approach tends to ignore the cost consequences of wrong decisions either in the nature of a false alarm or failure to act when warranted and consequently is often resisted by the facility production manager when confronted with a process shutdown strictly for safeguards purposes. Especially is this the case when history has established a pattern of the material being "found" either in process or in measurement errors once an investigation was initiated.

It would appear therefore that to properly address the problem one should utilize indices which consider the disutility and cost consequence of the facility during shutdown and the monetary and strategic cost consequence of undetected losses whatever the magnitude. There are other consequences having cost significance that might also be addressed; however, the above are the principal two.

Game theory (1) provides a technique by which "optimum" strategies can be determined. Minimax (2,3) methods can also be applied to obtain "optimum" indices. The approach to be presented in this paper provides still another "optimum" and is an adaptation of a technique that has been successfully used in quality control (4,5) to establish specification limits for a manufactured product when both process variation and measurement error are present and costs of misclassification are considered.

Risk Model

In the determination of the appropriate action to be taken on the basis of an observed value Y for a given index X (say, inventory difference), the following two classification errors are possible:

1. A Y can be judged to be indicative of loss/diversion when in fact none has occurred.
2. A Y can be judged to be non-indicative of loss/diversion when in fact such has occurred.

For any given true value x for X economic losses usually can be associated with misclassifications, since actions (including "doing nothing") are precipitated by estimates of x.

Assuming, therefore, a capability to quantify losses/diversions as a function of x, among other things, and a knowledge of the process and measurement variation, one can define the total risk associated with the classification and subsequent action as the expected value of the conditional expected loss resulting from misclassification, the condition being that the index is a given value x. The expectation is with regard to the distribution of x. The conditional expected loss for given x is the sum of the probabilities of misclassification times the losses resulting therefrom.

Definitions and Assumptions

Let

X = Index (Inventory Difference)
Y = Observation on X

Z = Process loss
z = Given value of Z
δ = Threshold quantity for the combined process and diversion losses
L = Trigger Index for X

Assume
Z is distributed as $N(\mu,\sigma_p^2)$ and that measurement errors are distributed as $N(0,\sigma_m^2)$.

Define loss functions
C_a = Cost consequence of not acting on a given Y (1)
C_r = Cost consequence of acting on a given Y (2)

where the decision rule is "if $Y < L$ no action will be taken; otherwise an action of cost consequence C will be taken."

Assuming the threshold quantity δ and linear cost consequence for losses represented by (1) above, the loss functions are

$$C_a = \begin{cases} 0 & \text{if } z + \theta < \delta \\ C_1 \dfrac{(z + \theta - \delta)}{\sigma_p} & \text{if } z + \theta > \delta \end{cases} \tag{3}$$

$$C_r = \begin{cases} C_2 & \text{if } z + \theta \leq \delta \\ 0 & \text{if } z + \theta > \delta \end{cases} \tag{4}$$

where θ is the unknown amount diverted.

Derivation of a Trigger Index

Let
$w = z + \theta$, then

$$\text{Prob}(Y < L|w) = F\left(\frac{L-w}{\sigma_m}\right)$$

and

$$\text{Prob}(Y \geq L|w) = 1 - F\left(\frac{L-w}{\sigma_m}\right)$$

where

$$f(t) = \frac{1}{\sqrt{2\pi}} e^{-t^2/2}$$

and

$$F(t) = \int_{-\infty}^{t} f(s)\, ds \quad .$$

Thus the conditional expected loss resulting from misclassification, the condition being that the true combined process and diversion loss is w, in a given case is

$$R(w) = F\left(\frac{L-w}{\sigma_m}\right) C_a + \left|1 - F\left(\frac{L-w}{\sigma_m}\right)\right| C_r \quad . \tag{5}$$

The total risk R to be associated with the decision rule is defined as the expected value of $R(w)$, that is

$$R = \int_{-\infty}^{\infty} R(w) \frac{1}{\sigma_p} f\left(\frac{w - (\theta+\mu)}{\sigma_p}\right) dw \tag{6}$$

which by (3) and (4) becomes

$$R = \frac{C_2}{\sigma_p} \int_{-\infty}^{\delta} \left\{1 - F\left(\frac{L-w}{\sigma_p}\right)\right\} f\left(\frac{w-\gamma}{\sigma_p}\right) dw$$

$$+ \frac{C_1}{\sigma_p} \int_{\delta}^{\infty} \left(\frac{w-\delta}{\sigma_p}\right) F\left(\frac{L-w}{\sigma_p}\right) f\left(\frac{w-\delta}{\sigma_p}\right) dw \tag{7}$$

where
$\gamma = \theta + \mu$.

By the change of variables

$$v = \frac{w-\gamma}{\sigma_p}$$

$$r = \sigma_p/\sigma_m$$

$$k = \frac{\delta-\gamma}{\sigma_p}$$

$$b = \frac{\delta-L}{\sigma_p}$$

equation (7) becomes

$$R = C_2 \int_{-\infty}^{k} \left\{1 - F[(k-v)\, r - b]\right\} f(v)\, dv$$

$$+ C_1 \int_{k}^{\infty} (v-k)\, F[(k-v)\, r - b]\, f(v)\, dv \ . \tag{8}$$

The condition imposed on equation (8) to arrive at a trigger index is that this function is a minimum, the minimization being with respect to b (and hence L). This is accomplished by taking the derivative of (8) with respect to b and equating the result to zero. The end result after some simplification is

$$C_1\left\{t[1 - F(t)] - f(t)\right\}$$

$$+ C_2 \sqrt{r^2 + 1}\ \ F(t) = 0 \tag{9}$$

where

$$t = \frac{1}{\sqrt{r^2 + 1}} (k + rb) \quad . \tag{10}$$

Equation (9) is equivalent to

$$\frac{f(t) - t[1 - F(t)]}{F(t)} = \frac{C_2}{C_1}\sqrt{r^2 + 1} \quad . \qquad (11)$$

A tabulation of $\frac{f(t) - t[1 - F(t)]}{F(t)}$ as a function of t is given in Table 1 to facilitate the solution of (11) in any given case.

It is to be noted that once having determinted b in a given application, one could numerically integrate equation (8) to determine the corresponding expected risk. Such information might be useful in cost-benefit analyses of actions which would change the input parameters, i.e., σ_m, σ_p, C_1 and C_2, say, resulting in a reduced expected risk.

Example

Suppose the material has value (monetary plus strategic) of $50/unit and the cost of shutdown (sweepdown) is projected at $100,000. It has been determined that process losses average 200 units for the given accounting period with standard deviation 2000 units. Measurement error for the accounting period has been estimated at 1000 units. Thus in the notation of this paper the parameters are

C_1 = 50(2000) = 100,000 (note that the cost model identifies C_1 in units of σ_p)

C_2 = $100,000

μ = 200

σ_p = 2000

σ_m = 1000

The selection of δ is keyed to the cost functions. Assuming any diversion is of significance, one would set $\delta = \mu$. In this case it would result in

$$\delta = 200 \quad .$$

From the above one computes

$$r = \frac{\sigma_p}{\sigma_m} = \frac{2000}{1000} = 2$$

$$\frac{C_2}{C_1}\sqrt{r^2 + 1} = \frac{100,000}{100,000}\sqrt{4 + 1} = 2.24$$

which by Table 1 corresponds to a t of -.5. Since

$$k = \frac{\gamma - \mu}{\sigma_p} = \frac{200-200}{2000} = 0 \quad .$$

b is obtained from (10) by the equation

$$b = \frac{t\sqrt{r^2 + 1}}{r} = \frac{-.5\sqrt{5}}{2} = -.56$$

and finally L from

$$L = \delta - b\sigma_m = 200 - (-.56)(1000) = 760 \text{ units}$$

TABLE 1

t	[f(t)-t(1-F(t))]/F(t)	t	[f(t)-t(1-F(t))]/F(
-3.0	2222.67318	.1	.65009
-2.9	1554.57230	.2	.52980
-2.8	1096.13238	.3	.43171
-2.7	779.08293	.4	.35159
-2.6	558.11177	.5	.28606
-2.5	402.92093	.6	.23241
-2.4	293.10276	.7	.18849
-2.3	214.81145	.8	.15252
-2.2	158.58563	.9	.12309
-2.1	117.91417	1.0	.09903
-2.0	88.28479	1.1	.07939
-1.9	66.54886	1.2	.06340
-1.8	50.49428	1.3	.05041
-1.7	38.55649	1.4	.03989
-1.6	29.62159	1.5	.03140
-1.5	22.89135	1.6	.02459
-1.4	17.79009	1.7	.01914
-1.3	13.90001	1.8	.01481
-1.2	10.91602	1.9	.01138
-1.1	8.61394	2.0	.00869
-1.0	6.82811	2.1	.00659
-.9	5.43535	2.2	.00496
-.8	4.34356	2.3	.00370
-.7	3.48350	2.4	.00274
-.6	2.80279	2.5	.00202
-.5	2.26163	2.6	.00147
-.4	1.82960	2.7	.00106
-.3	1.48332	2.8	.00076
-.2	1.20477	2.9	.00054
-.1	.97993	3.0	.0038
.0	.79788		

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Law Review Article Considers the Civil Liberties Implications of Nuclear Safeguards

The following is a technical note on an article entitled "Civil Liberties and Nuclear Energy Safeguards: the Inevitable Conflict," by *John N. O'Brien* of Brookhaven National Laboratory appearing in the Spring issue of the *Western New England Law Review*.

The article describes how safeguards measures may intrude upon constitutionally guaranteed civil liberties in the United States. The approach taken is to examine three broad areas of safeguards activity and draw conclusions concerning the use of safeguards measures.

The three safeguards areas examined are: access controls, employee screening, and recovery of contraband nuclear materials. The activities involved in collecting domestic intelligence on potential adversaries are not examined because these activities are strongly tied to other security functions of government and private industry.

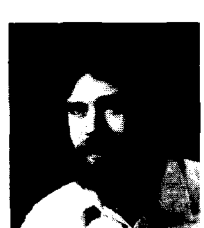
The consideration of access controls extends to all aspects of physical security and material accountancy. The major civil liberties concerns are searching and observation of individuals entering a facility. Both "hands-on" and "hands-off" detection methods are considered in light of all the relevant law. In addition, the law concerning a nuclear facility management's response to an apparent diversion or theft is reviewed. The potential problems associated with search and seizure, arrest and detention, and interrogation of employees during an emergency triggered by apparent malevolence are examined.

Employee screening, which has experienced recent rulemaking activity in NRC, is examined extensively. The legal authority of NRC to implement the proposed program is reviewed. The various techniques which have been considered, in addition to those recently proposed in rulemaking, are covered. Employee screening techniques such as compulsory disclosure questionnaires, national agency checks, and full field investigations are discussed along with psychological testing, polygraphy, and biophysical examinations. The history of employee screening activities is reviewed and general problems endemic to security screening are discussed.

The possible loss or theft of strategic nuclear material has spurred concern over the consequences of a warrantless area search. Those recovery plans which are publicly known are examined along with those measures which may have to be considered in the event of such an emergency. The three available legal justifications for a sweeping search national security, emergency, and ordinary crime, are discussed. Dentention and interrogation of suspects, electronic and physical surveillance of suspects and press censorship are considered.

Conclusions include recommendations for regulatory guides, rulemaking, and jurisdictional changes. In addition, national security and possible legislation are discussed to show how a more viable system may end the openended conjecture which has dominated the debate over civil liberties and nuclear energy safeguards.

O'Brien



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Editor's Note: For years 1972-1977 (Volumes 1-6 of **Nuclear Materials Management**), No. 3 is the proceedings issue of that year. Beginning in 1978 (Volume VII), No. 3 becomes a regular issue of the Journal. The proceedings of the 1978 annual meeting is designated, Volume VII, Proceedings Issue. Future issues of the INMM Proceedings will be designated similarly, i.e., in 1979 it will be designated Volume VIII, Proceedings Issue. Copies of the tables of contents for INMM Proceedings issues are available on written request to the editors.

EXXON Centrifuge Manufacturing Plant To Be Located In Oak Ridge Tennessee

BELLEVUE, Washington—Exxon Nuclear Company today confirmed that it plans to construct a centrifuge manufacturing plant in Oak Ridge, Tennessee if the Company's pending bid to become a principal supplier of centrifuge machines to the U.S. Department of Energy is accepted.

Exxon Nuclear submitted the bid on April 18 in response to a DOE Request for Proposals on the supply of centrifuges for Phase 1 of the Government's planned Gas Centrifuge Enrichment Plant (GCEP) near Portsmouth, Ohio. The Energy Department is scheduled to select two or three suppliers from among the competing bidders in late August 1979.

Exxon Nuclear's proposal to DOE said that if the Company were selected, it planned to consolidate its centrifuge manufacturing activities and facilities into an integrated operation in Oak Ridge. Currently Exxon Nuclear has centrifuge manufacturing and test facilities at Malta, N.Y. and Richland, Wash., respectively, and maintains associated offices in Oak Ridge.

Exxon Nuclear has obtained an option from the City of Oak Ridge to purchase a plant site located on Bear Creek Road in the Clinch River Industrial Park. However, the Company said no final decision had been made on the Bear Creek location and that other Oak Ridge sites were still under investigation.

Lee R. Raymond, President and Chief Executive Officer of Exxon Nuclear Company, said that Oak Ridge was chosen as the plant site following an extensive investigation of candidate sites in several states. He added: "Oak Ridge has been the principal seat of the nation's centrifuge technology development program. As such it has become a repository of trained manpower with all the relevant skills needed in this high technology field. Oak Ridge is also reasonably close to DOE's planned Gas Centrifuge Enrichment Plant at Portsmouth, Ohio, and this will facilitate delivery of the machines to the customer. Last but not least, the cooperation which Exxon Nuclear has received in the Oak Ridge area over the years has been outstanding. These were among the principal considerations influencing our siting decision."

Noting Exxon Nuclear's substantial commitments to nuclear fuel production and supply, Raymond observed: "Exxon believes that nuclear power is important to the vital U.S. objective of adequate and reliable energy supply. We feel strongly that a firm commitment to the timely use of the highly promising, more energy-efficient centrifuge technology will contribute to this goal. Our investments have been made in support of this firm belief."

Exxon Nuclear has been engaged in centrifuge development since 1972, when it was selected for the DOE Industrial Participation Program intended to permit private industry access to and participation in the uranium enrichment sector. Since that time the Company has spent over $40 million in centrifuge development, engineering and testing, and was among the first privately-funded companies to be qualified by DOE as a centrifuge manufacturer.

DOE's February 1979 Request for Proposals said that either two or three cost-type contracts will be awarded for Phase 1 of a two-phase program. A principal objective of Phase 1 is to develop the necessary supporting industry and establish the capability of two or three suppliers to provide centrifuges for the GCEP on a competitive, fixed-price Phase 2 procurement. Phase 1 will involve the manufacture of 1,000 to 1,500 GCEP-grade centrifuges per vendor during the period 1982 to 1984. Phase 2 is expected to begin in 1984 and to involve the production of tens of thousands of centrifuges. DOE estimates that the total cost of GCEP centrifuges will be in excess of $2 billion (in 1979 dollars) and that the total cost of the completed GCEP will be $5.1 billion (1979 dollars).

The planned Exxon Nuclear manufacturing facility in Oak Ridge is expected to cost about $30 million in the initial phase, with construction beginning early in 1980. It is anticipated the plant would result in over 300 new production, engineering and management jobs by the early 1980s.

While the Oak Ridge plant is under construction, Exxon Nuclear will manufacture the initial complement of centrifuges, destined for use at the DOE Centrifuge Plant Development Facility (CPDF) in Oak Ridge, using the existing Malta manufacturing facilities. During this same interim period, rotor balancing and centrifuge assembly and testing would continue to be performed at Exxon Nuclear's Richland facility. Once initial operation of the Oak Ridge plant has been achieved, probably in the 1982 time frame, Exxon Nuclear's present centrifuge facilities would be phased out and equipment transferred to Oak Ridge. It is expected that virtually all Exxon Nuclear personnel at Malta will transfer to Oak Ridge, whereas most of the small Richland centrifuge staff would be reassigned to other Exxon Nuclear activities in that area.

Exxon Nuclear, a wholly-owned affiliate of Exxon Corporation, is a supplier of fabricated fuel and uranium for nuclear electric power plants in the U.S., Western Europe, and the Far East. The firm is headquartered in Bellevue, Wash., with its main nuclear fuel manufac-

Raymond

turing operations and Research and Technology Center located in Richland. The Company commenced operation of a new nuclear fuel assembly and manufacturing facility in Lingen, West Germany earlier this year, and was recently selected by DOE for negotiation of a prime contract to operate the Chemical Processing Plant at the Idaho National Engineering Laboratory commencing on or about July 1, 1979. The Company is also involved in the development of laser enrichment technology, which is less well developed than centrifuge technology but which has the potential for eventually complementing gaseous diffusion and centrifuge enrichment facilities by enriching the partially depleted residue ("tails") from such facilities.

Background Information

Uranium, as found in nature, contains only about seven-tenths of 1 percent of the fissionable uranium isotope, uranium-235 (U-235). To be useful in most nuclear power plants currently in use throughout the world, this natural uranium must be "enriched" by increasing the concentration of U-235 from 0.7 percent to a level of 2 to 4 percent. Presently this enrichment process is accomplished in the U.S. and elsewhere for the most part by the gaseous diffusion process. Existing U.S. uranium enrichment capacity consists of three large gaseous diffusion plants constructed in the 1940s and early 1950s to support the U.S. national security program. Of roughly equal size, these plants are located at Oak Ridge, Tennessee; Paducah, Kentucky; and Portsmouth, Ohio, and are operated as an integrated complex by prime contractors for the DOE. The original design capacity of the three-plant complex was 17.2 million separative work units (SWUs) per year, a level that requires 6 million kilowatts of electric power. DOE is now engaged in the Cascade Improvement Program and the Cascade Upgrading Program, known as CIP/CUP, to incorporate the most modern technology into the complex. This program will increase the total capacity to 26.3 million SWUs/year at 7.0 million kilowatts by 1981.

The U.S. is the principal supplier of uranium enrichment services to the world nuclear power industry. In 1974, the existing and planned capacity was fully committed and the "order books" were closed. Consequently, in 1976, ERDA (DOE's predecessor) was authorized by Congress to begin a project which would double the Portsmouth plant capacity by adding 8.8 million SWUs per year of new gaseous diffusion capacity. In April of 1977, President Carter announced a decision to modify the project to utilize the newer, more energy-efficient centrifuge process for the Portsmouth expansion with centrifuges to be purchased from private industry.

A gas centrifuge for uranium enrichment consists of a long, hollow cylinder (rotor) spinning at a high speed. Gaseous uranium hexafluoride is fed into the rotor near its center. Centrifugal force causes the heavier uranium-238 molecules to move closer to the wall of the rotor, producing partial separation of the uranium-235 and uranium-238 isotopes. The enriched stream is withdrawn by a scoop near the top of the rotor. To achieve the desired level of throughput in a plant, many centrifuges must be connected in parallel. These parallel arrays are, in turn, connected in series in order to achieve the desired level of enrichment.

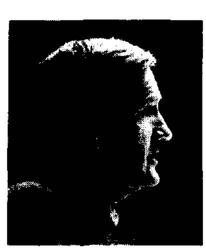
DOE's planned Gas Centrifuge Enrichment Plant (GCEP) at Portsmouth will produce a normal product of 3 percent U-235. Electric power consumption will be approximately 100 megawatts, or about 4 percent of that which would have been required for the gaseous diffusion project originally planned. Major construction activities are slated to commence during 1979. About 5,000 people are expected to be employed at the peak of the construction effort. A staff of over 2,000 will be required to operate the plant.

Agnew Named To National Academy

Harold M. Agnew, who has participated in recent INMM meetings, was named in April 1979 to the National Academy of Sciences.

Election to membership in the Academy is considered one of the highest honors that can be given to an American scientist or engineer. Agnew's election was announced at the Academy's 116th annual meeting April 24. The National Academy of Sciences is a private organization of scientists and engineers dedicated to the furtherance of science and its use for the general welfare. The Academy was established in 1863 by a Congressional Act of Incorporation signed by Abraham Lincoln. According to the Act, the Academy is an official advisor to the federal government, upon request, in any matter of science or technology. This provision accounts for the close ties that have always existed between the Academy and the Government, although the Academy is not a government agency.

Agnew was one of 60 new members elected to the Academy's membership. He was Director of Los Alamos Scientific Laboratory from 1970-1979 and is now president of the General Atomic Corporation, San Diego, Calif.



Agnew

INMM Officers Re-Elected

(Continued from Page 6)

There were many indications received with the ballots this year that there is dissatisfaction with the practice of nominating only one candidate for each of the officer's positions. This same complaint has been stated before and has been the subject of discussion in Executive Committee meetings. Those who feel strongly about this matter are reminded that a petition signed by fifteen (15) members can place additional names on the ballot as stated in Article III, Section 4 of the Bylaws. In addition, the Bylaws can be amended as indicated in Article VII, Section 1. When the original Bylaws were approved in 1958, Article III, Section 4 read that at least two names were to be submitted for each of the elective offices. This section was amended January 1, 1967 to read as it does now.

Institute Continues Growth, Vitality

(Continued from Page 2)

motivate plant operators who are clearly the first line of defense in any in-plant safeguards and security system. He calls for each facility to integrate safeguards instruction and training directly into its regular operator training program, and cites the new INMM safeguards awareness poster (cf. p. 22, this issue) as a useful contribution to an overall education program.

In closing, I'd like to call your attention to two upcoming INMM co-sponsored meetings: (1) the Topical Conference on "Measurement Technology for Safeguards and Materials Control," November 26-29, 1979 at Kiawah Island, S.C. (co-sponsored by INMM, ANS, and NBS), and (2) the "Third International Conference on Nondestructive Evaluation in the Nuclear Industry," February 11-13, 1980 in Salt Lake City, Utah (co-sponsored by INMM, ASM, ASTM and ASNT). Information on both of these timely Conferences can be found in this and subsequent issues of the Journal.

Safeguards Lessons Learned

(Continued from Page 1)

similar safeguards related incidents which should be collected and studied to achieve a better understanding of safeguards effectiveness: not necessarily real safeguards incidents, but incidents when safeguards measures fail or have been compromised. There are a lot of these which do not individually suggest any risk and tend to be ignored. We should take another look.

Like reactor accidents, the consequences of safeguards incidents range from the trivial to a 20-kiloton explosion in the N.Y.C. World Trade Center. Similarly, the probability that an annoyed employee may steal a can of low enriched uranium in order to embarass his employer is much more probable than that some extremely competant, dedicated, and crazy group will successfully steal plutonium oxide, design and construct an explosive, deliver and explode it. At a Congressional hearing a couple of years ago, **Ted Taylor** was asked if he expected the latter to happen. After thinking about it, he said that it was possible that this might happen in the next 20 to 30 years in the U.S.A. While I admit that this is conceivable, as is a thermonuclear war, what is much more likely is that much less destructive events will happen, which don't kill anyone, but could have a devastating impact on nuclear power. Imagine an enviornmental group, determined to shut off nuclear power, which managed to seize a nuclear power plant. Although this outfit had no intention to sabotage the reactor, and probably wouldn't know how, I guess that the NRC and the governor of the State would order evacuations. Another possibility is that some anti-social group might obtain a few grams of plutonium and claim to have enough for a bomb. The few grams would give credibility to the threat. I'll let you contemplate the reaction of Government and public.

At first, I was happy that I was not responsible for reactor safety or radiation measurements, like many of my friends and colleagues. But, then it occured to me that this time I was lucky, that next time it may be a safeguards failure. I don't like nuclear energy or the nuclear arms race. In spite of my emotional reactions, I feel that my children and the rest of the world will need nuclear energy. What are the lessons that INMM members should learn from the T.M.I. incident? Do comment.

IRT To Manufacture Portable Systems for Assaying Large Plutonium Samples

IRT's Nuclear Systems Division has been selected by the Los Alamos Scientific Laboratory (LASL) to manufacture a portable High-Level Neutron Coincidence Counter (HLNCC), developed at LASL, for the assay of plutonium by inspectors of the International Atomic Energy Agency (IAEA).

The HLNCC measures the effective ^{240}Pu mass in plutonium samples which in many cases may have a high plutonium content. (The term "high-level" refers to the neutron count rates produced by large, several kg, PuO_2 or plutonium metal samples.) In addition, it has the versatility to assay a large variety of plutonium sample types including oxide, mixed oxide, carbide, metal, fuel rods, fast critical assembly plates, solutions, scrap and waste.

The HLNCC performs its assay by detecting coincident fission neutrons from the plutonium in the presence of a random neutron background originating principally from (a,n) reactions in the material. The fission neutrons are primarily due to the spontaneous fission of the even-mass plutonium isotopes (^{238}Pu, ^{240}Pu, and ^{242}Pu) and to multiplication of (alpha,n) or spontaneous fission neutrons. The effective ^{240}Pu content of a sample is the mass of ^{240}Pu which would give the same corrected response to the measurement system as the actual ^{238}Pu, ^{240}Pu and ^{242}Pu content of the sample. After corrections are made for dead time, multiplication, matrix, and geometric effects, the HLNCC assay determines the effective ^{240}Pu content of the sample. The total plutonium content is then calculated from the plutonium isotopic composition, which is either assumed known or is estimated from gamma-ray measurements made with a germanium detector.

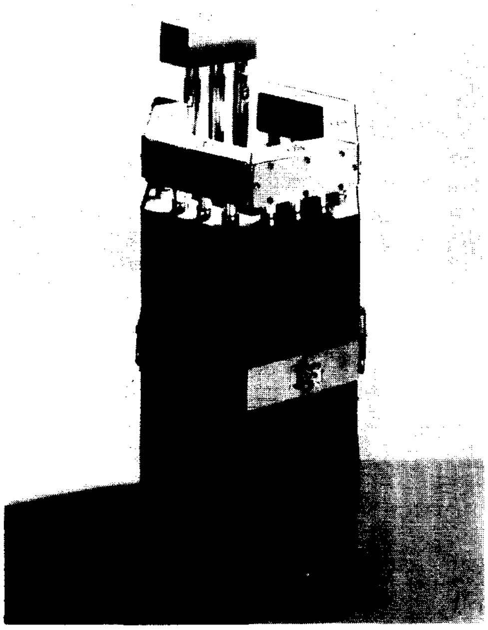


Fig. 2. The electronics portion of the HLNCC shown with a programmable calculator for on-line determination of the effective ^{240}Pu content of plutonium samples.

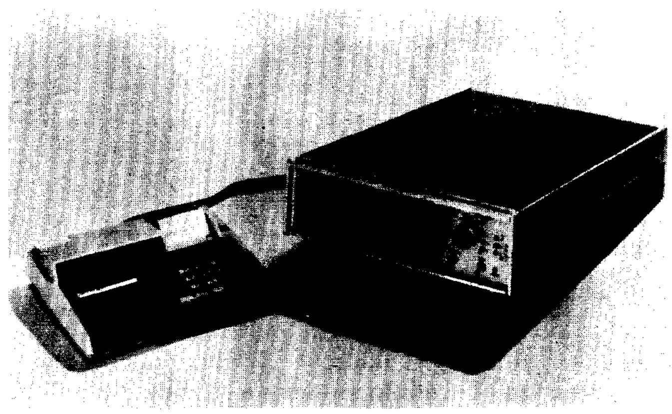


Fig. 1. The detector portion of the HLNCC, which is configurated to accept most PuO_2 sample containers. It can be further separated to accept larger containers, or two of the six sides can be used in a sandwich configuration to measure small samples of fuel rods.

Since the HLNCC was planned for field use, it has been configured for minimum size and weight. To provide the flexibility for accommodating a wide variety of sample containers, the detector portion is fabricated as six separate slabs that form a hexagonal well. The width of the well (18 cm minimum) accepts most PuO_2 sample cans, fast critical assembly fuel drawers, and some fuel rod assemblies. The slabs can be further separated to accept larger containers or, alternatively, two slabs can be used in a sandwich configuration to measure small samples or fuel rods.

The use of shift-register coincidence counting logic and six parallel analog signal processing channels permits neutron counting rates greater than 150 kHz with only moderate dead-time corrections. Data can be transferred automatically from the electronics package to a programmable calculator for on-line analysis. A standard RS-232 serial data communications output port is also provided for expedient interface to any computer.

IRT's contract with Los Alamos calls for the manufacture and delivery of seven complete HLNCC systems and five additional control units for existing well counters. IRT has another contract to manufacture five HLNCC control units which will be used to upgrade the performance of existing neutron well counters at a plutonium processing facility. All systems will be constructed to LASL-supplied manufacturing drawings.

ABOUT THE AUTHORS

Bishop

Cobb

Ensslin

Filss

Dennis M. Bishop (B.S., Metallurgical Engineering, California State Polytechnic University; MBA, Business, University of Santa Clara) is a senior engineer with the General Electric Company, San Jose, California. He has over 10 years experience in the nuclear energy field primarily in nuclear fuel cycle manufacturing process development, irradiation testing, and the development and application of safeguards systems and measurement techniques for both uranium and plutonium fuels. Mr. Bishop has been an active member of the INMM since 1971. He served on ANSI-INMM Standards Writing Group N15.8 (Calibration) which developed ANSI-INMM Standard N15.20. Since 1974 he has been chairman of ANSI-INMM Standards writing Group N15.9 (Non-Destructive Assay) which includes responsibility for six writing groups. He also serves on the 1975 Ad Hoc Writing Group to the INMM Safeguards Committee which developed the INMM publication "Assessment of Domestic Safeguards for Low-Enriched Uranium." Mr. Bishop has over 30 open literature publications in the areas of plutonium and uranium fuel measurement methods and safeguards, process development and irradiation performance.

Donald D. Cobb (Ph.D., Nuclear Physics, Iowa, 1970) is a staff member in the Safeguards Systems Studies Group, an interdisciplinary group at Los Alamos Scientific Laboratory engaged in the design, development, and evaluation of advanced materials control and accounting systems. He has participated in the design and analysis of measurement and accounting systems for a variety of nuclear research and production facilities. He is currently serving as a member of the ANSI INMM-5 (Measurement Control) Standards Committee.

Norbert Ensslin (Ph.D., Physics, MIT, 1972) is a member of the Nuclear Safeguards Research Group at the Los Alamos Scientific Laboratory. He is involved in the development of fast and thermal neutron coincidence counters. He is interested in the count rate, self-multiplication, and self-shielding problems associated with the nondestructive assay of large uranium and plutonium samples.

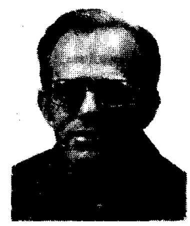
H. Peter Filss (Dipl. Phys., Neutron Physics, Dr. Phil. nat., Physical Chemistry University of Frankfurt, Germany, F.R., 1967) has worked since 1970 at the Nuclear Research Center (KFA) Juelich, Germany, F.R. in the field of Chemical Technology. He has been mainly involved in reprocessing of thorium based fuel, off-gas purification, and non-destructive measurement of irradiated fuel. At present time, he is delegated to the German INFCE Group at the Ministry of Research and Technology in Bonn, West Germany.

Hans E. Hansen (B.S., Washington State University, Pullman, 1952) is Supervisor of facility systems development in the Safeguards Facilities Protection Department at Sandia Laboratories, Albuquerque, New Mexico. Prior to 1976, he was Supervisor of a Sandia division responsible for high altitude test readiness involving high altitude sonic rocket experiments.

Jarsch

Hansen

Onnen



Sapir

Shipley

Stewart

Volker Jarsch (M.S., Computer Science, University of Karlsruhe, 1975) has been a staff member of the Institute of Data Processing at Karlsruhe Nuclear Research Center, West Germany since 1975. In the past two years, Mr. Jarsch has been involved in the development of the implementation of the nuclear materials control system.

Sanda Onnen (Dr. Ing., Technical University of Karlsruhe, 1972) is associated with the Institute of Data Processing at the Karlsruhe Nuclear Research Center in F.R. Germany. Her article, "MACSY—A Minicomputer Accounting and Control System," appears in this issue of the Journal. She is a co-author of a paper, "An Approach to a Generalized Real-Time Nuclear Materials Control System," given at the 1978 INMM Annual Meeting held in Cincinnati, Ohio.

Joe Sapir (M.S., Nuclear Engineering, Massachusetts Institute of Technology, 1959) is currently working as an inspector for the International Atomic Energy Agency, Vienna, Austria. He is on leave-of-absence from the Los Alamos Scientific Laboratory which he joined in 1962. His specialties are critical assemblies, Monte Carlo simulation of reactor physics problems, and nuclear safeguards measurements.

James P. Shipley (Ph.D., Electrical Engineering, University of New Mexico, 1973) is Alternate Group Leader of the Safeguards Sytems Studies Group at Los Alamos Scientific Laboratory. He is active in the systems aspects of nuclear safeguards, particularly advanced materials accounting techniques for both national and international safeguards.

James E. Stewart (Ph.D. Nuclear Engineering, University of Virginia, 1974) is a member of the Nuclear Safeguards Research and Development Group at the Los Alamos Scientific Laboratory. He has extensive experience and several publications in neutral particle transport computational methods. Currently, he is interested in transport problems of Nuclear Safeguards.

Fred H. Tingey (Ph.D., University of Washington, Seattle, 1951, Mathematical Statistics) is Assistant to the Manager at EG&G, Inc., Idaho Falls, Idaho. In this capacity, Dr. Tingey is in charge of the safeguards program and educational activities for the Company. He has been involved in the Institute for several years.