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INMM returns to Orlando in 1992

I'm sure that you know by now that the INMM is returning to the Stouffer Orlando Resort Hotel in Orlando, Fla., July 19-22 for the 1992 Annual Meeting (our 33rd) — and it looks like it may be our biggest ever. With the outstanding help of our brand-new technical divisions, the Technical Program Committee has selected more than 220 papers for presentation in 33 sessions.

The program is comprehensive and diverse, reflecting all of the areas in which the INMM has a demonstrated interest: safeguards and security, of course, including physical protection and MC&A; international safeguards, including containment/surveillance and NDA measurement technology; waste management; transportation and packaging; the environment and ES&H; arms control and verification; and several sessions on nuclear non-proliferation. In addition to the technical program, the many side meetings that usually are scheduled around the Annual Meeting already are being arranged.

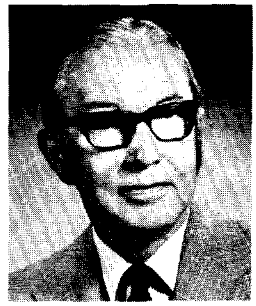
On the lighter side, Orlando is the home of many family-type theme attractions: Walt Disney World, Sea World, Disney/MGM Studios, Universal Studios, Cypress Gardens, Wet 'n Wild, the John F. Kennedy Space Center and many others. There's also downtown Orlando's Church Street Station, a nostalgic, one-of-a-kind entertainment complex, where you can sample unique shopping along the quaint walking mall; partake of an elegant three-course dinner at Lili Marlene's or enjoy simple fare at a variety of places; and enjoy Dixieland Jazz and old-time songs in Rosie O'Grady's Good Time Emporium.

Because there are so many excellent entertainment opportunities, we haven't planned a Monday evening event; however, we can help you make arrangements to visit one of the many spots available.

Most of the members I have talked to recently are planning to be in Orlando — I hope you are too. See you there.

Darryl B. Smith
Los Alamos National Laboratory
Los Alamos, New Mexico, U.S.A.

Safeguards and spent fuel



This issue contains six technical articles. Three were presented at the ninth annual INMM Seminar on Spent Fuel Management in January and three were presented at the Workshop on Near-Real-Time Accounting for Reprocessing Plants which was held at Los Alamos National Laboratory, Los Alamos, N.M., last fall. All of the papers were selected for publication because they should be of interest to others than just those concerned specifically with the management of spent fuel or safeguards for reprocessing plants.

Spent fuel must be stored and reprocessed or permanently disposed. National and international safeguards should be effectively applied to the material until it becomes consumed or diluted in such a way as to have no further safeguards significance. The radioactive spent fuel should be protected and the storage, transportation, processing and ultimate use or disposal activities should be performed safely and with minimum exposure of personnel. The three spent fuel articles discuss the subject of "burnup credit" with regard to transportation of the spent fuel. The design and loading of spent fuel shipping casks must ensure that no conceivable accident might result in a criticality which might disperse the radioactivity.

In most countries, the spent fuel which might be shipped has been removed from the reactor for five to ten years, so that the radioactivity has decayed significantly. Also, most of this fuel was irradiated for several years so that the reactivity is significantly less than was that of the fresh fuel before irradiation. In the United States and elsewhere, those designing spent fuel shipment casks would like to take advantage of the fact that the fuel to be shipped is less radioactive than when it was just discharged from a reactor and of the fact that the reactivity of the spent fuel is considerably less than that of fresh fuel. The three articles discuss the program which the U.S. Department of Energy is supporting to design shipping casks which take advantage of the reduced reactivity and radiation and to ensure that the design and the procedures to be used will be entirely safe.

One possibility, discussed here, is to make non-destructive measurements of the spent fuel assemblies before they are placed into a shipping cask so as to ensure that no errors in calculations or procedures could possibly permit a serious accident. It should be of interest to the readers of this Journal that the instrument is one that has been developed and used by the IAEA to verify that spent fuel assemblies have not been tampered with. This also suggests that there may be safety as well as safeguards reasons to make spent fuel measurements.

It should be noted that these articles refer to pressurized water reactor (PWR) spent fuel assemblies, and not to boiling water reactor (BWR) assemblies. The reason is that fresh PWR assemblies contain no poisons and have higher reactivity than the spent fuel assemblies after irradiation; while fresh BWR assemblies contain neutron poisons and the reactivity is but little changed with burnup.

The three papers from the near-real-time, reprocessing meeting should be of interest to anyone who is involved in accounting for nuclear materials. Whether or not near-real-time accounting is involved, it is necessary at any nuclear material processing facility to collect the accounting data for each material balance area and to calculate the material balances repeatedly at certain intervals. Modelling the measurement systems and automating the material balance calculations should be useful for any such activities. The authors of these papers have documented their programs and offer to make them available to other potential users. Such international cooperation is what makes safeguards R&D so valuable.

I would be happy to hear from any of you who have suggestions for topics which should be discussed in the future. Suggestions for authors would be even more appreciated.

Dr. William A. Higinbotham
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Upton, New York, U.S.A.

Progress in Spent Fuel Storage and Disposal

This issue of the *JNMM* includes several papers that were presented at the 9th INMM Spent Fuel Management Seminar that was held Jan. 15-17, 1992, in Washington, D.C.

It is becoming evident that this growing series of annual meetings is systematically documenting both the technology developments and the parallel evolution of the political process toward the ultimate common goal of waste disposal. The increments of political progress between annual meetings are also beginning to quantify the long-time constant of the political process.

This year, more than prior years, served to highlight that overall progress in nuclear waste disposal must be measured at least as much in political terms as in technical terms. This meeting was notable in that there was, for the first time in many years, direct evidence of tangible political progress. As a result, there was a definite tone of optimism that the new experimental political process crafted by Congress in 1987 might just be working — specifically the basic concept of the independent, presidentially-appointed Nuclear Waste Negotiator.

Much credit is due to the first Negotiator, David Leroy, because of his discerning implementation of that new office, including his clear articulation of what the voluntary siting process is, and the related activities that must, and must not, be pursued. One key to success using voluntary siting is to achieve mutual agreement on the conditions by which each of the effected parties will have a beneficial stake in the success of the project. Leroy, who was the luncheon speaker both last year and this year, informed the seminar attendees that his talk last year had been the first public speech he had made as the Waste Negotiator. He naturally took this fact as his starting point to reiterate, now with additional conviction, the basic principles under which he was, and is proceeding. By way of balance, he also noted an important adjunct of the voluntary siting process, that is worth repeating, verbatim: "I have steadfastly refused to make guarantees or commit to artificial time lines, not because of a lack of confidence or an expectation of a lengthy pursuit, but rather, in this process, artificial time lines and bold guarantees are inconsistent with the notion of a voluntary host-driven solution.

It is counter-productive and even somewhat arrogant to suggest that as this nation reaches out to seek the consultation and help of its states and Indian tribes, the national government or industry should predetermine a schedule." This reminds us that although DOE can, and must, have schedules and milestones for its own work, some of the major milestones upon which overall progress depends, are beyond DOE's reasonable control.

The real evidence of actual progress, however, was the presence and presentation of Fred Peso of the Mescalero Apache Tribe. In response to the Negotiator's initiative, the Mescaleros were the recipient of the first Phase 1 MRS Feasibility Study Grant. The dignity and direct, simple eloquence of Peso, and his conveyance of the profound importance of protecting the land and its people, provided the audience with both a factual and an emotional level of understanding. This is the reason for the Mescalero's strong preference for handling only clean, sealed containers, and no individual fuel assemblies at the MRS. To the extent that this is important to all candidate sites, a considerable change may be required in the MRS, as currently conceived, with additional potential impacts on the utilities.

In parallel with the progress and optimism on the political side, there was progress and optimism on the technical side. Dr. John Bartlett, director of OCRWM, recalling last year's appeal from Steven Kraft of EEI/UWASTE, to "Get on Yucca Mountain," began his talk by noting that "We are on Yucca Mountain." Bartlett reviewed the final dismissal of each of the three State of Nevada procedural lawsuits against DOE, the Supreme Court's denial of a hearing of Nevada's appeal of its unsuccessful challenge of the Nuclear Waste Policy Act, and the district court order, on May 13, 1991, that Nevada act on the three DOE permit applications necessary to proceed with Yucca Mountain site characterization. Nevada granted the air quality and underground injection control permits in mid-1991, and the water allocation permit, not granted as of the date of the seminar, was granted shortly thereafter. Bartlett also noted NRC's official approval of DOE's program-wide QA program. As a result of all of these events, the DOE characterization of the Yucca Mountain site is now well underway.

The talk given by Mary Louise Wagner of the Senate Energy Committee was particularly significant in the light of what she had noted in the Seminar two years ago. At that time she had given an excellent and provocative assessment of the OCRWM program, and among other things, had noted that "Coping with the political problem in Nevada — and the day-after-day rhetoric of Nevada politicians — will mean that DOE will have to get tough. Hard ball will be the name of the game if the Department is going to succeed with the program." This year she noted that Congress now recognizes both the importance of the project, and that DOE can mount an effective program. She observed that Bartlett has increased the technical credibility of DOE. She also applauds the industry's independent campaign in Nevada, to counter, in a responsible and factual way, the misinformation being put out by the opposition. All of these observations contribute to the evidence that the DOE program is now effectively led and has the support of both Congress and the industry.

With respect to spent fuel storage technology, the technology of dry storage continues its innovation and expansion. Seminar participants received updates on metal and concrete casks, vault storage, transportable storage casks and fuel consolidation. The new 40-PWR metal cask being licensed for Prairie Island is in licensing review which is expected to be completed well before the first of these casks have to be loaded in mid-1933. The Topical Report for the vertical concrete Ventilated Storage Cask for Palisades has been accepted by NRC, and a special exemption for the construction and loading of eight casks in mid-1992 has been given. This cask will also be used for storage at Point Beach. Two other concrete casks, the CONSTAR and GNSI-HDC, were described, and an update was provided on the continuing successful use of the horizontal concrete storage system, NUHOMS®, at the Oconee station, and its upcoming use at Calvert Cliffs. With respect to storage vault technology, the successful construction, completion and initial loading of Fort St. Vrain Modular Vault Dry Storage system for HTGR fuel was summarized.

Two other vault systems, the FUELSTOR and MACSTOR vaults, were described. The progress and complications of licensing the Storage-Transport Cask in parallel under Parts 71 and 72 were summarized. Participants were also shown impressive videos of the analysis and drop-testing of ductile cast iron casks in Japan.

Progress continues to be made with consolidation technology but is being hampered by the recent lack of opportunities for large-scale hot demonstrations of the most recent developments. In addition, there were several sessions on the MRS and its candidate technologies, including two papers dealing with dry casks transfer.

A special session of several presentations on the licensing of transport casks for burnup credit was held. Several interesting papers on a variety of special interest topics, including two papers on the process of accepting waste into the DOE system were presented. By way of overview, spent fuel storage technology continues to develop such that there are now, typically, several viable design alternatives available to match the individual circumstances at most sites. This provides the best evidence that, with proper planning, no utility is likely to encounter limitations on reactor operations arising from the lack of a suitable and viable spent fuel storage technology.

N. Barrie McLeod
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Standards: ANSI N14 Committee

The annual N14 meeting was held Nov. 20, 1991, at the U.S. Department of Transportation in Washington, D.C. Minutes of the meeting were mailed to individuals on the N14 roster.

The Ad Hoc Committee chaired by J.E. Stiegler, Sandia National Laboratories, relating to the N14 Scope Change has completed its assignment and has been disbanded. In its place a new committee is being established to review scopes of proposed hazardous materials for their merit, urgency and potential impact. The committee will be chaired by Robert E. Luna, and its members will be James R. Clark, Daniel Fisher, William H. Rucker and Michael E. Wangler. The committee will be effective the date the N14 Scope Change is approved.

Highlights of N14 standards development are:

- ANSI N14.2 — *Tiedowns for Transport of Fissile and Radioactive Container Greater Than One-Ton Truck Transport* (in process). The first draft has been completed. The need for examples and additional test information is currently being addressed. The examples include a flexible tiedown system similar to those of the NuPac 7D-3.0, an ISO corner tiedown, and a cradle and blocking-type tiedown. A firm date for a meeting of the writing group has not been set.
- ANSI N14.6 — *Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500kg) or More for Nuclear Material.* This standard is in the process of being revised or reaffirmed. George Townes, BE Inc., and chair of N14.6, is obtaining N14.6 Writing Group consensus.
- ANSI N14.7 — *Guide to the Design and Use of Shipping Packages for Type A Quantities of Radioactive Materials* (in process). Richard Rawl has agreed to take over this project. A high priority was suggested for this standard at the Annual Meeting.
- ANSI N14.10 — *Guide for Liability and Property Insurance Aspects in Shipping Nuclear Material.* Robert E. Luna has suggested the following title and scope to replace the existing title scope for N14.10:

  Title: Guide for Risk Management in Shipping Nuclear (Radioactive?) Materials

  Objective: This guide will discuss methods for assessing radiological and financial risks from the transport of nuclear (radioactive?) materials, risk management techniques, available coverages in the private sector, required coverage under state and federal law, nuclear liability exclusions and Price-Anderson coverage.

  This title scope change is currently being considered by the N14 Management Committee.
- ANSI N14.19 — *Ancillary Features of Irradiated Shipping Casks.* This standard will be revised or reaffirmed by the end of 1992. A ballot was sent to N14 members on Dec. 20, 1991, with a closing date of March 2, 1992.
- ANSI N14.23 — *Design Basics for Resistance to Shock and Vibration of Radioactive Material Packages Greater than One-Ton in Truck Transport.* Work on this draft is continuing within the Writing Group. A technical report which will be the basis for the draft standard has been completed.
- ANSI N14.24 — *Barge Transport of Radioactive Materials.* A new chair of the Writing Group is still being sought. This standard will be revised or reaffirmed by the end of 1992. A ballot was sent to N14 members on Dec. 20, 1991, with a closing date of March 2, 1992.
- ANSI N14.26 — *Guidance on Quality Control Activities as They Relate to the Inspection, Preventive Maintenance and Post-Incident Testing of Packages Used for the Shipment of Radioactive Material.* Work is continuing on preparation of a draft document.
- ANSI N14.27 — *Carrier and Shipper Responsibilities and Emergency Response Procedures for Highway Transportation Accidents Involving Truckload Quantities of Radioactive Material.* This standard will be revised or reaffirmed by the end of 1992. A ballot was sent to N14 members on Dec. 20, 1991, with a closing date of March 2, 1992.
- ANSI N14.30 — *Design, Fabrication and Maintenance of Semi-Trailers Employed in the Highway Transport of Weight-Concentrated Radioactive Loads.* This draft is in the process of being submitted to ANSI for approval.

John W. Arendt, Chairman
Oak Ridge Associated Universities
Oak Ridge, Tennessee, U.S.A.

Chapters: Pacific Northwest

Due to the extended off-site assignment of the Chapter Chairman, the election for 1991-92 officers was significantly delayed. The Chapter ultimately determined that the 1990-91 officers would continue to function until the end of the current fiscal year. At that time, elections will be held for the regular 1992-93 year.

The following officers were held over for 1991-92:

Chairman	Brian W. Smith
Vice-Chairman	Debbie A. Dickman
Secretary-Treasurer	Rich A. Hamilton
Executive Board	Ken R. Byers
	Jim Edgar

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Divisions: International Safeguards & Non-proliferation

On Nov. 22, 1991, the Institute of Nuclear Materials Management's (INMM) Division of International Safeguards and Non-proliferation (IS&NP) met at the IAEA headquarters, Vienna, Austria. The participants in the meeting included: Cecil Sonnier, Sandia National Laboratories, U.S.A., Chairman; Paul Ek, SKI, Sweden, Vice Chairman; Bjorn Dufva, ASO, Australia; Harold Stocker, AECB, Canada; Paul Gourlez, CEA, France; Andre Petit, Consultant, France; Bernd Richter, KFA Juelich, Germany; Yasuhiro Yokota, NMCC, Japan; Anita Nilsson, SKI, Sweden; Jon Jennekens, IAEA, Vienna, Austria; Raymond Parsick, IAEA, Vienna, Austria; Abdul Fattah, IAEA, Vienna, Austria; Dirk Schriefer, IAEA, Vienna, Austria; Robert Thiele, IAEA, Vienna, Austria; Norbert Jousten, Counsellor, CEC, Vienna Office.

Sonnier opened the meeting with a description of the changes in the INMM structure that were made at the November 1991 INMM Executive Committee Meeting. Basically, the INMM was restructured into six Divisions:

- Physical Protection
- Material Control & Accounting
- Waste Management
- Transportation Division
- International Safeguards & Non-proliferation (IS&NP)
- Arms Control Verification

The IS&NP Division replaces the original International Safeguards Subcommittee (ISSC) which was a part of the INMM Safeguards Committee. The meeting participants expressed pleasure with the new status of the IS&NP Division. The Charter of the IS&NP Division will be the same as that developed by the ISSC, with the addition of "Non-proliferation" where appropriate in the Charter. The officials of the IS&NP Division remained the same. Sonnier stated that at the next meeting, a secretary would be elected.

The 1992 INMM Annual Meeting will be held in Orlando, Fla., USA, July 19-22. The next meeting of the IS&NP Division will be on Sunday, July 19, 1300-1700 hours. Sonnier stated that, in the future, the IS&NP Division meeting may be expanded to a full day, if the members agree. This subject will be discussed further in the July 19 meeting.

One of the principal objectives of the Nov. 22, 1991 meeting was to consider the structure of the six (potentially eight, given acceptance of concurrent sessions) International Safeguards Sessions of the Annual Meeting. This subject was discussed, and it was proposed that a one-half hour panel be held to discuss the papers in the first two sessions covering general International Safeguards and Non-proliferation topics. As a result, the Division proposed the following program in its areas of responsibility:

INTERNATIONAL SAFEGUARDS I
Monday AM – Four to five papers on International Safeguards and Non-proliferation topics and issues

INTERNATIONAL SAFEGUARDS II
Monday PM – Five papers on International Safeguards and Non-proliferation topics and issues, followed by a Panel Discussion on the previous papers and/or other topics and issues

INTERNATIONAL SAFEGUARDS III
Tuesday AM – Eight to nine papers on Containment and Surveillance (C/S) Technology

INTERNATIONAL SAFEGUARDS IV
Tuesday PM – Four to five papers on C/S Technology

INTERNATIONAL SAFEGUARDS V
Wednesday AM – Eight or nine papers on Measurement Technology

INTERNATIONAL SAFEGUARDS VI
Wednesday PM – Eight or nine papers on Integrated Safeguards Systems

Sonnier added that, as has been the practice for the past seven years, the Informal C/S Meeting will be held on July 23, 1992, 1300-1700 hours, followed by a social 1800-1900 hours.

There was considerable discussion on the new name of the Division regarding the combination of International Safeguards and Non-proliferation in the same title. It was generally recognized that Non-proliferation is a much broader subject than International Safeguards, and that International Safeguards is only a part of Non-proliferation.

The IAEA expressed interest in participating in the IS&NP, and stated that all practical efforts would be made to prepare papers of interest, and provide attendance at the 1992 Annual Meeting.

A number of other participants indicated that they and their colleagues would be very interested in the IS&NP Division activities, and certainly would give serious thought to preparing papers for the Annual Meeting. Sonnier requested that he or Paul Ek be contacted by Jan. 17, 1992, regarding the titles, abstracts and authors for proposed papers for the IS&NP sessions.

Cecil S. Sonnier
Chairman
INMM International Safeguards and Non-proliferation Division
Sandia National Laboratories
Albuquerque, New Mexico, U.S.A.

Divisions: Waste Management

The Division successfully organized and held the INMM Spent Fuel Management Seminar VIX at Loew's L'Enfant Plaza Hotel in Washington D.C. on Jan. 15-17, 1992. There were approximately 170 persons in attendance, our largest turnout to date. Included among the participants were persons from five countries (United States, Spain, Germany, Japan and France), representatives of nine utilities and utility organizations and three members of the press. Wall Street Journal Television covered the initial session of the seminar which heard from DOE and contractor management, EEI, the staff of the Senate Committee on Energy and Natural Resources, and the Mescalero Apache tribe. David LeRoy, the U.S. Nuclear Waste Negotiator, was the luncheon speaker. A total of 31 papers were presented.

The Division is continuing to provide INMM co-sponsor representation on the Steering Committee for the 1992 International High-Level Radioactive Waste Management Conference, to be held in Las Vegas, Nev., in April 1992.

The Division is currently organizing the Waste Management Sessions and confirming speakers for the 1992 INMM Annual Meeting to be held in Orlando, Fla., July 19-22, 1992.

The Division is developing a proposal to produce a monograph on spent fuel storage technology. This will be presented to the Executive Committee prior to the Annual INMM Meeting in Orlando.

E.R. Johnson
Chairman
INMM Waste Management Division
E.R. Johnson Associates
Fairfax, Virginia, U.S.A.

Committees: Constitution and Bylaws

After long and arduous discussions, the need to provide the technical working groups with a more permanent title, representation and visibility has been provided by elevating them to Technical Division status. These groups have been a major source of our success and accomplishment.

The resolution in the minutes of the last Executive Committee meeting is, of course, quite proper as far as it goes, elevating the present groups to Divisions and generally outlining the intent of the Executive Committee. This is a major change in our structure. We are formally elevating TWGs to Division status, creating groups with authority to create their own boards and to operate as a separate — albeit a subsidiary — unit within the INMM. Because it is such a change and will affect many members of the Institute, I believe an Amendment to the INMM Bylaws is indicated.

We are creating a new level above our committees, and those committees are covered in detail in the Bylaws. In addition, there is no provision in the resolution for the term of the Chairman, or for changes in the general structure and future number of the Divisions. I would also like to suggest that the status of these new divisions will be such that their Chairmen should report directly to the Executive Committee and not through an Executive Committee member oversight.

Therefore, I propose that the Bylaws be amended to include the following new Articles:

1. The Executive Committee may, at its discretion, create and establish Technical Divisions composed of technical working groups dedicated to specific major disciplines and activities of the Institute as defined by the Executive Committee.

2. Each Division may structure its own membership and, at its own discretion, create a board of directors for its administration. Actions and activities of any division requiring Institute approval shall be presented to the Executive Committee by the Division Chairman for consideration and action.

Roy Cardwell
Chairman
INMM Constitution and Bylaws Committee
Lenoir City, Tennessee, U.S.A.

Pacific Northwest *continued*

John H. Ellis
Past Chairman Don E. Six

The Chapter will continue to be represented on the Tri-City Technical Council by Curtis A. Colvin.

The spring meeting will be held on April 9, 1992, at the West Richland Golf Course. George Westsik, Westinghouse Hanford Company, will give a presentation on the status of the Westinghouse Plutonium Finishing Plant activities relative to Order compliance and facility startup.

Other future meetings will include presentations by Battelle, Pacific Northwest Laboratory staff on the development of the Internal Review and Assessment Program at the Savannah River Site, and the training matrix and career progression activities at the Central Training Academy.

Debbie A. Dickman
Vice Chairman
INMM Pacific Northwest Chapter
Pacific Northwest Laboratory
Richland, Washington, U.S.A.

Committees: Long Range Planning

Charter

Responsible for maintaining a framework for developing future actions of the INMM consistent with the Institute's goal to encourage efficient management and safeguards of nuclear materials.

Activities

Provide long-range planning advice and guidance to the officers and the Executive Committee. Report at each Executive Committee meeting.

Review of old long range planning files shows interest in increasing membership, broadening the scope of membership to include all aspects of nuclear materials management, conducting workshops, training programs, publishing monographs and maintaining an adequate financial position. Current objectives stated in the last report (May 1990) are to raise the level of awareness and recognition of all areas currently embodied in the INMM; to help promote a stable financial position for the INMM to assure its continued viability; and to establish additional fields of commitment that are of interest and within the expertise of the membership such as environmental restoration, nuclear safety and arms control verification.

Issues

World changes leading to increased interest in and importance of non-proliferation of nuclear weapons.

INMM is a strong player in international safeguards. How can we build on this role and provide a forum and meeting ground for the wider non-proliferation community?

Involve the export control community. Is there a forum to discuss export control policies and technologies today?

Ask the same questions regarding the intelligence community. Is it possible to discuss the role of Intel in international safeguards and non-proliferation in an open forum?

Invite papers from the remote monitoring technology development community.

Generally, try to broaden the international safeguards aspects of the annual meeting, the publications in the *JNMM*, and the workshops to include non-proliferation topics.

Action: International Safeguards and Arms Control Divisions and the technical program committee.

Massive changes in the nuclear weapons production and deployment postures of the United States and the former Soviet Union.

U.S. domestic safeguards will be impacted by the down-sizing and streamlining of the weapons complex. INMM can play its traditional role in providing a forum for the exchange of information among the U.S. players.

What can the INMM do to foster domestic safeguards with the new Confederation of Independent States (CIS)? Can we promote members from the CIS republics in addition to the members who come via the IAEA? Are there contacts from the government (OACN) or outside (NRDC) who could help foster these intentions?

Action: Members remain in contact with DOE/OACN, NRC, and each other to monitor activities involving the former Soviet Union (FSU). Request DOE funding for FSU attendees to the annual meeting.

Arms control will continue to be important. The arms control importance of world-wide control of fissile materials may grow. The boundaries between arms control and non-proliferation may become blurred as the Commonwealth of Independent States republics evolve.

INMM should continue to be involved in arms control issues. We must ensure coordination with the non-proliferation community. Actions similar to those above concerning OACN are needed.

Nuclear environmental, safety, health protection, monitoring and cleanup will grow in world-wide importance. This may be especially true in the former East Block and the former Soviet Union.

INMM should continue to play a role in this area, but keep the focus on nuclear materials.

How do we solidify a role for INMM in ES&H? How do we make it international? Do we need an ad hoc working group?

Transportation and Waste Management: Are there issues here involving the domestic and international communities in which the INMM can play a role?

Re-affirm the mission of the INMM. To promote communications, professional development and the exchange of technologies among the world-wide community responsible for the management of nuclear materials, including non-proliferation, safeguards, security and protecting the environment and public health. Advance nuclear materials management in all its aspects; promote related research; establish standards consistent with professional norms; improve qualifications through high standards, education and recognition of those who meet such standards; and disseminate information through meetings, professional contacts and publications.

INMM Management Issues

Membership; Journal subsidy; Size and length of the annual meeting; Selectivity of the technical program committee; Management support services contract; Leadership pool (will organizations outside the national labs provide the support for their personnel to be active leaders in the INMM?); and Workshops.

Jim Tape
Chairman
INMM Long Range Planning Committee
Los Alamos National Laboratory
Los Alamos, New Mexico, U.S.A.

Strategies For Certifying A Burnup Credit Cask

William H. Lake
U.S. Department of Energy
Washington, D.C., U.S.A.

Jack R. Boshoven
General Atomics
San Diego, California, U.S.A.

Lawrence A. Hassler
Babcock & Wilcox
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ABSTRACT

A new generation of high capacity spent fuel transport casks is being developed by the U.S. Department of Energy (DOE) as part of the Federal Waste Management System (FWMS). Burnup credit, which recognizes the reduced reactivity of spent fuel is being used for these casks. Both cask designs being developed for DOE by Babcock & Wilcox and General Atomics use burnup credit. The cask designs must be certified by the Nuclear Regulatory commission (NRC) if they are to be used in the FWMS. Certification of these casks by the NRC would not require any change in transport regulations, and would be consistent with past practices. To support certification, DOE has identified the technical issues related to burnup credit, and embarked on a development program to resolve them. Following a background discussion of criticality safety for spent fuel transport, an approach to design and use of a burnup credit cask is presented. It is concluded that an adequate technical basis is being developed for spent fuel casks to demonstrate compliance with the NRC criticality safety requirements.

INTRODUCTION

The Department of Energy's (DOE's) Office of Civilian Radioactive Waste Management (OCRWM) is in the process of developing a new generation of high capacity casks to transport spent fuel from commercial nuclear reactor facilities to federal waste facilities. The DOE's role in the Federal Waste Management System (FWMS) is defined in the 1982 Nuclear Waste Policy Act (NWPA) and its 1987 amendment (NWPAA). The NWPAA requires DOE to use spent fuel and high level radioactive waste casks certified by the Nuclear Regulatory Commission (NRC). Because of the high shipping rates anticipated, and since cooling times of spent fuel to be shipped significantly exceeds the design cooling times of existing casks, a decision was made to develop new higher capacity casks. The potential benefit of higher cask capacities, is fewer shipments. Fewer shipments result in health and safety benefits as well as cost benefits. In evaluating the needs of the cask development program a number of technical issues were identified that would further support improved cask capacities. Burnup credit is one of these technical issues.[1]

Burnup credit is the practice of accounting for the reduced reactivity of spent fuel in evaluating criticality safety. The NRC transportation regulations[2] (10 CFR 71) require subcriticality of transport systems. The regulations do not elaborate on how subcriticality should be assured, nor do they prohibit the use of burnup credit for criticality safety. The NRC has, in the past, approved one cask which uses burnup credit. It is the Model NLI-6502[3] (NRC certificate of compliance No. 9103) which is used to ship highly enriched research reactor fuel. However, in the case of commercial light water reactor (LWR) spent fuel, the NRC has established a long standing precedent of assuming that fuel is unburned or fresh (i.e., the fresh fuel assumption) for the purpose of evaluating criticality safety.

Since burnup credit has not been considered in the past for criticality safety analysis of spent LWR fuel, it has been necessary to develop additional technical data to supplement the data used for the fresh fuel assumption. Other areas of interest being pursued are verification of analytic methods and verification of procedures to assure proper loading for casks using burnup credit.

THE DOE/OCRWM BURNUP CREDIT PROGRAM

The OCRWM burnup credit activities are performed cooperatively by two separate groups. The base technology for burnup credit is being developed by the Burnup Credit Task Group (BCTG), lead by Sandia National Laboratories (SNL). The implementation of burnup credit for use in spent fuel cask design is the responsibility of the second group, consisting of the OCRWM cask contractors.

The BCTG activities include identifying and resolving generic technical issues associated with the design of burnup credit casks. The issues which have been identified by SNL and others within the OCRWM transportation program,[1] have also been identified more recently by an independent group of experts. [4]

Both of the OCRWM Initiative 1 cask contractors are planning to use burnup credit for criticality safety. Both of the contractors, General Atomics (GA) and Babcock & Wilcox (B&W), are using burnup credit for their pressurized water reactor (PWR) spent fuel cask designs. Neither contractor is currently using burnup credit for their boiling water reactor (BWR) spent fuel cask designs. Both contractors have met with the NRC on several occasions to discuss their approaches to using burnup credit. The BCTG has met with the NRC separately, and has supported the cask contractors in their efforts to gain NRC approval for the use of burnup credit.

CRITICALITY AND CRITICALITY SAFETY

Criticality is the achievement of a self-sustaining nuclear chain reaction. The chain reaction proceeds as atoms of a fissile material absorb thermal neutrons, fission into new lighter atoms (i.e., fission products), and emit additional neutrons which interact with more fissile atoms. When the process continues on its own, the system of atoms of fissile material is said to be critical. The measure of criticality is the multiplication factor, k. The multiplication factor is the ratio of the rates of neutron production to neutron loss. When $k < 1$, we say the system is subcritical. Criticality is achieved when $k = 1$, and a system is said to be supercritical if $k > 1$. In theory we may consider an unbounded system of fissile material (i.e., infinite system), in which case k_{inf} is used as the measure of criticality. In practice we are interested in real systems which have finite size, in which case k_{eff} is used as the measure of criticality.

Nuclear reactors are designed to achieve criticality. The results of a reactor's operation include the conversion of fissile material to its lighter elements called fission products and heat which is used to generate electric power. About once a year, 1/3 of the fuel in a reactor is replaced. The spent fuel is removed because its reactivity is too low to effectively contribute to power generation in the reactor environment.

Spent fuel casks are designed to be subcritical. This is accomplished by using one or more of the following approaches: (1) limit the quantity of fissile material in the system, (2) remove thermal neutrons by using neutron absorbers (poisons), (3) control the population of thermal neutrons by moderator and/or reflector materials, and (4) control the spacing of the fissile elements of the system to reduce reactivity. Although the spent fuel is no longer very effective for power generation it is still somewhat reactive. Furthermore, under the assumed worst case flooded conditions of 10 CFR 71, and under transport conditions which are cooler and lack the boron control of PWR water, the spent fuel would be somewhat more reactive.

To obtain an NRC certificate of compliance the cask designer must demonstrate subcriticality of the spent fuel cask under the requirements of 10 CFR 71. Criticality safety must be demonstrated for a single package assumed to contain water, and be surrounded by water (this provides moderation and reflection of neutrons). Criticality safety must also be demonstrated for arrays of casks in their most reactive credible condition following both normal and hypothetical accident damage conditions of 10 CFR 71. For dry spent fuel casks which are water tight under normal and hypothetical accident conditions, the single package which assumes a water flooded cask is most reactive. Furthermore, since water is necessary for criticality in a LWR system only the single package case can achieve criticality for such a dry cask system. The analytic conditions described above represents a worst case approach to assuring criticality safety. In addition, it has become a customary practice to design transport casks to a 5% criticality safety margin. That is, the cask under its most reactive circumstance must be shown to have a $k_{eff} \leq 0.95$. For OCRWM casks which will be used to transport spent fuel to a repository that will be licensed by NRC under 10 CFR 60 a cask $k_{eff} \leq 0.95$ is required by those regulations.[5]

CASK DESIGN FOR CRITICALITY SAFETY

Casks are designed and used to specific limits of fissile content and internal configuration. For multi-assembly PWR casks, fuel baskets are used to limit neutron interaction between assemblies by controlling geometry and by the use of external (i.e., outside the fuel) poisons. Baskets may also use flux traps to control neutron interaction between adjacent fuel assemblies. A flux trap is basically a gap built into a basket which is activated for a water flooded cask by forming a sandwich of water surrounded by neutron poisons to separate adjacent fuel assemblies. The flux trap configuration traps neutrons travelling between fuel assemblies.

Under the fresh fuel assumption for criticality safety analysis, the fissile content of the fuel is assumed to be the same as the unused levels, and fission products that may act as internal poisons are ignored. For casks designed using burnup credit for criticality safety, the reduced fissile content of the fuel is considered along with the internal poisons present in the burned fuel.

A substantial amount of data and experience exists for criticality safety in transportation under the fresh fuel assumption. This information is directly applicable to criticality safety design for burnup credit casks. However, the use of burnup credit introduces several new variables and issues that require additional information and resolution. These include: (1) fuel characteristics and criticality analysis methods, (2) effects of fuel in-core burnup history on average and local characteristics (e.g., the so-called end effects), (3) assurance of loading burnup casks with fuel having sufficient minimum burnup characteristics, and (4) uncertainties associated with the new variables.

The predictability of spent fuel characteristics and criticality by analysis is being addressed by the BCTG headed by SNL. This activity is referred to as benchmarking.[6] Once the basic benchmarking efforts are completed by the BCTG, the cask contractors will be able to incorporate the information into their specific cask designs. Similarly the end-effects issue is being addressed by the BCTG,[7] and will be incorporated into the specific cask design activities by the cask contractors. Finally, the issue of assuring proper cask loading will be primarily addressed by the cask contractor, with support from the BCTG who will develop loading verification measurement methods.[8]

Figure 1 presents a graphical description of an approach to criticality design safety. The graph provides a useful quantitative description of criticality safety design for a cask using either the fresh fuel assumption or burnup credit.

Curve A represents k_{inf} for an infinite array (or perfectly reflected finite array) of spent fuel assemblies having various initial enrichments, no external criticality controls, and no burnup (i.e., fresh fuel assumption). Curve B represents the k_{eff} for essentially the same system, but of finite size. The difference in k between curves A and B is due to neutron leakage. For very large arrays the leakage is small, and for small arrays the leakage would be larger (e.g., a rail-barge or R/B cask vs. a legal-weight truck or LWT cask).

Curve C_o is the k_{eff} for an externally controlled version of the system represented by curve B. The external criticality controls may include poisons as well as flux traps which are part of the fuel basket. The k_{eff}s represented by curves C_1 through C_5 correspond to the system represented by C_o, but with increasing burnup credit assumed, and corresponding reduced reactivity.

The multiplication factor for our hypothetical cask design, $k_{eff'D}$, is represented by curve C_o up to initial enrichment e_o, and curve D between e_o and e_m. The increasing $k_{eff'D}$ (up to e_o) is the fresh fuel portion of the criticality safety design curve. The decreasing portion between e_o and e_m is the burnup credit portion. If there were no uncertainties associated with burnup credit the burnup portion of the curve would coincide with a design multiplication factor $k_{eff'D} = 0.95$ throughout its range. The difference in k_{eff} between curve D and 0.95 represents the increase in uncertainty as more burnup credit is taken. Basically, we see that for a cask designed using the fresh fuel assumption or burnup credit the peak k_{eff} occurs at the maximum enrichment under the fresh fuel assumption. Although uncertainties can be reduced for burnup credit, they can never be reduced to zero; furthermore, they tend to increase with increased burnup credit. These factors are reflected by curve D.

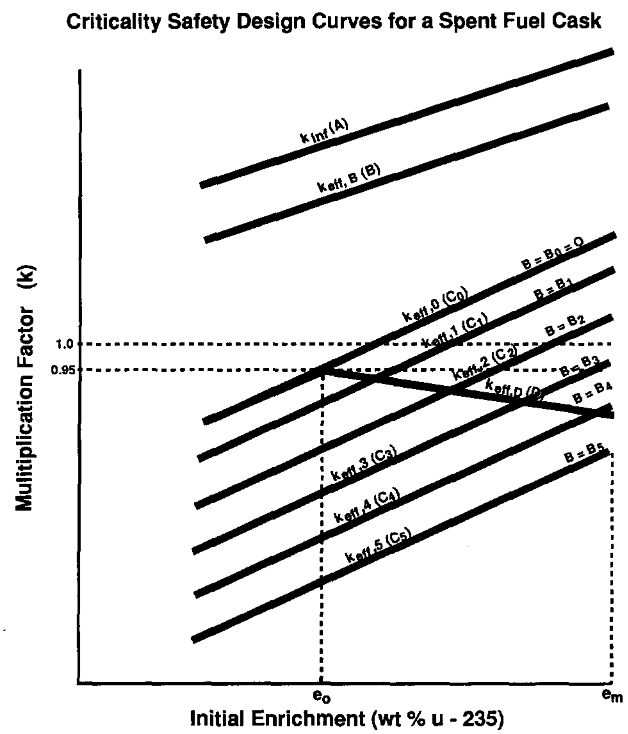

Figure 1
Criticality Safety Design Curve
Curves are for a worst case (criticality safety) condition (water filled cask with full water reflection).

USE OF A BURNUP CREDIT CASK

Operation and use of a burnup credit cask is nearly the same as operation and use of a cask designed using the fresh fuel assumption. The difference is that for fuel falling into the region where criticality safety relies on burnup credit, the loading process must assure that the additional burnup conditions are met. For proper loading of a burnup credit cask we need to know the amount of burnup the fuel has undergone, its age and initial enrichment. Of those, only the initial enrichment is needed for a fresh fuel cask loading. Figure 1 provides a design curve for a specific fuel type in a specific cask design along with specific age, initial enrichments and burnups. The curve representing the cask design k_{eff} in Figure 1 can be used to develop the spent fuel loading curves shown in Figure 2. The family of loading curves, designated L_o, ... L_3, represent loading curves for different fuel types with different reactivities. Curve L_o represents the most reactive of those considered. Curve L_3 represents the least reactive.

Spent fuel with burnup and initial enrichment above and to the left of the curve representing (or bounding) its fuel type in Figure 2, may be loaded to full capacity. Spent fuel with initial enrichment less than the enrichment designated $e_{o'i}$ (where i = 1, 2, 3, or 4) for its fuel type is loaded as a fresh fuel array, and minimum burnup is not a concern. Spent fuel with burnup and initial enrichment below and to the right of the curve representing (or bounding) its fuel type cannot be loaded without additional evaluation and possibly additional actions to control reactivity, and assure a $k_{eff} \leq 0.95$. Additional control could include reduced capacity (less fissile mass) or use of additional neutron poisons (increased external control).

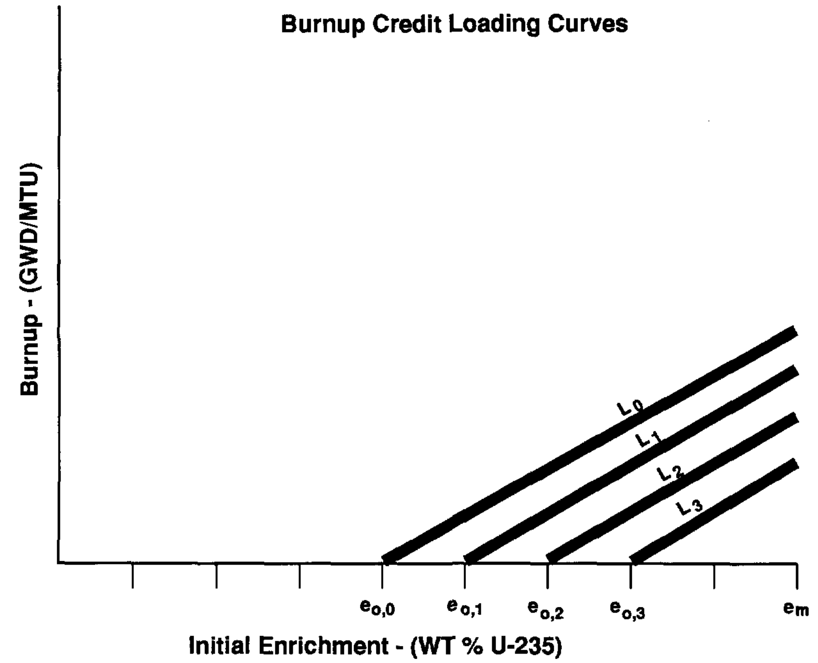

Figure 2
Burnup Credit Loading Curve
Loading curves meet $k_{eff} \leq 0.95$ with calculational biases and burnup credit uncertainties included. Curves are for a worst case (criticality safety) condition (water filled cask with full water reflection).

OCRWM CASK DESIGNS USING BURNUP CREDIT

The two spent fuel casks currently being designed for OCRWM are the GA LWT casks and the B&W R/B cask. Both GA and B&W use burnup credit in addition to external poisons in the fuel baskets for criticality safety of their PWR loadings. Both use external poisons alone for their BWR loadings (i.e., fresh fuel assumption). Neither GA nor B&W use flux traps in their basket designs. Both GA and B&W will use loading curves similar to that shown in Figure 2. The loading operations for the OCRWM casks will rely on utility fuel management practices which are regulated by the NRC, and verification of cask loading which may include physical measurements.

The GA-4 Legal Weight Truck Cask

GA uses two separate cask bodies for their LWT cask system. The GA-4 is used for spent PWR fuel. The GA-9 is used for spent BWR fuel. The GA-4 has a capacity of up to four PWR assemblies with initial enrichments of up to 4.5 wt% U-235. The GA-4 uses a cruciform stainless steel fuel support structure (FSS or basket) which has boron carbide (B_4C) rods held in the FSS plates to provide external criticality control. GA takes credit for only a small portion of the available fission products for demonstrating criticality safety. Those fission products used have been well characterized, and represent about 80% of the control available from fission products.

GA is currently considering a single curve for use in assuring safe loading of a burnup credit cask. The single curve is conservatively based on the most reactive fuel types to be approved for the GA-4 cask design. The GA-4 fuel loading curve is convex outward rather than a straight line as shown in Figure 2. For the GA-4 cask, burnup credit is taken for spent PWR fuel with initial enrichments between 3 wt% U-235 and the maximum initial enrichment for the cask, 4.5 wt% U-235. At the maximum enrichment, the minimum burnup is 25 gigawatt-day/metric ton uranium (GWD/MTU). The loading curve is based on a $k_{eff} \leq 0.95$ with all calculational biases and uncertainties due to the use of burnup credit taken into account.

The B&W Rail/Barge Cask

B&W uses a single cask body for their BR-100 R/B cask. Separate interchangeable baskets are used for PWR or BWR spent fuel in the BR-100. Only the PWR configuration of the BR-100 uses burnup credit. The BR-100 has a capacity of up to 21 PWR assemblies with initial enrichments of up to 4.5 wt % U-235. The BR-100 basket is a stainless steel structure with Boral (borated aluminum) plates to provide external criticality control. The basket also includes copper plates to enhance heat transfer. Like GA, B&W takes credit for only a small portion of the available fission products for demonstrating criticality safety. Those fission products used have been well characterized, and represent about 80% of the control available from fission products.

B&W is currently considering a single curve for use in assuring safe loading of a burnup credit cask. The single curve is conservatively based on the most reactive PWR fuel types to be approved for the BR-100 cask design. The BR-100 fuel loading curve has a slightly convex outward shape rather than the straight line shown in Figure 2. For the BR-100 cask, burnup credit is taken for spent PWR fuel with initial enrichments between 2.2 wt% U-235 and the maximum initial enrichment for the cask, 4.5 wt% U-235. At the maximum enrichment, the minimum burnup is 30 GWD/MTU. The loading curve is based on a $k_{eff} \leq 0.95$ with all calculational biases and uncertainties due to the use of burnup credit taken into account.

CONCLUSIONS

Although no LWR spent fuel casks using burnup credit have been certified by the NRC, the regulations do not prohibit such an action. Furthermore, the NRC has certified a cask for burnup credit under the condition of verification of the loaded cask by measurement. It is clear that the use of burnup credit as part of the criticality control for a spent fuel cask introduces new variables in evaluating criticality safety. It is also clear that for burnup credit casks, loading is somewhat more important for criticality safety than loading of a cask that is designed and used on the fresh fuel assumption. For the loading of a fresh fuel cask, only initial enrichment needs to be considered to assure criticality control. For a burnup credit cask, initial enrichment, age and burnup must be considered to assure criticality control.

The uncertainties associated with the introduction of burnup credit for criticality control of spent LWR fuel in transportation casks have been identified. Furthermore, these uncertainties are being addressed adequately, and technical issues are being resolved in a manner that will assure criticality safety for burnup credit cask designs. In addition, the use of utility fuel management practices coupled with verification measurements will assure proper loading of burnup credit casks. It is believed that a strong basis is being developed for NRC's eventual approval of spent LWR fuel casks that use burnup credit as part of their criticality safety design.

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Lawrence A. Hassler, Ph.D., has been involved in the area of physics and radiation transport at the B&W Fuel Company for twenty years. His work also has included methods development, vessel fluence analyses and criticality evaluations. During the past ten years he has concentrated on disrupted core and the failed fuel shipping canisters, spent fuel storage rack analyses and licensing analyses for low level waste shipping containers. Currently, he is the lead criticality analyst for the BR-100 ton rail/barge shipping cask.

Burnup Credit Issues In Spent Fuel Transport

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ABSTRACT

Burnup credit has been successfully applied to spent fuel storage pools in the United States, resulting in increased capacity and permitting the storage of spent fuel with higher initial enrichments. The regulatory acceptance of transport cask design incorporating burnup credit and its benefits depends on the identification and resolution of issues that affect the determination of criticality safety margins. The factors contributing to the determination of criticality safety have been examined by fault-tree analysis to compare the issues involved in both the "fresh fuel" and "burnup credit" approaches to criticality safety.

INTRODUCTION

A traditional assumption used in evaluating the criticality safety of a spent fuel storage or transport cask is that the spent fuel is as reactive as fresh fuel. This is known as the "fresh fuel assumption." This assumption avoids a number of calculational and verification problems, but takes a heavy toll in decreased cask capacity to accommodate neutron absorbers, flux traps and other criticality controls. An alternative to the fresh fuel assumption is called "burnup credit." That is, the reduced reactivity of spent fuel that occurs from the net depletion of fissile nuclides and the net increase in fission and activation product neutron absorbers (poisons) is considered.

Burnup credit has been successfully applied to spent fuel storage pools in the United States, resulting in increased capacity and permitting the storage of spent fuel with higher initial enrichments. Both of the commercial transport cask designers supporting the Cask Systems Development Program (CSDP) of the U.S. Department of Energy's Office of Civilian Radioactive Waste Management (OCRWM) — General Atomics and Babcock & Wilcox — have cask designs incorporating burnup credit.

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The implementation of burnup credit in the design of transport casks must be accomplished while still maintaining or enhancing individual cask safety and system safety. It is recognized that the use of burnup credit will increase the amount of unacceptable fuel that could be misloaded. To implement a burnup credit design and operational strategy, critical issues must be identified and quantified, and steps taken to reduce the uncertainties involved.

CRITICALITY SAFETY

The criticality safety of an array of spent fuel assemblies depends on the reactivity of the fuel. Spent fuel reactivity can be specified as a function of four variables: (1) the initial enrichment of the fresh fuel, (2) the geometry of the fuel, (3) the in-core burnup history of the fuel, and (4) the decay time since the fuel was discharged as "spent" from the operating reactor.

The fresh fuel assumption considers reactivity to be a function of a single variable — initial enrichment. Requiring the cask criticality designs to be based on the maximum available initial enrichment of the fresh fuel provides sufficient excess design margin to preclude a criticality event from occurring under any foreseeable circumstance. While the need for a criticality safety design margin is acknowledged, the actual margin provided by the fresh fuel assumption is not explicitly defined.

Criticality can only occur in an array of light-water reactor (LWR) fuel if (1) sufficient fissile material is available in an appropriate geometry, (2) a moderator (such as water) is present, and (3) the criticality control features are compromised. No single event, loss, or failure, whether operational or component-related, should result in loss of criticality safety. Under the fresh fuel assumption, cask criticality control system depends on "external" components that include neutron absorbers (poisons) incorporated in the cask or basket web, void spaces or "flux traps" incorporated in the basket for moderator requirements, and structural support members.

These features are "external" to the fuel. When, a cask is flooded, the water filling the flux traps will thermalize neutrons in the vicinity of the poisons, increasing the probability that neutrons will be absorbed in the basket poison rather than in the fuel. These basket features are "hardware" subcomponents of the criticality control system. Loss of any hardware subcomponent could result in reduced reliability.

The criticality safety analysis normally assumes maximum water moderation and reflection, although shipments are intended to be dry. The absence of moderator results in additional negative reactivity of 30 to 40%. Also, the reactivity of the loaded spent fuel is significantly less than that of the fresh fuel. The reliability of the criticality safety margin associated with the fresh fuel assumption is generally assumed to be independent of the reliability of fuel loading operations. This is indeed true if, in the future, fuel is designed to the same reactivity limit over the life of the cask. If future fuel is made more reactive (higher initial enrichment) active operational requirements will be necessary to preclude loading nonspecification fuel into the cask. Nonspecification fuel is any fuel (fresh or irradiated) that exceeds the design basis reactivity.

Three individual events, when combined, might result in a reduced subcritical margin during cask loading even for the fresh fuel assumption. Excessive fuel reactivity and inadequate criticality controls in a cask, combined with an absence of soluble poison in pool water, could result in reduced safety margin as illustrated in Figure 1. Excessive fresh fuel reactivity may already exist because current generation casks were designed for maximum fuel enrichments about 3.0 to 3.8 wt % U-235, and fresh fuel with enrichments up to 4.5 wt % U-235 are available at numerous reactor facilities. Future fuel designs are expected to be based on even higher enrichments and possibly varying enrichments within an assembly design.

Inadequate criticality controls can result from design error, defective fabrication processes, incorrect basket or cask selection, or use of a damaged cask, as illustrated by the fault tree in Figure 2. Any of these could independently affect the reliability of the external criticality control system. While soluble poison is generally present in pool water at pressurized water reactor (PWR) plants, its reactivity effect may vary from site to site. Soluble poison control is not used at boiling water reactor (BWR) plants.

Error sources which could lead to a defective cask design or fabrication are illustrated by the fault tree in Figure 3. A faulty component can result from incorrect fabrication procedures or errors during the fabrication process. An incorrect fabrication procedure may be developed because of design or analysis errors. These can occur because of analyst error, an error in the fuel enrichment chosen for the design basis, or errors associated with benchmark or experimental data or methodology. Fabrication errors can also result from inadequate material inspections or a procedural error in the inspection or forming processes. Finally, for a defective design or hardware to be placed in service, a faulty acceptance test must fail to detect the defects.

For criticality to occur during transport, moderator and fissile material must be present in the cask in a critical geometry. During normal transport, the casks are shipped dry, and criticality is not possible. The fault tree in Figure 4 illustrates the conditions necessary for criticality to occur during a transport accident. Severe damage to external criticality control features such as the basket supports, absorbers, or flux traps could render most fuel design margins inadequate. Excessive fuel reactivity could result from rearrangement during an accident and/or preshipment loading errors. Presence of a moderator could arise from two situations; an accident could occur that leaves a cask where a moderator is present (such as submerged in a river), or the cask could begin transport in a moderated (flooded) condition. Similarly, moderator in-leakage can only occur in two ways. First, operator error during cask loading could result in an improper seal, or second, the containment could be severely ruptured during an accident. The net result of each path is a reduction in the subcritical margin of the cask/fuel system.

It should be noted that an accidental criticality event in transport requires at least two highly unlikely, independent, and concurrent or sequential changes in the conditions essential to criticality safety. Because LWR fuel must be moderated to achieve criticality, the most significant condition for criticality safety comes from the possibility that the accident could result in a moderated condition. If a cask is indeed "dry" and sealed properly before shipment, three independent events are required for criticality to be credible: first, an accident; second, severe containment failure; and third, moderator presence.

BURNUP CREDIT ISSUES

In the case of burnup credit, the criticality control system consists of two separate components. The first is an "external" control component, similar to that used in a fresh fuel assumption design basis, that includes poisons in the cask or basket web and geometric spacing and support. The second is an "internal" control component — the compensation of the loaded spent fuel. Burned fuel reduces external criticality control requirements due to net depletion of the fissile material and the production of poisons.

The major events that could lead to reduced subcritical margin during cask loading or transport are unchanged with burnup credit. However, the number of opportunities for error leading to one of those events, excessive fuel reactivity, will increase because the populations of nonspecification fuel will be larger and the characteristics of spent fuel must be included in the cask design basis. These affect the error sources in Figure 2.

The isotopic composition of the spent fuel is the critical element in determining fuel reactivity. The capability of the calculational codes to predict accurately isotopic composition must be validated by comparison to experimental isotope assays — chemical and radiochemical — of spent fuel rods.

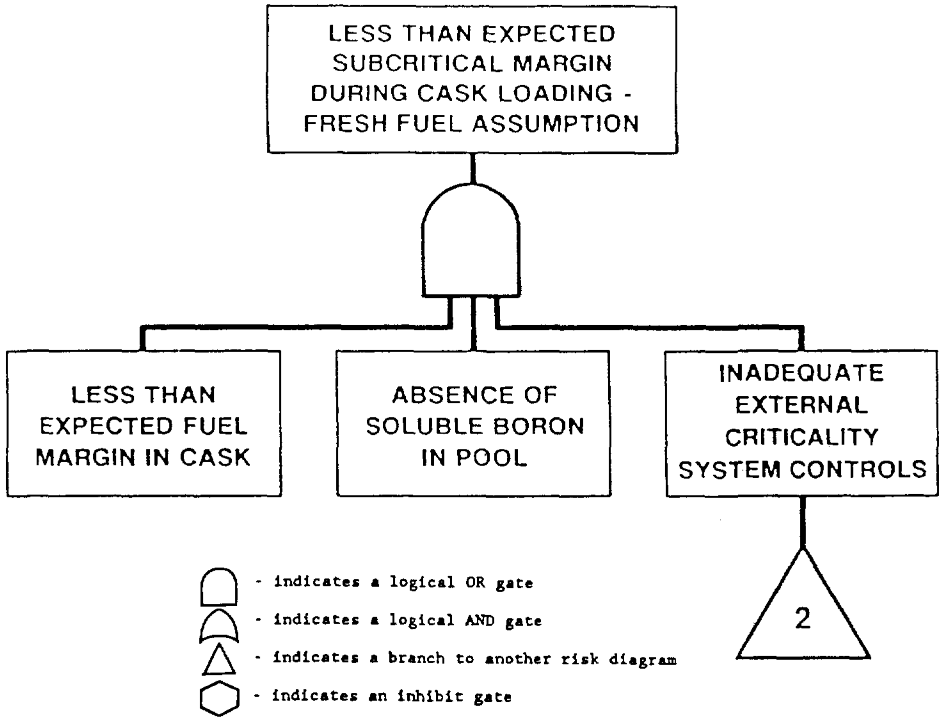

Figure 1
Factors Affecting the Criticality Safety Margin During Cask Loading for the Fresh-Fuel Assumption Case

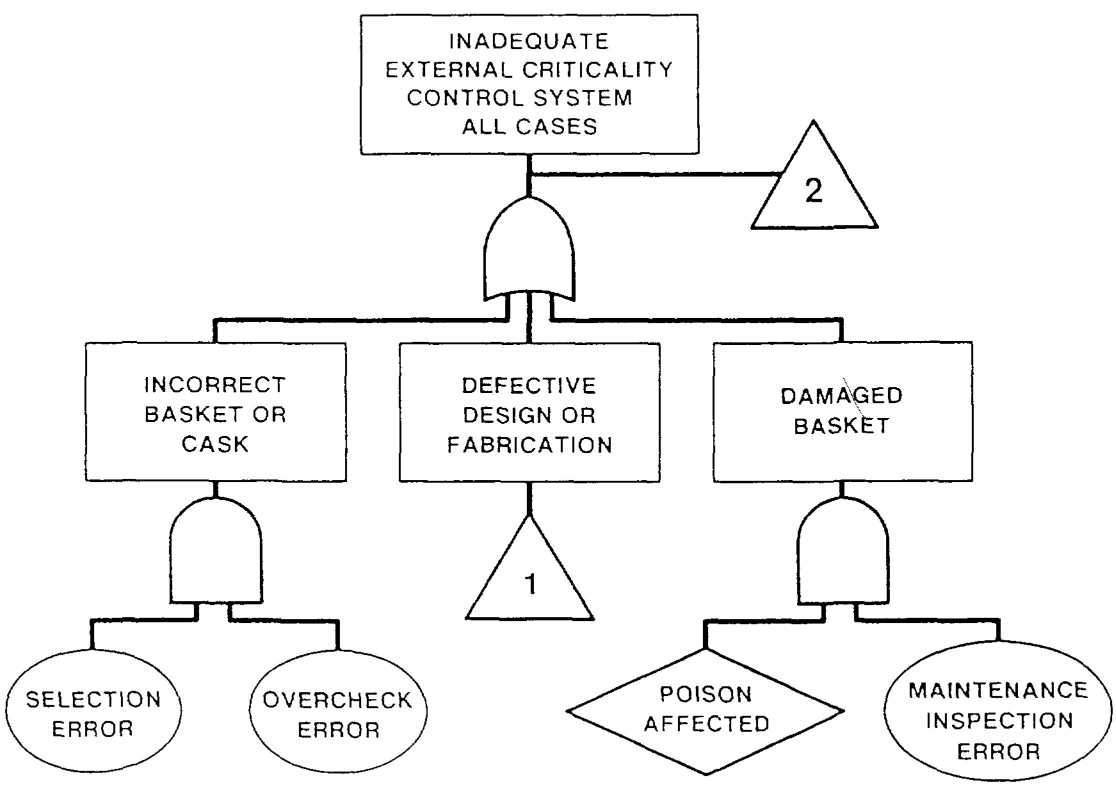

Figure 2
Factors Affecting the Reliability of the External Criticality Controls System

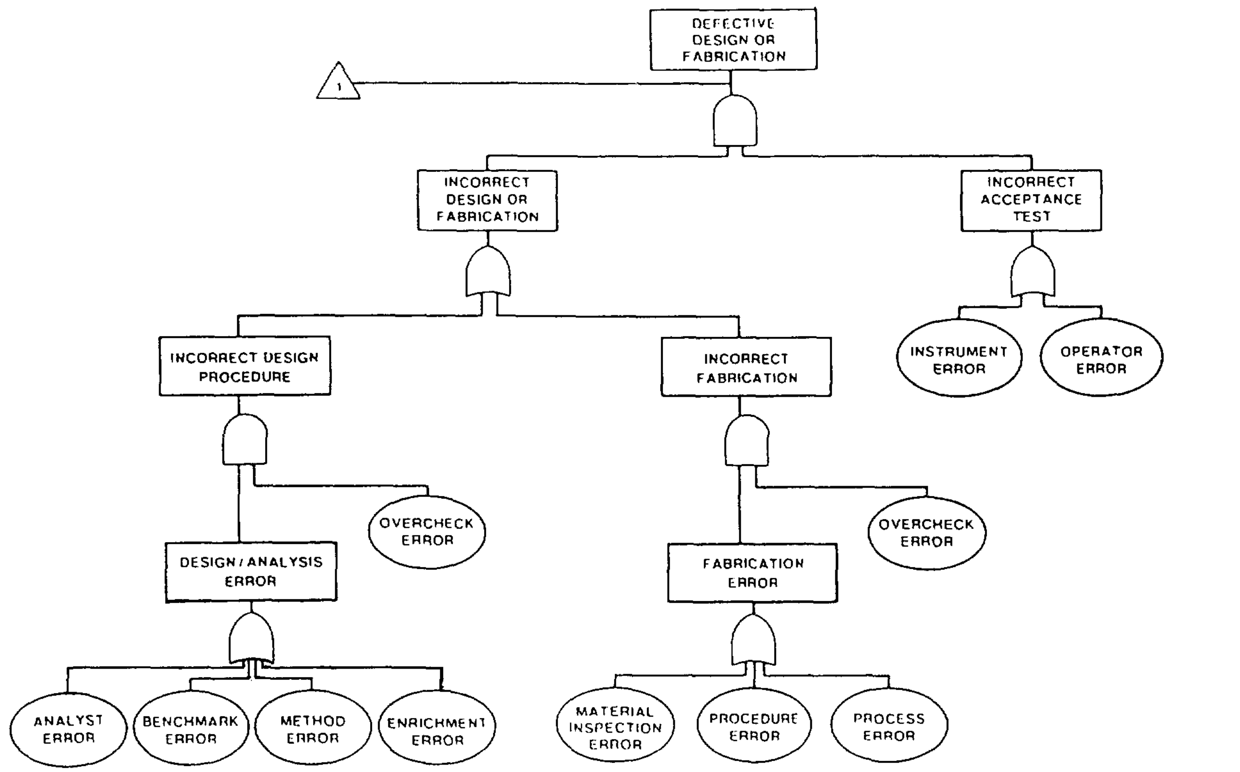

Figure 3
Failure Modes for a Defective Cask Design or Fabrication

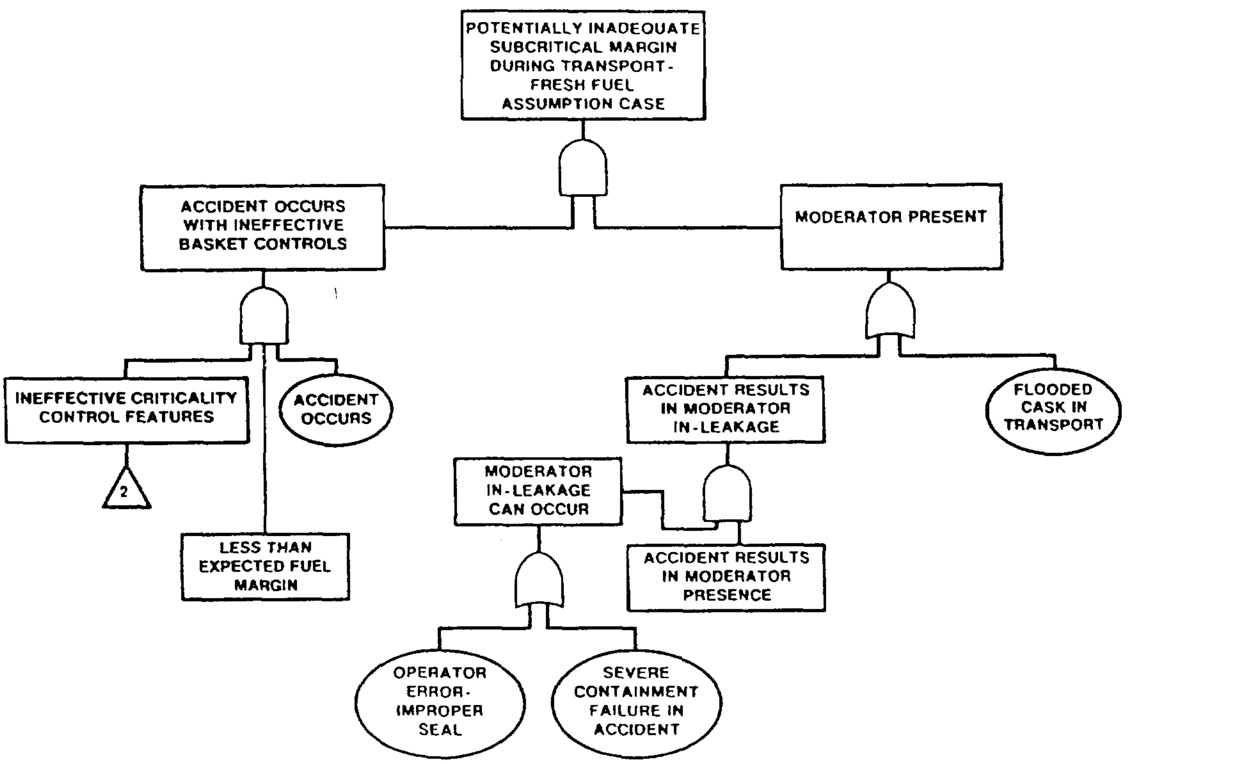

Figure 4
Factors Affecting Criticality Safety During Transport for the Fresh-Fuel Assumption Case

The fissile and dominant actinide isotopes have well-characterized yields and cross-sections because of their importance to reactor control. Only a few fission products need to be considered because of their dominance of neutron absorption.

As illustrated in Figure 5, exceeding fuel reactivity limits could result from a system assembly error, an error in the analysis used to develop fuel-loading procedures, or an error in the burnup characterization of the spent fuel (from error in in-core measurements or subsequent analyses). Events that can lead to in-core analysis error are illustrated in Figure 6. An erroneous loading procedure (Figure 7) could occur if nonspecification fuel is included in the spent fuel inventory. Such an error must be repeated several times (i.e., several wrong assemblies must be chosen) to achieve sufficient excessive fuel reactivity for criticality to occur. For example, analyses indicated that at least three to four misloadings of highly enriched fresh PWR fuel are required to approach criticality in an 18-assembly burnup credit spent fuel cask.[1] A system assembly error (Figure 8) could result if incorrect fuel is loaded, or an incorrect cask or basket is used. Loading errors affect criticality safety only H there is nonspecification fuel for the particular cask in the pool. Each error must be identically repeated if an independent overcheck of each loading operation is required.

The effect of fuel-related failure modes on criticality safety depends on the nonspecification fuel inventory available for misloading. The size of this inventory depends on the burnup and enrichment specifications used as the reactivity design basis for a cask. Analyses indicate that the minimum burnup necessary for a given enrichment will be significantly lower than the design burnup value for that enrichment, for any fuel type, because there is an upper limit on the potential benefits of burnup credit for a given cask design. As more assemblies are added to the cask, weight or shielding restrictions will become the capacity limiting factor rather than criticality. For this reason, it is not necessary to take full credit for the design burnup of a given assembly before sufficient negative reactivity is present to reduce the need for external criticality control features in a cask design. This is particularly true for truck casks. Model analyses indicate that burnups as low as 5 to 10 GWD/MTU (at 5.5 wt % U-235 initial enrichment) are sufficient to reduce the reactivity of a 3 to 4 PWR assembly array such that truck cask capacities can be increased from 2 to 4 PWR assemblies.[1] For rail casks, burnups in the range of 25 GWD/MTU (5.5 wt % U-235) appear to be necessary to achieve a benefit from burnup credit. For a given cask design, lower enrichments and burnup combinations yield lower reactivities and thus can also be accommodated. In fact, the reactivities associated with fresh fuels of much lower enrichments could also be accommodated.

SPENT FUEL INVENTORY

Preliminary analyses of the existing spent fuel inventory have been completed and indicate several important points. First, less than about 2% of the existing spent fuel inventory appears to consist of fuel that would have reactivity in excess of a typical maximum enrichment/minimum burnup specification for a rail cask. Second, only about 50 individual assemblies in the current inventory are unacceptable for loading into a four-PWR assembly truck cask. Third, a significant fraction of the existing inventory of nonspecification spent fuel consists of older-generation stainless-steel-clad fuel with high enrichment-to-design burnup ratios. Much of this fuel was prematurely discharged because of in-core failures or other reasons that may require special handling and transport conditions. Fourth, the majority of the existing inventory of nonspecification spent fuel appears to be located at a small number of older reactor facilities.

A trend analysis of historical premature fuel discharges from reactors was conducted to investigate the reasons for those discharges. The results indicate that the sciences of fuel management and plant chemistry controls have matured considerably. Standardization and improvements in fuel designs, operational efficiencies, and chemistry management practices have occurred during the intervening years, resulting in fewer projected premature fuel discharges. The nonspecification spent fuel inventory of the future could still be dominated by older spent fuel discharged during the late 1960s and early 1970s. This fuel will comprise a very small fraction of the future inventory, and could be removed first.

The only difference between the fresh fuel and burnup credit cases is the addition of the spent fuel criticality control component as illustrated in Figure 9. It is important to recognize that this component is also a "system," consisting of many design, development, fabrication and operational activities that are very similar to those that result in the external criticality control features of a cask. The external features are also affected by the spent fuel characteristics. Some minimal acceptance criteria for demonstrating the reliability of spent fuel analysis and operational activities is needed. This does not mean that the reliability or quality of current spent fuel operations is questionable; however, the reliability associated with those operations needs to be defined. The fact that the subsystem activities in both the fresh fuel and burnup credit cases are similar indicates that the root cause error probabilities are likely of similar magnitude, although the opportunities for error increase with burnup credit because of additional benchmark requirements and analyses.

CONCLUSIONS

The regulatory acceptance of cask design incorporating burnup credit depends on the identification and resolution of issues that affect the determination of criticality safety margins. Fault-tree analyses of nuclear criticality safety issues indicate that both the "fresh fuel" and "burnup credit" approaches to calculating criticality safety follow similar pathways, and both involve risks. In theory, the possibility of misloaded (non-specification) fuel is increased in casks using burnup credit. However, an analysis of the actual inventory of spent fuel in the U. S. indicates that only a small fraction (less than

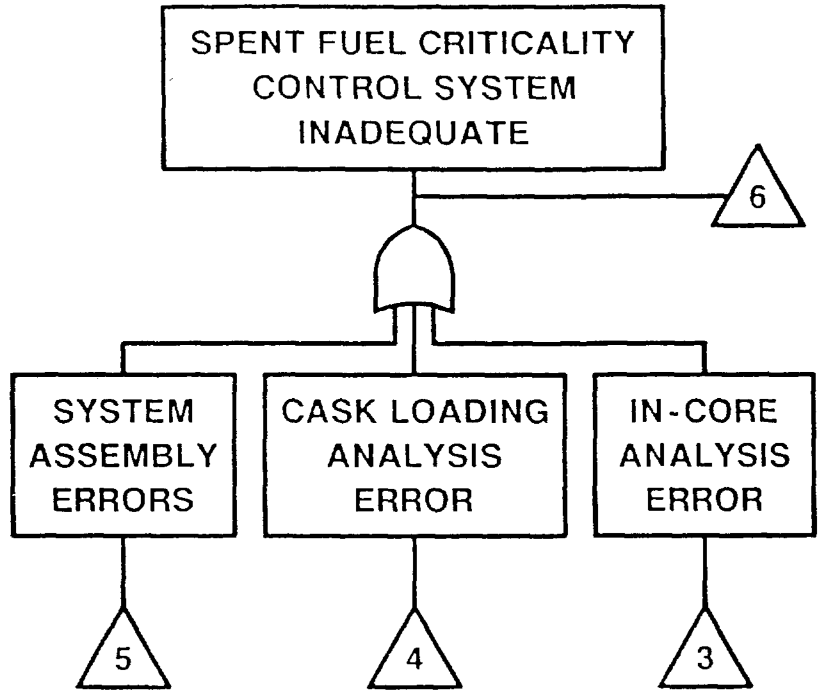

Figure 5
Failure Modes Affecting the Reliability of the Spent-Fuel Criticality Control System

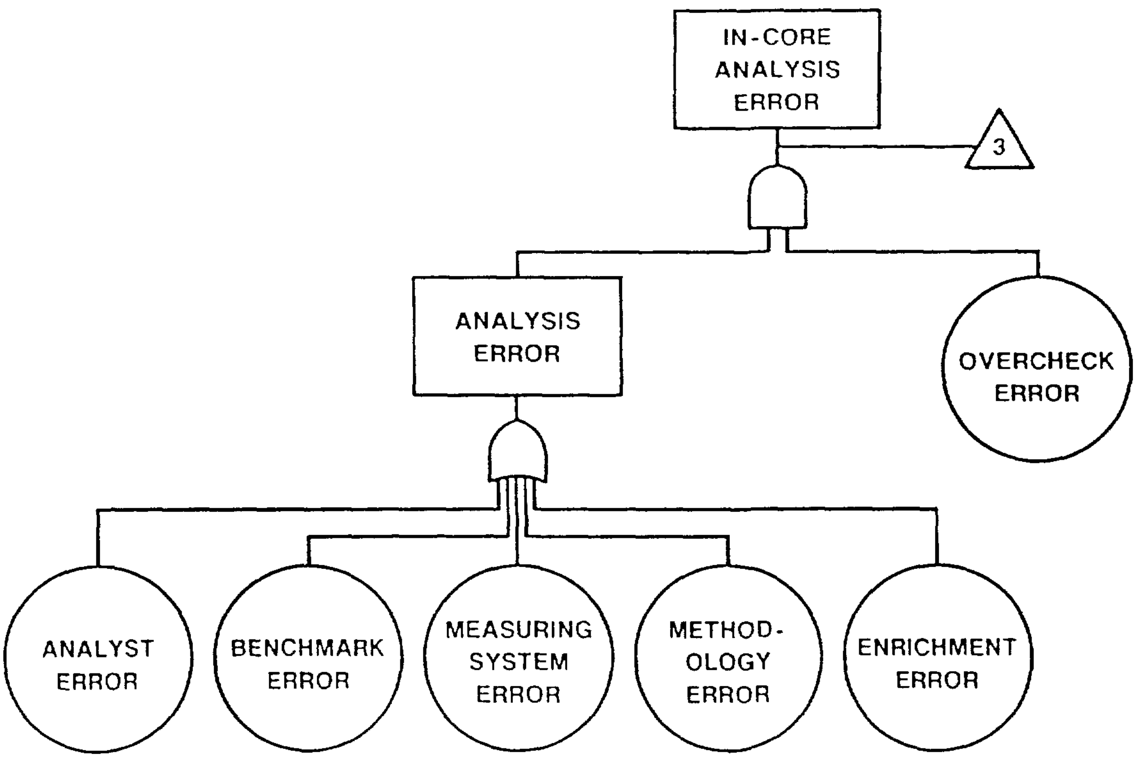

Figure 6
Failure Modes Affecting Spent-Fuel Characterization

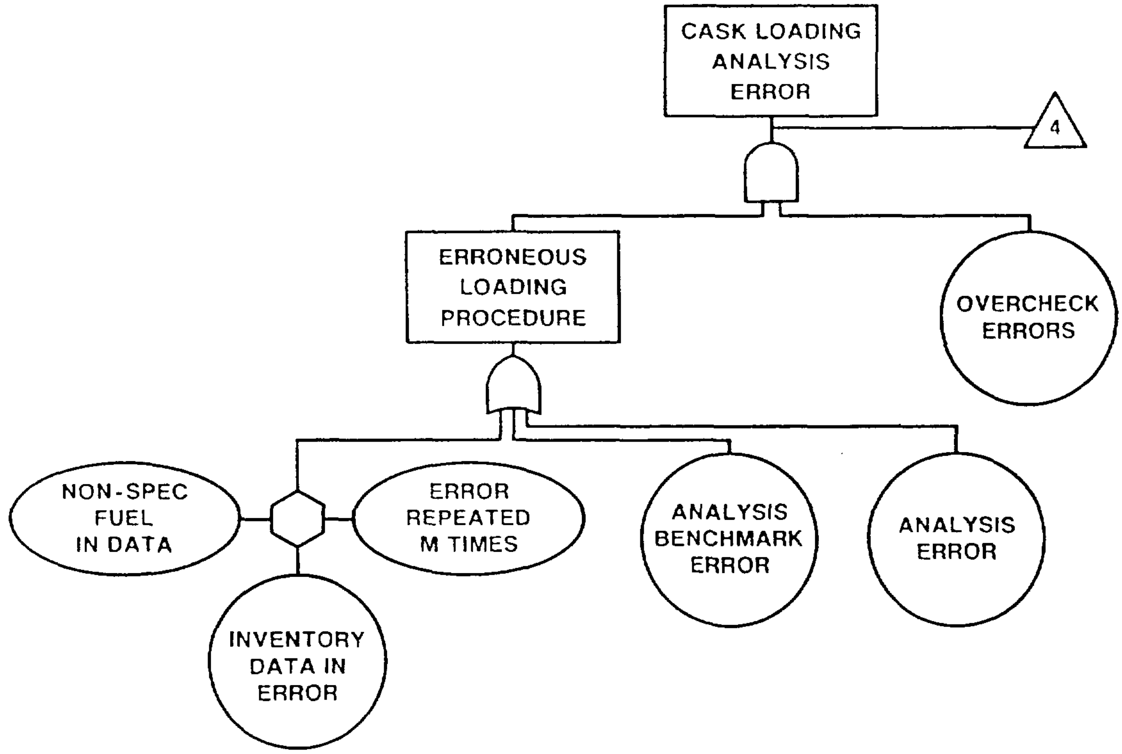

Figure 7
Failure Modes Affecting Fuel-Loading Procedures

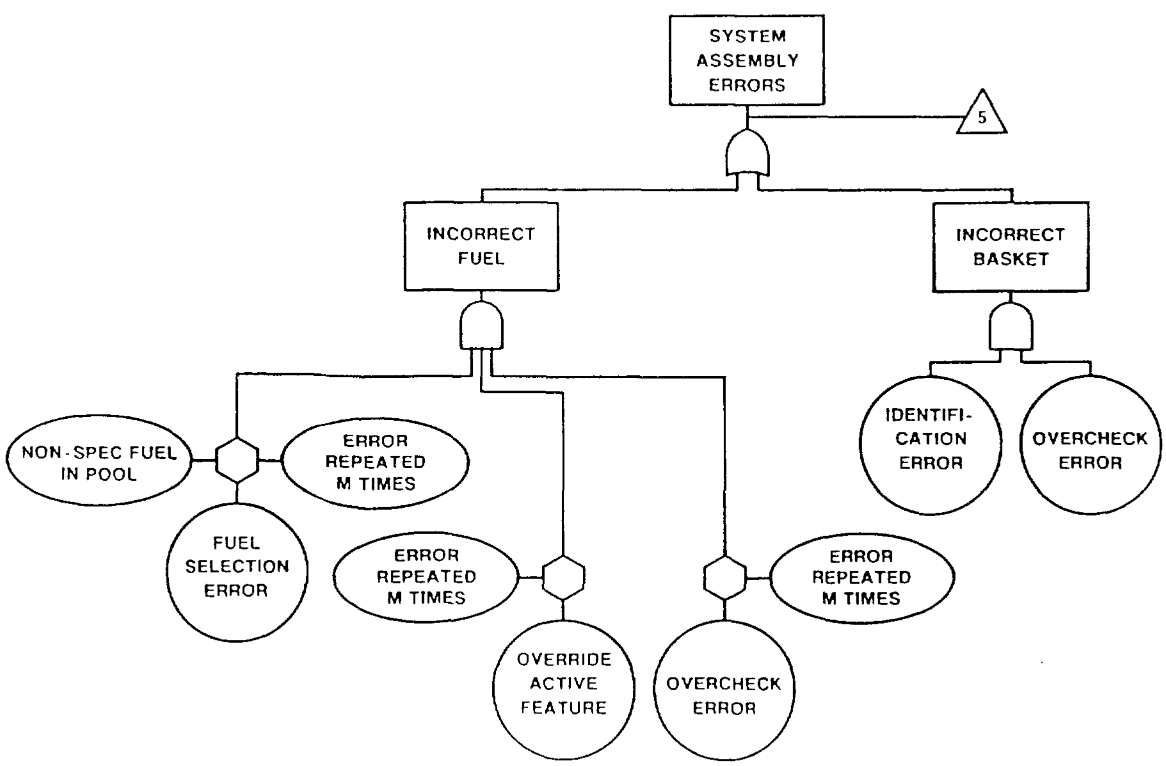

Figure 8
Failure Modes Affecting the Reliability of Fuel-Loading Operations and Systems Assembly

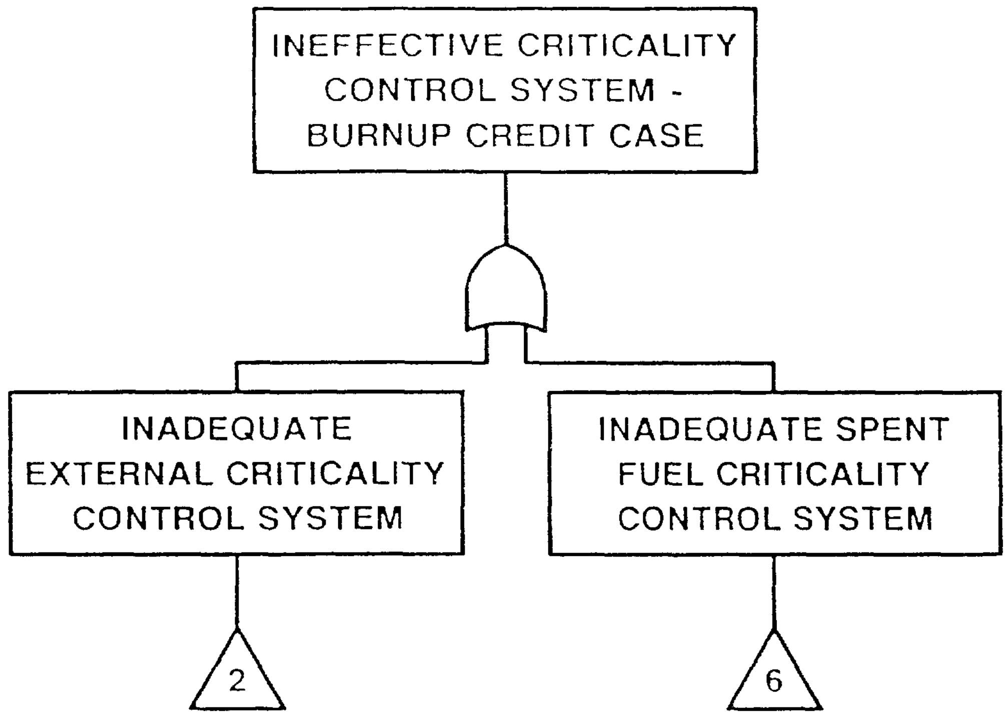

Figure 9
Failure Modes Affecting the Reliability of a Burnup Credit Criticality Control System

one percent) of the existing spent fuel inventory is likely to be non-specification for typical cask designs using burnup credit. Several misloading errors and a sequence of unlikely (accident) events would be required to significantly reduce criticality safety margins if burnup credit is allowed.

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T. L. Sanders entered the nuclear industry through the Navy Nuclear Submarine Program where he ultimately qualified as a supervisor of submarine reactor plant operations. He has held an NRC Senior Operators License and has earned a bachelor's, master's and doctorate in mechanical/nuclear engineering from the University of Texas at Austin. He currently supervises Sandia National Laboratories' Transportation Systems Development Division which supports cask development and technical issue resolution for the Department of Energy's Office of Civilian Radioactive Waste Management.

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Measurement Techniques For Verifying Burnup

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ABSTRACT

Measurements of the nuclear radiation from spent reactor fuel are being considered to qualify assemblies for loading into casks that will be used to transport spent fuel from utility sites to a federal storage facility. To ensure nuclear criticality safety, the casks are being designed to accept assemblies that meet restrictions as to burnup, initial enrichment and cooling time. Measurements could be used to ensure that only fuel assemblies that meet the restrictions are selected for loading. The "Fork" measurement system, designed at Los Alamos National Laboratory and used by the International Atomic Energy Agency to verify burnup and age in international safeguards applications, is being investigated for this application.

INTRODUCTION

Radiation measurements have been used for many years to aid in the characterization, handling and processing of spent nuclear fuel. Applications have included radiation protection, international safeguards, fissile content estimation for reprocessing and verification of records and calculations. The application of radiation measurements to support the identification of spent fuel assemblies for loading into "burnup credit" transport casks is an outstanding issue in the cask development program. Transport casks are being designed to accept assemblies that meet certain restrictions as to burnup, initial enrichment, and cooling time. These restrictions arise from considerations of nuclear criticality safety, particularly under severe accident conditions. For a critical event to occur, the cask would have to be breached and flooded with water of very low neutron absorber content. While the necessary sequence of events is highly unlikely, the consequences and regulatory concerns require that flooded criticality be addressed in cask design.

Previous studies have concluded that the utility-supplied data on burnup, age and initial enrichment is of greater accuracy and reliability than could be provided by additional radiation measurements on spent fuel assemblies.[1] A possible role for measurements in burnup credit operations is to help prevent misloading of unacceptable fuel assemblies, either by confirming reactor records prior to cask loading, or by detecting operator error at the time of loading. A possible alternative to measurements is to make use of the administrative controls and operational procedures that have been used at reactor sites that incorporate burnup credit in spent fuel storage. Experience at such sites needs to be carefully analyzed for its applicability to the misloading and misidentification probabilities.

Since there are over 40,000 spent fuel assemblies stored at more than one hundred locations in the United States, it is important to carefully determine the necessity for and applicability of any measurement requirement. It is imperative that any measurement system selected be as simple, inexpensive, quick and non-intrusive as possible.

BURNUP CREDIT CASKS

The characteristics of fuel acceptable for loading into a burnup credit cask are determined by the design of the cask, and can be specified by a loading curve, as shown in Figure 1. The theoretical cask design to which this Figure applies is part of a comprehensive study in progress to evaluate the impact of the existing spent fuel inventory on transport cask design. The cask would transport four assemblies of the Westinghouse 17 X 17 pressurized water reactor (PWR) fuel design. The criticality (k_{eff}) of the cask was calculated using computer programs and cross section data contained in the Standardized Computer Analyses for Licensing Evaluation (SCALE) system, assuming that the cask was completely flooded with pure water.[2] This system was developed for the NRC by Oak Ridge

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2. Work supported under DOE contract DE-AC06-76RL01830. Pacific Northwest Laboratory is operated for DOE by Battelle Memorial Institute.

National Laboratory to perform standardized criticality, shielding and heat transfer analyses. In this example, the acceptable assemblies (cooled for a minimum of two years) are configured in the cask so that, under flooded conditions, the system is less than 95% of critical ($k_{eff} < 0.95$). The curve separating the "acceptable fuel region" from the "unacceptable fuel region" is determined by $k_{eff} = 0.95$. Fuel for which the combination of initial enrichment and burnup place it in the acceptable fuel region will result in $k_{eff} < 0.95$ when loaded into the cask. The curve also delineates the minimum burnup credit required for a particular initial enrichment. In Figure 1, the cask can accommodate spent fuel of initial enrichment 4.5 wt % U-235 with burnup greater than 7 GWD/MTU and maintain $k_{eff} < 0.95$. This cask design can accommodate fresh fuel (burnup = zero) for initial enrichments less than 3.5 wt % U-235. Another significant result from the criticality calculations for this cask is that a 25% change in burnup produces a change of less than 3% in k_{eff}, over the range of burnup values of interest.

RADIATION FROM SPENT FUEL

The spent fuel assemblies of initial interest in burnup credit loading operations are those that have been cooling for the longest period of time. Most assemblies that are likely to be loaded into burnup credit casks in the first years of transport cask operations will have cooling times greater than 10 years. This long cooling time results in simplified radiation spectra from the spent fuel. Some important gamma-ray and neutron emitting nuclides for aged assemblies are listed in Table 1. For shorter cooling times, many more isotopes are significant emitters, but most have decayed to insignificance after 10 years because of the predominance of short half-lives in the fission and activation products.

Table 1

NUCLIDE	HALF-LIFE (yr)	RADIATION
242_{Cm}	0.45	n, spontaneous fission (sf)
244_{Cm}	18.1	n, sf
240_{Pu}	6.6×10^3	n, sf
238_{Pu}	87.7	n, (α, n) reactions, sf
134_{Cs}	2.06	γ, 605, 796 keV
137_{Cs}	30.0	γ, 662 keV

The cesium isotopes are produced as fission and activation products while in the reactor. After the fuel is removed from the reactor, the isotopes decay with the indicated half-lives. The spontaneously fissioning isotopes are produced by successive neutron capture, beginning with the uranium in the fresh fuel. ^{244}Cm production initially proceeds as the sixth power of the integrated flux. Initially, burnup is directly related to the flux times the initial enrichment. For higher burnups, ^{244}Cm production is found to be proportional to the fourth power of burnup, still a very strong function. ^{242}Cm and ^{240}Pu are less sensitive, but also very strong functions of burnup. This strong dependence of neutron emission on burnup means that uncertainties in measuring the neutron emission result in even smaller uncertainties in the burnup, identifying neutron emission as a very sensitive and accurate means of inferring burnup. At 10 years cooling, for low burnups, the Pu isotopes dominate the neutron emission. For higher burnups, ^{244}Cm dominates. Gross gamma emission follows a decreasing function of cooling time that is complicated for short cooling times; with many isotopes contributing, but after ten years, ^{137}Cs dominates. For cooling times out to about eight years, the ratio of ^{134}Cs to ^{137}Cs gamma-rays provides a means of determining cooling time, but after 10 years, the technique is no longer viable due to the decay of ^{134}Cs.

The information that can be inferred from passive neutron measurements on spent fuel that has cooled for more than 10 years is limited to burnup/enrichment ratios, with a weak dependence on age. Gamma-ray measurements provide an indication of age and a less sensitive indication of burnup.

MEASUREMENT SYSTEMS

In a study being prepared for publication at Sandia National Laboratories, measurement systems from around the world were examined for applicability to burnup credit cask operations. The study considers both passive techniques, that detect the radiation generated internally in the spent fuel, and active techniques, that employ external radiation sources to interrogate the spent fuel. Only one measurement system will be considered in detail here, because it combines the features considered most desirable in this application. This system depends on the availability of standard calibration assemblies of the same design as the assemblies to be examined. The pertinent characteristics of each calibration assembly are assumed to be known. In the measurements under consideration, an assembly or group of assemblies of well-documented characteristics would be selected as the reference standard, and all measurements would be referenced to the chosen standard.

THE "FORK" SYSTEM

The spent fuel measurement system designated "Fork" because of its shape, was developed at Los Alamos National Laboratory for use in safeguards applications for the international Atomic Energy Agency (IAEA).[3] The system is diagrammed in an operational arrangement in Figure 2. It is portable, and can be moved in the storage pool to the spent fuel assembly to be examined. It requires that the assembly be raised in the storage rack so that the midpoint is about 50 cm above the top of the rack. The detector head is located at the midpoint of the assembly for the measurement. A cadmium-covered fission chamber is used to measure epithermal neutrons from the assembly, and an ion chamber is used to measure gammas. An additional fission chamber, without a cadmium cover, is used to measure thermal neutrons. The

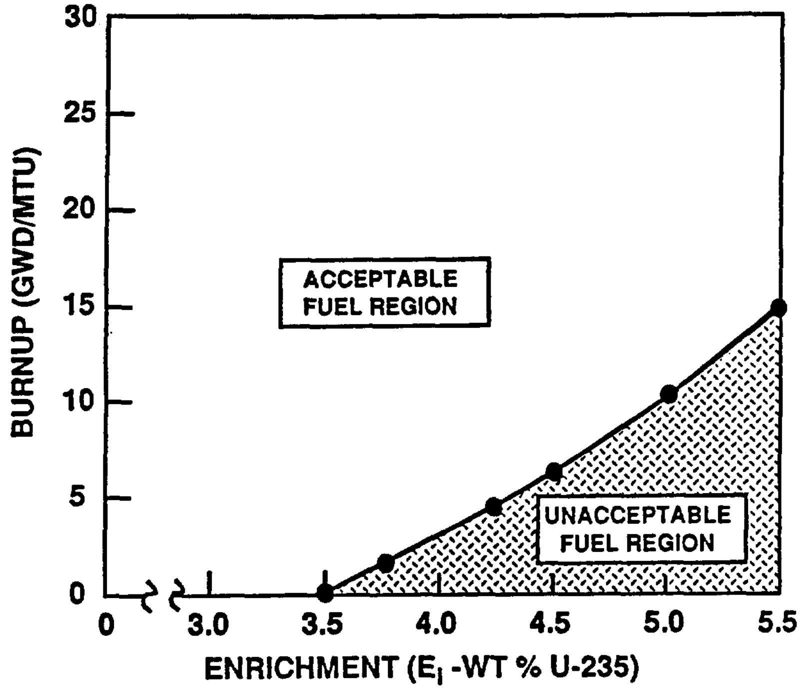

Figure 1
Loading Curve for a Burnup Credit Cask

ratio of the two fission chamber measurements permits the determination of the boron content in the storage pool. The boron content is important in the safeguards application, since the value of that parameter could vary widely. The detector head is made of polyethylene and has identical three-detector arrays in each of the two tines of the fork, for redundancy. A battery-powered electronics unit and microprocessor are used to supply all power to the detectors, collect and analyze the detector outputs, and perform necessary calculations and documentation. The unit has been used to examine spent fuel assemblies at storage facilities around the world. The measurements have required less than 100 seconds measuring time per assembly, and a considerable database has been established. The users state that with proper calibration standard assemblies, burnup has been determined to an average accuracy of about 5%, for burnup in excess of 10 GWD/MTU. The gross gamma-ray measurements have been shown to be consistent with operator-declared values for burnup and age to about 10%. This result has proved to be adequate to eliminate the need for more complex active or high-resolution measurement techniques.[4]

PLANS

The safeguard measurement requirement for validation of reactor records is similar to the requirement for measurements in burnup credit operations, so it is likely that the Fork system, or a similar design, would be appropriate for the burnup application. In that application, a qualified spent fuel assembly would produce neutrons above some level corresponding to the ratio of burnup credit to enrichment required by the cask design. The gross gamma measurement would have to be consistent with the indicated burnup and age.

The database generated to date using the Fork system will be examined for applicability to 10-year-plus fuel. Modifications of the Fork system that might possibly be desirable in the burnup credit application include changes in the type and number of detectors, and a rearranged geometry. Operational requirements and restrictions will be more clearly defined by obtaining utility input, and applying the experience of operators using burnup credit in storage facilities.

CONCLUSIONS

Measurements can be used in burnup credit operations to help prevent misloading of fuel that does not meet the minimum specifications for a particular cask design. Passive neutron and gross gamma-ray measurements are proposed as a means of qualifying spent fuel assemblies. Active systems to measure reactivity or fissile content are necessarily more complex and appear to offer no obvious advantage to burnup credit applications over simpler systems. Plans are underway to produce a prototype measurement system and generate a database of spent fuel measurements, making use of experience with the "Fork" design used by the IAEA for safeguards inspections.

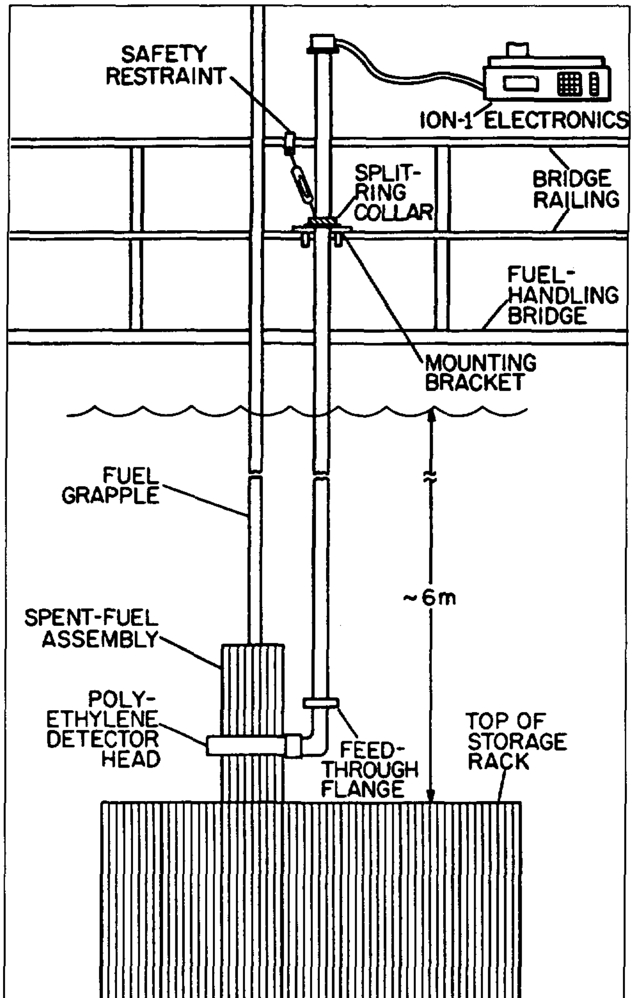

Figure 2
Sketch of Fork Detector in Operation

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R.I. Ewing is a senior member of the technical staff at Sandia National Laboratories, Albuquerque, N.M. He received his doctorate in physics from Rice Institute in Houston, Texas. His publications involving radiation detector design and applications include the neutron spectrometer system known as Bonner spheres, a fruitless search for cold fusion, and verification of the INF and START arms control treaties. Other experimental studies have included the collection of atomic particles from the aurora borealis, the solar wind experiments on the Apollo 16 and 17 lunar missions, and severe lightning strikes to Naval aircraft. He is currently involved in the nuclear criticality aspects of spent fuel transport. He is a member of the American Physical Society, the American Nuclear Society, Phi Beta Kappa and Sigma Xi.

Sidney R. Bierman is a staff scientist at the Battelle Pacific Northwest Laboratory, Richland, Wash. He received a B.S. in chemical engineering from Texas Technology University in 1956 and a M.S. in nuclear engineering from the University of Washington in 1963. Presently, he is providing technical assistance to Sandia National Laboratory staff on burnup credit issues in the transport of spent reactor fuel. He was a senior criticality experimenter at the Hanford Critical Mass Laboratory from 1963 until its closing in December 1988. Prior to that, he was a chemical process engineer for Exxon Corp. and General Electric Co. He is a past chairman of the ANS Nuclear Criticality Safety Division and has authored over 60 professional papers on nuclear criticality. He is a member of the American Nuclear Society, Tau Beta Pi and Kappa Mu Epsilon.

MEMO — A PC Program To Establish Measurement Models For NRTA Evaluation Procedures

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ABSTRACT

The PC program MEMO for determining the dispersion matrix (statistical measurement model) of a series of materials balance results is introduced and described, including theoretical aspects of establishing measurement models. MEMO is a useful tool in international safeguards and provides the essential input data for the near-real-time accountancy *evaluation procedures as implemented in such programs like PROSA.*

1. INTRODUCTION

International Safeguards is an important matter not only for the international safeguards authorities but also for the public acceptance of the peaceful use of nuclear energy. The aim of international safeguards is the timely detection of diversion of significant quantities of nuclear materials, and deterrence of such diversion by the risk of early detection.

In order to achieve these goals, as primary method nuclear materials balancing is applied, which essentially consists of the following principle: The so-called book inventory is the beginning inventory I_B plus net transfers T, which is compared with the physical inventory or the ending inventory I_E. This leads to the well-known materials balance equation:

$$(1) \qquad MUF = I_B + T - I_E,$$

where MUF means material unaccounted for.

This so-called "classical" approach is capable of detecting a possible diversion with high probability.

But it is evident that this approach is not capable to meet the goal of timely detection due to the fact that the balance is closed only at the end of the reference time which is usually in order of half a year to one year.

To meet the criteria of timely detection advanced methods need to be applied which are usually referred to as near-real-time accountancy (NRTA).

The basic idea is to subdivide the total reference time into N several balance periods and to close the balance after each period. This procedure leads to a sequence of material balance results MUF(i), i=l,..,N with:

$$(2) \qquad MUF(i) = I(i-1) + T(i) - I(i),$$

where I(i) is the ending inventory of period i as well as the beginning inventory of period (i+l), and T(i) is the net transfer in period i.

Using this approach, at the end of the i-th balance period, a decision can be taken on whether or not there is a loss of material, based on all i balance results MUF(l),...,MUF(i). Thus, trends can be recognized at an early stage and losses may be detected timely related to their occurrence.

But now a further problem occurs. On the first glance one tends to assume that each MUF value in Formula (2) should be zero if no loss of material has occurred, or visa versa, if there are MUF values greater than zero, nuclear material is missing. However, as the terms in the materials balance equation are the results of several measurements — each of them connected to characteristic measurement uncertainties — the MUF values generally do not amount to zero even in the case of no loss of material.

That means, at the end of each balance period i, a decision has to be made whether or not the deviations of the sequence MUF(l),...,MUF(i) from the expection values zero can be explained by (known) measurement uncertainties.

To evaluate the series of MUF values in an objective and convenient manner the computer program PROSA[1, 2] was developed. Input of PROSA are the sequence of material balance results MUF(l),...,MUF(i) and the measurement model of the facility considered. Using sequential statistical test procedures PROSA evaluates the series of MUF values based on the measurement model. That means the output of PROSA is a decision on whether the series of materials balance results can be explained by the assumed measurement uncertainties or, if not, which among other reasons may be a hint of a possible diversion.

That means the determination of the measurement model is an essential step in data evaluation. This measurement model contains all individual measurement uncertainties with random and systematic components, including the appropriate propagation of variances. From the statistical point of view, the measurement model is described by a matrix which can be regarded as the variance/covariance matrix or dispersion matrix of the MUF series.

In the last several years a lot of R&D effort went into theoretical considerations about measurement models,[3,4] into establishing detailed measurement models based on real process data[5,6] and into developing related computerized tools.[7] In the following, theoretical and fundamental aspects in this framework are discussed and the PC program MEMO for computerized establishing of measurement models is introduced.

2. SOME REMARKS TO THE MEASUREMENT MODEL

To determine the measurement model, first of all it is necessary to define the facility model. This is to declare the number of inventory components in which the considered material is contained, further the number of input and output batches per balance period. That means, for each balance period the facility model consists of a triple of integer values ni,nr,no where ni refers to the number of inventory components, nr to the number of input batches, and no to the number of output batches.

Next the error model has to be established. This model includes all measurement uncertainties for volume measurements as well as for concentration measurements.

Each single measurement — volume as well as concentration — is described by the relative standard deviation (rsd) of:

- the multiplicative component of the systematic measurement uncertainty,
- the multiplicative component of the random measurement uncertainty,

and the standard deviation (sa) of:

- additive component of the systematic measurement uncertainty,
- additive component of the random measurement uncertainty.

The facility model as well as the error model are constant as long as the facility design is not changed. The facility design could be changed either if a new component is added (change of facility model) or a new measurement procedure is introduced (change of error model).

The measurement model itself is affected by the measurement data through the relative error model. That means the measurement model is a function of the (long-term constant) facility model, of the (long-term constant) error model and of the (variable) measurement values, and, therefore, is not constant at all.

3. ASSUMPTIONS ON THE MEASUREMENT UNCERTAINTIES

We assume that the considered amount of material in each inventory component and in each transfer batch can be determined as the mathematical product of a volume determination and a concentration determination:

(3) amount = vol * con.

Further we assume that each single determination (volume and concentration as well) can be conceived as a "measurement" with a systematic and a random error component as well.

Both, systematic and random error component, are assumed to belong to the relative error model; that means they have a multiplicative error component and an additive error component.

For example,

(4) con = E(con) + E(con) * (msc + mrc) + asc + arc,

with E(con): the true (but unknown) concentration value
- ms : the multiplicative component of the systematic error of the concentration determination
- asc : the additive component of the systematic error of the concentration determination
- mrc : the multiplicative component of the random error of the concentration determination
- arc : the additive component of the random error of the concentration determination.

The next assumption is that all error components are normally distributed with known standard deviation and mean zero. Mean zero means that all measurements are unbiased.

Volume measurements and concentration measurements are assumed to be mutually uncorrelated and also systematic error components and random components as well. Correlations may occur when the same measurement method is used several times (propagation of variances for the systematic error component) or when the same measurement is used to determine the content of material in two or more components (propagation of variances for the random error component). This may occur in the case that the concentration is measured in a vessel and the same measurement value also relates to the concentration in the connected pipe.

Another assumption is that the systematic error is constant during the reference time; that means, there is no recalibration of measurement methods and instruments.

4. DETERMINATION OF THE MEASUREMENT MODEL

In the considered facility, each amount of material of each process component and each transfer batch is calculated by the mathematical product of a volume measurement (vol) and of a concentration measurement (con).

Using truncated Taylor series expansion in the mean values E(vol) and E(con) leads to

(5) vol * con = E(vol) * E(con) + (con - E(con)) * E(vol) + (vol - E(vol)) * E(con),

which results in the reasonable equation

(6) E(amount) = E(vol) * E(con).

With regard to the variances, the following expression holds:

(7) var(amount) = $(E(vol))^2$ * var(con) + $(E(con))^2$ * var(vol).

On the right hand of Formula (7) the terms E(vol) and E(con) appear which represent the true but unknown values of volume and concentration measurements.

These unknown values can be substituted by the measurement values themselves, because each measurement is an unbiased estimate of the true value. With this procedure, the variance of each batch of material can be calculated. Furthermore, the variances can be split into random components and into systematic components.

With flow-sheet information provided by the operator all variance and covariance calculations can be summarized into the so-called dispersion matrix of the MUF series. For example, the (i,j) -th element of the matrix describes the covariance between MUF(i) and MUF(j). The dispersion matrix is the condensed form of the measurement model and provides all necessary information to evaluate the sequential MUF series.

5. THE COMPUTER PROGRAM MEMO

To facilitate the determination of the dispersion matrix, computerized procedures are unavoidable. Already in the early '80s, at the Kernforschungszentrum Karlsruhe a host version of a computer program to determine the measurement model was developed. But for routine field use a PC version of such a computer program is necessary. Therefore, in the last months, it was tried to transpose the host version into a PC version.[7] But some difficulties occurred. The greatest problem was the huge amount of memory needed by the old host version. The next problem was a run-time problem. The old program, fast on the host, was very slow on the PC. The third problem was that the old program was not able — neither on host nor on PC — to cover all correlations which may occur in practice.

Due to these reasons a completely new version was realized called MEMO 2.0 (Measurement Model). The computer program MEMO 2.0 runs on personal computers which need at minimum the following configuration of hardware system:

- AT compatible computer system
- mathematical co-processor
- at least 256 kbyte memory
- hard-disk drive
- operating system DOS version 3.0 or higher.

To make MEMO user-friendly for routine field application, the MEMO program modules are covered in a menu-guided user-shell. The entry to MEMO application is shown in Figure 1 which displays the main menu.

MEMO-Version 2.0
MEMO Program:
Enter the Number of Your Choice

1	==	**Browse/Read/Edit/Print Input Data**
2	==	**Run MEMO**
3	==	**Browse Results**
4	==	**Data Transfer to PROSA**
Q	==	**Quit the Session**

Figure 1
The Main Menu of MEMO

There are only a few possibilities to handle MEMO in an incorrect manner, because the user is menu-guided. Examples of incorrect use of MEMO may be the missing of initial data sets. In these cases, the program system generates self-explanatory messages which give the user advice how to proceed correctly.

As mentioned above, the measurement model is a function of the facility model, of the error model and of the actual measurement data. These input information can be imported to the MEMO program via two input data files. The first input data file, usually called "design.dat," contains all measurement methods (volume and concentration measurements) with their specific multiplicative/additive, and systematic/random error components.

The second input data file, usually called "measure.dat," contains in the first line the facility model and in the following lines the actual values of volume and concentration measurements. These values are complemented by key parameters in order to link the various measurement data with the associated measurement errors. Output of MEMO is the sequential MUF series which is computed from the several single measurements. A further output is the dispersion matrix of the MUF series. Both output components of MEMO, MUF series as well as dispersion matrix, are essential input components of the evaluation program PROSA.

In the current version 2.0, MEMO is able to determine the sequential measurement model of up to 50 balance periods as long as the sum of inventory components, input batches per period and output batches per period does not exceed a number of 100 items.

A further very comfortable feature of MEMO 2.0 is the possibility to built up the measurement model from period to period. That means that only the last line of the dispersion matrix has actually to be calculated, which then is added to the former matrix. This leads to a considerable reduction of the run time.

6. CONCLUSION

The PC program MEMO 2.0 is capable of computing the sequential material balance results (MUF) and the related dispersion matrix of the MUF series which is the statistical measurement model of the facility considered. Therefore, MEMO is the connecting piece between data resulting from the design of the facility and actual measurements, and an evaluation program like PROSA. The menu-guided user-shell of MEMO makes the application of MEMO user-friendly. Therefore, MEMO is very suitable in routine field use. A comprehensive manual is in preparation and will be made available before long.

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Experience In Establishing Detailed Measurement Models For Real NRTA Balance Data

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ABSTRACT

The applicability and performance of NRTA procedures essentially depend on the statistical measurement model which includes the measurement uncertainties of the various inventory and transfer determinations in the material balance area and their mutual correlations. The iterative process of establishing a detailed measurement model based on real process data using the PC programs MEMO and PROSA is described in detail.

1. INTRODUCTION

Near-real-time accountancy (NRTA) appears to be an adequate tool to achieve the International Atomic Energy Agency's safeguards criteria in future large scale plants. Research and development work related to this subject is going on in several countries. In order to demonstrate the applicability of NRTA measures for reprocessing plants the German BMFT (Federal Ministry for Research and Technology) has sponsored a research program for computerized inventory taking nuclear materials during ongoing reprocessing process at the Wiederaufarbeitungsanlage Karlsruhe (WAK). The basic idea was to collect the measurement data on process inventories which are primarily generated for process control purposes, and to use these data to establish short-time material balances. Sequential statistical tests were applied to these data in order to decide whether or not the observed differences between book inventories and measured inventories can be assigned to measurement errors.

The evaluation of the material balance results, the so-called material unaccounted for (MUF) values, is based on the measurement model of the facility considered. It must be emphasized that the determination of the measurement model is an essential step in the application of NRTA measures. The measurement model includes all relevant measurement uncertainties and the related propagation of variances. The measurement model is a function of the facility model, of the error model of the various measurement methods, and of the actual measurement data. This measurement model is represented by a matrix, which, from the statistical point of view, is the variance/covariance matrix or dispersion matrix of the MUF series. This report describes the iterative process of establishing a detailed measurement model for NRTA measures of a WAK campaign using real process data.

2. DESCRIPTION OF THE PLANT/ CAMPAIGN

The Wiederaufarbeitungsanlage Karlsruhe served as a test bed in order to get experience with establishing real measurement models. The amount of Plutonium, the material under consideration, is contained in about 70 process components and is calculated from about 100 single measurements. To determine the variance of the various inventory and transfer measurements, operational information as well as target values were used for the measurement uncertainties.

The investigation was applied to data from the reprocessing campaign 2/83 of the WAK plant. Figure 1 shows the increasing Pu- inventory over the time as a result of the more than 100 single measurements in the about 70 process components. By means of Figure 1, it is obvious that the data do not represent a steady-state operation of the facility. Furthermore, the transfers differed considerably from period to period.

With the help of the PC program MEMO,[1] the MUF series (see Figure 2) and the related measurement model were established. To do so, a data handling program was necessary to transpose the process data generated for process control purposes into a proper input format for MEMO. In the next step, the sequential material balance data were evaluated by the PC program PROSA[2,3] based on the dispersion matrix to investigate whether or not the data can be explained by the measurement model.

3. PRACTICAL RESULTS OF THE INVESTIGATION

A first approach to determine a measurement model led to a so-called "one-block-model." That means that the total inventory of the facility is assumed to be contained in one single component. However, evaluations with real process data revealed that the balance data were not in accordance with this measurement model. This was due to the fact that the distribution of the inventory among the various plant components was not at all taken into account.

The next step establishes a more detailed measurement model. But in this approach some inventory components with only small amounts of Plutonium, such as pipes, were not taken into consideration. Again the evaluation with PROSA caused an alarm due to the following. At the beginning of the campaign all plant components, including the pipes, were almost empty. With the increasing inventory, our model showed more and more material unaccounted for had "disappeared" in the pipes causing positive MUF values and consistently an alarm.

A further approach took all Plutonium-containing components into account. However, surprisingly, the test procedures contained in PROSA gave an alarm again as can be seen in Figures 3a to 3c.

But what was the reason for this alarm? All materials were accounted for. This could not be the reason for the alarm. The fault was that the facility design was not modelled properly enough.

At first it was recognized that the reading errors of various volume measurements were modelled as multiplicative errors instead of additive errors. Furthermore, the strong correlation between some concentration measurements were not taken into account. For example, the concentration is measured in a vessel and the same measurement value is assigned to the following pipe. In this case the two concentration values are not only correlated through the systematic error component (same measurement method) but also correlated through the random error component (identical measurement).

Incorporating these findings into the measurement model, the application of PROSA no longer gave any alarm, which

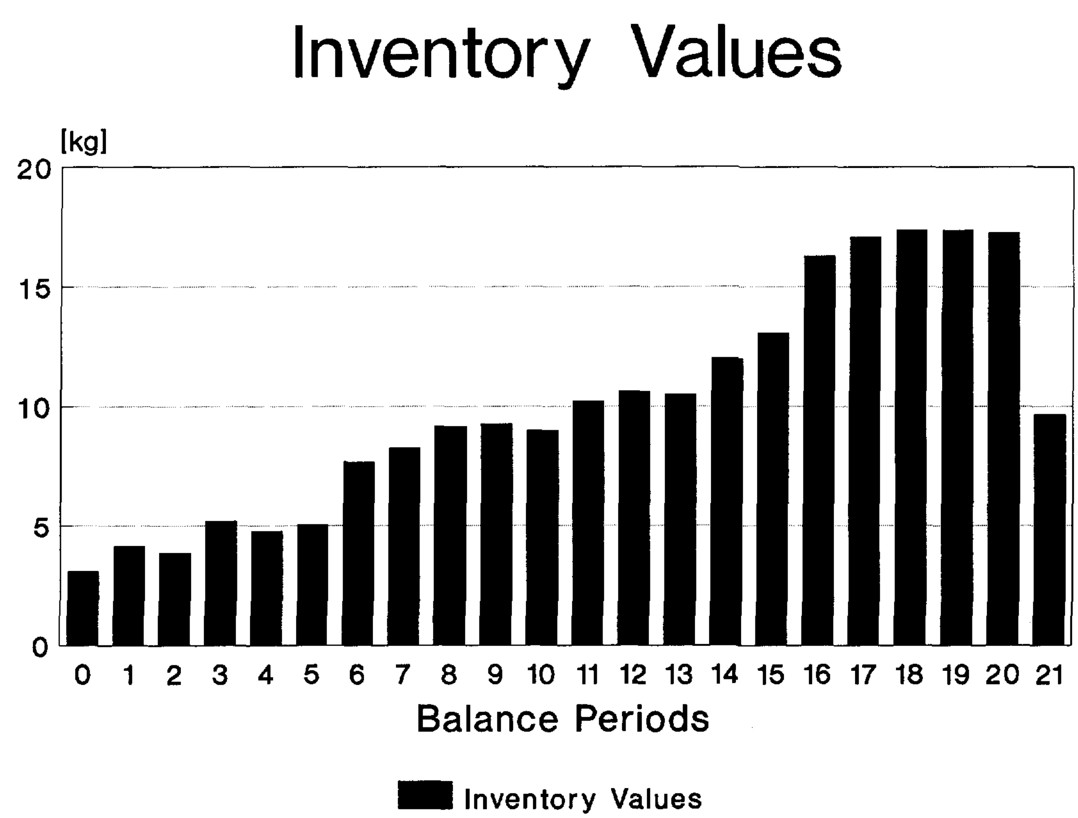

Figure 1
Pu-Inventory of the Campaign

meant that the balance data were in accordance with the underlain measurement model. Figures 4a to 4c show the results of the NRTA tests.

Now the question arises, whether the NRTA measures included in PROSA could detect a loss of material based on this detailed measurement model. To investigate this question, a loss of 0.125 kg Plutonium each from period 1 to period 4 was assumed and added to the original balance data. These modified MUF data, see Figure 5, were also evaluated with PROSA, again based on the same detailed measurement model. In Figures 6a to 6c, the NRTA-test results are illustrated. The CUMUF test alarms in the second period (Figure 6c), and the GEMUF test in the forth one (Figure 6b), whereas the Page's test does not alarm at all (Figure 6a).

These results of the data evaluation demonstrate that the established detailed measurement model is very close to the real situation: The original balance data can be explained by the measurement model, but the modified MUF series — the original MUF series was overlaid by the hypothetical loss pattern as mentioned above — is indicated to be not in accordance with the underlying measurement model.

4. CONCLUSIONS

The determination of the detailed measurement model for real process data with the computer program MEMO leads to very realistic results. The evaluation of the original balance data with the NRTA measures included in PROSA shows that these data are in accordance with the established detailed measurement model. On the other hand, the NRTA measures give an alarm in the example where a loss pattern was added to the MUF values. This demonstrates the capability of NRTA measures based on the described approach of establishing the detailed measurement model. Furthermore, it seems to be useful to investigate further campaigns with real process data in order to demonstrate the routine applicability of MEMO and PROSA.

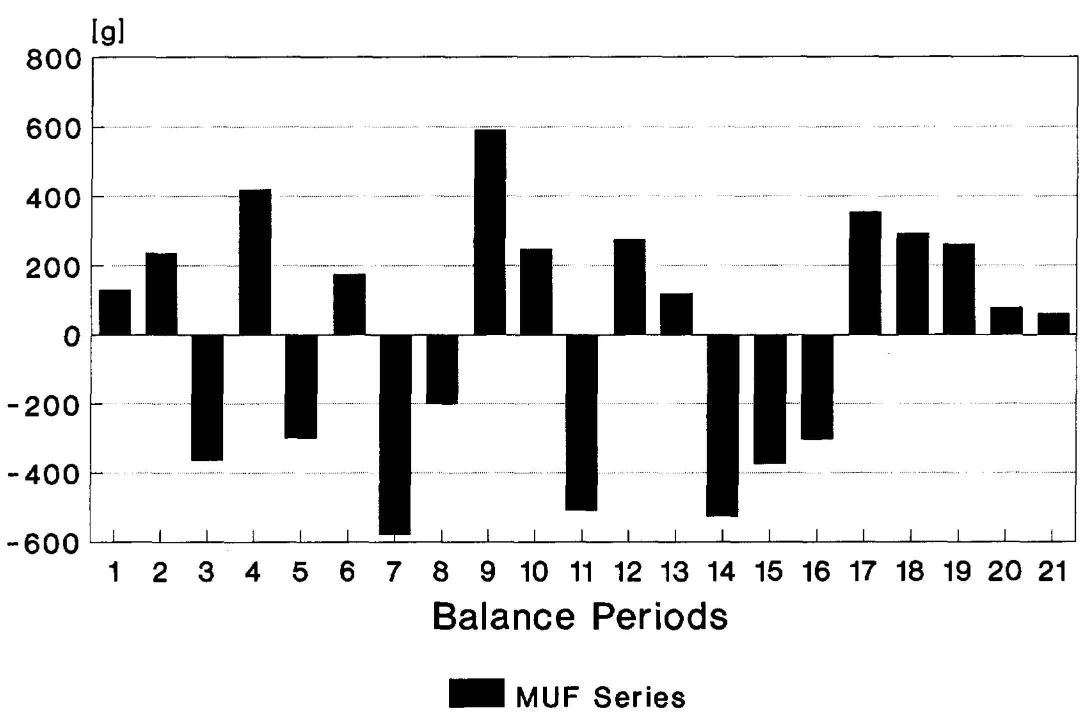

Figure 2
MUF Series of the Campaign

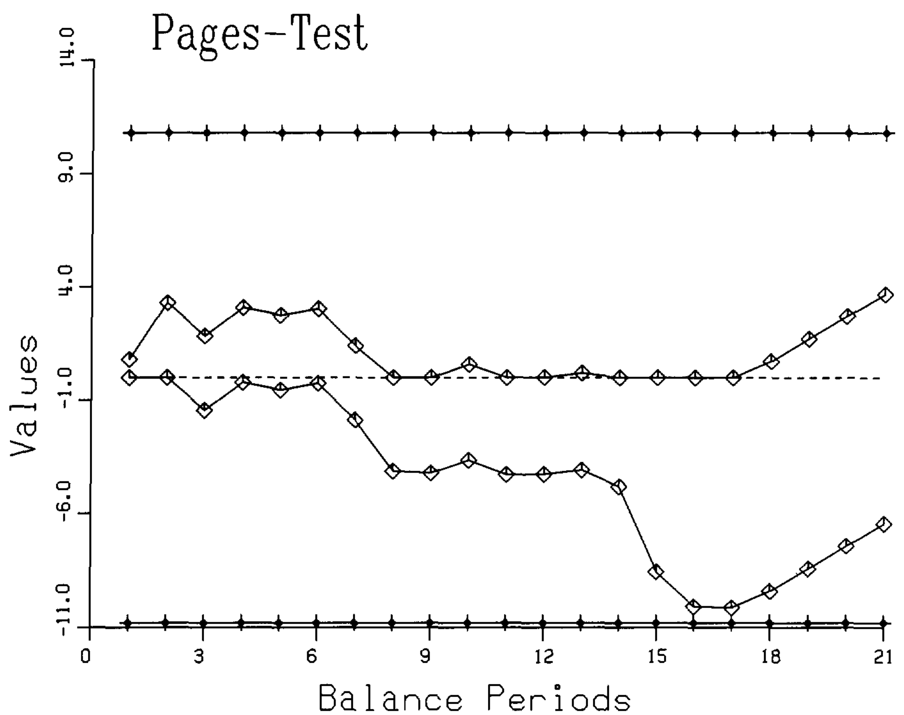

Figure 3a
Results of the Page's Test Based on an Only Multiplicative Error Model

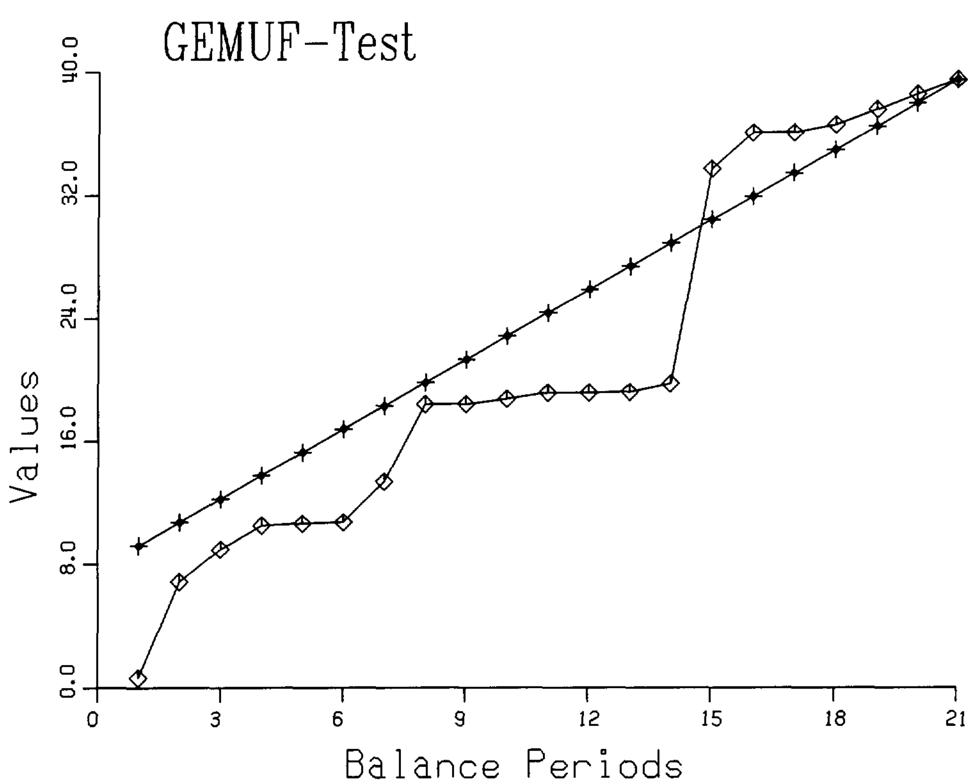

Figure 3b
Results of the GEMUF Test Based on an Only Multiplicative Error Model

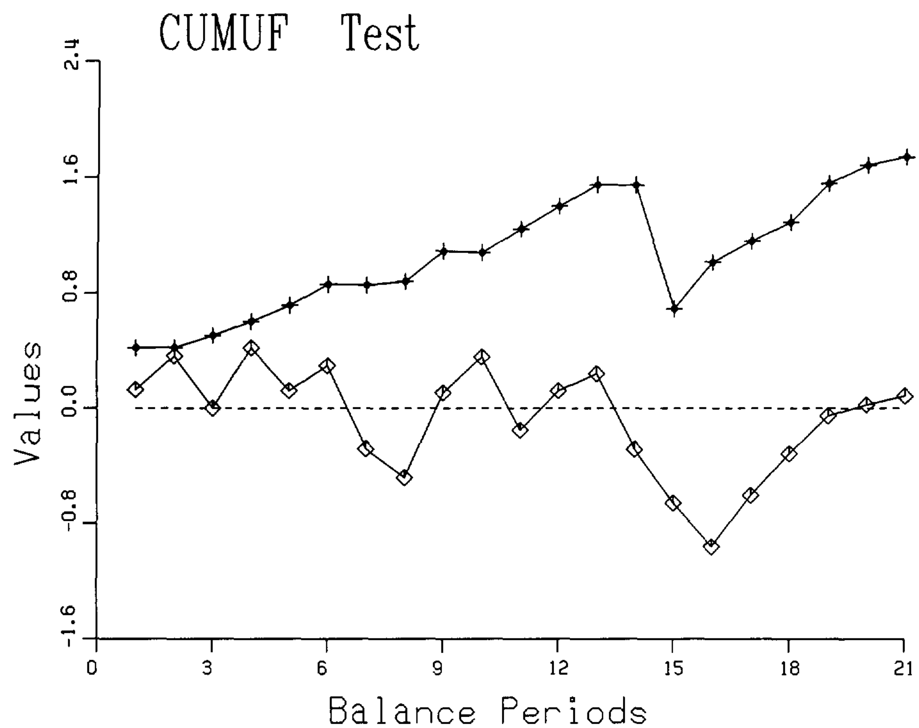

Figure 3c
Results of the CUMUF Test Based on an Only Multiplicative Error Model

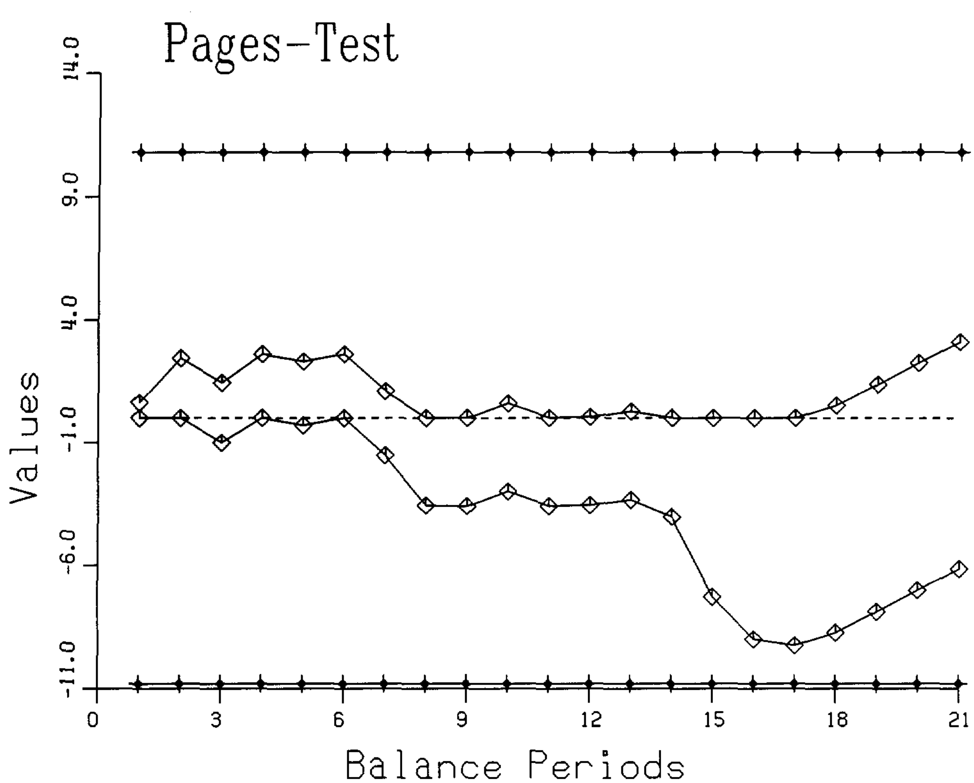

Figure 4a
Results of the Page's Test Based on a Mixed Measurement Model; Correlations are also Considered

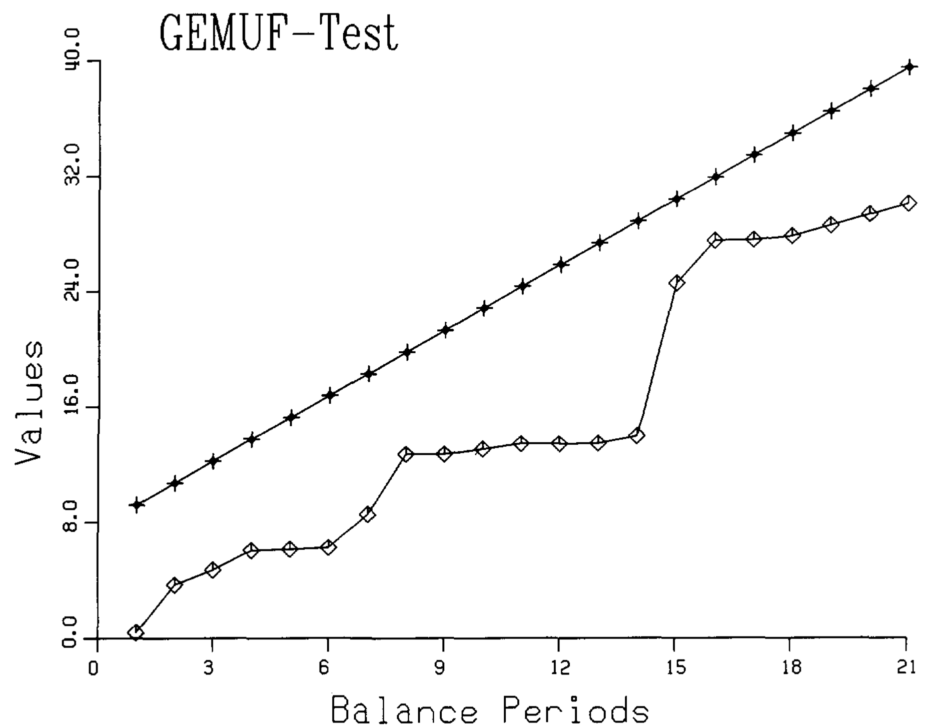

Figure 4b
Results of the GEMUF Test Based on a Mixed Measurement Model; Correlations are also Considered

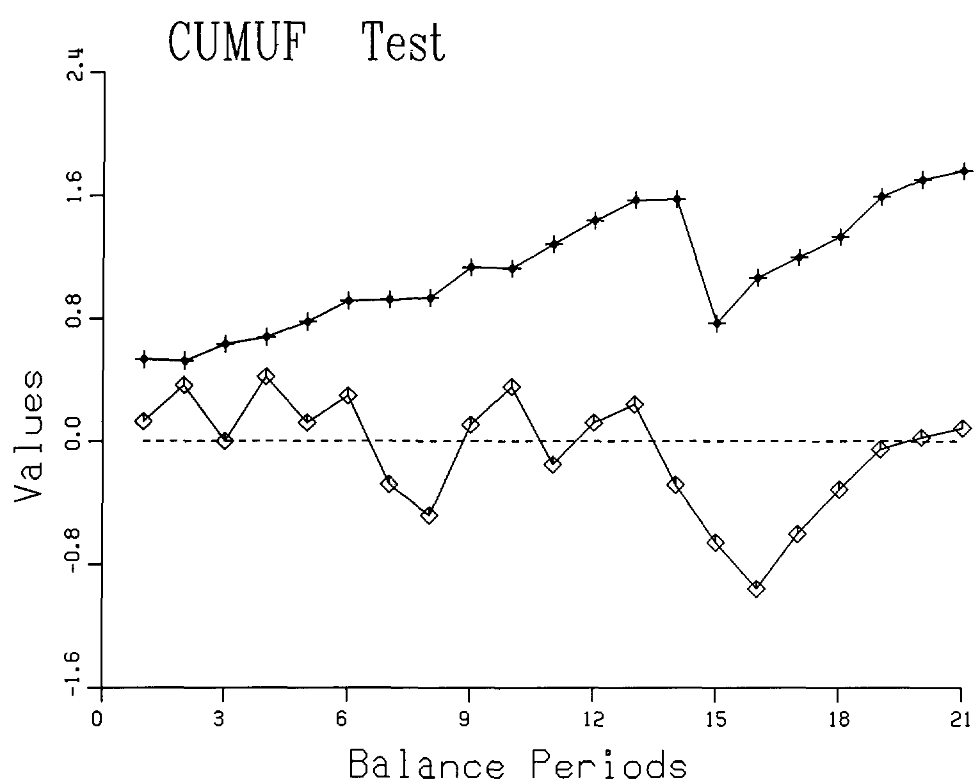

Figure 4c
Results of the CUMUF Test Based on a Mixed Measurement Model; Correlations are also Considered

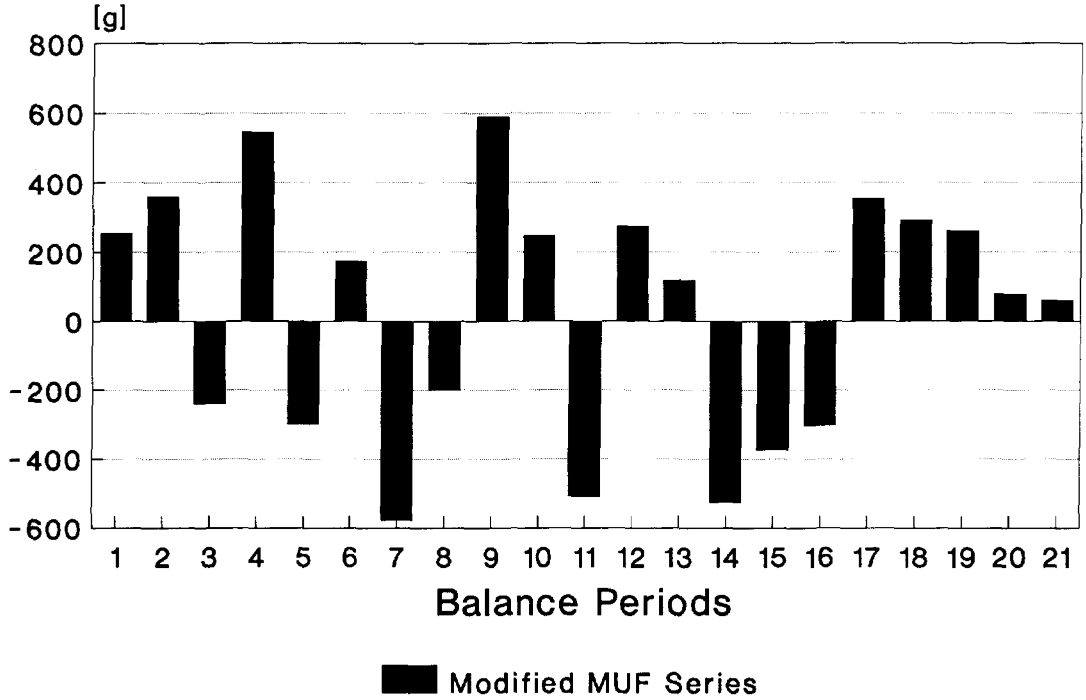

Figure 5
Modified MUF Series

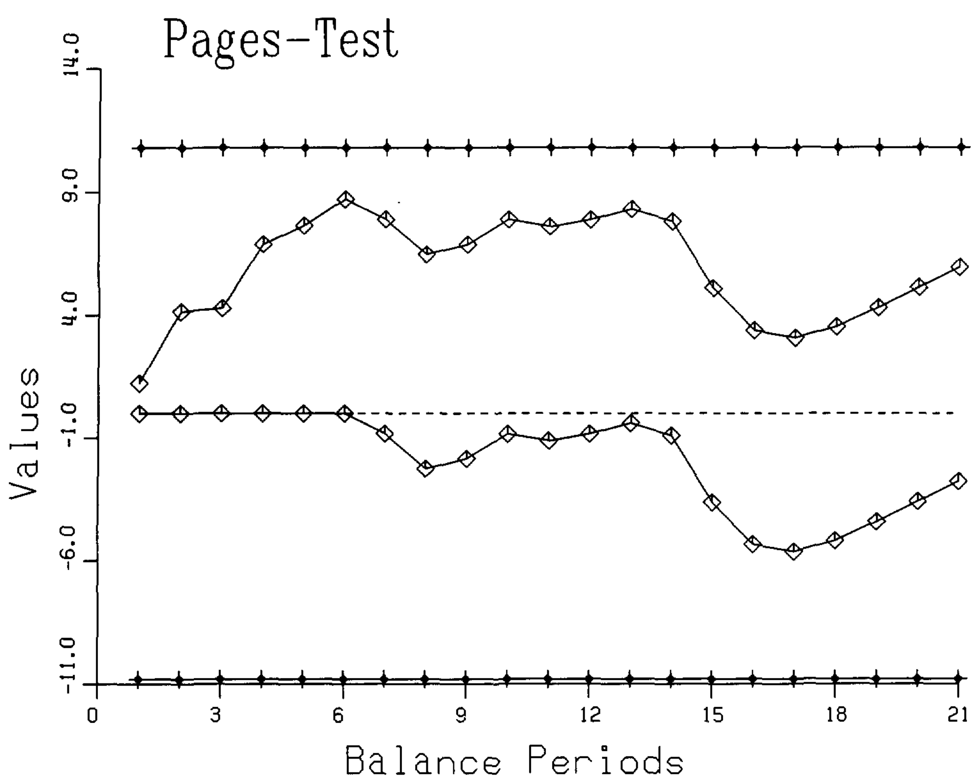

Figure 6a
Results of the Page's Test with the Modified MUF Series

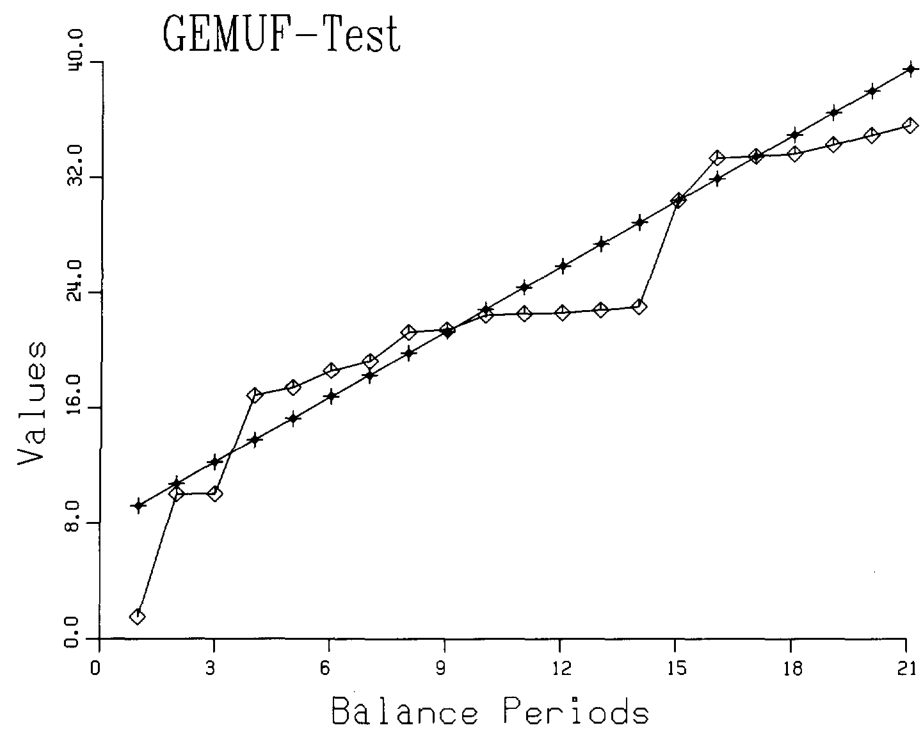

Figure 6b
Results of the GEMUF Test with the Modified MUF Series

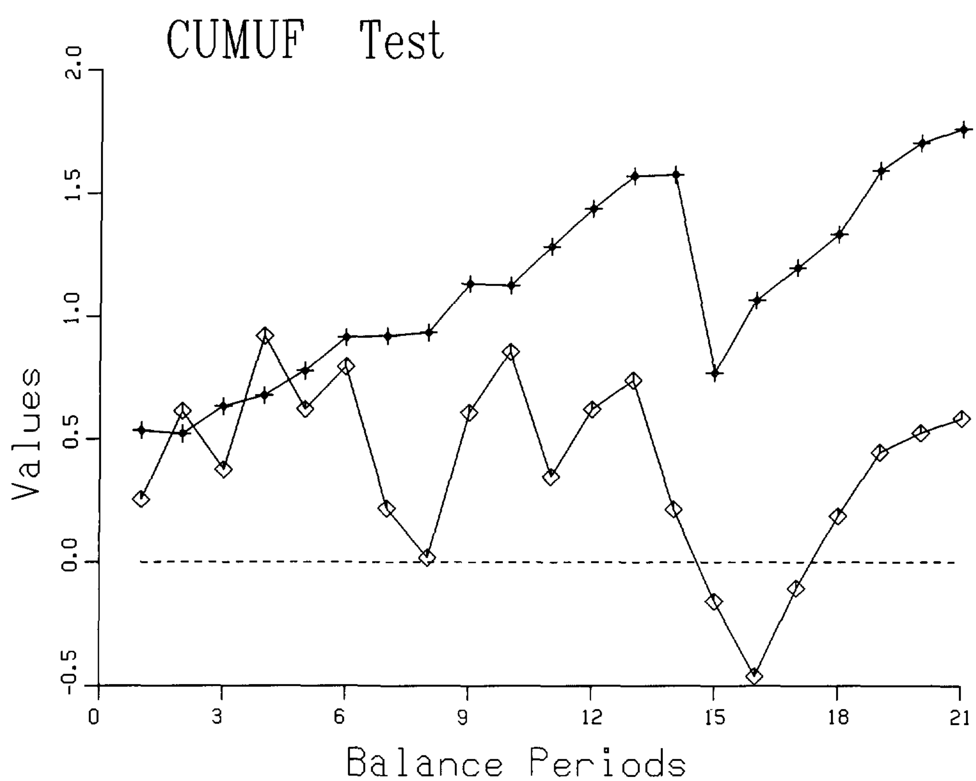

Figure 6c
Results of the CUMUF Test with the Modified MUF Series

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D. Schmidt received his physics degree in 1987 from the Technical University Berlin, where he conducted research in the field of numerical simulation of energy distribution in large stellar systems. He joined WAK in 1988. As deputy head of the MC&A Group, Schmidt has worked on problems of statistical significance of material balance differences. Presently, he is in charge of nuclear safety aspects with particular involvement in unusual operation occurrences.

Automating Large-Scale LEMUF Calculations

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ABSTRACT

To better understand material unaccounted for (MUFs) and, in some cases, to comply with formal regulatory requirements, many facilities are paying increasing attention to software for MUF evaluation. Activities related to improving understanding of MUFs are generic (including the identification, by name, of individual measured values and individual special nuclear material (SNM) items in a data base, and the handling of a wide variety of accounting problems) as well as facility-specific (including interfacing a facility's data base to a computational "engine" and subsequent uses of that engine). Los Alamos efforts to develop a practical engine are reviewed and some of the lessons learned during that development are described. Major obstacles to implementation do not involve lack of software or statistical theory, but the lack of resources required for implementation and (in some cases) realistic measurement uncertainties.

1. BACKGROUND

For many years, nuclear facilities have computed MUFs for materials accounting. Interpretation of these MUFs can be very difficult. The familiar complaints: measurement and sampling uncertainties are sometimes poorly quantified, effects of using "historical factors" and "nominal values" in lieu of direct measurements can be incompletely understood, post-period bookkeeping adjustments must be considered to reflect analytical measurements received after the close of the accounting period or to correct for past clerical mistakes, impacts of unmeasured holdup may be nontrivial, and so on.

Note that the above accounting problems are magnified in a near-real-time environment with frequent balance closures. Results from an analytical laboratory are less likely to be received during an accounting period when that period is short, increasing the number of post-period adjustments of nominal values. It becomes impractical to make holdup measurements at the close of each accounting period. And evaluation for material loss in a near-real-time environment extends beyond a single period's MUF to sequential testing over several periods to detect possible protracted losses. All of these issues complicate MUF evaluation.

Nonetheless, it is impossible to understand MUFs without understanding the role of measurement uncertainties. In the United States, recent efforts have made the understanding of uncertainty a regulatory requirement. Section II.6.b(2) of Department of Energy Order 5633.3 discusses MUF (domestically called an "inventory difference") and LEMUF (defining associated "control limits"). That section states that "control limits shall be based on variance propagation or any other statistically valid technique." Increased auditing activities, such as by inspections and enforcement teams in the United States, may require facilities to explain their control limit determinations.

Automating large-scale LEMUF calculations is nontrivial. Some problems to be overcome relate to incomplete knowledge of needed measurement uncertainties. Others relate to automating access to information required for propagation of uncertainties. Still others relate to the role of an error propagation "engine" to provide desired calculations. And still others relate, more generally, to establishing a positive mindset about improved safeguards. This paper reviews these problems and suggests approaches to resolving them.

2. COMPUTATIONAL GENERALITY

The evaluation of uncertainty in observed quantities is a problem common to many scientific endeavors. Several of those areas have developed their own terminology for this evaluation. Labels such as "the delta method," "the method of statistical differentials," and "statistical tolerancing" are used in other fields. The phrase "error propagation" is of physics and engineering origin, is standard to International Atomic Energy Agency (IAEA) documentation, and until recently had been politically correct in the United States[1]; owing to the nature of this forum, international terminology is used herein.

Despite the widespread interest in propagation, large-scale calculations are apparently unique to safeguards. Consequently, there are no generic software packages that can be readily applied to facility-specific problems. In response to the needs of the safeguards community, several implementations have evolved.[1] Approaches to the problem embodied in those implementations may provide insights to those interested in developing or improving their own software. Approaches to error modeling (e.g., Chapter 5 of Reference 2) also should be studied.

The single most important design issue concerns the degree of sophistication to be built into the system. On one hand, complete generality in terms of measurement structures and error modeling would require the user to specify individual variances and covariances for (potentially) thousands of measurement values each accounting period. Automated data input, provided in a standardized format, is not feasible for large problems. On the other hand, systems-study codes with very simple input and simple propagation formulas are easy to use, but force-fitting realistic problems into those codes may lead to poor results. The trade-off between computational generality and manageable input drives many subsequent software decisions, and there is a considerable range in the degrees of sophistication present in codes described in the literature.

Computational generality also affects the ability to handle propagation problems not envisioned at the time of code development. Such surprises are usually less common at production facilities, where the types of activities change comparatively slowly, than elsewhere. Limited, inflexible software leads eventually to code modification, with accompanying validation efforts and revised documentation. My experience is that it is impossible to completely anticipate all propagation needs and, over the long term, generality is very important.

Another issue concerns the nature of desired output. In some ways, it is more simple if report writing can be automated, thus involving minimal effort on the part of facility personnel. In other ways, it is often desirable to insist personnel examine input data and confirm that calculations are performed as desired. Also, it is beneficial to include analysis and output options so that users can extract results they want from the engine — no more and no less. The former approach is similar to batch computing, the latter to interactive computing.

3. THE COMPUTATIONAL ENGINE

It is useful to distinguish facility-specific requirements from generic propagation needs. Requirements such as the desired input data format (tailored to the nature of accounting at the facility) and output of results (report writing, graphics, and so on) should not be formally incorporated into a computational "engine."

The role of an engine is to handle generic calculations. An engine can be developed and improved in-house or externally. Advantages of local involvement include the familiarity of on-site personnel with local needs and problems, which is helpful in adapting a site-specific engine. Disadvantages of in-house development include some aspects of reinventing the wheel, almost unavoidable in first-time efforts. Also, at many facilities the personnel having expertise in statistics, accounting, and computer science may be few, and removing them from existing responsibilities to develop and document software may be difficult — especially in an era where many safeguards organizations are understaffed.

The engine should be separated from the other components of a larger system. The separation has several advantages. An engine can be validated once (for internal edification and external auditors) instead of repeatedly (as would be done if a specialized code were created for each new computational problem to come along). From a security standpoint, input data and output results may be classified, but the engine itself need not be, thereby simplifying handling of software.

Effective functioning of an engine requires the accounting data base to supply necessary information in the proper format and in a timely fashion. Some accounting systems in the United States, for example, can not provide such information without considerable manual effort by operations and accountability personnel. This is because many computerized accounting systems were designed years ago, when the interest in uncertainty evaluation was very different than at present.

Two issues arise with respect to older systems. First, there is a natural aversion to change when such change involves substantial effort. Resistance may arise from those who see no need for understanding uncertainties because such understanding was not required in the past. Conquering such resistance and establishing a positive mindset can be nontrivial. As physicist Max Planck once wrote[3] regarding the progress of science, "A new scientific truth does not triumph by convincing the opponents and making them see the light, but rather because its opponents eventually die, and a new generation grows up that is familiar with it."

A second, more technical aspect of older systems concerns their emphasis on materials control functions. Queries regarding an item's location (Where in the facility is it?), custodianship (Who possesses it?), authorization (Is the custodian authorized to take certain actions with it?), and audit trails (What is the history of past activities?) can be promptly resolved. Integration of data error checking (e.g., is the instrument code attached to a measured value a legal one?) and containment/surveillance indicators are also useful, of course.

Materials control information is important for its intended purpose but, unfortunately, has little relevance to LEMUF evaluation. A very different information base is required for uncertainty propagation. Each item's SNM content must be described in terms of "raw measurements," such as gross and tare weights, concentrations, isotopics, bias corrections, manometer readings, calibration constants, nominal values used pending receipt of results from analytical chemistry, and so on. To propagate correctly, the engine must know which measurement instruments, with which uncertainty parameters, are used for each raw measurement. It must also know if a given measurement is used multiple times, as when a batch concentration is applied to several individual items or the same bias correction is used for a specific instrument. Static items (e.g., items that do not affect materials unaccounted for such as those in both beginning and ending inventory) must also be identified.

A separate piece of software, called an interface, is required to extract propagation information from the accounting data base and input that information to a calculational engine. For large-scale calculations (thousands of measured values per accounting period), a manual interface is time consuming, awkward and prone to clerical errors. As such, the interfacing should be automated. Automation could also expedite transfer of information to domestic regulators and to the IAEA. Such modernization may be nontrivial, especially for older systems designed for materials control and not for propagation.

Each facility's data base has its own eccentricities, which makes an interface highly facility-dependent. But because the core of information required for LEMUF calculation is determined by the MUF equation and associated measurement error modeling (and not by the specific engine used), the interface issue is distinct from the engine issue.

Input files constructed by an interface should uniquely identify individual measurements, such as weights, concentrations, and so on. For each measurement, the corresponding measurement instrument should be identified and cross referenced to a file containing measurement uncertainties. Variances and covariances can then be computed. Use of separate input files for measured values and for uncertainty information allows for those files to be easily edited to correct mistakes in the data; calculations can then be redone for the corrected input.

As a side benefit of single-period propagation, information needed for propagating uncertainties for each individual MUF in a series of MUFs is sufficient to propagate for the entire series. At that point, sequential tests that have appeared in the safeguards literature can be implemented. As noted previously, such testing is valuable in a near-real-time setting with short accounting periods.

Capabilities for sequential testing can also be exploited to better allocate computer runtime and to identify anomalies. For example, an individual MUF can be expressed as a sum of (sequential) MUFs over shorter time periods or over batch operations. By examining sequential MUFs as they become available, potential anomalies can be dealt with before the end of an accounting period. Also, the variance of the larger MUF is simply the sum of the elements in the covariance matrix for the sequential MUFs, portions of which can be computed as the sequential MUFs are observed. Thus, the entire LEMUF calculation need not be done in a single run at the end of the accounting period.

4. LOS ALAMOS EFFORTS

A few years ago, the Safeguards System Group at Los Alamos began to build materials accounting with sequential testing (MAWST), a computational engine designed along the lines of the previous section. That engine, christened "Mae West" by Tom Marr, was designed to handle difficult, real-world propagation problems. That is, the simplifying assumptions common to calculations in systems studies are avoided — assumptions such as steady-state operation involving homogeneous sets of items. A drawback to this approach is that sophisticated codes can run comparatively slowly when a simplified structure exists and can be exploited, but this is a small price to pay for maintaining generality and preserving expandability to unforeseen problems.

MAWST's formal treatment of propagation is described in detail elsewhere.[1,4] Many lessons have been learned from using MAWST on a variety of problems at a variety of locations. Some of these lessons are reviewed below.

The programming of the engine should be done by computer scientists and not by statisticians. It is not only necessary that the engine produce correct results (an area with which statisticians are familiar), but also that runtime is minimized by computationally efficient handling of large data arrays, file structures, and so on (areas with which they often are not). Code users experienced in MUF evaluation, perhaps from use of their own specialized propagation software, are very helpful in formulating required capabilities for an engine and in validating an engine.

Validation is essential. Here, validation is interpreted as confirming that output results are correct. External review of a source code is helpful and practical (for example, the original MAWST had less that 1,000 lines of executable FORTRAN), and provides one check on the algorithms used. Running test cases is a must. Many regulatory bodies, such as the U.S. Department of Energy, have no formal certification process for propagation software, leaving facilities on their own. More generally, regulators and inspectors are not in a position to take extensive facility data and reproduce claimed results using their own software. Published QA guidelines exist, such as by Institute of Electrical and Electronic Engineers,[5] and private companies may have their own internal policies. But explicit guidance on validation is usually lacking.

Input file structure affects computer runtime for large problems. That is, different data files can correspond to the same problem and lead to the same end result, but some of those files may run more quickly than others. For example, removal of static items from the input file is much more efficient than grinding through the related variance and covariance terms for those items. The process of computing measurement covariances by comparing two measured values' corresponding measurement instruments and then cross referencing to uncertainty values can be expedited by sorting the relevant input files. This sorting is done to more efficiently search through arrays.

Runtime can also be improved by manipulating code-related bookkeeping. For example, measured values from different measurement instruments are often independent. Under such independence, the usual propagation leads to a variance decomposition where the variance of MUF is a sum of contributions attributable to individual measurement instruments. This decomposition is useful in understanding variability. A sophisticated engine is capable of carrying out the decomposition, though the bookkeeping is nontrivial compared to other calculations.

One situation where such bookkeeping is beneficial concerns poorly quantified measurement uncertainties. Measurement control activities are sometimes designed to ensure instrument stability, and not to provide estimates of needed uncertainty parameters. Garbage-in-garbage-out numbers result from the use of poor quality uncertainty values. By computing variance contributions from individual instruments and, in the manner of a sensitivity study, recomputing those contributions for different uncertainty values, it can be determined whether LEMUF is sensitive to assumed uncertainties. Such bookkeeping should be an option for the user, who can decide if the additional information is worth the additional runtime.

With respect to Los Alamos efforts, John Hafer is working to improve the MAWST engine (avoiding FORTRAN and linear arrays for storing and retrieving information, for example) to run large problems on a personal computer. As hardware improves, the need for coding efficiency should diminish. Improvements in software (newer versions of the C language, for example) should also help.

5. FINAL REMARKS

Retrofitting accounting systems not designed for uncertainty evaluation presents technical, economic and political difficulties. From a technical standpoint, obtaining realistic uncertainty values for a facility's accounting data base and constructing an automated interface between that data base and a calculational engine are difficult. On the economic side, developing a large-scale, computerized accounting system requires time, money and people. Finally, in what is basically a political issue, the will to solve the above problems must be summoned.

Importantly, these obstacles are by no means insurmountable and headway is being made. Two of the more advanced systems, located at the Savannah River Site (in the United States) and at the Plutonium Fuel Production Facility (in Japan) have automated systems capable of supplying information for LEMUF calculations.

Automated MUF evaluation is still in its infancy. Time and experience with the subject will produce better software.

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R. R. Picard joined the Los Alamos National Laboratory in 1981 after earning his B.A. in mathematics from Carleton College and a Ph.D. in applied statistics from the University of Minnesota. He has since worked on a variety of safeguards problems with the Safeguards Systems Group and on arms control problems with the Statistics Group.

Victoreen introduces Model 700 Spectrometer

Victoreen Inc. recently introduced the MICROSPEC-1 Model 700 Portable MCA Spectrometer System. The system includes a two-inch Nal detector, an on-board notepad computer with applicable software and rechargeable batteries.

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