JNMM

Journal of Nuclear Materials Management

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More than spent fuel

It would be a great help to me if some of you who receive this *Journal* would tell me what subjects interest you. As it is, the technical articles we publish are those contributed or solicited which appear to the editors to be of interest to at least some of our members. I have heard some grumbling, recently, because the *Journal* fell behind schedule. This suggests that some of you look at it. We hope to be back on schedule with this issue and to remain so. However, it would be nice to hear more comments more frequently.

This issue contains papers on an assortment of topics. Two of them are contributions. The other three were presented at the annual INMM Spent Fuel Management Seminar in January. It may interest you to know why I consider the five to be of broad interest to several of the subject areas which the Institute represents.

The subject of IAEA safeguards is one such topical area which impinges on the national safeguards, spent fuel management and arms control verification areas. One subject which the IAEA has had difficulty in dealing with is nuclear waste discards. As is explained in the article by A. Fattah and N. Khlebnikov, the documents which define the Agency-State agreements for NPT and non-NPT States specify that safeguards may be terminated on nuclear materials which have been consumed or diluted or otherwise no longer have safeguards significance. The problem has been to decide what these criteria mean in practice. As a start, the Agency has concluded that certain small quantities, defined in facility attachments, may be discarded without notifying the Agency in advance. In the fall of 1988, it convened the first of several meetings of representatives of interested member States to define practical criteria for termination of safeguards on more significant quantities. The authors describe what has resulted so far from this effort. The member's representatives have agreed on some of the criteria but not on all of them. That is, some States wish to be more strict than others. The authors suggest that the Agency might adopt the criteria on which all agree and deal with those cases where there is disagreement, individually, as they arise. This appears to be a sensible proposal until wider agreement can be attained among the interested parties. Of course, all agree that waste discards should be measured by the operators of the facility where they are generated and that the IAEA should have the right to verify such measurements. If safeguards are not terminated on waste discards, continuing effort would be required of the Agency and of the facility operators.

McKenzie, Hartwigsen and Lowe have contributed a very interesting paper on ultrasonic seals. The idea of designing seals with unique ultrasonic signatures is quite old. Scientists at the Euratom Ispra Research Center began developing them for use on LWR fuel assemblies in about 1969. The seals were attached to the object in such a manner that removal would be evident. A seal consisted, for example, of a lighter metal with heavier particles randomly distributed within it. A seal is interrogated with pulses of ultrasound, and the pattern of the reflected sound should be very similar to that previously recorded.

One of the problems has been how to design the seal and the interrogation or reading instrument so that it would be easy to make good acoustical coupling between them and to reliably distinguish between seal signatures in the field. As the article describes, several different types of seals have been developed for different applications, one of which, as designed, needed some means to decide when to start

reading the signature. This stimulated the idea of converting the signature which arrived sequentially to a frequency signature by means of a Fourier transform. This not only solved that problem, but also appears to be an improved method for distinguishing the signals from other acoustic seal designs.

Seals have increasing importance for national, as well as international, safeguards. They probably will be increasingly important for verifying the identity of spent fuel items destined for dry storage or burial in a repository. They have become of great interest for a number of applications in nuclear arms reduction agreements.

In addition to the technical articles, this issue contains a summary of the information presented at the Spent Fuel Management Seminar in January. This seminar is of particular interest to those involved in developing the U.S. civil radioactive waste disposal program and to the nuclear power reactor operators who are paying for this in the face of considerable uncertainty as to its political and technical future.

We have selected three papers from the Spent Fuel Management Seminar which should be of interest to other Institute members, as well.

One paper concerns the electric power plants in the United States owned and operated by private utilities. Some are owned and operated by public utilities, that is, by cities. The city of Sacramento, Calif., constructed a nuclear power plant, as described in the paper by Bowser, Keuter and Miller. Last year, the citizens of Sacramento decided that the city should sell or dispose of its nuclear power plant. No one was willing to buy it, so the public utility had to decide how to dispose of the spent fuel and the reactor. This is an interesting example of how public opinion may affect nuclear power. It is also interesting to learn how those faced with this problem decided what to do with the reactor and the spent fuel. It is uncertain as to when the government will take responsibility for the spent fuel and other wastes.

The other two papers from the Spent Fuel Seminar are technical papers concerning "allowance for burnup" and "non-fuel-bearing wastes." The first of these has to do with the costs and effort required to ship or to handle spent fuel. A primary concern is to avoid criticality accidents by geometric designs or by the use of neutron poisons. Since reactivity is reduced considerably by burnup in a reactor, more fuel assemblies could be placed in a cask for shipment or storage without adding poisons. The IAEA is interested in burnup calculations as a means of verifying the input to a reprocessing plant and also as regards how many fuel assemblies may be placed in a shipping or storage cask.

It has finally dawned on me that non-fuel bearing wastes may present some problems for national and international safeguards. The reason is that spent fuel and the other wastes are to be disposed of in a geological repository in the United States and in several other countries. As was noted above, the IAEA may decide to terminate safeguards on nuclear wastes which meet certain criteria. However, if such wastes are to be handled along with spent fuel items for transportation, processing and burial, it will probably be necessary for the Agency to distinguish them from each other or to confirm that there is no significant amount of fissile material in the non-fuel-bearing wastes. This possibility should be of interest to many of those engaged in the development of safeguards measures and techniques.

Finally, I wish to thank Michael Franklin, who has asked to be relieved of his duties as an associate editor, for his many years of service. He has provided especially valuable advice regarding a number of the papers we have received relating to material accounting.

Dr. William A. Higinbotham
Brookhaven National Laboratory
Upton, New York U.S.A.

Potpourri

I'm excited — really excited — about the 32nd Annual Meeting in New Orleans. Despite all of the gloomy forecasts, I'm convinced we're going to have another great meeting. The Technical Program Committee has met and selected more than 220 papers in all of the areas that 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; arms control and disarmament; and the environment and ES&H. The many side meetings that usually are scheduled around the Annual Meeting are being arranged. An unusually large number of exhibitors have already signed up. And, on the lighter side, the local arrangements are well under way: Monday evening on the Mississippi on a sternwheeler and a spouses' program that includes a Cajun cooking class. Most everyone I have talked to recently is planning to be there — I hope you are too.

As I mentioned in my last column, our Long-Range Planning Committee recommended that INMM consider reorganizing into divisions to facilitate more fully incorporating elements of nuclear materials management other than safeguards and security (transportation and waste management, for example). At its March meeting, after several hours of discussion, the Executive Committee concluded that reorganizing into divisions might be too divisive (pun more or less intended) and could further separate us rather than draw us closer together.

The Committee decided to investigate alternatives — perhaps creating sections with appropriate representation on the Executive Committee. Jim Tape's ad hoc committee will examine possibilities and report to the Executive Committee in July. Please share your ideas with Jim or me. Either of us can be reached at Los Alamos National Laboratory, MS E550, Los Alamos, NM, 87554, U.S.A.

As I draft these thoughts, the Technical Workshop on Materials Control and Accountability is just finishing in Atlanta, with more than 80 safeguards professionals participating. And by the time you read this, the workshop on Mass Measurements will have also been completed. These technical workshops always address topics and issues at the leading edge of our technology. They are an important part of the Institute's program and provide a means of professional growth for INMM members (and others). I hope you take advantage of the time and effort that goes into each of these workshops.

If it's not too early to start thinking about next year, we most likely will return to the Stouffer's Resort Hotel in Orlando, Fla., for our 33rd Annual Meeting. If so, the dates will be July 19-22.

See you in New Orleans.

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

Progress in spent fuel storage and disposal

The eighth INMM Seminar on Spent Fuel Management was held Jan. 16-18, 1991 in Washington, D.C., and several of the significant papers from that meeting are included in this issue. This Seminar added to the continuing chronicles on the evolving technology and status of the back end of the nuclear fuel cycle, including the storage, transport and disposal of spent nuclear fuel.

It has been the conventional practice to view progress in nuclear waste disposal in largely technical terms: how close are we to setting rebar, pouring concrete and accepting spent fuel. Measured in these narrow terms, we are not much closer to the goal than we were 8 years ago, when the Nuclear Waste Policy Act was passed. However, it is becoming more and more apparent that we have defined progress too narrowly. We are in fact developing something broader and much more fundamental: a process by which a democratic society sites and operates a major national facility that is perceived by the general public as, at best, a liability. This process involves a complex combination of technical and political issues. In this process, technology is the apparent issue, good technology is essential, but even a perfect technology (if such exists) could not possibly result, by itself, in acceptance: good technology, though essential, is not the primary issue. The overriding issues are political: siting and operation will not be achievable until the political process moves forward to resolution.

Assuming that we are engaged in such a process, we have made real progress in a number of ways. With the passage of the Nuclear Waste Policy Act Amendments in late 1987, including the designation of a single repository site for evaluation, and the authorization of a broadly empowered waste negotiator, we have begun the second iteration of the political part of the process. In effect, Congress has selected two parallel possibilities: a process of federal imminent domain at the federal-state level and a process of negotiation including possible benefits in compensation for the perceived liabilities. Either of these could succeed independently, or in combination.

In the technical area, our understanding of the technical issues has increased. In program management, DOE has become much more articulate in distinguishing between schedule milestones that DOE can control (which tend to be technical, and those that it cannot (which tend to be political). And finally, the remarkable fertility of the judicial process in creating diverse sources of deliberate delay has become obvious and a source of frustration for even the federal legislative branch.

Participants in the Spent Fuel Management Seminar heard from two of the most important leaders in this evolving process, both of whom have been appointed within the past year: Dr. John Bartlett, director of OCRWM, and David Leroy, the nuclear waste negotiator. After describing the technical and management improvements being initiated, John Bartlett focused on the key critical path item in the repository schedule: obtaining the state permits necessary to characterize the candidate Yucca Mountain repository site. In September 1990, the 9th Circuit Court rejected all of the State of Nevada claims that underlay its refusal to grant access permits to DOE. On Dec. 15, 1990, Nevada filed an appeal to the U.S. Supreme Court. (On March 4, 1991, the Supreme Court declined to review the case. Presumably, DOE's suit in the District Court to compel Nevada to issue the required permits will now proceed to a positive outcome.) In parallel, DOE has drafted and is proposing legislation to assure access for site characterization. John Bartlett also noted that, on the technical side, DOE has everything ready to proceed with site characterization, once procedural barriers are removed.

David Leroy, the recently-appointed nuclear waste negotiator, described his role and his approach to finding an MRS and/or repository site on terms that are acceptable to the host state and to the local government and/or Indian tribe. It is quite evident that David Leroy sees his task in the largest context — not just the finding of suitable sites, but also the establishment of a sound process for doing this in the future. He emphasized his independence from DOE, including the fact that he will not be a promoter of an MRS or repository; he will, however, be both a promoter and an advocate of an agreement for the siting of an MRS and/or repository. He intends to use all available resources of the Federal government, including the possible commitment of favorably perceived federal facilities to states or regions, in order to achieve that end.

When the definition of progress is expanded to include the political process, we have made significant progress, predominantly in the political area. However, because this process does not have a predictable end point, we cannot know how far along we are to that end point. Hopes and expectations are high for the success of the currently defined process and for the waste negotiator in particular. However, based on history, we probably can anticipate continuing surprises and court-related delays.

With respect to progress in spent fuel storage technology, the past year has continued the prior history of technical innovation and cost cutting among the competing storage technologies and service firms. A notable event of the past year is the return of the metal storage cask to the winner's circle of cost-competitive storage technologies. Jon Kapitz of Northern States Power described NSP's procurement approach and its outcome: the commitment to use up to 48 metal storage casks of 40-assembly capacity, supplied by Transnuclear. The sources of the low evaluated unit storage cost appear to be: the large size of the procurement, the large capacity of the cask and the inexpensive basket design. The latter is notable in that although licensable for storage, it could not likely be licensed for transport for both structural and criticality safety reasons. This serves to emphasize, again, the growing divergence between the goals of low cost storage and the goal of ultimate transportability of the metal storage casks.

Progress continues with the concrete storage cask. The first loadings of the large (24 PWR) NMHOMS modules were successfully completed at Duke Power's Oconee site. Licensing of the storage facility continues at Baltimore Gas & Electric's Calvert Cliffs station, using that same technology. NRC licensing review of the vertical concrete cask continues, although the license application has been withdrawn for the lead site to use this cask. Spent Fuel Management Seminar participants also received updates on other concrete casks, modular vault storage and the transportable storage cask and on recent operations at the original Interim Spent Fuel Storage Installation at Virginia Power's Surry site. Spent fuel consolidation continues to be the primary intended storage technology of some utilities, and progress in fuel consolidation equipment was described, including the use of commercial robots.

Chris Kouts, chief of DOE's Transportation Branch, chaired the session on transportation status and gave the lead paper, noting the goal of having DOE's truck and rail casks certified by NRC in 1994, with prototype models completed by 1995 and a transportation system in place by 1998 to support MRS waste acceptance. Descriptions and updates were also provided on the DOE truck and rail casks, including burnup credit issues, operational planning, cask maintenance and the interfacing of the DOE's transport system with the specific transportation and on-site cask handling capabilities at each utility site.

Although much of its work and issue resolution is still in the future, DOE's Transportation Program appears to have good prospects for being both ready and capable of handling the wide diversity of waste transport circumstances at the various utility sites, when DOE begins its initial acceptance of spent fuel.

N. Barrie McLeod
E. R. Johnson Associates Inc.
Oakton, Virginia U.S.A.

Technical Working Group: Physical Protection

The currently scheduled and planned activities of the Technical Working Group on Physical Protection are listed below:

• 32nd Annual Meeting of the INMM will be held July 28-31, 1991, at the Fairmont Hotel, New Orleans. Approximately 55 physical security related papers will be presented.

• No workshops are scheduled at this time due to the smaller attendance at recent workshops. This decrease in our attendance is attributed to the increase in similar workshops by ASIS, ANS and the trade magazines.

• A workshop, "Package Search Techniques," is currently being considered but has not been scheduled. Such a workshop would concentrate on better and more effective methods of searching packages which enter restricted areas. If you have an interest in such a workshop, please contact Donald Rasum, Nuclear Regulatory Commission, (301) 492-3379.

Workshops on other subjects of interest to physical protection personnel will be considered if enough interest is expressed. Additional details about group activities are given below.

General

The 32nd Annual Meeting of the Institute of Nuclear Materials Management will be held July 28-31, 1991, at the Fairmont Hotel, New Orleans. A broad range of physical protection papers will be presented.

We have been working with personnel from the nuclear power industry to follow up on a proposal submitted to the Executive Committee about one year ago. In particular, we are proposing a "consortium" of the operating companies which would be assisted by INMM in getting organized and in conducting workshops, etc., in the physical protection area. After several false starts, I finally contacted Barry Saunders, Nuclear Security Administrator, Commonwealth Edison, c/o Dresden Nuclear Power Station, R.R. #1, Morris, IL 60450. His telephone number is (815) 942-2920 Ext. 2744.

Security Personnel Training

The next workshop on this topic will probably be scheduled in the spring of 1992.

James D. Williams, Chairman
Sandia National Laboratories
Albuquerque, New Mexico U.S.A.

Technical Working Group: Radioactive Waste Management

The following summarizes the activities of the Technical Working Group (TWG) on Radioactive Waste Management for the period November 1990 through March 1991.

• The TWG organized and held the INMM Spent Fuel Management Seminar VIII at Loew's L'Enfant Plaza Hotel in Washington, D.C. on Jan. 16-18, 1991. Dr. John Bartlett, director of the DOE Office of Civilian Radioactive Waste Management, participated in the seminar as our kick-off speaker the first morning. David Leroy, who was recently appointed as the nuclear waste negotiator, was our luncheon speaker on Thursday, Jan. 17. Approximately 145 seminar attendees enjoyed these two presentations as well as those of the 28 other speakers giving presentations. Preliminary arrangements have already been initiated for the INMM Spent Fuel Management Seminar IX, which has the tentative dates of Jan. 15-17, 1992.

• The TWG is continuing to provide INMM co-sponsor representation on the Steering Committee for the 1991 International High Level Radioactive Waste Management Conference. This conference is scheduled for April 28-May 3, 1991 at Caesar's Palace in Las Vegas.

• The TWG is in the process of organizing the Waste Management Session of the 1991 INMM Annual Meeting to be held July 28-31, 1991 at The Fairmont Hotel in New Orleans. The Waste Management Session includes four different subsessions — Waste Management Systems and Technology, Spent Fuel Burnup Measurements, Transportation/Waste Acceptance Infrastructure and Panel on Material Control and Accountability for Spent Fuel. Speakers have been contacted and session chairs assigned.

E.R. Johnson, Chairman
E.R. Johnson Associates
Oakton, Virginia U.S.A.

Technical Working Group: Transportation

In July 1990, legislation instituting the Commercial Driver License (CDL) nationwide became effective. All states have two years to implement a testing process and procedures to develop and administer the program.

This will impact all U.S. DOE contractors because it requires all drivers of vehicles carrying hazardous materials to obtain a CDL. Also, drivers of vehicles capable of carrying more than 15 passengers and drivers of vehicles over 25,000 pounds gross vehicle weight (GVW) must obtain a CDL.

For more than five years, the Department of Energy has been evolving an official policy for how its contractors will either meet Department of Transportation (DOT) standards or provide equivalent safety while performing on-site movements, transfer or shipments. Many drafts of an order have been circulated for comment, and many things have been defined by DOE to facilitate implementation of the order when it is promulgated. Release is imminent, and full compliance by Oct. 1, 1992, is expected.

A workshop for all contractors was held by DOE during August 1990 to discuss this latest order draft and to provide an opportunity for comments, criticism and suggestions about the significant impact which is expected to occur.

A National Transportation Safety Committee was formed by DOE in September 1990 to address issues which evolved from the August meeting and to permit continual input from contractors and DOE operations offices.

A National Transportation Operations Committee was formed by DOE in November 1990 to discuss operational problems and to provide professionals an opportunity to aid DOE in implementing the new order for on-site movements, transfer and shipments.

In November 1990, the DOE held workshops around the country to instruct contractors about its forthcoming rules for vehicle inspection. These inspections must be conducted regularly for all DOE-licensed vehicles and performed on all non-DOE-licensed vehicles whenever they enter a DOE facility.

HM 181 was enacted into law on Dec. 21, 1990. This legislation will have major impact upon almost all United States specification packages by requiring performance-based testing criteria to be made.

Most Type B radioactive material packages meet the new standard, but the new requirements will cause all other packages to be certified before use. A transition period of five years has been allowed, but much work must be done
1) to teach the industry about these rules and how to apply them
2) to test packages to assure all criteria are me and
3) to develop new packages for those not meeting the new standards.

The transportation of radioactive materials is important to the U.S. DOE because it is in transport that all of its programs become subject to public scrutiny. No life has ever been lost as a result of the transportation of radioactive materials, and the foregoing initiatives will help to assure that the record will continue.

As professionals serving the U.S. DOE, we must embrace, enact and ensure compliance with the new DOE orders and congressional legislation.

The Institute of Nuclear Materials Management is considering methods of addressing transportation issues and the importance that DOE has placed upon transportation. We look forward to transportation playing a greater role in the Institute.

Francis Kovak, Chairman
Oak Ridge National Laboratory
Piketon, Ohio U.S.A

Committees: Safeguards

The INMM Safeguards Committee met in the Nuclear Regulatory Commission (NRC) offices in Rockville, Md., on Thursday, March 7,1991. The following is an agenda of that meeting:

Introduction
R. Burnett, NRC
Regulatory Effectiveness
D. Orrik, M. Warren, NRC
Regulatory Development
T. Sherr/P. Ting, NRC
- Enrichment Rule
- Physical Fitness Rule
- Fuel Cycle Fitness for Duty Rule
- Category I Transportation Rule
- Rule Making Petition
- Design Basis Threat for Sabotage
- Miscellaneous Amendment 10 CFR Part 73 & 74

DOE Safeguards Activities
D. Myers, DOE
Safeguards Committee Discussions

R. Burnett, NRC, presented an overview of recent activities including the impacts surrounding Desert Storm to both Cat I and reactor facilities. The Commission is in the process of responding to a congressional petition for changing the base case threat at nuclear facilities. This response to Congress will occur in the May to June time frame. Burnett also discussed the Regulatory Recovery Act in which the government recovers their cost through fees to their licensees. This process is in the early stages, but will certainly impact industry budgets for FY 91.

Dave Orrik, NRC, discussed regulatory effectiveness activities. Three areas of emphasis were
l) armed response,
2) intrusion detection and
3) assessment. Armed response included target analysis, weapons capabilities and the use of weapons under stress conditions. A videotape was presented which illustrated these activities. Shared learning experience among sites is an important element of

this program. Ted Sherr and Phil Ting provided an excellent overview of domestic safeguards activities. These included the enrichment rule, the physical fitness rule, fuel cycle fitness for duty rule, Category I transportation rule, the rule-making petition for a new design basis threat and miscellaneous amendments to 10 CFR Part 73 and 74.

David Myers, DOE, International Safeguards Branch, provided an excellent overview of current safeguards activities within the DOE. The DOE is going to performance-oriented rules and is defining performance measures, broadening the scope of the requirements, setting minimum performances and redefining the nuclear description of attractive material levels. They also are performing a systemwide evaluation of the material control and accounting systems. All of the DOE orders and standards and criteria are undergoing changes. The Office of Safeguards and Security is developing a five-year integrated safeguards plan. The safeguards organization within DOE is in the process of a major new reorganization.

Industry representatives provide feedback to the NRC on current issues or prior problems. This meeting was well worth the time invested to better understand NRC's positions.

The next Safeguards Committee meeting will be held at the upcoming INMM Annual Meeting.

Leon D. Chapman, Chairman
Sandia National Laboratories
Albuquerque, New Mexico U.S.A.

Committees: International Safeguards Subcommittee

Last year, the INMM International Safeguards Subcommittee was organized and held its first organizational meeting at the conclusion of the 1990 INMM Annual Meeting. Based on the discussions at the July 1990 meeting, a revised Provisional Charter for this committee has been prepared. In this charter, there is a list of potential topics which we agreed could be considered for study and discussions.

The committee met in November 1990 to discuss its potential activities, including the potential of having a meeting in Europe in early 1991. We concluded that it would be best to have the next meeting immediately before the 1991 INMM Annual Meeting. We believe that the next decade of international safeguards will present a number of significant challenges which this committee can very effectively address.

We plan to hold the next meeting of this committee on Sunday, July 28, 1991, at the New Orleans Fairmont Hotel, from 1:00 to 5:00 p.m. It is our hope that in this meeting we can establish a limited list (one to three) of topics which we feel should be examined, and identify volunteers to study the selected topic(s). We believe that one of the prime topics that could be considered would be integrated safeguards technology (radiation detectors, optical surveillance, electronic seals, electronic transmission of tamper-protected safeguards data, etc.) that could support future safeguards effectiveness and potential facility operator/state acceptance of this technology.

We continue to believe that this committee, because of the vast experience of the participants, can make significant contributions in the promotion of international safeguards.

Cecil S. Sonnier, Provisional Chairman
Sandia National Laboratories
Albuquerque, New Mexico U.S.A.

International Safeguards Subcommittee Provisional Charter

Goal

The goal of the Subcommittee on International Safeguards is to promote International Safeguards within the INMM and the International Safeguards Community.

Charter

The INMM Subcommittee on International Safeguards has the responsibility to provide an informal forum for exchange of information related to further development of selected aspects of international safeguards and for enhancement of a broader understanding of these topics. The Subcommittee will examine the various technical issues so as to promote exchange of related information, e.g., through INMM Annual Meetings and workshops. The Subcommittee will endeavor to coordinate its activities with other groups involved in international safeguards.

Structure and Operation

The INMM Subcommittee on International Safeguards shall be initially comprised of the following:

- Chairperson — C. Sonnier
- Vice Chairperson — P. Ek

In subsequent meetings, a secretary for this Subcommittee will be selected.

The Subcommittee will meet at least two times per year, with one meeting being at the time of the INMM Annual Meeting and at a mutually agreed upon location where a significant number of participants will be present (e.g., IAEA Symposium Meetings, etc.).

The Subcommittee's initial endeavors will be to consider and recommend the structure and topics for international safeguards sessions for the subsequent INMM Annual Meeting and to study the topics which it should address, including:

- Integration of NDA and C/S technologies.
- Furtherance of the IAEA objective of characterizing the performance of containment and surveillance (C/S)

equipment.

• Furtherance of the IAEA objective of characterizing the performance of nondestructive assay (NDA) equipment.

• The potential effect of the proposed IAEA 1991-1995 Safeguards Criteria on future safeguards technology requirements and developments.

• Technology to support IAEA Safeguards Criteria beyond 1995.

• Experience of facility operators involved in the application of IAEA safeguards, and potential technology to improve safeguards effectiveness and efficiency for both the inspectorates and facility operators.

• International safeguards of transportation of nuclear materials and technological advances required.

• Challenges in application and implementation of state/operator-supplied equipment/data for safeguards use.

• State/operator assistance in performing inspector activities.

• Potential use of remote monitoring of safeguards information.

• Transfer and use of international safeguards experience and technology to other treaty verification applications.

• Means to strengthen the recognition of the INMM as a forum to discuss technology for international safeguards.

• Periodic workshops on international safeguards.

Coordination of the Subcommittee activities with the international safeguards community participants, meeting arrangements, notifications of all meetings and reports to the INMM Safeguards Committee will be the responsibility of the Subcommittee chairperson and vice chairperson.

It is anticipated that, once the Subcommittee has had several meetings, consideration will be given to the INMM sponsoring periodic workshops on selected international safeguards topics.

Committees: N14 Standards

1. An N14 Management Committee meeting was held April 24, 1991, in Germantown, Md. Plans were made for the INMM Annual Meeting to be held in July, in addition to standards planning activities.

2. There are currently 69 members, including alternates, which constitute the Balloting Committee. In addition, there are approximately 30 people from various organizations who receive copies of N14 activities.

3. The final report on the N14 scope change was reviewed at the April 24 management meeting and forwarded to ANSI for approval.

4. Highlights of N14 standards development are:

ANSI N14.1 - 1990 - Packaging of Uranium Hexaflouride for Transport - Approved by ANSI on June 21, 1990 and available from ANSI.

ANSI N14.2 - Tiedowns of Fissile and Radioactive Containers Greater Than One-Ton Truck Transport - Work is continuing on preparing a draft document for Writing Group approval.

ANSI N14.6 - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More for Nuclear Materials - This standard must be revised or reaffirmed in 1991. Planning on this activity has started.

ANSI N14.7 - Guide to the Design and Use of Shipping Packages for Type A Quantities of Radioactive Materials - Continued work on this draft is being re-evaluated including the selection of a new Writing Group chair.

ANSI N14.19 - Ancillary Features of Irradiated Shipping Casks - This standard must be revised or reaffirmed in 1991. Planning on this activity has started.

ANSI N14.23 - Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater Than One-Ton Truck Transport - Work on this draft is continuing within the Writing Group.

ANSI N14.24 - Barge Transport of Radioactive Materials - Plans for reaffirmation or revision of this standard have started with a standard completion date in 1991.

ANSI N14.25 - No activity

ANSI N14. 26 - Fabrication, Inspection and Preventative Maintenance of Packaging for Radioactive Materials - Work has started on preparing 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 must be revised or reaffirmed in 1991. Planning on this activity has started.

ANSI N14.30 - Design, Fabrication and Maintenance of Semi-Trailers Employed in the Highway Transport of Weight-Concentrated Radioactive Loads - A revised draft has been prepared and approved by the Writing Group.

The ANSI N14 Subcommittee for development of a numerical model for thermal evaluation of UF_6 cylinders is in process. A risk-benefit analysis for the transport of bare 10- and 14-ton cylinders containing UF_6 less than 1% ^{235}U has been approved by DOE and is awaiting funding.

Plans to revise the Standard Matrix for Light-Water Reactor Spent-Fuel Transportation are under way.

John W. Arendt, Chairman
Oak Ridge Associated Universities
Oak Ridge, Tennessee

Chapters: Pacific Northwest

The INMM Pacific Northwest Chapter held 1991 elections for officers. The following officers were elected:

Chairman	Bryan Smith
Vice Chairman	Debra Dickman
Secretary/Treasurer	Richard Hamilton

Ken Byers and John Ellis were elected to the board of directors. James Edgar remains on the board. Donald Six is past chairman.

The Chapter held a dinner meeting on March 19. The topic of discussion was waste management.

Curtis Colvin was appointed to the Tri City Technical Council as a representative of INMM.

Donald Six, Past Chairman
Pacific Northwest Laboratories
Richland, Washington

Spent Fuel Storage: A Decommissioning Perspective

■

Rita W. Bowser, Dan R. Keuter, Ken R. Miller
Sacramento Municipal Utility District
Sacramento, California U.S.A.

■

ABSTRACT

On June 6, 1989, the public voted to recommend that the Sacramento Municipal Utility District (District) no longer operate Rancho Seco Nuclear Generating Station as a nuclear power plant (although the vote did not prohibit sale to or operation by a qualified operator). Plant operators shut down the plant on June 7, 1989. Reactor defueling was subsequently completed on Dec. 8, 1989. A total of 493 fuel assemblies are now stored in the Spent Fuel Pool (in the Fuel Storage Building).

The original 10 CFR 50 operating license for Rancho Seco expires in the year 2008. Although the premature decommissioning of Rancho Seco presents some unique problems regarding the management of spent fuel, the District's decisions and lessons learned may benefit all facilities eventually facing decommissioning. The results of a review of this perspective well before decommissioning begins can be integrated into the decision-making process while still operating.

To close and ultimately decommission Rancho Seco as safely and economically as possible, the District developed objectives to support the related disposition of the spent fuel. The objectives are:

1) Minimize occupational and public radiation exposure,

2) Minimize decommissioning costs, including the need to maintain the spent fuel pool and

3) Prepare the fuel for Department of Energy (DOE) acceptance.

The District plans to use dual-purpose (combined storage/transport) casks to meet these objectives. This plan is contingent upon a successful DOE demonstration program that will resolve any outstanding NRC issues and provide sufficient evaluation to permit licensing of existing large DOE-compatible shipping casks as dual-purpose casks.

INTRODUCTION

On June 6, 1989, the public voted to recommend that the Sacramento Municipal Utility District no longer operate Rancho Seco Nuclear Generating Station as a nuclear power plant (although the vote did not prohibit sale to or operation by a qualified operator). Plant operators shut down the plant on June 7, 1989. Reactor defueling was subsequently completed on Dec. 8, 1989. A total of 493 fuel assemblies are now stored in the Spent Fuel Pool (in the Fuel Storage Building). Although the premature decommissioning of Rancho Seco presents some unique problems regarding the management of spent fuel, the District's decisions and lessons learned may benefit all facilities eventually facing decommissioning. The results of a review of this perspective well before decommissioning begins can be integrated into the decision making process while still operating.

While fuel off-loading was in progress, the District developed a Rancho Seco closure mission, to "close and ultimately decommission Rancho Seco as safely and economically as possible, consistent with Nuclear Regulatory Commission (NRC) License Requirements." This mission led directly to a planning effort to evaluate selection of the most appropriate decommissioning alternative.

DECOMMISSIONING ALTERNATIVES

The three decommissioning alternatives considered by the District include:[1]

DECON
Site and facilities are decontaminated and/or dismantled to achieve residual radioactivity levels meeting unrestricted release criteria.

ENTOMB
The reactor containment building acts as a containment for radioactive materials for the lifetime of the structure. Radioactive decay reduces residual radioactivity to a level permitting unconditional release.

SAFSTOR
The facilities are placed in layup, then a dormancy period, followed by deferred decontamination, and subsequent free release of the site.

DECON is the most likely alternative for plant decommissioning at the end of license when a repository is available to accept their spent fuel. To relinquish a 10 CFR Part 50 license[2] (and release the reactor site for unrestricted use), either the fuel must be gone from the site or the fuel may be stored in a 10 CFR 72 licensed Independent Spent Fuel Storage Installation (ISFSI) with no reliance on other facilities for maintenance, repair, or reloading casks for shipment.

The ENTOMB alternative is not a very likely alternative for large commercial reactors since the long-lived activation products associated with the reactor vessel internals will not decay to acceptable levels within the required time period.

SAFSTOR, although more costly than DECON, is the most likely option for prematurely decommissioning plants since neither a Monitored Retrievable Storage (MRS) facility nor a permanent repository is available for Department of Energy (DOE) acceptance of spent fuel (and is not expected to be available until at least 1998). SAFSTOR can consist of different layup modes:

- Custodial SAFSTOR is a mode of layup where the spent fuel is stored wet in the spent fuel pool. Systems required for the safe storage of spent fuel (e.g., water purification, security) must be maintained.
- For Hardened SAFSTOR, the fuel is stored dry (at an ISFSI). The remainder of the facility is placed in a more permanent layup mode (e.g., doors welded shut, pipes blanked off).

At Rancho Seco, tentative plans call for Custodial SAFSTOR until 1998, followed by Hardened SAFSTOR for up to 10 years.

FUEL DISPOSITION OBJECTIVES

Once dependency between selection of a decommissioning alternative and disposition of the spent fuel became obvious, the District developed objectives to support fuel disposition. The objectives are:

1) Minimize occupational and public radiation exposure,

2) Minimize decommissioning costs, including the need to maintain the spent fuel pool, and

3) Prepare the fuel for DOE acceptance.

A consultant, S. Levy, Inc., evaluated a total of 15 options for disposal of Rancho Seco's fuel.[3] Rancho Seco is a single-unit site, and the District has only the one nuclear plant. This eliminated the possibility of intrautility storage. Transfer to another utility, company or county for storage until DOE acceptance also proved costly or risky. The study concluded that the use of dual-purpose (combined storage and transport) casks would enable the District to meet its objectives.

The District agrees. In addition, the NRC said in a letter to DOE, "The Commission believes that radiation exposure and other handling risks should be minimized in the entire process from removing fuel from the pool the first time to its ultimate disposal."[4] As shown in Figure 1, the use of dual-purpose casks eliminates the need to reload casks at the site prior to shipment, and possibly at an MRS, thus reducing handling risks.

Minimizing costs using dual-purpose casks depends not only on the initial investment for the casks, but also on the annual SAFSTOR costs and ISFSI decommissioning costs as well.

Rancho Seco is in a unique position when compared to operating plants that are selecting casks for expanded dry storage only. The District has a finite number of fuel assemblies (493) to cask. The District has no need to maintain the spent fuel pool for subsequent refuelings. The District can save money by reducing staff such as licensed operators needed to maintain liquid systems. The difference between Custodial SAFSTOR (wet storage of spent fuel) and Hardened SAFSTOR (dry storage, no spent fuel pool) is approximately $7.5 million/year.[5]

Figure 2 shows the areas of estimated cost reductions. With a projected 10-year Hardened SAFSTOR period, the initial investment for the more expensive dual-purpose casks is easily recovered. The costs of decommissioning an ISFSI designed for dual purpose casks are insignificant. No radioactive contamination remains once the casks are transported. The only decommissioning costs are related to a confirmatory site characterization survey.

The issue of acceptance of the fuel by DOE has several facets. The DOE is currently developing large rail/barge casks for eventual transport of spent fuel to an MRS or final repository. These casks will be compatible with the DOE system. However, for many of the dry storage modules, there is currently no guarantee of compatibility with the DOE system, either using the baskets themselves or with some sort of dry transfer device to load directly into the DOE cask. Use of one of these systems would almost certainly result in the need to maintain the spent fuel pool with associated costs.

In addition, acceptance of fuel by DOE is expected to occur in the order of "oldest fuel first."[6] None of the fuel burned in the Rancho Seco reactor would be expected to be accepted until at least 1998. The fuel most recently burned in the Rancho Seco reactor would not be expected to be accepted by the DOE well beyond the time when the District would like to have the plant in Hardened SAFSTOR (unless special provisions are made for decommissioning plants). To not interfere with the Hardened SAFSTOR, the fuel would have to be stored, ready to ship, in an on-site ISFSI.

To ensure that the dual-purpose casks are acceptable for both storage and transport, they must be certified in accordance with 10 CFR 71 and 72. However, no large dual-purpose casks are currently certified under parts 10 CFR 71 and 72 by the NRC.[7] As a result, the District is pursuing a demonstration program with the DOE to resolve outstanding NRC issues and provide sufficient evaluation to permit licensing of existing large, DOE-compatible shipping casks as dual-purpose casks. To accomplish this, the District has pursued congressional support to obtain appropriations to fund the demonstration program.[8]

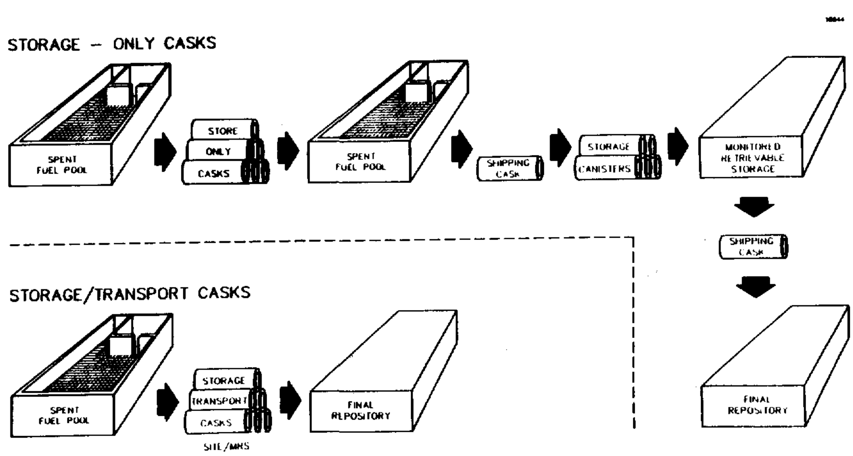
The DOE demonstration program is expected to be limited to approximately 72 of the 493 fuel assemblies available at Rancho Seco. (This is the amount required to fill three casks.) Based on the success of the demonstration program, the District plans to cask the remaining fuel. A total of 17 to 20 casks will be required based on cask capacity and fuel mix. It would be possible for the DOE to use these dual-purpose casks for lag storage at an MRS or even reuse them for shipping.

CONCLUSION

For Rancho Seco Nuclear Generating Station, to fulfill the District's overall decommissioning mission of closing the facility as safely and economically as possible, the use of dual-purpose casks is the best option.

This option supports the fuel disposition objectives of minimizing exposures, minimizing decommissioning costs (through abandonment of the spent fuel pool) and preparing the fuel for DOE acceptance.

Although the solution to Rancho Seco's fuel disposition is not universally applicable, the decommissioning mission and related fuel disposition objectives are. By understanding the District's decommissioning perspective, those not yet ready to decommission (or even ready to think about decommissioning) can make more informed decisions that may save later work and expense.



CASK HANDLING COMPARISON

Figure 1

REFERENCES

1. NUREG 0586, "Final Generic Environmental Impact Statement Decommissioning of Nuclear Facility"

2. Title 10, Code of Federal Regulations.

3. SLI-9006, "Rancho Seco Nuclear Generation Station Nuclear Fuel Storage Options," prepared by S. Levy, H. Klepfer, J. Heald, and S. Smith, March 31, 1990.

4. Nuclear Regulatory Commission Comments on the Initial Version Dry Cask Storage Study, from Lando W. Zech, Jr., to Sam Russo, sent to DOE November 18, 1988.

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7. "Final Version Dry Cask Storage Study," DOE/RW-0220, U.S. Department of Energy, Office of Civilian Radioactive Waste Management, Washington, D.C., February, 1989.

8. Energy and Water Development Appropriations Bill, 1991.

Rita W. Bowser is the fuel disposition project manager at Rancho Seco Nuclear Generating Station. She has 13 years of nuclear experience in radiation protection and engineering, licensing and project management. She earned a B.S. in mathematics from Clarion University, and an M.S. in Health Physics from the Georgia Institute of Technology.

Dan R. Keuter is the assistant general manager, nuclear, for Rancho Seco Nuclear Generating Station. He has more than 16 years of nuclear experience which includes management of operations, maintenance, constructio, and radiological engineering. Keuter received his B.S. from Oregon State University. He holds Professional Engineering and Senior Reactor Operations certifications.

Ken R. Miller is a senior project manager for decommissioning the Sacramento Municipal Utility District's Rancho Seco Nuclear Generating Station. Prior to this assignment, he was a consultant to the nuclear and computer industries. Additionally, he was a manager and principal engineer with General Electric's Nuclear Energy Division. He is a graduate of San Jose State University and is a Professional Engineer.

WET versus DRY
SPENT FUEL STORAGE

	ESTIMATED ANNUAL COST REDUCTIONS
• UTILITY STAFF	$ 5.1 MILLION
• SECURITY STAFF	$ 1.2 MILLION
• ISFSI SUPPORT	$ 0.86 MILLION
• INSURANCE PREMIUMS	$ 0.23 MILLION
• PLANT ENERGY COSTS	$ 0.16 MILLION
TOTAL	$ 7.5 MILLION

Figure 2

Activation of Metal in Reactors: Program Status

A. T. Luksic
Pacific Northwest Laboratory
Richland, Washington U.S.A.

ABSTRACT

In order to properly package and dispose of spent fuel, a complete characterization of the waste stream is necessary. Part of the spent fuel is the structural hardware including end fittings, grid spacers, and guide tubes. This paper presents the results of research conducted at the Pacific Northwest Laboratory for the U.S. Department of Energy. The research included obtaining samples of irradiated spent fuel hardware and determining the radionuclide and elemental composition of those samples by laboratory analysis.

The Pacific Northwest Laboratory (PNL) has been investigating the activation of metals in nuclear reactors for a number of years. In recent years, the work has focused on the relationship between the concentration of radioactive activation products in metals and regulatory limits regarding disposal in shallow land burial sites.

Title 10 of the Code of Federal Regulations, Part 61 (10 CFR 61), regulates the disposal of radioactive materials in shallow land burial sites. The regulation provides a classification scheme for radioactive waste, based on the concentration of a number of radionuclides. The waste categories are Class A, B and C, with C having the highest concentration. Waste having concentration in excess of the Class C limit are referred to as Greater Than Class C (GTCC). They are generally not acceptable for disposal in a shallow land burial site licensed under 10 CFR 61, though exceptions do exist.

A study was begun at PNL in 1985 to determine the waste classification of spent fuel assembly hardware, i.e., the structural portion of the fuel assembly remaining after the fuel pins had been removed. This included the end fittings (top and bottom), grid spacers, guide tubes and fuel channels. After a literature search revealed a lack of data regarding the radionuclides identified in 10 CFR 61, calculations were performed to predict activation levels. These included neutronic modeling of major reactor types and ORIGEN2 calculations. This initial work indicated that much of the hardware on a fuel assembly that has experienced a normal irradiation could be considered GTCC. Details of this study are documented in reference 1.

Calculations indicated that components that were made from Inconel (springs, grid spacers) had concentrations of Ni-59 and Ni-63 several orders of magnitude in excess of Class C limits. Components made from stainless steel (end fittings, guide tubes) and Zircaloy (grid spacers, fuel channels, guide tubes) could have concentrations of Nb-94 in excess of the Class C limit. This result is highly dependent on the amount of initial niobium in the metal. The initial level only needed to be on the order of 100 ppm to present a disposal problem. At this level, it is not of any metallurgical concern, and actual data regarding the impurity level are not generally available. In order to determine whether these materials were above or below the Class C limits, more information was required on the actual amounts of impurities in the materials (specifically niobium) and the activation rate.

A program was established whereby PNL would obtain samples of irradiated hardware and measure both the elemental composition of the samples and the concentration of radionuclides of interest. Concurrently, once the samples were identified, a calculation specific to those samples would be done to predict the activation rate in the samples. In this manner, PNL hopes to

1) gather raw data and
2) prove a calculational technique that could be applied to other materials.

Within the Material Characterization Center (MCC) at PNL, there are available several spent fuel assemblies which were irradiated in boiling water reactors (BWR) and pressurized water reactors (PWR). These assemblies had been selected by the MCC because they represented major fuel types that the Department of Energy would have to dispose of and which had burnups typical for their design. PNL obtained a total of 38 samples from three separate assemblies. The samples represented each type of material found in the assemblies at a variety of locations.

Table 1: Spent Fuel Assemblies Sampled

Westinghouse 14 x 14	**32,000**	**MWD/MTU**
Combustion Engineering 14 x 14	**41,800**	**MWD/MTU**
General Electric 8 x 8	**27,500**	**MWD/MTU**

Table 2: Laboratory Measurements

Element: **Mn, Fe, Cr, Ni, Co, Nb, Cu, Mo**

Radionuclide: **^{54}Mn, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{60}Co, ^{94}Nb, ^{93m}Mb**

For each fuel assembly that was sampled, a neutronic model was developed and the neutron flux and a multi-group spectrum were calculated to provide reaction rates during irradiation, as a function of axial position on the fuel assembly. This information was used to develop one-group cross-sections that were in turn utilized in adjusting the results of ORIGEN2 calculations, to predict as accurately as possible the expected activation level in these assemblies. The ORIGEN2 calculations took into account the actual power history of each fuel assembly sampled, including decay times, and provided estimates of radionuclide concentrations. The results of these calculations, laboratory measurements and the subsequent comparisons are detailed in reference 2.

Much of the uncertainty in the measurements was introduced due to problems encountered in measuring elemental compositions in radioactive samples. A task is currently being funded by the Electric Power Research Institute (EPRI) to address this issue. A number of unirradiated samples of materials used in fuel assembly construction have been obtained. The samples obtained are traceable to their purchase specifications. These samples will be analyzed for their elemental composition, including niobium. Since these samples are unirradiated, the measurements that will be done will be more precise than those previously performed.

The program at PNL has been expanded to include non-fuel bearing components such as control rods and burnable poison rod assemblies. Though the materials used in the construction of these components are nominally similar to those used in fuel assemblies, they are purchased to less stringent specifications. They also undergo very different irradiation histories. Control rods remain in reactors for significantly longer periods of time than do fuel assemblies, while burnable poison rods can stay in for as little as one cycle. At the same time, the neutron flux and spectrum vary significantly from that experienced by a fuel assembly. This is due to location (i.e., control rods in PWRs are generally situated above the core) and the fact that many of these components have neutron-absorbing material that can dramatically change the local neutron spectrum.

PNL has obtained three non-fuel-bearing components. They are a control rod blade from a boiling water reactor, a rod cluster control assembly and a burnable poison rod assembly from a pressurized water reactor. Approximately 10 samples will be taken from each component. Samples will represent the structural portion of the components as well as the absorber materials. Each individual sample will be analyzed both elementally and radiochemically. In this manner, a relationship between activation product and parent isotope will be established. At the same time, calculations will be performed to predict radionuclide production in each component, based on its irradiation history. If the predictions and measurements can be correlated, then the activation in components not sampled can be predicted.

The specific results obtained to date are described in the following references. Additional reports will be published as more data are collected and analyzed. The preliminary results may be illustrated by the following. All of the assemblies analyzed employed stainless steel, zirconel and Inconel for some of the hardware, but the quantities, locations and niobium content varied considerably. The induced radioactivities were related to the niobium content and to the neutron exposure at the different locations on an assembly. The niobium/inconel ratio was in the range .008 to .046, while it was substantially less for the stainless steel and zirconel materials. The curies per gram of the three materials and of the niobium contained in each of them showed similar wide variations.

REFERENCES

1. PNL-6046, *Spent Fuel Disassembly Hardware and Other Non-Fuel Bearing Components: Characterization, Disposal Cost Estimates, and Proposed Regulatory Acceptance Requirements*, Luksic, A.T., et al, October 1986

2. PNL-6906, *Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal*, Luksic, A.T., et al, June 1989

Andrzej T. Luksic received his bachelor's degree in mathematics from the University of New York at Cortland in 1973 and his master's degree in nuclear engineering from Brooklyn Polytechnic Institute in 1976. Since then he has worked in the nuclear field, for Burns & Roe Engineering and Westinghouse Hanford. Luksic has worked at the Pacific Northwest Laboratory since 1983. He has been investigating the activation of hardware in light water reactors, and the effect that this hardware will have on the federal waste management systems, since 1984.

Burnup Credit for Transportation and Storage

Thomas L. Sanders
Ronald I. Ewing
Sandia National Laboratories
Albuquerque, New Mexico U.S.A.

William H. Lake
U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Washington, D.C. U.S.A.

ABSTRACT

After spent fuel has cooled for several years, its efficient transportation and storage tend to be limited by nuclear criticality safety design considerations. This differs from experience with current generation spent-fuel systems, which have been designed for short cooling times and are generally limited by heat transfer or shielding considerations. Consideration of the reduced reactivity of spent fuel due to burnup in the reactor is termed burnup credit. The use of burnup credit in nuclear criticality analysis can increase capacities of storage facilities and transport casks, resulting in reduced handling and shipments of spent fuel. These factors reduce worker and public exposure and result in decreased risk and cost. The acceptance of burnup credit for the design of transport casks and dry storage is contingent on the resolution of certain system safety issues. These issues are being addressed by using an integrated approach to reduce the uncertainties involved and to provide calculational and operational guidance.

INTRODUCTION

Burnup credit is the application of the effects of fuel burnup to nuclear criticality design. When burnup credit is considered in the design of storage facilities and transportation casks for spent fuel, the objectives are to reduce the requirements for storage space and to increase the payload of casks with acceptable nuclear criticality safety margins.

As nuclear fuel is burned in a reactor, the net fissile content decreases and neutron absorbers are produced. These effects reduce the nuclear reactivity of the fuel assemblies, and in most cases the fuel remains in the reactor until it is no longer useful. Because nuclear reactivity is reduced, spent fuel assemblies that have been removed from a reactor can be packed together more closely than unburned (fresh) fuel assemblies before any possibility of nuclear criticality (a self-sustaining nuclear chain reaction) is approached.

Burnup credit has been successfully applied to spent fuel storage pools, resulting in increased capacity and the storage of spent fuel with higher initial enrichments.[1] Efforts are under way to allow burnup credit in dry storage casks. The essential considerations are the same for both transport and storage casks.[2]

The spent-fuel carrying capacities of previous-generation transport casks have been limited primarily by requirements to remove heat and/or to provide shielding. Shielding and heat transfer requirements for casks designed to transport older spent fuel with longer decay times are reduced significantly. Thus a considerable weight margin is available to the designer for increasing the payload capacity. One method to achieve an increase in capacity is to reduce fuel assembly spacing. The amount of reduction in assembly spacing is limited by criticality and fuel support structural concerns. The optimum fuel assembly spacing provides maximum cask loading within a basket that has adequate criticality control and sufficient structural integrity for regulatory accident scenarios.

The traditional assumption used in evaluating criticality safety of a spent-fuel cask is that the spent fuel is as reactive as fresh fuel. This is known as the fresh fuel assumption. It avoids a number of calculational and verification problems but takes a heavy toll in decreased transport efficiency. Burnup credit is an alternative to the fresh fuel assumption that provides a more efficient design for spent-fuel casks.

The incorporation of burnup credit in cask designs could result in considerable benefits in the transport of spent fuel. Increasing cask capacity results in lower public and occupational exposures to ionizing radiation by reducing the number of shipments necessary to transport a given amount of fuel. Additional benefits result from reduced non-radiological risks to both the public and the nuclear occupational sector. In addition, economic benefits result from lower in-transit shipping costs, lower transportation fleet capital costs and a reduced number of cask-handling operations at both shipping and receiving facilities. Studies have shown that cask capacities could be increased by as much as 400%. This could amount to a significant cost savings and a significant reduction of risk to the public.[3]

The acceptance of burnup credit for the design of transport casks depends on the resolution of system safety issues and the uncertainties that affect the determination of criticality safety margins.[4,5] The remainder of this report will examine these issues and the integrated approach under way to resolve them.

CRITICALITY SAFETY AND CASK DESIGN

The criticality safety of a spent-fuel system is determined by the reactivity of the spent fuel. Spent-fuel reactivity is a function of four variables: (1) the initial enrichment of the 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 from the operating reactor as spent fuel.

Criticality, $k_{eff} = 1.0$, can occur in an array of light-water reactor (LWR) fuel only if (1) sufficient fissile material is available in a nearly optimum geometry, (2) a moderator is present, and (3) the criticality control features, if present, are compromised. No array of LWR fuel can achieve criticality without water present in the array.

Criticality control of any array of fissile material is accomplished by one or more of the following methods: (1) the overall mass quantity of fissile material may be limited, (2) thermal neutron absorbers (poisons) may be introduced, (3) the energy spectrum and thermal neutron population may be controlled by moderator and/or reflector materials and (4) geometry controls may be implemented by maintaining specific assembly-to-assembly spacings that reduce reactivity. Both the International Atomic Energy Agency (IAEA) and the Nuclear Regulatory Commission (NRC) regulations allow some combination of the above for criticality control.[6,7] Casks are required to remain subcritical within a specific margin. A 5% margin is generally used on k_{eff}, the measure of criticality ($k_{eff} = 0.95$).

Both the NRC and IAEA regulations allow taking credit for a reduction in the reactivity of spent fuel (burnup credit) as a result of irradiation. The regulations require three major reactivity considerations that are subsets of the criticality control measures listed above: "The most reactive credible configuration consistent with the chemical and physical form of the material . . .," "Moderation by water to the most reactive credible extent . . .," and "Close reflection by water on all sides . . ."[6,7]

The IAEA regulations specifically allow the use of burnup credit in cask criticality analyses provided that the "degree of irradiation is known with appropriate accuracy".[7] However, the following conditions are recommended:[8]

"a. Procedures should be adopted to prevent the package having a higher reactivity than the calculated value under any foreseeable circumstances. Among the contingencies to be considered are the possibility of misidentification of the fuel at the time of loading into the transport container, and any possible errors in the evaluation of burn-up;

b. The validity of the method used for evaluating the effect of irradiation on the fuel composition should be established. One of the validated computer codes used for this purpose could be used;

c. The assessment should include an evaluation of any inherent uncertainties, so that the probability of criticality is demonstrated to be acceptably small."

Although the NRC transportation regulations in 10 CFR 71 do not specify a criticality safety margin, general NRC criticality control requirements for all radioactive waste are specified by 10 CFR 60.131:

> "All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation".[9]

To illustrate the effect of burnup credit on cask design, consider a hypothetical cask whose characteristics are described by Figure 1. The k_{eff} applicable to the cask is examined as a function of the initial enrichment, the burnup of the spent fuel it is to carry and an assumed uncertainty of 38% in the available burnup credit. This value (38%) was determined by adding a 5% operational uncertainty to the 33% (an upper limit) assumed by others.[4]

Figure 1 shows the criticality (k_{eff}) vs. initial fuel enrichment for fuel with burnup of 0, 10 and 20 gigawatt days/metric ton of uranium (GWD/MTU). It is assumed that the spent fuel has cooled for two years. The hypothetical cask is designed with a criticality margin of 5% (i.e., maximum $k_{eff} = 0.95$) for initial enrichments of up to 2.5% without burnup credit . For enrichments greater than 2.5%, burnup credit must be taken. The minimum burnup needed for any initial enrichment above 2.5% can be found by reading the value on the curves where the initial enrichment of interest crosses $k_{eff} = 0.95$. For example, spent fuel with an initial enrichment of 4% would need a minimum burnup of approximately 12 GWD/MTU (for fuel cooled a minimum of two years) for $k_{eff} < 0.95$.

The effect of uncertainties associated with using burnup credit are illustrated by curve C, the burnup load limit, which assumes an uncertainty equal to 38% of the burnup credit. Figure 2 is an operational loading graph for the hypothetical cask. Curve L in Figure 2 (a transposition of curve C from Figure 1) forms a limit curve for safe loading of the cask for burnup credit. Figure 2 is interpreted as follows: spent fuel

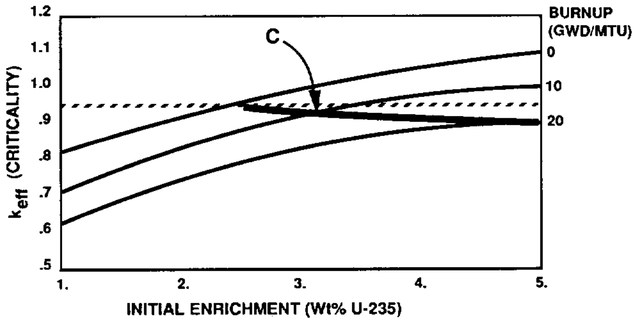

Figure 1. Hypothetical Cask Characteristics

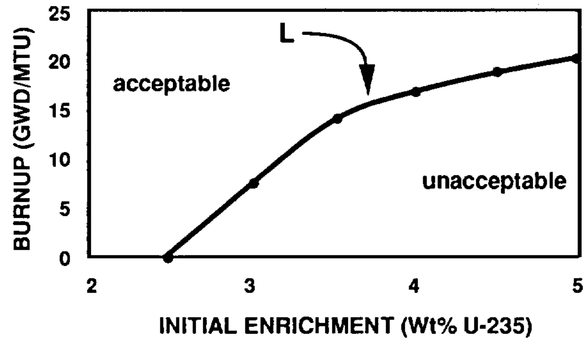

Figure 2. Loading Graph

(with minimum two year cooling time) can be loaded in the cask without additional criticality safety control if the burnup is equal to or greater than that shown on curve L, for the particular initial enrichment of the fuel (acceptable region). Fuel not satisfying the above specification (unacceptable region) would need some additional criticality control (e.g., poisons, additional mass limits, etc.).

Some general observations can be made from the hypothetical cask design. In Figure 1, the curve for burnup = 0 represents k_{eff} for a cask designed with a fresh fuel assumption for an initial enrichment of up to 2.5%. For higher enrichment, the cask k_{eff} follows curve C with an increased criticality safety margin corresponding to the increase in burnup credit. Along curve C, the value of k_{eff} decreases from 0.95 at initial enrichment of 2.5% to approximately 0.90 at initial enrichment of 5%. Curve C may be interpreted as the maximum k_{eff} for a cask design incorporating burnup credit and its uncertainties. For a cask using a fresh fuel assumption, the maximum k_{eff} occurs at the design initial enrichment; for a burnup credit cask this maximum occurs at the initial enrichment where burnup credit is first used. Although the uncertainty is expected to be reduced significantly below 38% as the burnup credit approach is developed, it is expected to continue to be non-zero and to increase with increased burnup credit, as shown in Figure 1.

Where burnup credit is used, the criticality control system will consist of two separate components with the reliability of each being important. The first is an external control component similar to that used in a fresh fuel assumption design basis. The external control component includes poisons in the cask or basket web, and geometric spacing and support. The second internal component is the loaded spent fuel. Burned fuel reduces external criticality control requirements due to the net depletion of the fissile material and the production of poisons that deprive remaining fissile nuclei of available neutrons.

From a broad perspective, 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. Exceeding fuel reactivity limits could result from a fuel-loading 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). Some minimal acceptance criteria for demonstrating the reliability of spent-fuel analysis and operational activities are needed. This does not mean that the reliability or quality of current spent-fuel operations is questionable; however, any uncertainties associated with those operations need to be defined.

ISSUES IN THE USE OF BURNUP CREDIT

Uncertainty in the predicted criticality safety using burnup credit arises from the cumulative uncertainties in (1) the identification of average and local variation in burnup, (2) nuclear data upon which calculations are based, (3) reactor operation details and (4) certain other operational and analytical characteristics.[5]

An integrated approach to specifying and reducing the uncertainties in the application of burnup credit is being actively pursued. The questions addressed are, Which design and operational factors dominate criticality safety, and which combination of measures to reduce uncertainties results in optimal criticality safety? This approach to uncertainty reduction will provide design guidance by determining and prioritizing factors that can affect design and safety, reducing data uncertainties and providing validation methods, developing reference benchmark data and reference problems, and developing design recommendations. Guidance will be provided by developing procedures for user validation, fuel acceptance and other operational activities, as needed.

The calculational methods for predicting criticality are being validated by comparing independent methods and using reactor critical data and fresh-fuel critical experiments as input.[10] A reference problem set is being established to facilitate intercomparisons and benchmarking.[11] Analyses have been performed to determine the sensitivity of the calculational methods to various factors, including the isotopic composition of the fuel, axial variation of burnup, low density moderation, assembly design, variations in burnup, initial enrichment, cooling time out of the reactor and the operating history while in the reactor.

Isotope assays have been obtained by chemical analyses and radiation measurements.[11] Isotopic distributions are being obtained from destructive assays of spent fuel, which can be compared with operating histories and criticality data reports.[12-14] Criticality data from reactor cycles has been obtained from Surry, North Anna, Three Mile Island and Sequoia. These data are being incorporated into a reference set of analysis problems that are being analyzed by using several independent, complementary methods.[10,15]

The reference set of isotopes is also being evaluated by several means and will be updated as additional assay measurements become available. 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 must be considered because of their dominance of neutron absorption. Ten isotopes account for over 80% of the neutron absorption. These products are also well-characterized, predominantly stable, and have well-known cross-sections.[16]

The sensitivity of criticality safety to other factors involved has been independently assessed by several methodologies. These analyses have shown that nuclear criticality depends primarily on initial enrichment and burnup and only marginally on cooling time, operating history, axial burnup effects, assembly design and low-density moderation (without flux traps).[17] Uncertainties in these minor issues have little effect on the determination of k_{eff}. The small uncertainties in initial enrichment have little effect on predicted inventories and the resulting reactivities for the enrichment ranges of interest. With the present set of isotopes, burnup can be predicted with an uncertainty less than 10%.

OPERATIONAL VALIDATION

The approach to operational validation is to identify, prioritize and resolve the uncertainties in operational parameters. Methods and procedures will be developed for validating user operations. The cask designs being considered by cask contractors of the Office of Civilian Radioactive Waste Management make use of external criticality controls; therefore, the amount of burnup credit required is quite low. An example is the General Atomics GA-4 cask.[18] The acceptable contents for the cask will be described by a burnup vs. enrichment curve that includes uncertainties. This approach reduces the importance of operational uncertainties, the population of under-irradiated fuel that could be misloaded and the complexity of on-site verification measurements.[19]

Since the uncertainty in the discharge characteristics of a specific spent-fuel assembly is very low, and because cooling time (age) is not a significant effect, the only credible source of error is misloaded spent fuel. This could occur due to an error in the cask-loading procedure or a misidentification of spent fuel during cask-loading operations. The misidentification is the dominant error mode because of the time elapsed between fuel discharge and cask loading.[16]

The significance of operational uncertainties is reduced by minimizing the population of spent fuel that could possibly be misloaded (non-specification fuel). Statistical analyses of the existing spent-fuel inventory[20] indicate several important points. First, less than approximately 1% of the existing inventory of spent-fuel assemblies consists of fresh or irradiated fuel that would have reactivity in excess of a typical maximum enrichment and minimum burnup specification for a rail cask. Second, a preliminary analysis indicates that only four individual assemblies are available in the current inventory which could not be shipped in a typical truck cask designed for four pressurized water reactor assemblies. Third, a significant fraction of the existing inventory of non-specification spent fuel consists of older generation fuel clad with stainless steel that contains high enrichment-to-design burnup ratios. Much of this fuel was prematurely discharged because of in-core failures or other reasons that may have required special handling and transport considerations. Fourth, the majority of the existing inventory of non-specification spent fuel is located at a small number of older reactor facilities.

A few operational guidelines should be emphasized that can significantly benefit the safety of casks designed using burnup credit. First, because the size of the non-specification spent-fuel inventory at the start of shipments from reactors will not be large, this fuel could be removed first by using existing casks or down-loaded casks using burnup credit. Second, any prematurely discharged fuel following the onset of burnup credit cask operations could be tagged or locked in place. Third, burnup credit cask operations could either be prohibited when fresh fuel is available in pool storage locations, or a gamma measurement of each assembly could be performed to ensure that fresh fuel is not loaded into a burnup credit cask. Careful system design that makes use of these guidelines can significantly reduce the likelihood of misloading spent fuel from the non-specification inventory (less than 0.1% of the total inventory).

Spent-fuel characteristics can be verified by on-site measurements. A preliminary evaluation of several fuel verification techniques has been carried out. These techniques include reactivity measurement concepts, neutron source-driven measurements, passive neutron measurements, gamma spectroscopy, gross gamma, thermoluminescent dosimeters and Cerenkov light measurements. Evaluated parameters include accuracy, complexity, flexibility, compatibility with reactor constraints, verifiability and calibration, and independence from operating factors.

Some preliminary conclusions arise as a result of these evaluations. The best application of measurements in burnup credit operations is to employ a simple concept that (1) can ensure that a minimum burnup value has been achieved, (2) does not significantly interfere with cask-loading operations and (3) can be easily calibrated to recently discharged fuel. The best candidates for these purposes are those based on light intensity or gross and spectroscopic gamma measurement concepts.

CONCLUSIONS

The storage and transportation of spent fuel are integral parts of the nuclear energy supply system far into the future. The demands of public safety and cost reduction require the thorough evaluation and application of potentially beneficial concepts such as burnup credit.

The important uncertainties involved with the use of burnup credit in transport cask design have been identified, and activities are under way to establish priority for investigation, reduce the significant uncertainties and adopt conservative design assumptions. A comprehensive cooperative validation effort is under way that complies with current standards. A systematic method is being used to develop burnup credit guidelines. Specific implementation requirements will be developed in cooperation with utilities and cask designers.

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Thomas 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 B.S., M.S. and Ph.D. degrees 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.

Ronald I. Ewing is a senior member of the technical staff at Sandia National Laboratories, Albuquerque, N.M. He received his Ph.D. in physics from Rice Institute in Houston, Texas in 1959. He has been at Sandia National Laboratories since that time. He has published in a wide range of fields, including the development of neutron spectrometers, experimental nuclear physics, radiation detector design, severe lightning tests on aircraft, cold fusion, and verification of arms control treaties. He is currently involved in the design of transport casks for spent nuclear fuel. He is a member of the American Physical Society, the American Nuclear Society, Phi Beta Kappa, and Sigma Xi.

William Lake received his B.S.M.E. (1967) and M.S.M.E. (1970) from Polytechnic Institute of Brooklyn. He began his career as a thermal engineer with Grumman Aerospace Corp. He has had years of experience in the area of packaging for transportation of radioactive materials. That experience was gained at the Atomic Energy Commission, the Nuclear Regulatory Commission and the Department of Energy. In his present position, Lake is responsible for the DOE headquarters activities in the Office of Civilian Radioactive Waste Management's cask system development program.

A New Approach to Comparing Ultrasonic Seal Signatures

J.M. McKenzie
C.C. Hartwigsen
D.L. Lowe
Sandia National Laboratories
Albuquerque, New Mexico U.S.A.

ABSTRACT

Sandia National Laboratories since the late 1970s has been developing ultrasonic sealing systems for international safeguards applications on reactor fuel assemblies. The Seal Pattern Reader (SPAR) is a product of this effort. The SPAR instrument reads the identities of ultrasonic seals such as the Sandia Fuel Assembly Identification Device (FAID), the Atomic Energy of Canada Limited Random Coil (ARC) seal, the Sellafield, UK Multielement Bottle (MEB) seal and the JRC-Ispra VAKIII seal. The unique acoustic pattern of an ultrasonic seal, when compared to the previously recorded pattern of the same seal, is used to confirm the identity of the seal. Previously, such comparisons or correlations were performed in the time frame or special configuration unique to the seal. The random start feature of the VAKIII seal required a different approach. The approach chosen was to employ the Fourier transform of the acoustic pattern obtained by the seal pattern reader, rather than the special pattern as it was read, in this case. Other applications of this aproach are presented.

INTRODUCTION

The International Atomic Energy Agency (IAEA) considers the accountability of nuclear fuel assemblies to be very important. If the identity and the integrity of these fuel assemblies are verified before, during and after the fuel is irradiated, then the material accountability issue is addressed effectively.

Atomic Energy of Canada Limited (AECL) and Sandia National Laboratories (SNL) jointly developed such a verification system. The AECL Random Coil (ARC) seal and the Sandia-developed Seal pattern Reader (SPAR) are the basis of this system for sealing Canadian Deuterium Uranium (CANDU) spent fuel stacks. This ARC/SPAR system was jointly developed during the early 1980s and in May 1988 was approved by the IAEA for routine inspections.

Another such sealing system based upon today's newest technology is currently being developed as a comprehensive sealing system for Mixed Oxide (MOX) fuel assemblies [1]. SNL designed and built a VAKIII SPAR [2] to investigate the identity correlation algorithms. The microprocessor within this SPAR is programed in the FORTH language, giving it the name VAKIII FORTH SPAR. The random start feature of the VAKIII identity was the challenge which resulted in the Fourier Transform Correlation Coefficient, FTCC.[5] The FORTH language made the change from a 100-point to the 128-point identity required by the Cooley-Tukey Fast Fourier Transform (FFT) algorithm very easy. The Fourier Transform is start invariant, and thus the VAKIII FORTH SPAR is "operator independent." Hundreds of VAKIII seal insert identity readings have been performed. These readings were required to evaluate the FTCC system. The correlation coefficient gives a quantitative measure of how closely two variables are related. This measure is used to compare identities. It does however require the same byte position for each variable or identity to have reflectance measurements from the same region. There is some statistical variation. Evaluation of a system requires examining the autocorrelation, r_a, and the crosscorrelation, r_c, population distributions. The r_a population is obtained by correlating many reading pairs of the same seal. It has a maximum value of 1.0. The r_c population is obtained by correlating many reading pairs of different seals. A random population is distributed about 0.0. A practical system requires $r_a\text{min} > r_c\text{max}$. Fitting known distributions to the r_a and r_c populations gives a method to calculate the probability at some r value of false acceptance of an identity.

One hundred FTCCs of VAKIII seal pairs gave an autocorrelation population with a mean of 0.995 and a standard deviation of 0.004. This population is very well confined. The crosscorrelation population is now dominant.

The SPAR reads the seal identities in both the ARC/SPAR and in the VAKIII/FORTH/SPAR systems. These identities are collected in real time and stored in memory. The interrogation identities are compared with stored reference identities. The seal identity and the seal integrity are determined. Pearson's correlation formula[3] gives the comparison for both ARC/SPAR and the VAKIII/FORTH/SPAR. Pearsons correlation is gain invariant.

Consecutive VAKIII identity readings do not have a 1:1 correspondence between byte position and reflectance measurement region. The VAKIII identity is measured over 360 degrees. Readings are visually similar but are phase-shifted. A quick and accurate method of comparing the readings was required. The simple expedient of taking the Fast Fourier Transform (FFT)[4] of the 128 points and then comparing them is the method developed. The fundamental and harmonics of the 360 degree measurements remain the same within acceptable statistical variation. Viewing a problem in the frequency domain often leads to a solution simply by examining the problem from a different viewpoint.

The ARC and the VAKIII sealing systems are described. The additional advantages of correlation in the frequency domain or of using FTCC are investigated. The ability to select the examining frequencies extends the ARC system.

THE VAKIII ULTRASONIC SEAL SYSTEM

The JRC, Ispra VAKIII system, Figure 1, accurately positions the interrogating transducer over the identity region. The SPAR starts the motor and pulses the transducer to get the appropriate number of reflectance measurements in 360 degrees. The voltage measured by the analog to digital converter is a function of angle

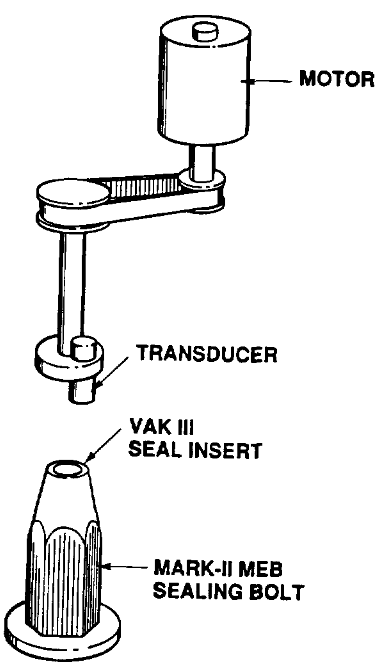

Figure 1: VAKIII Ultrasonic System

$$V(\mathrm{ADC}) = h(\theta)$$

where $h(\theta)$ is the spatial function. There is, however, no spatial fiducial mark; the reflectance pattern starts randomly at the pulse start position. This is indicated by the signatures shown in Figures 2 and 3. Obtaining the amplitude or Fourier spectrum $H(f)$ of $h(\theta)$ is a quick, operator-independent method of using these data. The SPAR, controlled by a laptop computer, obtains 128 reflectance values in 360 degrees. It then transfers this file $h(\theta)$ to the laptop computer which calculates the FFT with the Cooley-Tukey algorithm. The 64 point complex transform is used to calculate $H(f)$. Pearson's correlation coefficient, r, is then computed and used to compare previous and present $H(f)$'s. $H(f)$ is $h(\theta)$ start invariant. The Fourier transform Correlation Coefficient is start and gain invariant. Frequencies 2 to 16 are used for the r calculation. This system is called Fourier Transform Correlation Coefficient or FTCC.[5] The calculated FTCC for the patterns in Figures 2 and 3 is 0.997.

The probability of false acceptance or of false rejection of an identity depends on the r level. Since the SPAR output is the r value, an inspector by choice of the critical r value can accent either false acceptance or false rejection. Readings for 20 seals taken over a six-month period and correlated in pairs, autocorrelations, give r_a min = 0.901. The reading head was removed and replaced for reading. Mere removal and replacement of a reading head gives r_a min = 0.981. One hundred ninety FTCCs of different seal readings pairs, crosscorrelations, gives r_c max = 0.858. This makes a practical system $r_a$min > $r_c$max. Although the identities are visually a little different, FTCC correctly identifies the identity. FTCC is small change invariant. This, together with operator independence, makes this system a strong candidate for this application.

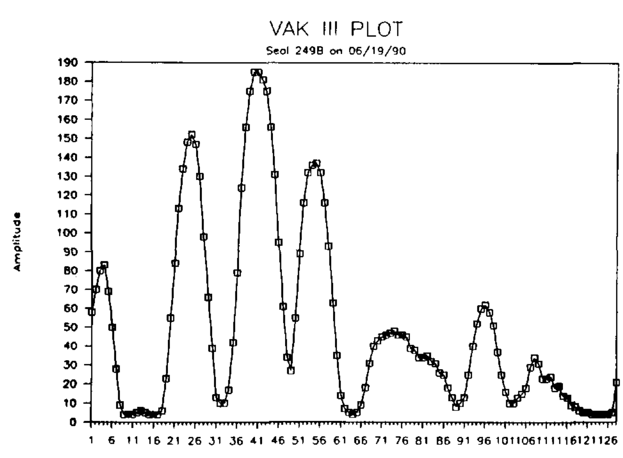

Figure 2: Pattern Start 1

THE ARC ULTRASONIC SEAL SYSTEM

The AECL Random Coil, ARC, is a 5 mm-diameter, 20 mm long coil of pretwisted 0.25-mm stainless wire, as depicted in Figure 4. The interrogating transducer is positioned above the coil. The SPAR pulses the transducer and records the reflectance as a function of time

$$V(\mathrm{ADC}) = h(t)$$

The transducer pulse provides a time fiducial mark. Pearson's correlation coefficient is used to compare present and previous $h(t)$s. The probabilities of false acceptance or of false rejection are essentially similar (10^{-3}) to those of the VAKIII.[6] In today's system, $h(t)$ is reading head-transducer dependent. This system, correlating in the time domain h(t), is approved by the IAEA for routine inspection use. The system is designed for sealing spent fuel in CANDU type spent fuel bays. Reading the seals with two reading heads and storing both reference identities gives the necessary confidence in the system. If FTCC is used with the lower 512 bytes of the 640 byte signature, then a reading head invariant system results. FTCC is reading head invariant. Table 1 compares, for some old files, FTCC or the frequency domain correlations with the time domain correlations. Thirteen autocorrelations, r_a, same seal with different reading heads, were calculated. This same seal population yielded 91 crosscorrelations, r_c, for different seals and different reading heads. The correlation frequency range of 2 to 64 for FTCC gave the maximum separation of r_a from r_c to identify a seal regardless of reading head. This is not necessarily the best frequency range to identify a seal with the same reading head.

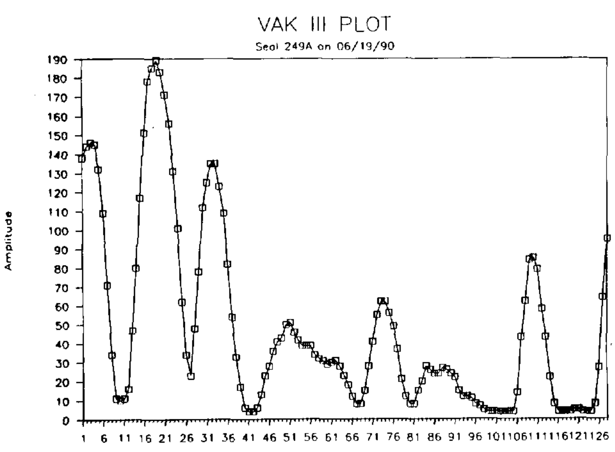

Figure 3: Pattern Start 2

Table 1

System	$r_c$max	$r_a$min
Time domain	0.511	0.356
FTCC	0.784	0.815

The time domain results show that the ARC system is impractical when using different reading heads. A more realistic case is to compare the results for one seal and two different reading heads. Eleven values of the autocorrelation, r_a, for the same seal and two reading heads were obtained. One hundred values of crosscorrelations, r_c, that is, the seal read with one reading head correlated with the readings from the other head, were calculated. The correlation frequency range was 2 to 32.

$$\mathrm{FTCC}\ r_c\mathrm{max} = 0.667\ r_a\mathrm{min} = 0.689$$

Either analysis shows the ARC FTCC system as practical, $r_a\mathrm{min} > r_c\mathrm{max}$. The VAKIII FORTH system was specifically designed for FTCC. These very unsophisticated ARC FTCC results strongly indicate that a specifically designed ARC FTCC system would be reading head invariant. Reference identities from only one reading head would be required for the ARC system. If the original reading head-transducer were to fail, then a second reading head-transducer could be used, and the reading head invariant characteristic would recover

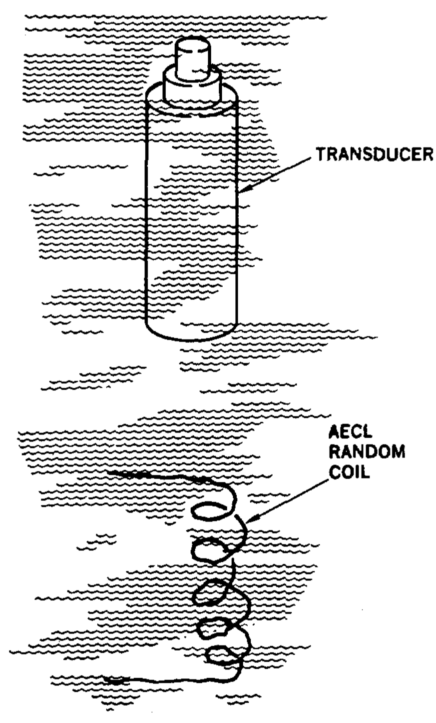

Figure 4: ARC Ultrasonic System

the seal signatures. One reading head greatly simplifies seal reading procedures and equipment and reduces the time involved in the safeguards operation. The ARC SPAR system uses a Harwell MB algorithm to detect a proposed counterfeit. The Fourier transform provides the pattern recognition, and the correlation coefficient gives a measure of the similarity. A specifically designed ARC FTCC system should not require the MB algorithm.

CONCLUSION

Fourier Transform Correlation Coefficient (FTCC) results have been described. The Fourier transform provides a means for pattern recognition. The pattern is changed to Fourier frequency components or workable components. The correlation coefficient is a measure of the similarity of the components. The experimenter can choose those components that are needed. For the VAKIII random start, similar frequency components mean similar patterns.

a.) The transformed identity of the VAKIII is start invariant. The VAKIII FORTH SPAR uses FTCC. This is the best method found to date. The system was evaluated with 20 seals giving 190 r_c's. The system is r_c dominant. Forty seals with their 780 r_c's are sufficient for a population distribution. The VAKIII FORTH SPAR with an adequate setup could read these seals in an hour.

b.) The ARC system with FTCC and selecting correlation frequencies eliminates the requirement for reference identities read with a back-up calibrated transducer.

c.) The two systems ARC and VAKIII with FTCC are similar in their ability to verify an identity. They were competitive developments. Both are in situ verifiable and operator independent. The choice of a system will probably depend on the availability and convenience of the SPAR.

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J.M. McKenzie completed his Ph.D. and MS. studies in New Zealand. He is a chartered engineer and a member of the Institution for Electrical and Electronic Engineers in Great Britain. He was employed by Atomic Energy of Canada, Ltd. for five years as a scientist and by AT&T Sandia Laboratories as a distinguished member of the technical staff for 25 years. He is now retired

Christian C. Hartwigsen earned his BSME at Valpariso University in 1964. He earned his MS at the University of New Mexico in 1967. He has been employed at Sandia National Laboratories since May 1967 in the following areas: 1967-1974, testing of explosive components; 1974-1980, physical security systems development; 1980-1982, beneficial uses of waste radioisotopes; 1982-present, physical security systems development.

Daniel L. Lowe received his A.S. in electric engineering technology from Blue Mountain community College, Pendleton, Ore., in 1986. He joined the Advanced Systems Department of Sandia National Laboratories in 1987. Currently he is working in the development of alarm communication processing and display systems for nuclear facilities in the United States. He has received recognition for is contributions in Sandia's internal self-assessment of environmental, safety and health issues.

A Proposal for Technical Criteria for Termination of Safeguards for Materials Characterized as Measured Discards*

A. Fattah
N. Khlebnikov
Department of Safeguards
International Atomic Energy Agency
Vienna, Austria

1. INTRODUCTION

The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC/153 (corrected),[1] paragraph 11, states:

"The agreement should provide that safeguards shall terminate on nuclear material subject to safeguards thereunder upon determination by the Agency that it has been consumed, or has been diluted in such a way that it is no longer usable for any nuclear activity relevant from the point of view of safeguards, or has become practicably irrecoverable."

The Agency's safeguards system, INFCIRC/66/Rev. 2, paragraph 26c, provides:

"Nuclear material shall no longer be subject to safeguards after the Agency has determined that it has been consumed, or has been diluted in such a way that it is no longer usable for any nuclear activity relevant from the point of view of safeguards, or has become practicably irrecoverable."

Hence, the basis of termination of safeguards primarily depends on a determination that the nuclear material in question has been consumed or diluted or has become practicably irrecoverable. It is necessary for the Agency to develop a workable, technical definition for effective implementation of the provision foreseen in these agreements.

In this context it will be useful to note paragraph 35 of INFCIRC/153. After first specifying that safeguards should terminate under the conditions defined in paragraph 11, it notes that where the conditions of paragraph 11 "are not met, but the State considers that the recovery of safeguards nuclear material from residues is not for the time being practicable or desirable, the Agency and the State shall consult on the appropriate safeguards measures to be applied."

Furthermore, measured discards have been defined in INFCIRC/153 as "nuclear material which has been measured or estimated on the basis of measurements and disposed of in such a way that it is not suitable for further use." No such definition could be found in INFCIRC/66/Rev. 2.

2. ADVISORY AND CONSULTANTS' MEETINGS

The safeguards community should apply safeguards measures to low concentration residues from fuel cycle manufacturing facilities or determine the circumstances under which INFCIRC/153 paragraph 11 or INFCIRC/66/Rev. 2 paragraph 26c should be invoked to terminate safeguards.

This requires technical criteria which consider the nature of the material, the conditions under which termination of safeguards would be appropriate and the feasibility of implementation. An Advisory Group Meeting (AGM) convened in September 1988 to address these issues provided recommendations were used to guide further work.[3] These recommendations, which serve as basic guidelines to develop criteria, are paraphrased below.

a.) The Agency should undertake, in consultation with Member States, to define specific criteria for termination of safeguards from waste material.

b.) Most waste generated under normal operating conditions might be described as being practicably irrecoverable, and accordingly might also qualify for classification as "measured discard" and termination of safeguards.

c.) The criteria for making determinations of "practicably irrecoverable" include waste material type, nuclear

* The views expressed are those of the authors alone and do not reflect those of the IAEA.

material concentration, chemical and physical form, and waste quality (e.g., the presence or absence of fission products). Total quantity, facility-specific technical parameters and intended method of eventual disposal should also be considered.

d.) The Agency should verify both nuclear material content and other factors which qualify a waste material for the determination of practicably irrecoverable.

e.) The determination that a candidate material qualifies as being practicably irrecoverable should be made at the earliest practicable point in the process, and plant operators should be encouraged to provide appropriate measurement procedures to facilitate these determinations.

f.) Waste which meets the resulting criteria and which has been verified by the Agency should be considered to have no further safeguards relevance. Safeguards should be terminated on transfer from the MBA.

Subsequent consultants' meetings were organized in June and October 1989. The consultants held[4] that the Agency should have criteria which can be used in the field and which can be used by an inspector to answer questions of whether a particular batch of material presented for termination of safeguards does, in fact, qualify for the termination of safeguards. To maintain the credibility of safeguards, the criteria of termination should be such that termination of waste is not the weak link in the safeguards system.

It was recognized that determining criteria for the termination of safeguards on nuclear material in waste in the spirit of paragraph 11 must be based on the undertaking of INFCIRC/153 namely:

a.) To verify that such material is not diverted to nuclear weapons or other nuclear explosive devices (paragraph l) and

b.) To avoid hampering the economic and technological development of the nuclear industry and to be consistent with prudent management practices (paragraph 4).

Based on a Secretariat working paper,[5] the consultants prepared a report[4] that provides details of discussions, conclusions and recommendations for application of termination of safeguards from nuclear material classified as measured discards. Since the conditions relating to INFCIRC/66/Rev. 2 type agreements were sufficiently different and could not be addressed simultaneously, the recommendation is considered to be applicable only to INFCIRC/153-type agreements.

Details of the recommendation will not be quoted here; however, it forms the basis for the following discussion on the proposal for technical criteria for termination of safeguards. It should be noted that the Agency's effort toward the development of a technical definition for determination of "practicably irrecoverable" waste categories was strongly endorsed by the Member States. During the meetings, the consultants identified candidate criteria for termination based on their commitment to a strong international safeguards system. The interpretation of measured discards is obviously influenced by various experiences with the operation of a wide range of nuclear fuel cycle facilities, national regulations and practice as well as ever expanding technological innovation. Some agreement on a higher limit, e.g., Pu in high active waste, was naturally endorsed by some consultants with great reluctance due to concern on its impact toward maintaining effective safeguards.

It is the Secretariat's responsibility to draft guidelines, including the criteria associated therewith, on the basis of the recommendations placing particular emphasis on those areas in which a clear consensus is believed to exist and exercising its judgment in those areas where a divergence of views occurs.

3. THE CRITERIA PROPOSAL

On the basis of these discussions and recommendations, the following proposal is presented for consideration. Criteria for termination of safeguards on nuclear materials in waste must not degrade the effectiveness of the international safeguards system. Criteria should permit the termination of safeguards on genuine waste but must assure that effective safeguards continue to be implemented. In other words, the criteria should provide a working definition of practicably irrecoverable.

3.1. Safeguards considerations.

There are three basic safeguards considerations in determining whether safeguards should be terminated on waste nuclear materials. The first one is to provide assurance that the waste itself has not been diverted and processed to recover the nuclear material for subsequent use. A basis for this concern is that if safeguards are terminated, the Agency has no further purview of the material unless the state voluntarily requests that safeguards be reinstated. Thus, practicability of recovering nuclear material from such waste must be considered in developing criteria. While it may be theoretically possible to recover material from various forms of waste, recovering significant quantities may not be feasible. Factors including engineering technology, cost, chemical complexity of waste forms, and the dilute concentrations of nuclear material in waste could make recovery of material from this source very unattractive.

The second is that the deliberate overstatement of the nuclear material content of the waste could provide a means to conceal diversion of material from other sources in a facility. The purpose of such an overstatement would be to conceal the fact that some more concentrated nuclear material had been removed from some other point(s) within the facility. The diversion of nuclear material would be indicated by anomalies and through evaluation of the material unaccounted for (MUF) during material balance periods. If waste discard data falsified, the divertor must still be concerned with safeguards verification measurements. If the falsification is not detected through such verifications, then it will never be detected, because the quantity represented by the falsification no longer exists in the inventory. The total nuclear material content of waste discards might not be overly important as long as the stated nuclear material content is measured and verified and is practicably irrecoverable.

The third safeguard consideration recognizes the possible resubmission of materials on which safeguards have been previously terminated as a concealment for a diversion of other nuclear materials. The INFCIRC/153 agreement requires that the material reintroduced into the nuclear facility be declared to the Agency (but it is not necessarily the case for INFCIRC/66/Rev. 2). Therefore this possibility should be addressed in the context of safeguards procedures related to waste material present for termination.

However, adequate protection against resubmission is an important element for the credibility of IAEA safeguards. Replacement of potentially recoverable scrap with equivalent practicably irrecoverable wastes could hinder detection of diversion of significant quantities of potentially recoverable material from safeguards. If the facility did not fully falsify its records to reflect the waste flow, it could be detected during the Agency's audits of the facility records. The analysis and verification of process materials that were precursors to the true waste materials might also provide an element for the detection of resubmitted wastes. Protection against concealment of diversion by resubmission could be achieved by withholding termination of safeguards until the time of conditioning, which assures that the material is practicably irrecoverable.

3.2. Maximum limit without verification.

Under the existing agreement of specific provisions and criteria for termination of safeguards on nuclear material, safeguards on measured discards on a certain agreed amount per month can be considered to be terminated upon receipt by the Agency of the inventory change report pertaining to such discards. In case of quantities of material exceeding the agreed amount per month, the operator and State are obliged to consult the Agency before such discards take place. The agreed amount depends on the type of facility and its operating condition which varies from zero effective kilogram (Ekg) to 0.1 Ekg per month. Any exception requires material to be kept for Agency verification prior to termination of safeguards. The degree of verification to be applied is dictated, among other things, by the evaluation criteria in force, availability of methods, procedures, instruments and personpower. The experience gained so far has been satisfactory, since the amount reported was small and facilities involved were either R & D type or medium sized commercial facilities. However, with the advent of large industrial scale facilities there is a need to re-evaluate the concept of monthly provision without verification. For a plutonium processing facility, the termination of safeguards on, for example, 0.1 Ekg per month (i.e., 1.2 Ekg per year) of plutonium would permit the removal from safeguards of 1.2 kg of Pu without the opportunity for IAEA verification. This is 15% of a significant quantity of plutonium. The aforementioned raises the concern about termination of safeguards on a quantity that is a significant fraction of a significant quantity.

Considering the safeguards concerns and vulnerability to diversion, particularly for large bulk handling facilities, it would be advantageous to adopt criteria to terminate safeguards on nuclear materials in waste of up to a maximum of 0.01 Ekg/facility/month without verification.

3.3 Maximum limit with verification.

The consultants made recommendations[4] on the limit of the maximum amount of nuclear material in waste for which safeguards can be terminated, if the amount of nuclear material is verified and if the waste characteristics meet other criteria presented herewith. However, they were reluctant to agree on a higher limit due to their concern for its impact on safeguards. It implies criteria can be formulated by placing emphasis on those areas in which a clear consensus is believed to exist and exercising judgment in those areas where a divergence of views occurs. This means a step-by-step solution of the issues involved which could eventually lead to a unified criteria once enough experience has been gained. At this stage, therefore, it is reasonable to divide the waste into two categories:

- Waste which contains all waste characteristics agreed upon by all parties (Type A in Table l) and

- Waste which contains waste characteristics recommended by the majority of the consultants and the point of contention by others, while both held the credibility of IAEA safeguards as a major concern (Type B in Table 1).

Such an approach will allow the immediate application of a technical definition of practicably irrecoverable while keeping the option to include a higher limit open. Until sufficient experience has been gathered and a review takes place, a higher limit could be considered only on a case-by-case basis. In accordance with this approach, the attached Table 1, Proposed Maximum Limits for Termination of Safeguards from Material Categorized as Measured Discards, has been constructed.

The following procedure and material characteristics would be applied in conjunction with Table 1 to determine if the material presented qualified as practicably irrecoverable. a.) *Individual batches.* Termination of safeguards on nuclear material in an individual batch of waste of a particular waste stream would be considered if it has a concentration of nuclear material less than that listed in column 2 of Table 1. The appropriate authority should be consulted if the concentration exceeds that in column 2 of Table 1.

Type A waste could be terminated at the point of transfer. For Type B waste, limits should be considered on a case-by-case basis, which may require, depending on the circumstances, a certain degree of conditioning in accordance with the safeguards agreement and would require a management decision for the most logical termination point.

b.) *Total content of a waste stream.* A control chart and projection of expected total nuclear material as a percent of dissolved feed for each of the waste streams listed in Table 1 will be maintained. Once a month a projection should be made of the material balance period, i.e., the total material expected as a percent of dissolved feed for each of the waste streams, based on the data provided so far for the current material balance period. If the projection is greater than the percentage given in column 3 of Table 1, the appropriate authorities would be consulted.

c.) *Evaluation of design information.* The range of concentration of each waste stream from the design information should be determined. Based on experience utilising the best design features of existing technology, it can be reasonably expected that such a waste rate is compatible with column 3 of Table 1.

3.4. Termination point.

In order to minimise the risk of resubmission, it is advantageous to terminate safeguards on waste after it has been converted into a form that is different from that present in the process area of a facility. As pointed out earlier, the Advisory Group recommended that "the determination that a candidate material qualifies as being practicably irrecoverable should be made at the earliest logical point in the process, and plant operators should be encouraged to provide appropriate measurement procedures to facilitate these determinations." Once the appropriate nuclear material quantity measurements have been performed, and it has been established that a given material meets established criteria and the material has been transferred from the MBA, the contained nuclear material could be removed from the accountancy records as measured discards.

Consistent with the proposal of dividing the waste into two categories, it may be recommended that safeguards on Type A waste should be terminated at the boundary of the process. Type B waste, however, would need further consideration in order that a decision may be made on the termination point. Depending on the material concentration and the type of safeguards agreement, Type B waste might require a certain degree of conditioning depending on the circumstances and safeguards agreements in force to ensure that some material is not resubmitted again.

3.5. Review of the criteria.

Periodical review and appraisal of these criteria are required if circumstances change to make recovery more practicable. These include a change in available recovery techniques which materially reduce the effort for recovery of nuclear material from waste. (for example, the availability of a much more efficient extraction solvent and development of very efficient dry reprocessing), a change in effort for production of direct-use material from ore such as a major increase in the world price of ore or low effort laser enrichment, a change in available verification methods.

4. WASTE THAT DOES NOT MEET CRITERIA

The Advisory Group Meeting on "Safeguards Related to Final Disposal of Nuclear Material in Waste and Spent Fuel" recommended that "the Agency, in consultation with Member States, should give further consideration to the question of waste which does not meet established criteria for a termination of safeguards." The consultants did not discuss this category during the meeting. However, following the criteria described above, termination of safeguards for an isolated event can be considered on a case-by-case basis provided that the termination point of safeguards of such waste takes place after the waste has been converted into a form that is different from that in the process area in order to minimise the risk of safeguards concern described in 3.1. To reiterate, depending on the case, such conditioned waste may not qualify for termination of safeguards and thus safeguards would have to be continued.

5. CONCLUSION

Specific quantity limits have been proposed[5] by considering such factors as waste material type, concentration, chemical and physical form, quality, total quantity and facility-specific technical parameters. These factors have been evaluated in developing a set of proposed termination criteria. It appears to be unrealistic to establish universally applicable criteria governing a wide range of physical forms, quantities, concentrations and compositions of waste. A divergence of views in setting a quantity limit is evident and unlikely to be resolved soon. However, a proposed limit can be considered, placing particular emphasis on the area in which a clear consensus is believed to exist and exercising judgment in the area in which a divergence of views occurs. This implies consideration of a tw- tier limit (specified as Type A and Type B in Table l) that will hopefully converge the differences with increased experience in handling safeguards termination issues relating to waste. Accordingly, the following proposal could be considered:

1) Termination of safeguards on nuclear material in waste up to 0.01 Ekg per facility per month without verification.

2) Termination of safeguards on nuclear material in waste without limit if the Agency has the opportunity to verify (a) the nuclear material content and concen-

tration and (b) that the waste meets the characteristics specified in Table 1 and the conditions in 4 and 5 below.

3) Termination of safeguards on nuclear material in waste, for Type A, on transfer from the material balance area in which it meets the requirements specified in Table 1 for termination of safeguards and for Type B wastes, at the time of conditioning.

4) Termination of safeguards on nuclear material in an individual batch of waste of a particular waste stream if its nuclear material concentration is within the limit specified in Table 1.

5) Termination of safeguards on the nuclear material content in waste streams if the Agency's projection of the expected total nuclear material as percent of feed based on currently and previously submitted waste for the material balance period remains within the limit established in Table 1.

6) Review the above criteria 1 to 5 if circumstances change.

TABLE 1
Proposed Maximum Limits for Termination of Safeguards from Material Categorized As Measured Dicards

Waste stream		Maximum permitted concentration of Pu or U for a batch in ppm	Expected percentage of feed
		Plutonium Waste	
		(a) Spent Fuel Reprocessing Plant	
Hulls		200	0.5
Feed clariflcation sludges	Type A	500	0.5
	Type B	5000	1.0
Highly active liquid		0.2	
Liquid		5	
Fraction of solids		500	
Medium active liquid		0.1	
Liquid		5	
Fraction of solids		500	
Low level solid		0.1	0.1
		(b) Pu/MOX Conversion Plant	
Solids	Type A	20	0.1
	Type B	1000	0.3
Liquid		N/A	-
Sludge	Type A	20	.04
	Type B	25000	.04
		(c) Uranium Waste for All Types of Plants	
Final extraction residuals		2000	< 0.02
Combustible waste		200	<< 0.01
Incombustible waste		200	< 0.02
Air filters		100	<< 0.01
Chemical residues		200	<< 0.01
Aqueous solutions		15	< 0.01

Note:

1) ppm is expressed as the ratio weight/volume in grams/cubic meters for plutonium and weight/weight in grams/metric tonnes for uranium.

2) Termination for Type B waste is to be established on a case-by-case basis and may, depending upon the circumstances, require a certain degree of conditioning before termination of safeguards.

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A. Fattah has been a staff member of the International Atomic Energy Agency since 1974. Currently he is working as a system analyst in the Division of Concepts and Planning at the Department of Safeguards.

Since 1986, Dr. Khlebnikov has been the head of the System Studies Section, Department of Safeguards, International Atomic Energy Agency (IAEA). In 1976 he earned a Ph.D. in physical chemistry from the State Research Institute of Rare Metals in Moscow. Prior to joining the Department of Safeguards, Dr. Khlebnikov was the head of the Safeguards Laboratory at the Central Research Institute of Atomic Information in Moscow. He was responsible for the Laboratory's systems analysis activities including: non-proliferation issues, international safeguards systems design and state systems of accounting for and control of nuclear material.

Khlebnikov has more than 40 publications, including papers presented during the third and sixth ESARDA Symposia in 1981 and 1984, respectively, the IAEA Safeguards Symposium in 1982, the 1989 and 1990 Institute of Nuclear Materials Management meetings and the 1989 and 1990 American Nuclear Society meetings. He co-authored a book entitled International Control Systems in the Field of Atomic Energy Utilization *which was published in Moscow in 1986.*

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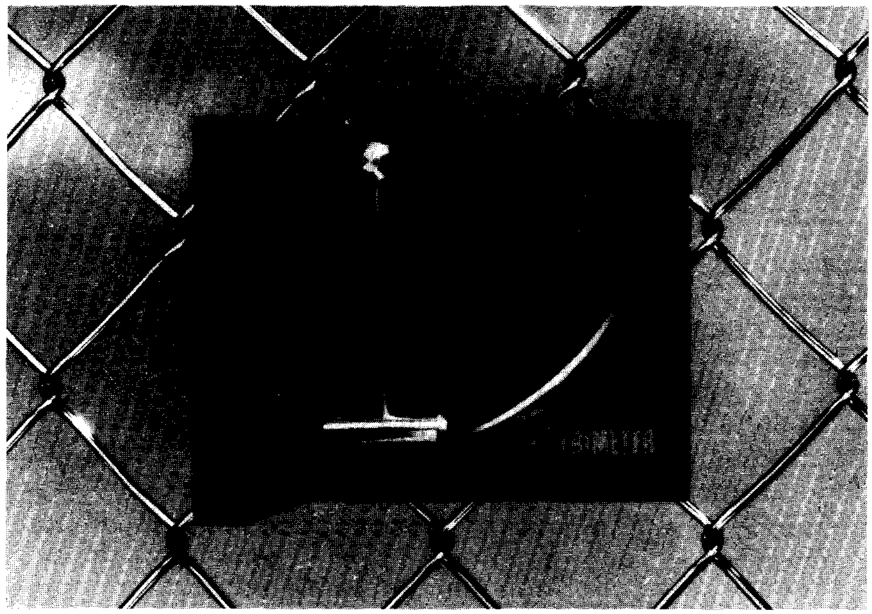

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Fourth International Conference on Facility Operations-Safeguards Interface, Albuquerque, New Mexico *Sponsor:* American Nuclear Society Isotopes and Radiation Division, ANS Fuel Cycle and Waste Management Division, Trinity Section of ANS, and the Institute of Nuclear Materials Management *Contact:* ANS Meetings Dept., 555 N. Kensington Ave., La Grange Park, IL 60525 U.S.A.; phone (708) 579-8258.

October 1 - 3, 1991
Emerging Technologies for Hazardous Waste Treatment, Atlanta, Georgia *Sponsor:* American Chemical Society, Division of Industrial and Engineering Chemistry *Contact:* Dr. D. William Tedder, I&EC Symposium Chair, School of Engineering, Georgia Institute of Technology, Atlanta, GA 30332-0100 U.S.A.

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32nd Conference on Analytical Chemistry in Energy Technology, Gaitlinburg, Tenn. U.S.A. *Sponsor:* Oak Ridge National Laboratory, U.S. Department of Energy *Contact:* R.D. Laing, Oak Ridge National Laboratory, P.O. Box 2008, MS 6127, Oak Ridge, TN 37831.

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1991 Annual Calorimetric Assay Training School, EG&G Mound, Miamisburg, Ohio *Sponsor:* U.S. Department of Energy *Contact:* Lina Di Girolamo, EG&G Mound, Miamisburg, Ohio 45343 U.S.A.; phone (513) 865-3753; fax (513) 847-5264.

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