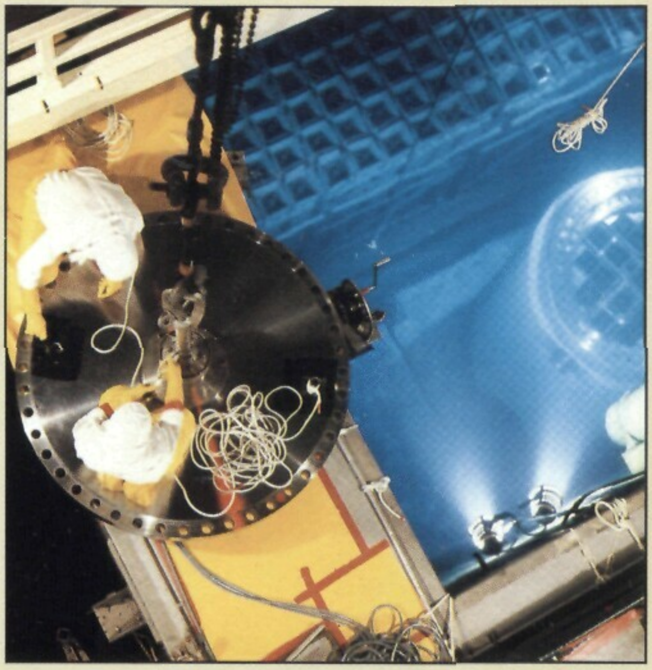
JNMM

JOURNAL OF
THE INSTITUTE OF
NUCLEAR MATERIALS MANAGEMENT



Volume XVII, Number 3 ▪ April 1989

JNMM is published four times a year by the Institute of Nuclear Materials Management, Inc., a not-for-profit membership organization with the purpose of advancing and promoting efficient management and safeguards of nuclear materials.

SUBSCRIPTION RATES: Annual (U.S., Canada and Mexico) $100.00, annual (other countries) $135.00; (shipped via air mail printed matter); single copy regular issues (U.S. and other countries) $25.00; single copy of the proceedings of the annual meeting (U.S. and other countries) $65.00. Mail subscription requests to *JNMM*, 60 Revere Drive, Suite 500, Northbrook, Illinois 60062 U.S.A. Make checks payable to INMM.

ADVERTISING, distribution and delivery inquiries should be directed to *JNMM*, 60 Revere Drive, Suite 500, Northbrook, Illinois 60062 U.S.A. or contact Thalia Keeys at (312) 480-9573, telecopy (312) 480-9282. Allow eight weeks for a change of address to be implemented.

ISSN 0893-6188

SOCIETY OF NATIONAL ASSOCIATION PUBLICATIONS

INMM
INSTITUTE OF NUCLEAR MATERIALS MANAGEMENT

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On the Cover:
Preparing the cask lid for installation on a loaded spent fuel cask at the Surry Power Station. Photo courtesy of Virginia Power.

TECHNICAL EDITOR'S NOTE

Inspiring Contributions

This issue includes several invited papers on management of spent fuel and two contributed papers on safeguards measurements.

The invited papers were selected from those presented at our annual January seminar on spent fuel management. The United States has decided to postpone reprocessing indefinitely and to dispose of spent fuel along with high level radioactive wastes in a geological repository. The papers and the discussions at this seminar were primarily concerned with how the U.S. government is implementing its plans to receive spent fuel from the nuclear utilities around the year 2000 and what the utilities can do to store their spent fuel until then. Although other countries may have quite different policies regarding reprocessing, spent fuel storage and high level waste disposal policies and techniques are of concern to every country with a nuclear power program. It is hoped that the papers presented here will be of interest to our members in other countries as well as to those in the United States.

Storage units and shipping casks for spent fuel are being designed and marketed in several countries. Little that is new was reported at this year's meeting. However, the paper contributed by British Nuclear Fuels Ltd., one of these suppliers, is included because it is a concise review of 33 years of spent fuel management in the United Kingdom involving Magnox, AGR, and recently LWR fuels.

Highlights of this year's symposium are summarized in the paper which introduces the technical section (see JNMM Comment).

It will be interesting to see how the nuclear policies develop in our different countries. The United States has decided that reprocessing and breeders will not be economically attractive for many years and has established a program to develop a safe geological repository by the year 2002. Although the plan is to bury spent fuel, the design is to permit recovery for 50 years after that. Sweden has decided to phase out its nuclear power plants and is developing facilities for temporary storage and permanent disposal of the spent fuel. Every nuclear country is designing means to safely dispose of low and high level nuclear wastes. A few countries have or are developing reprocessing facilities and fast breeders.

Some of these policies are more far-sighted than others. It is tempting for nuclear advocates to emphasize that nuclear power does not contribute to the greenhouse effect, which may create serious problems for future generations. But it is not clear how serious the greenhouse effect may be, or even that the consumption of fossil fuels is a major factor. Although the records are not very good, the global temperature has gone up and down significantly throughout recorded history. An exceptional long-range plan would be to learn about all of the factors which affect the temperature and the other weather features of importance to society and to plan, over a period of many years, to control it for the benefit of mankind. Nuclear energy has a vital role whether or not it affects global temperature.

We also have two contributing papers. One describes a new method for calibrating accountability tanks. The other, from Australia, is about an enrichment monitor which might be used by an inspectorate at an enrichment plant. These papers should be of interest to members who are not involved in the subjects covered by our special meeting papers. I hope that these papers will inspire others to contribute. You could make the *Journal* a more valuable publication.

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

CHAIRMAN'S MESSAGE

Orlando's Edge

Have you made your reservations for the INMM's 30th Annual Meeting? The meeting will be held at the Stouffer Orlando Hotel in Orlando, Florida, July 9-12, 1989. More than 225 technical presentations are scheduled covering all facets of safeguards technology. More than 500 of the world's leading nuclear materials management, safeguards, security, transportation and waste management professionals will be on hand to participate in this tremendous technical information exchange.

To round out the extensive technical program, leading organizations offering safeguards-related technology and services will take part in the annual INMM exhibit. In addition, a number of ANSI writing committees will meet to discuss updating current standards and the need for potential new standards. This is a meeting you should not—*cannot*—miss. If you are going to be on the leading edge of safeguards, you will be in Orlando.

The INMM Executive Committee continues to work hard. The Committee is reviewing and in some cases revising or creating charters for all of the Institute's standing committees. This work includes developing operating plans to make it easier for more members to become involved. If you are asked to help, please try to find the time. You will be helping yourself as well as the Institute. If you would like to serve please contact me. There are committees, standards writing groups, technical workshops and meeting planning groups which need *your* ideas and involvement to make the Institute more influential and effective in safeguards.

John F. Lemming
EG&G Mound Applied Technologies
Miamisburg, Ohio

Can We Amend the Way We Act?

The Nuclear Waste Policy Amendents Act appears to have had a secondary effect which was neither technical nor institutional in substance. One could sense that the atmosphere at the sixth annual INMM Spent Fuel Management Seminar was different from the typical get-togethers of this type. There was no public name calling or finger pointing, and very little hallway whispering. This is noteworthy, since the majority of the 118 attendees consisted of the usual "us" and "them" crowd. It was almost as if "them" were finally beginning to understand the problems that "us" were facing.

Two presentations were key for setting and maintaining the tone of the meeting—one by "us" and one by "them" (interchangable, depending on which side you're on). Steve Kraft, of EEI's Utility Nuclear Waste Management Group, was the first speaker. The room was near capacity—everyone waiting for Mr. Kraft to talk about "them." What Mr. Kraft said was that although he did not always agree with the approach that "them" had used to solve the nation's nuclear waste problems, there were, sometimes, extenuating circumstances (political and institutional) which were responsible for impeding progress. He commended "them" for their efforts of trying to manage a technical problem guided by political rules. Later, the upbeat tone was maintained by the keynote speaker, Ralph Stein of the DOE Office of Civilian Radioactive Waste Management (OCRWM). Mr. Stein highlighted the successes to date and delineated current efforts to solve the many problems facing "us" and to resolve any differences with "them," given the rules.

Not all the attendees were one of "us" or "them." There were representatives from seven foreign countries interested in the latest technical developments and systems approaches; the Nuclear Regulatory Commission; the National Academy of Sciences; the National Conference of State Legislatures; the press; as well as the various service and hardware vendors and interested independents.

This year's Seminar coincided with the first anniversary of the implementation of the Nuclear Waste Policy Amendments Act. Consequently, many of the presentations considered the impacts that this Act has had and continues to have on various parts of the waste management system. Most significant of these impacts was the imposed linkage of the MRS schedule to that of the repository—especially since there is strong evidence that the repository schedule will to be delayed and the earliest start date will be in the year 2003.

From a utility's perspective, the time at which the Federal waste management services will be available not only dictates the quantity of additional storage required in the interim, but can also affect the specific storage technology selected. This was evidenced in the presentation describing technology considerations at Baltimore Gas and Electric's Calvert Cliffs site, as well as ongoing interim storage activities by Virginia Power and Northern States Power. Delays in the Federal waste management operation also increases the total number of utilities that must evaluate the various alternatives for interim storage.

Provisions of the Amendments Act also had a significant impact on many of the DOE programs. Specific among these was the MRS program. In addition to the MRS/repository linkage, the Amendments Act established an MRS Review Commission to assess the need for an MRS and provide a recommendation to Congress as to whether or not an MRS facility should be included in the Federal waste management system. We were honored to have one of the Congressionally appointed commissioners, Dr. Frank Parker, report on the status of some of the activities of the Commission. Related to the MRS Review Commission's activities are the ongoing studies by the DOE in support of the Commission. Jeff Williams, DOE-OCRWM, outlined the assumptions and bases for the studies and provided a good summary of the provisions of the Amendments Act as they pertain to the MRS.

The Amendments Act also required the DOE to study and evaluate the use of dry cask storage at reactor sites as a means of meeting the utilities' storage needs until the Federal waste management system is available. A draft of the results of this evaluation was completed and released for public comment in September 1988. Charles Head, DOE-OCRWM, provided a summary of the study and status of the final report publication schedule. The MRS Review Commission is also required to compare an MRS facility to the alternative of at-reactor storage.

Not all of the pre-Act waste management activities suffered the reconnoitering like the MRS program did. The provisions of the Act had little effect on, if only to solidify, some of the DOE-OCRWM programs. Chris Kouts' update of the OCRWM transportation program showed one such example.

In this issue of the *Journal* you will find papers that are a good representation of the tone of Spent Fuel Management VI. The complete proceedings are available from INMM Headquarters. While the technical and institutional changes which the Amendments Act brought about are significant, the apparent change in attitude which it brought about is even more promising.

Updates, and the latest developments relating to nuclear waste management, will be presented at the INMM Annual Meeting in Orlando in July. Plan to attend.

Billy M. Cole
E.R. Johnson Associates
Oakton, Virginia

Japan

1. 1989-1991 Officers of Japan Chapter
The following officers were elected for the 1989-1991 and approved at the 29th Executive Committee Meeting which was held on October 3, 1988 in Tokyo.

Chairman
Hirata, Mitsuho (JAERI)

Vice Chairman
Haginoya, Tohru (NMCC)

Secretary
Osabe, Takeshi
(Japan Nuclear Fuel, Co. Ltd.)

Treasurer
Seki, Yoshinobu
(Mitsubishi Metal Corp.)

Members at Large
Mori, Kazuhisa (JAIF)
Kurihara, Hiroyoshi (PNC)
Hara, Reinosuke
(Seiko Instrument Inc.)
Iwamoto, Harumitsu
(Nuclear Fuel Transport Co. Ltd.)

2. Executive Committee Meeting
The 30th Executive Committee Meetings were held at NMCC Headquarters in Tokyo on November 7, December 23, 1988, and March 10, 1989. The plan of chapter activities as listed below for 1989-1990 has been discussed and approved. Mr. Hiroyoshi Kurihara was elected as a Program Chairman of the 10th Annual Meeting of the Chapter.

3. Plan of the activities for 1989-1990

3.1 Annual Meeting
The 10th Annual Meeting will be held in Tokyo on June 9, 1989. The Annual Meeting is programmed as follows:

A. Plenary Session
Four guest speakers including Mr. C. Vaughan, former INMM Chairman, will be invited.

B. Technical Session
Three papers on Containment/Surveillance, two on Measurement Techniques, seven on MC&A, four on Safeguards System, and two on Inspection Techniques will be presented.

C. Business Meeting
Mr. R. Cardwell will be invited as a guest speaker.
The Banquet will be held following the Business Meeting.

3.2 INMM Journal Translation Services
Abstracts of the papers and messages from the INMM of the *Journal* will be translated into Japanese and distributed to members as a chapter's new activity to be served from Volume XVII, Number 1, October 1988 issue.

Mitsuho Hirata, Chairman
INMM Japan Chapter
Japan Atomic Energy
Research Institute
Tokyo, Japan

Physical Protection

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

- 30th Annual Meeting of the INMM will be held July 9-12, 1989 at the Stouffer Resort, Orlando, Florida. Approximately 60 physical security related papers are anticipated.
- Workshop, "Detecting Outsiders and Insiders by Integrating the Elements of Delay, Intrusion Detection, and Entry Control into Physical Security Systems," tentatively scheduled for the fall of 1989 in the mid-East Coast area.
- Workshop on "Package Search Techniques," is currently being considered, but has not been tentatively 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 Kasum, 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 the group activities are given below.

General

The 30th Annual Meeting of the Institute of Nuclear Materials Management will be held July 9-12, 1989 at the Stouffer Resort, Orlando, Florida. In addition to the usual sessions we hope to have sessions on computer security and live fire range risk analysis.

Integrated Safeguards

This workshop was held at the Marriott Hotel in Albuquerque, NM, October 17-20, 1988. This workshop focused on administrative, technical and operational problems relating to integration of safeguard systems. The program provided participants with the opportunity to present, discuss and exchange information on the problems associated with this topic. The workshop was very well attended with approximately 90 participants. The workshop was jointly sponsored by the INMM Materials Control and Accountability and Physical Protection Working Groups. The Co-chairmen were Jack Markin, Los Alamos National Laboratory and Ivan Waddoups, Sandia National Laboratories.

Detecting Outsiders and Insiders by Integrating the Elements of Delay, Intrusion Detection, and Entry Control into Physical Security Systems

Workshops in this general topic area have been very interesting and well attended in the past. The most recent one was held in November 1987 in Kerrville, Texas. The next one is tentatively scheduled for the fall of 1989 in the mid-East Coast area. Douglas Kunze, (703) 934-4038, PSC, Inc., and James Hamilton, (614) 289-2331 ext 2204 or 2109, Martin Marietta Energy Systems, are the workshop co-Chairmen.

James D. Williams, Chairman
INMM Technical Working Group on Physical Protection
Sandia National Laboratories
Albuquerque, New Mexico

N14 Standards

PATRAM 89

Larry Harmon reported on the progress of the PATRAM 89 Conference. The conference will be held June 12-16, 1989 at the Crystal Gateway Marriott Hotel in Arlington, Va. Sponsorship is by the U.S. Department of Energy and the IAEA. Over 200 papers will be presented in four concurrent sessions. Attendance is expected to be 600-1000 participants.

N14 Annual Meeting

The annual meeting of the N14 Committee is set for June 11, 1989, immediately preceding the PATRAM conference. N14 attendees will be able to obtain conference rates at the Marriott. The meeting will start at 1:30 pm and last approximately four hours. M. Welch will coordinate with the logistics chairman for PATRAM on meeting space. Details will be sent to all N14 members as soon as possible.

INMM Annual Meeting

The INMM 1989 Annual Meeting will be held in Orlando, Florida, July 9-12. Only one session will be devoted to transportation this year as the INMM meeting follows the PATRAM meeting. There will be a summary of the PATRAM conference presented. Of interest to N14 will be a one-day session on standards at the INMM meeting. Presentations at the morning standards session will be made by ISO, ANSI, ASTM, N14, IEE, ANS, ASME and others. The afternoon session will be a panel discussion on general understanding of standards.

Utility Participation

The Nuclear Standards Board recently completed a survey on utility participation in standards activities. Out of 54 utilities:

- 6 utilities provide 50% of utility/ participation on standards committees,
- 15 utilities provide 25% of utility/ participation, and

• 37 utilities provide the remainder of participation.

N14 Scope

The Management Committee approved submitting for N14 ballot changing the scope of the N14 Committee to include "non-nuclear hazardous waste and combinations thereof." Ballot and background information will be mailed to all N14 members. If the change in scope is approved by the membership, a task force will be established to address the committee's future actions.

Membership Information Forms

By diligently pursuing the members, Arendt has received forms from all but six members. The summary of information shows that the N14 membership includes six Ph.Ds and 16 professional engineers (PEs) that are licensed in at least one state. The membership tabulation of disciplines is as follows:

Discipline	*Number*
Mechanical Engineer	15
Physics	14
Chemical Engineer	9
Business Admin.	7
Nuclear Engineer	3
Civil Engineer	2
Ceramic Engineer	1
Electrical Engineer	1
Chemistry	1
Law	1
Law Enforcement	1
Total	55

Federal Agency News

DOT—Mike Wangler. Notice of proposed rulemaking will be published in the Federal Register in March 1989, regarding conformance to new IAEA regulations. The Final Rule should be ready by January 1990. Major changes are: (1) three categories of LSA, (2) strong package gives way to IP 1, 2, and 3, and (3) new occupational radiation protection standards.

NRC—C. MacDonald. NRC is in a new home near White Flint Mall in Rockville, Md. NRC published its proposed rule in June 1988, regarding changes to conform to IAEA regulations. NRC is extending the comment period to coincide with DOT rulemaking. Most comments have been on changes related to LSA material, the venting rule and use of metric system. NRC has several new publications of interest to N14:
Guide for Preparing Operating Procedures for Shipping Packages, NUREG/CR-4775 (LLL).
Public Information Circular for Shipments of Irradiated Reactor Fuel, NUREG-0725, Rev. 6.
Directory of Certificates of Compliance for Radioactive Materials Packages, NUREG-0383, Rev. 11, Vols. I, II, and III.

DOE—Larry Harmon. DOE now has a policy that all packagings must be certified by the NRC. Number one problem area in transportation is still public acceptance. DOE is addressing this by being prepared for accidents and through emergency response planning. There is pressure to comply with NEPA and DOE plans to update NUREG-0170, the EIS for transportation of nuclear materials. Special trains are still a major issue and DOE is documenting the history of all shipments of radioactive materials by rail and the associated safety record. The Environmental Restoration Budget will increase to cover all Defense Program areas.

Other

The next Management Committee meeting will be in October, 1989. The N14 Annual Meeting for 1990 will be held after the INMM Annual Meeting.

John W. Arendt, Chairman
INMM/ANSI N14 Committee
Consultant
Oak Ridge, Tennessee

N15 Standards

Recent N15 highlights include: approval of one revised standard; a Nuclear Standard/Project Initiation Notice and Data Sheet(s) (NUSIND) was submitted; and an ERRATA Sheet was issued to correct errors in a standard. On the negative side, a NUSIND for one standard has yet to be approved; the chairman of a subcommittee resigned and a couple more need to be replaced. Details of the positive and negative, as well as a status list of all standards is provided below.

Approved

A revised N15.19-1988, "Volume Calibration Techniques for Nuclear Material Control," was approved. Writing Group Chairman Al Liebetrau and the entire group deserve to be congratulated.

NUSIND

The INMM 2.2 subcommittee submitted a NUSIND for proposed standard N15.1, "Unirradiated Uranium Scrap Classification". Chairman Bill Hopwood has formed a diversified writing group to pursue development.

Barbi Wilt has continued to diligently pursue the approval of the NUSIND for proposed standard N15.28, "Guide for Qualification and Certification of Safeguards Personnel". Numerous questions have been addressed but all issues have not yet been resolved.

Errata Sheet

An errata sheet for ANSI N15.10-1987 was issued to correct one error and three inconsistencies noted after publication.

Chairmanships

Gary Kodman resigned as Chairman of INMM-1, "Accountability". A replacement has yet to be identified and all suggestions are welcome.

Obie P. Amacker, Jr., Chairman
INMM/ANSI N15 Committee
Pacific Northwest Laboratories
Richland, Washington

Bylaws Update

Publication and distribution of the recently revised INMM bylaws has been postponed so that charters for all standing committees may be included. This project was initiated subsequent to the approval of the revised bylaws and is still in progress. We hope to have the new printing of the bylaws with the committee charters by the July 1989 INMM Annual Meeting.

Awards and Fellows

Reminder: Nominations for INMM Awards and Fellowship are now being accepted. Contact INMM Headquarters for details, (312) 480-9573, 60 Revere Dr., Suite 500, Northbrook, IL 60062 U.S.A.

Safeguards

Robert F. Burnett, Director, Division of Safeguards and Transportation, Nuclear Regulatory Commission (NRC) provided the introduction and facilities for the INMM Safeguards Committee Meeting. Major safeguards activities at the NRC included threat awareness, consolidation of inspections from regions to the Headquarters (Fuel Cycle only), the Japanese Bilateral agreement, and a reorganization of NRC.

In Burnett's reorganization, he has created three branches: 1) International Safeguards Branch headed by M. Smith, 2) Domestic Safeguards and Regional Oversight Branch headed by D. Kasun (Acting), and 3) Transportation Branch headed by C. MacDonald. This reflects the current work and activities within his branch.

D. Kasun discussed the current work in the NRC/DOE (Department of Energy) comparability rule. It is based upon a 1986 NRC/DOE review and applies to Category I material fuel facilities. A final rule was published in November 1988, and plan changes are due in June 1989, and full implementation by June 1990. A major thrust of the upgrades include tactical response team exercises, night firing qualifications, searches, and improved barriers. There will also be rule making in the area of transportation which deals with CAT I and II material.

Material Control and Accounting licensing will be sent to Headquarters for approval in the near future. Also, inspections will be consolidated into Burnett's office for Fuel Cycle Facilities.

Jim Cook, NRC, provided an update of 0170 EIS Transportation actions. They are looking at the environmental statement for all modes of transportation. Spent fuel, radiological safety and safeguards, routine and accidents in transport are some of the elements they are considering. A document published in 1977 based upon data in the 1975 time frame is being updated to be effective during the 1995 to 2005 time frame. The new document will be published in 1991.

Bob Dube, Office of Nuclear Reactor Regulation (NRR), reviewed the current RER program. This is a Headquarters team of about 7 people that visit nine reactor sites each year to evaluate the effectiveness of safeguards and contingency plans. They have completed 80 tests at 51 sites. The utilities are now hiring consultants to do testing before and after the NRC team arrives. It appears this has been very useful for both the NRC and the utilities in identification of problems and associated solutions.

Don Emon, DOE, provided a complete update on DOE MC&A work at the DOE. They issued new orders at the end of 1986 and early 1987. The facilities have until mid-April to be in compliance with the new orders. Within these orders is a graded safeguards table which is used for performance standards on MC&A systems. The details of the DOE program were discussed and reviewed with us. Attachment H provides more detail in this area. My observation is that the approach being utilized by the DOE in this area has been well-formulated and highly structured to provide a comprehensive way to evaluate the effectiveness of MC&A systems. It is the best approach that I have seen to the MC&A problem in the last 15 years. Don Emon deserves recognition for his work in bringing a tough problem into focus with a logical approach for evaluating the effectiveness of these systems.

Leon D. Chapman, Chairman
INMM Safeguards Committee
BDM Corporation
Albuquerque, New Mexico

Yucca Mountain, Nevada: Is It a Safe Place for Isolation of High-Level Radioactive Waste?

Carl P. Gertz, Project Manager
U.S. Department of Energy Yucca Mountain Project
Las Vegas, Nevada U.S.A.

ABSTRACT

As mandated by Congress in the amended Nuclear Waste Policy Act, Yucca Mountain, Nevada is currently being studied by the U.S. Department of Energy to determine if it is a suitable location for the nation's first geologic repository for disposal of high-level radioactive waste and spent nuclear fuel. While the site has attributes that suggest it may be suitable, DOE must spend the next five to seven years and up to $2 billion to find out if the site would be able to isolate radioactive materials for 10,000 years. Site characterization studies are being conducted to determine the geologic, hydrologic and environmental characteristics of the Yucca Mountain site. The DOE is planning to start construction of an underground laboratory in 1989 and to implement an extensive surface-based study program. The next five to seven years of study will determine if Yucca Mountain can meet regulations for waste isolation and qualify for a Nuclear Regulatory Commission license for repository construction and operation.

The answer to the question posed in the title of this paper runs the gamut from yes to no, depending on who's telling the story. Some would suggest the U.S. Department of Energy and other interested parties say "yes", and that they are proceeding to build a high-level radioactive waste repository at Yucca Mountain, some 100 miles northwest of Las Vegas, Nevada. In a recent article critical of the repository project, a respected major newspaper stated that there remain serious questions about the suitability of the proposed site. On that point, they were correct: there are indeed many questions to be answered. Nobody knows if Yucca Mountain is the right place to build a repository. Based on the knowledge we have now, the proposed site appears to be a favorable candidate, however we are nowhere near a final conclusion. The next five to seven years and $1 billion to $2 billion of study at Yucca Mountain will provide the answers to many questions. If it is not a safe location, we certainly do not want to recommend construction of a repository at Yucca Mountain.

The objective of a repository is to utilize natural and engineered barriers to isolate radioactive materials from the environment for 10,000 years. By regulation, we cannot count on the integrity of engineered barriers beyond 1,000 years. Therefore, the performance of a repository is going to rely heavily on natural barriers, primarily a formation of volcanic welded tuff at Yucca Mountain.

We must prove that radionuclides packaged approximately 1,000 feet below the surface and approximately 1,000 feet above the water table will not migrate to the accessible environment (5 km from repository) for 10,000 years. We are looking at the tuff formation and dry climate of Yucca Mountain to meet this objective. Water travel is the mechanism that could carry radioactive materials to the accessible environment. We presently believe it takes from 20,000 to 80,000 years for water to move from the unsaturated geology in the proposed repository horizon to the water table below. Site characterization studies are planned to give us an understanding of the unsaturated zone and other components of the proposed repository.

The Yucca Mountain Project site characterization phase officially began May 28, 1986, when President Reagan recommended Yucca Mountain be studied as a potential site for a high-level nuclear repository. The program schedule calls for issuance of an environmental impact statement (EIS) in 1994, submission of a license application in 1995, and U.S. Nuclear Regulatory Commission (NRC) review from 1995 to 1998, followed by a five-year construction period and the capability to receive waste shipments in 2003. This represents an extremely aggressive schedule, with many challenges along the way. If at any point it is determined that Yucca Mountain will not meet NRC regulations, DOE must turn to Congress for further direction.

The Yucca Mountain Project employs 1,400 scientists, engineers, and support personnel working with a fiscal year 1989 budget of over $200 million. Less than 100 project staff are DOE employees. The majority of project employees work for contractor companies and national laboratories. Sandia National Laboratories is responsible for repository facility design and performance assessment. Lawrence Livermore National Laboratory is working on waste package design, while Los Alamos National Laboratory is conducting geochemical investigations, volcanism studies and is responsible for underground test implementation. The U.S. Geological Survey is performing geologic, hydrologic and climate investigations. Design of the mined exploratory shaft facility (ESF), the underground laboratory used in site characterization, is being prepared by Fenix & Scisson and Holmes & Narver. Construction of the ESF and general site support will be provided by Reynolds Electrical and Engineering Company. Science Applications International Corporation is responsible for project management and integration, regulatory compliance, institutional affairs and project quality assurance. Mac Technical Services provides quality assurance consulting services.

In December 1988, a major program milestone was achieved when the Site Characterization Plan (SCP) was released on schedule to the NRC, State of Nevada, and the public. The SCP contains a description of the site as we know it today and an outline of design for the repository facility and waste package. Chapter 8 comprises about one half of the document's 6,300 pages, describing the studies necessary to determine if Yucca Mountain is a suitable place for a high-level nuclear waste repository and to provide the needed information for a construction and operation license application to the NRC. The document is essentially a map of where we are, and a blueprint of where we're going and how we're going to spend up to $2 billion to understand the Yucca Mountain site.

The release of the SCP was noted in the Federal Register on December 28, 1988. The public, NRC, State of Nevada, local governments, and nuclear utilities are expected to comment on the SCP in writing. The Yucca Mountain Project Office held four public update meetings in Nevada in February, which included discussion of the SCP. At those meetings, project scientists answered citizens' questions about the document. Public hearings will be held at three locations next month as part of the formal comment process.

The site characterization testing program includes activities to establish the geologic conditions and history of the site. We intend to break ground for construction of the ESF in 1989, and we are planning to conduct extensive surface-based tests.

The ESF will be an underground laboratory where tests will be conducted to study the processes and phenomena contributing to waste isolation performance, the effects of ESF construction, and underground conditions at the ESF location. The ESF will consist of two 12 foot diameter, 1100 foot deep shafts, two miles of horizontal drifts and tunnels, and a variety of demonstration and testing rooms. The facility will be constructed using drill, blast, and muck mining techniques. While the ESF construction activities will appear much like a conventional mining operation, there are unique and important differences. Progress will be slow, with miners working one shift followed by scientists who will take two shifts or longer to study the geology as the excavation progresses. It is going to be a mining operation with high-technology science in between, where science takes precedent over mining activities. Figure 1 shows what types of test will be performed at various levels of the ESF.

The surface-based testing program is a series of investigations designed to characterize the geologic environment throughout the repository area. The types of investigations include geology, volcanology, hydrology, tectonics, and geo-engineering studies. The investigations will be conducted through a series of drill holes, trenches, nondestructive geophysical measurements, seismic reflection surveys, and laboratory work.

The major part of surface-based testing is the drilling program. There are 329 drill holes planned, with emphasis placed on the unsatured zone that lies above the proposed repository depth. The holes will be from 4 inches to 12 inches in diameter and will be bored as deep as 5000 feet below the surface of Yucca Mountain. The surface-based program includes seismic, stream flow, and meteorological monitoring activities that are currently under way.

Another activity planned for the surface-based testing program involves trenching to investigate surface traces of geologic faults. Data from the trenches will characterize the magnitude and history of past movement on faults throughout the site area that may have been active in the last 10,000 to 2 million years. Trenching will also be used to investigate the nature of potential faulting at the proposed site for repository surface facilities.

While limited field work has been performed, most activity to date on the Yucca Mountain Project has involved planning and preparation for actual hands-on work. Our primary short range goal is to "move dirt" in 1989, an objective that has several hurdles to cross before it is real-

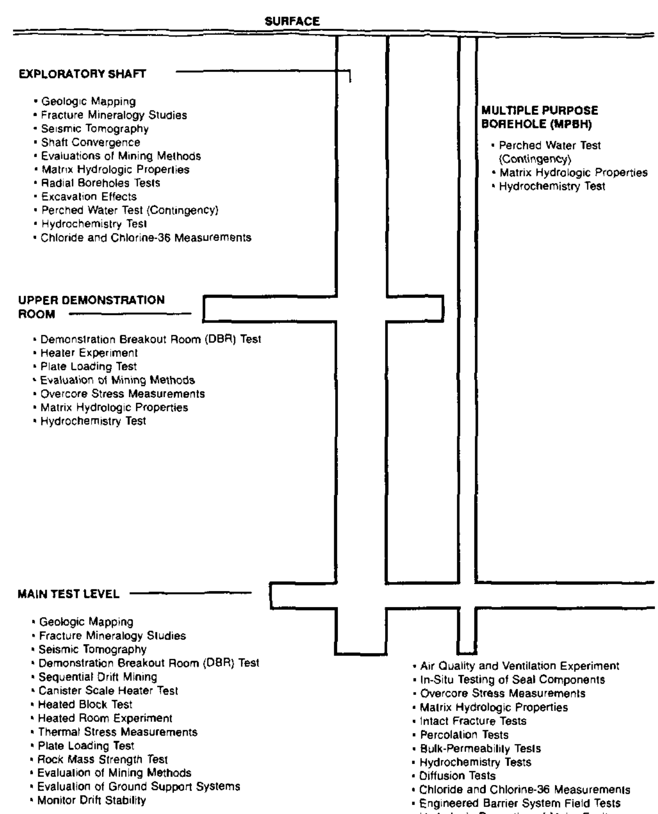

Figure I. Location and type of test in the exploratory shaft facility.

ized. In order to break ground for the exploratory shaft facility, there are six prerequisites that must be met.

1. The DOE must gain full legal access to the federal lands on which the Yucca Mountain site lies. We already have a right-of-way (ROW) reservation for Bureau of Land Management (BLM) property, and the Nevada Test Site (NTS) portion has never posed an access problem. However, we are waiting for the BLM to grant a ROW for Nellis Air Force Base Bombing and Gunnery Range property. The U.S. Air Force must complete an environmental assessment and submit it to the BLM prior to a ROW being issued. We anticipate the process being completed in the near future and expect to have full land access shortly. ROW reservation access will permit studies to be conducted in the proposed repository area, but will not assure DOE exclusive use of the properties. DOE has submitted a land withdrawal application to the BLM, which would allow DOE to control the properties. The State of Nevada filed suit against the BLM and the Department of Interior, asserting state approval is necessary for the DOE land withdrawal. A ruling last month by a U.S. District Court Judge denied Nevada's right to approve the land withdrawal and the case was dismissed. The state attorney general's office has indicated the ruling will be challenged in the 9th U.S. Circuit Court of Appeals.

2. We are having difficulty obtaining environmental compliance permission from the State of Nevada. DOE applied for an air quality permit for land disturbance over a year ago, but the state has taken no action as of this time.

3. Nuclear Regulatory Commission comments on the SCP must be considered prior to commencing ESF construction. NRC said recently it needs until mid-August 1989 to review and comment on the SCP. Actual construction cannot begin until DOE receives and considers NRC comments, although site preparation work can start earlier.

4. The DOE must create Study Plans to help define an appropriate level of detail for characterization studies described in the SCP. The Study Plans must be reviewed by the NRC before those specific activities are undertaken. Five ESF Study Plans have been submitted to the NRC.

5. Another area requiring NRC acceptance is quality assurance. The NRC recently approved the Yucca Mountain Quality Assurance Plan. A fully qualified QA program should be in place by summertime, once all project participant organizations implement procedures and personnel training that will assure adherence to the overall project QA plan. With a fully-qualified QA program in place, the project will be prepared to collect new site characterization data that is admissible for NRC licensing purposes. When entering the licensing arena, exemplary scientific work alone is not enough. The work must be of the highest quality and be performed under strict adherence to quality assurance and technical procedures. The bottom line is that for licensing activities, our data is not data unless the NRC says it's data. We are implementing measures that will assure our scientific activities produce usable data.

6. The final prerequisite DOE must meet is completion of the ESF design. An ESF Title 1 Design Report has been completed and Title 2 Design work is now under way. We should be able to start site preparation in May with road and pad construction.

As part of the effort to provide usable scientific data, DOE opened the Yucca Mountain Project Sample Management Facility (SMF) in July 1988. The facility is housed in two renovated, access-controlled 14,000 square foot warehouses near Yucca Mountain on the NTS. The SMF will use advanced technology to process, document, and preserve borehole core samples and samples taken from the ESF. The facility is designed to document where a geologic sample came from, who had access to it and what research was performed with the sample.

The national repository program is one of the most closely reviewed projects ever undertaken by the Federal government. The extensive oversight was called for by Congress and is necessary to assure safety and public confidence. One of the strictest and most important oversight roles is served by the NRC. Not only does the NRC consider a license application when and if Yucca Mountain is found suitable, they are a key player at all stages of the program. No major steps are taken without consultation with the NRC. The DOE interacts with the State of Nevada on a continuous basis. The state is operating on an $11 million grant fund in 1989 for independent oversight, and three Nevada counties are receiving $5 million for oversight. The state has a legislative committee and a nuclear projects commission that follow the program. A Nuclear Waste Technical Review Board has been nominated by the National Academy of Sciences and approved by the President to oversee the repository program. The nuclear electricity generation utility companies who are paying for the program have a significant role in monitoring the program's progress through the Edison Electrical Institute (EEI). Additionally, the U.S. General Accounting Office (GAO) is an active participant and provides quarterly and yearly reports to Congress.

Is Yucca Mountain a safe place for a high-level nuclear waste repository? Data gathered in preliminary studies suggests that Yucca Mountain possesses many of the attributes that a repository site must have, but we can't begin to answer that question until we move forward with site characterization studies. In summary, the release of the SCP is the springboard for Yucca Mountain site characterization activities in 1989. We're looking forward with great anticipation to moving dirt at the site this year, so the process of answering questions and testing theories can begin.

Carl P. Gertz received his B.S. in civil engineering from Michigan State University and an M.S. in Systems Management from the University of Southern California. He is currently a project manager of the Yucca Mountain Project Office at the Department of Energy's Nevada Operations Office. Before going to Nevada, Gertz was manager of the Special Isotope Separation Project Office at DOE's Idaho Operations Office where he headed the effort to design, develop and operate a facility that uses lasers to separate isotopes of plutonium. Prior to joining DOE, he worked 16 years in the intermountain states for The Boeing Company in missile site development, installation and construction management.

Status and Projected Activities of the Monitored Retrievable Storage Review Commission

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ABSTRACT

The Nuclear Waste Policy Amendments Act of 1987 established a Monitored Retrievable Storage Review Commission. The nature of this assignment, the commission's activities, and its plans for completing the assignment are described.

Federal monitored retrievable storage was first proposed as a retrievable surface storage facility (RSSF) by the Atomic Energy Commission (AEC) as early as 1972. It was originally proposed to provide back-up storage capacity until a repository was ready to receive waste. Because of the controversy it engendered, no federal facility for interim storage has been built to date. However, the concept was discussed repeatedly over the years until Congress, in the Nuclear Waste Policy Act of 1982, directed DOE to submit a proposal by June 1, 1985 "for the construction of one or more monitored retrievable storage facilities for high-level radioactive waste and spent nuclear fuel."

DOE developed a proposal and in April 1985, identified three candidate sites for an MRS facility, all of which were located in Tennessee. Tennessee filed suit in U.S. District Court to prevent submittal of the proposal to Congress alleging that in not consulting with the state, DOE failed to comply with the Nuclear Waste Policy Act. Extensive court proceedings ensued but the way was finally cleared for DOE to submit the proposal to Congress in March 1987. At that time, DOE formally proposed to site the facility in Oak Ridge at the site formerly proposed for the Clinch River Breeder Reactor. Later that year, however, Congress revoked the DOE proposal to site the facility at Oak Ridge in the Nuclear Waste Policy Amendments Act of 1987 and directed that a study of the need for an MRS be conducted before DOE could begin again the search for a site for such a facility.

ARGUMENTS FOR AN MRS

The provisions in the 1987 amendments regarding the MRS were among the most controversial in the act. Some say the difficulty in reaching agreement on those provisions threatened passage of the entire bill. Proponents of the MRS (principally DOE and many utilities) argued that, in addition to providing needed spent fuel storage capacity, a central integrated monitored retrievable storage facility would (1) accelerate the waste disposal program by allowing early licensing and implementation of substantial portions of the program, including establishment of a transportation system, and (2) increase system flexibility and reliability.

Others, for example the State of Tennessee, environmental groups, and the General Accounting Office (GAO), argued that DOE did not present a fair comparison of an MRS and a no-MRS option. They asserted that DOE had (1) underestimated MRS costs and risks, (2) ignored risks associated with rod consolidation, and (3) used sub-optimal transportation strategies for the no-MRS options.

The utilities took the position that linking the MRS to the repository would severely limit the usefulness of the MRS while environmental groups claimed that an MRS not tied inextricably to the repository would become a de facto repository because it would provide an opportunity for indefinitely delaying the hard political decisions needed to put a permanent geologic repository into operation.

THE MRS REVIEW COMMISSION

It was in this environment of complex and controversial issues that Congress decided in the Nuclear Waste Policy Amendments Act of 1987 to create the Monitored Retrievable Storage Review Commission.

The Commission's primary objective, to produce a report assessing the need for an MRS facility, relates closely to several other provisions of the 1987 amendments to the NWPA. While Congress revoked the Department of Energy's decision to site an MRS on the Clinch River in Roane County, Tennessee, Congress did authorize the Department to site, construct, and operate one monitored retrievable storage facility subject to certain conditions. Even if those conditions are met, of course, MRS could be challenged during the appropriations process. However, the conditions are of particular interest to the Commission.

One of those conditions is that the Secretary of DOE must conduct a survey and evaluation of potentially suitable sites for an MRS facility according to certain criteria—but may not do so until the MRS Commission submits its report to Congress. Also, in response to the concern expressed continuously over the years that an MRS facility might become a permanent repository by default, Congress placed certain restrictions—more commonly called "linkages"—on the operation of any MRS facility.

These "linkages" state that construction of any MRS facility may not begin until the Nuclear Regulatory Commission has issued a license for the construction of a repository and that construction of any MRS facility or acceptance of spent nuclear fuel or high-level radioactive waste shall be prohibited if the repository license is revoked by the NRC or if construction of the repository ceases.

The "linkages" also limit the amount of spent fuel processed or stored at a facility. A maximum of 10,000 mtu of spent fuel may be received at any MRS facility prior to the acceptance of spent fuel or high-level waste at the repository. And a maximum of 15,000 mtu of spent nuclear fuel may be stored at any one time at any MRS facility.

MRS COMMISSION, NWPAA PROVISIONS

The duties and limitations of the MRS Commission are specified in the NWP Amendments Act of 1987, too. I would like to focus on just a few points of possible interest. The Nuclear Waste Policy Amendments Act requires us to review the adequacy of the Department of Energy's evaluation of the advantages and disadvantages of creating an MRS, obtain comment and data from affected parties, evaluate the need for an MRS from a technical perspective, and make a recommendation to Congress as to whether an MRS facility should be included in the national nuclear waste management system.

We are also required to compare an MRS facility to the alternative of at-reactor storage of spent fuel. In doing so, the law states that we shall take into consideration repository design and construction, waste package design, waste preparation, the waste transportation systems, the reliability of each option to ensure waste disposal, and economic factors, including the cost impact imposed on the nation's ratepayers.

The statute originally required us to deliver our report to Congress on June 1, 1989. However, in October of last year, to compensate for the additional months it took Congress to appoint the three Commissioners, we sought and obtained a legislative extension to submit the report on November 1, 1989. My fellow commissioners are Alex Radin of Radin and Associates in Washington, D.C. and Dr. Dale Klein of the University of Texas. Alex Radin is serving as Chairman.

THE COMMISSION'S ACTIVITIES

To help us prepare the report, we have hired a small staff of professionals including a transportation specialist, an economist, and two systems analysts. The day-to-day activities of the staff are managed by our Executive Director and General Counsel, Jane Axelrad, who is on loan to us from the Nuclear Regulatory Commission. The Commissioners are operating as a collegial body and will be very involved in directing the research and writing of the report.

To make our evaluation as professional and unbiased as possible, we are reading and evaluating the work that has been done on this subject to date, and where necessary, conducting our own studies and evaluations. As part of this process, we have visited the H.B. Robinson Nuclear Project in Hartsville, South Carolina and the Surry Nuclear Power Station in Surry, Virginia, the only commercial plants in the country which have already developed dry storage facilities for spent fuel. (Other utilities are exploring the possibility of dry cask storage). At the Surry and Robinson facilities we observed two different types of dry storage, examined spent fuel handling and transportation facilities, and discussed with the two utilities their plans and concerns regarding the need for a federal MRS.

In addition to our U.S. site visits, we have visited four countries in Europe to examine possible components of an MRS system. There, we saw many different approaches to spent fuel management ranging from wet storage at CLAB in Sweden to the dry storage of vitrified high-level wastes from reprocessing at La Hague. We visited the Gorleben facility in the Federal Republic of Germany and the Grimsel underground laboratory in Switzerland.

During our visits to these countries we were struck by the difference in emphasis on monitored retrievable storage versus permanent disposal. Some of the European countries we visited seem to have solved their interim spent fuel storage problems with relative ease by building centralized storage facilities. Most are now addressing the problem of permanent disposal on a more attenuated schedule than is the United States.

It must be remembered, however, that European programs are much smaller in magnitude than ours, that distances within countries are shorter, and that population density is greater. Consequently, while we have learned and will continue to learn a great deal from the European experience, their experience is not necessarily directly applicable here.

In addition to gathering data by site visits and through evaluations being conducted by the staff and ourselves, we have also decided to provide as much opportunity as possible to listen to the views of all parties interested in the issue. Toward this goal, last July, we held a three-day series of public briefings during which we heard from some of the parties which were previously involved in the debate over the need for an MRS facility, including the Department of Energy, members of Congress and Congressional staff, the General Accounting Office, the Nuclear Regulatory Commission, the nuclear industry, the State of Tennessee, and environmental groups. We have since expanded the opportunity for input to receive testimony from a broader spectrum of people and organizations through a series of public hearings held in different sections of the country. We have held hearings in Washington, D.C., Denver, Colorado, and San Francisco, California and will be holding our last hearing in this series in Atlanta, Georgia on January 17 and 18. The purpose of the hearings was to solicit the views of interested persons on the need for an MRS and, particularly, on any of the topics specified in the Nuclear Waste Policy Amendments Act. I am happy to say that our goal of expanding public participation through this process was realized. We heard a very wide variety of views expressed during the hearings and look forward to receiving additional written testimony before the record of the hearings closes. Of course, we will welcome the views of interested persons throughout the period of our evaluation.

In addition, as we identify the need for specific information on particular topics that require briefings from other agencies or interested parties, we shall request them. The scheduled briefings will be open to the public so that interested persons can follow what we are doing. All of our hearings and public meetings are transcribed and the transcripts are available for review in our Public Document Room.

DESCRIPTION OF EVALUATION

Our thoughts on the direction our evaluation will take are very preliminary at this point, but I would like to mention how we plan to structure our evaluation and some of the issues we intend to address. One of our first tasks will be to define possible MRS configurations, including possible functions one or more facilities might perform. We will then describe the generic assumptions underlying such issues as fuel characteristics, projected fuel inventories, available rod consolidation technologies, wet and dry storage possibilities at reactors and at MRS facilities, storage and transport cask capabilities, truck and rail capabilities, and repository locations that will form the basis for our evaluation. We are working now to define a representative series of spent fuel management strategies that include both MRS and no-MRS options and various scenarios under which the alternative strategies will be evaluated and have begun to discuss the methodology we will use to conduct our evaluation.

Among the topics we are considering in our development of criteria are:

- Safety and licensing considerations and technology;
- Systems analysis considerations, including the public policy benefits and costs associated with MRS and no-MRS options; the effects of the statutory linkages on those benefits and costs; and generic siting considerations including public perceptions;
- Transportation issues, including transportation mode choice; routing; and cask capacities and functions;
- Economic issues, including facility construction and operating costs; transportation costs; and potential benefits and costs to local economies.

In the next few weeks, we expect to announce the conclusion of contract negotiations for studies to augment our own staff's efforts. We intend to hire contractors to develop some models and to assemble data bases regarding the need for storage, rod consolidation, economic costs and transportation risks and costs. Both the Department of Energy and the State of Tennessee are doing further studies on MRS facilities and we look forward to receiving the results of those studies. We are also looking forward to a briefing by the Department of Energy by mid-March on the results of their systems studies of alternative spent fuel management strategies.

Since our real work is just beginning, I am unable to share with you any likely findings or recommendations but I can assure you we intend to take a good hard look at the technical data that are available and consider all of the public policy implications of the various strategies before we reach any decisions. Key to our findings will be the assumptions we deem most creditable and the weight we accord to the various criteria used in judging each of the scenarios.

Frank L. Parker is the Alexander Heard Distinguished Service Professor at Vanderbilt University. Prior to joining Vanderbilt, he was at the Oak Ridge National Academy for 10 years and at the International Atomic Energy Agency for a year and a half. In addition to being a Commissioner of the Monitored Retrievable Storage Review Commission, he is Chairman of the Board of Radioactive Waste Management of the National Academy of Sciences and Harvard University, and is a member of the National Academy of Engineering.

Spent Fuel Storage and Management in the United Kingdom

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ABSTRACT

During the past 33 years, fuel of various types have been stored, transported and reprocessed in the United Kingdom.

This paper provides an overview of those programs starting from the Magnox stations, through the AGR program and the move to LWR fuel. Throughout this time BNFL has provided services for fuel storage, reprocessing, transportation and the enrichment and fabrication of new fuel.

The development of new plants and processes to handle the changing fuel types and the associated waste management schemes will be addressed. A description of future plans for fuel storage and reprocessing will be discussed.

1. MAGNOX PROGRAM

The Calder Hall Magnox station went on line in 1956, representing the world's first commercial scale nuclear station. The success with this program lead to the development of 26 reactors. The program is named for the fuel cladding which is an alloy of Aluminum and Magnesium. The fuel transfers its heat to a circulating CO_2 gas during operation. The plants are refueled on line and the spent fuel is stored under water in skips awaiting transport. This underwater storage is of short duration, primarily to allow for the decay of fission product gases. The chemically reactive nature of both the uranium fuel, and its cladding in water, limits the long term potential for wet storage. The most economical solution to this problem is reprocessing.

After cooling, the fuel is loaded into transport casks for road and rail transport to BNFL's reprocessing plant at Sellafield. The cask for Magnox fuel is designed to transport about 3 MTU of fuel in a wet condition. Each package weighs about 45 tons and is capable of heat loadings to about 6 kw. It is this transport package that has become so well recognized through the CEGB crash test with the 100 mph locomotive.

At Sellafield, the cask is unloaded underwater and the skip, containing the fuel, is removed and placed in wet storage for approximately 140 days. To prepare the fuel for reprocessing the alloy cladding is mechanically removed from the uranium rod. The cladding is stripped from the uranium by forcing the fuel element through a die and cutter. This method, as opposed to complete dissolution of the entire assembly, was chosen as it minimizes the quantity of waste produced and allows for continuous feeding of a dissolver, rather than a batch process.

The declad uranium rod is fed to a nitric acid dissolver and traditional reprocessing flow sheet is used to recover uranium, plutonium and to separate the high active waste from the two valuable elements.

The waste cladding from the Magnox element will be encapsulated in a cement/blast furnace slag matrix in 500 liter drums. The high active waste will be vitrified in borosilicate glass. The uranium is recovered for reuse in new fuel elements and, to date, over 15,000 MTU have been recycled. 1500 MTU have been recycled for the Advanced Gas Cooled reactor program. The plutonium is stored and will be used to support the Fast Breeder Program and Mixed Oxide Fuel for other systems.

To date, over 30,000 MTU have been reprocessed. The plant for the initial program was built in 1952 and after 12 years of operation was shut down. The new plant came on line in 1964 and was designed for higher throughput to match the growing number of Magnox stations and to meet the increasingly stringent effluent release requirements. The excellent reliability and availability (95%) have validated the design philosophy and engineering concepts. A potential for effecting availability was the unexpected corrosion of a dissolver after 14 years of service. However, a standby, duplicate unit was placed in service while decontamination and refurbishment work proceeded. Similarly, corrosion required the regular planned repair of medium active evaporators.

A new fuel storage and decanning plant came on line in 1985. The building is approximately 1000 ft. long, contains three Magnox storage pools and two fully automated decanning lines. Fuel from the pool is transferred to the decanning cell by a remote automated process. The plant is intended to meet the needs of the Magnox program until 2005.

2. THE ADVANCED GAS COOLED REACTOR PROGRAM

During the 1970's, a program to determine the future reactor mix for the UK was started. The choices were a Light Water Reactor, a Steam Generating Heavy Water Reactor and the Advanced Gas Cooled reactor. The latter was chosen as it had elements that were similar to the existing Magnox program. Additionally, training and technology developed during the past 20 years could be directly applied. There were some important differences from a fuel management view. The new fuel would be stainless steel clad, uranium oxide contained in a graphite wrapper.

The AGR program to date has had a varied performance. The designs, like the Magnox stations are fairly diverse, although all are of a similar concept. Some stations have operated at very high load factors and others have been plagued with construction and safety problems. The Central Electricity Generating Board (CEGB) has not had good results with this program and has elected not to build any more AGR's. The South of Scotland Electric Board (SSEB), however, believes that the Hunterson Station is a success and is glad to have another station, Torness.

The total program to date is 14 reactors with about a 10,000 MWe output.

While the fuel appears similar to LWR fuel, at least in pin construction, there was insufficient long term data to provide assurances of long term integrity during pond storage. With this background, the SSEB and CEGB have elected to reprocess about 1850 MTU of AGR fuel at Sellafield in the new THORP facility. At the same time, they are considering the development of a dry fuel store for irradiated AGR fuel. The reason for examining this additional fuel management path is to decouple power plants from reprocessing plant operations. They hope that this will provide:

1. a buffer against reprocessing plant outages,
2. the option of reprocessing after longer storage periods and,
3. the option of not reprocessing if direct disposal becomes an economically attractive or otherwise available option in the future.

AGR fuel is transported to Sellafield by rail in transport packages similar to those used for Magnox fuel. At Sellafield, the fuel is unloaded and stored underwater. Before reprocessing can take place, the fuel must be consolidated. The graphite wrapper around the fuel pins does not lend itself to straight forward shearing and dissolving as LWR fuel does. In addition, the open structure of the fuel assembly does not shear well. A new facility on the end of the storage pools has been built to remove the graphite sleeve and grid spacers and repackage the AGR fuel pins. The process involves cracking the graphite to remove it and then collecting the pins from three elements and placing them in a slotted can. This consolidated fuel is then returned to the pool for storage awaiting reprocessing. In addition this process also increases the efficiency of pool storage space utilization. AGR fuel is expected to cool about three years following reactor discharge before reprocessing. Magnox fuel only requires about six months total cooling.

The waste from this consolidation program will be treated as intermediate level waste and immobilized in cement.

3. LIGHT WATER REACTORS

The recent decision to build Sizewell B and the Inquiry into the Hinkley LWR station shifts the burden of fuel storage to the reactor. These new stations are designed for 18 years of spent fuel storage and no decision has been reached about reprocessing of this fuel.

However, LWR fuel reprocessing will occur in the UK as a result of contracts with overseas customers. LWR fuel will be reprocessed in THORP (Thermal Oxide Reprocessing Plant) which will come on line in 1992. This plant is expected to reprocess about 7000 MTU during its first 10 years of service. To date contracts for 6,000 MTU of fuel have been secured.

The fuel for THORP from overseas customers is transported by road, rail and sea.

In Europe, the fuel is transported by Nuclear Transport Limited (NTL) a company formed between BNFL and TN for the purpose of moving fuel to Cogema's facilities at LaHague and BNFL's plants at Sellafield. In general, this operation is quite similar, except in magnitude, to fuel movements in the US. The spent fuel casks are licensed by IAEA Competent Authorities in the countries the casks move through and carry a B(U) license. These packages are similar in size to ones that would be used in the NWPA repository program. NTL was formed to coordinate transports and to insure that fuel could be economically moved to support reprocessing. BNFL also operates a wholly owned transport company for moving Magnox and other fuels from Europe and Japan to the UK or other locations.

The LWR fuel from Japan moves by ship. The transport company is Pacific Nuclear Transport Limited (PNTL), with BNFL the major share holder and the Japanese utilities and Cogema as minor shareholders. PNTL has 5 purpose built ships that can transport up to 24 spent fuel casks on each trip. The ships load empty casks in Barrow, England and at LeHavre, France.

They travel to Japan via the Panama Canal and arrive at the Japanese reactors to coincide with planned outages, weather conditions and other requirements. The shipping schedules are often determined years in advance of the planned fuel movements. The ships leave Japan, returning to Barrow, where the casks are off loaded to rail cars and then transported to Sellafield. The ship will also call in France, off load casks, which are transported to La Hague by rail and then into the plant by truck.

The companies maintain a transport fleet of over 200 spent fuel casks, railcars, and trucks with dedicated cask maintenance, repair and decon facilities.

PNTL has to date transported over 2500 MTU of fuel, NTL in excess of 4000 MTU and BNFL Transport about 2000 MTU. The three transport companies have over 25 years of experience and have collectively transported in excess of 8500 MTU of irradiated nuclear fuel.

THORP differs from MAGNOX reprocessing in that it relies upon a "chop-leach" batch method while Magnox relies upon a continuous dissolver. Fuel elements, both canned AGR and intact LWR fuel will be sheared into sections between 1 and 4 inches long and pass down a chute into a basket suspended in 90°C Nitric acid. The fuel is dissolved and the liquid is passed to a centrifuge for solids/fines removal and then to a PUR cycle, pulsed column solvent extraction process for uranium/plutonium separation from fission products. The recovered uranium will be converted to uranium tri-oxide prospectively for recycle in a fuel manufacturing plant now being designed. The plutonium, like that from the Magnox program will be used in MOX or fast reactors.

The "hulls", end fittings, guide tubes and other non-fuel components are removed from the dissolver and treated as intermediate level waste. The fission product waste will be evaporated to reduce its volume and then stored in double shell tanks for an interim period. The liquid in these tanks will be vitrified in a plant now undergoing commissioning.

4. WASTE MANAGEMENT

The overall waste management activities relating to spent fuel storage and reprocessing can be summed up as follows:

1. Liquid effluents are treated with the best available technology to reduce discharges to well below government limits, with a long term goal of keeping technology ahead of expected discharge limit reductions.
2. Low level radioactive solid wastes are disposed at the Drigg burial site, owned and operated by BNFL. Engineering improvements will be made to Drigg to ensure its long term availability until a new NIREX developed burial site is available or until a geologic repository is opened. Additional treatment of wastes will be carried out by the generators to reduce the demands on Drigg space.
3. Intermediate level wastes will be encapsulated in a cement matrix. A standardized container for these wastes has been selected and the first of the encapsulation plants is nearing completion. This waste will be stored on site until an intermediate waste repository is opened. BNFL is seeking local planning authority approval to conduct research on the suitability of the Sellafield site geology to host such a repository.
4. High active wastes will be vitrified and stored in an above ground engineered facility for at least 50 years to gain the benefit of radioactive decay. Waste arising from the reprocessing of off shore fuel will be returned to the customer. The vitrification plant is undergoing cold check-out and commissioning and active operation is expected in 1990.

R.J. Sills is currently head of marketing and sales for BNFL Engineering. He received his B.S. in chemistry and his Ph.D from Birmingham University. He joined BNFL in 1970 as a researcher at the Capenhurst gaseous diffusion plant. He also worked in a supervisory capacity in R&D and operations for the gas centrifuge process. In 1975, he moved to the Sellafield reprocessing site and worked on the operating plant and THORP R&D.

Status of Spent Fuel Storage Expansion at Prairie Island

■
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■

ABSTRACT

The Prairie Island Nuclear Generating Plant contains two 560 MWe, two-loop Westinghouse designed PWRs. These two units have been in commercial operation since 1973 and 1974. The current spent fuel storage capacity will be exhausted in the 1994-1995 time frame. In 1987 Prairie Island performed a demonstration of rod consolidation technology in the spent fuel pool. The demonstration included waste classification and empty cage volume reduction. The results of this demonstration and Prairie Islands future plans for spent fuel storage expansion are discussed.

The Prairie Island Nuclear Generating Plant is a two unit station owned and operated by Northern States Power Company, located on the Mississippi River approximately 30 miles southeast of Minneapolis/St. Paul. The site contains two Westinghouse designed two-loop PWR's, each rated at 560 MWe. Unit 1 began commercial operation in December 1973, and Unit 2 followed beginning in December 1974. Since beginning operations, Prairie Island has achieved an excellent operating record with a station capacity factor of 80%. Among the more notable accomplishments during this time frame have been a 19 day refueling outage, 5 runs of over 300 consecutive days online including one of 407 days, and three calendar years on Unit 2 without a reactor trip. This record has been achieved while maintaining one of the lowest electric production costs of any steam-electric plants in the nation.

PRAIRIE ISLAND FUEL MANAGEMENT

Each Prairie Island unit contains 121 fuel assemblies. The fuel design includes a 12 foot active core, with a 14 × 14 fuel rod array. Each assembly contains approximately 360 Kg U. Prairie Island has used several different fuel designs in the past including Westinghouse Standard, Exxon Standard, Exxon TOPROD, and Westinghouse Improved Optimized Fuel Assemblies (OFA). Prairie Island has used Gadolinia as a integral fuel absorber since 1979. All use of BPRA, thimble plug and source assembly inserts has been discontinued and only control rod assemblies remain to be moved during refuelings. The current fuel management strategy includes 48 fresh assemblies per reload at 3.8 w/o enrichment, with no axial blankets and Gadolinia as the burnable absorber. This strategy yields cycle lengths of approximately 16,500 MWD/MTU (17 months), with batch average discharge exposures of 42,000 MWD/MTU maximum assembly exposures of 50 MWD/MTU.

PRAIRIE ISLAND SPENT FUEL STORAGE

The two Prairie Island units are served by a common spent fuel pool storage area. This area includes a new fuel pit (dry), a small pool, a large pool, and a transfer canal. When the plant was initially designed it was expected to ship spent fuel off for reprocessing after a 2-4 month cooling time. As such, the small pool was intended to hold a shipping cask, while the large pool contained racks with a total capacity of 210 fuel assemblies. Since the initial plant construction, the pools have be re-racked on two different occasions. In the mid-1970's, new racks were installed that provided 132 storage spaces in the small pool for a fuel core off-load capability, and 555 storage spaces in the large pool. It was felt at this time that this would provide sufficient space to store fuel until reprocessing became operational. When President Carter deferred reprocessing in 1977, Prairie Island initiated a new modification that resulted in new high density racks being installed in the pools that have now maximized the capacity of the pools with regard to storage of intact spent fuel assemblies. These racks use Boraflex as a neutron absorber and provide space for 1386 permanent spent fuel storage spaces, plus space for a full core off-load (121 assemblies). These racks were installed in 1981.

PLANS FOR ADDITIONAL STORAGE

Currently, there are 937 spent fuel assemblies stored in the Prairie Island spent fuel pool. The existing storage space will be completely full in the 1994-1995 time frame. Because of this situation, NSP has maintained an active program evaluating the various methods for increasing onsite fuel storage as they develop. In 1986, NSP performed an evaluation of the various technologies that were either available or envisioned to be available in the near future and determined that rod consolidation was potentially the lowest cost method available, but had yet to be demonstrated as a viable production technology. NSP decided that it would perform a rod consolidation demonstration at Prairie Island, and use the knowledge learned from the demonstration as a basis for selecting a method for increasing the Prairie Island storage capacity.

In early 1987, NSP contracted with Westinghouse Electric to perform a consolidation demonstration of up to 50 fuel assemblies. In late September '87 the equipment and crews arrived on-site, and between mid October and mid November a total of 36 fuel assemblies were consolidated into 18 canisters.

The Westinghouse scope did not include compaction of the non-fuel bearing components. This work was performed by WasteChem Corp. in the spring of 1988. Using the WasteChem Underwater Shear Compactor (USC), 9 of the empty cages were compacted and packaged into storage liners. As part of this compaction work, direct sample measurements were taken of the fuel cage components for waste burial classification according to 10CFR61.

EVALUATION OF CONSOLIDATION

The consolidation demonstration provided NSP with a firm grasp of both the strengths and weaknesses of the process. The fact that 36 assemblies were consolidated proved invaluable in assessing the performance of both the equipment involved and of the personnel required for large scale production. As with any demonstration of a relatively new technology, not all of the information obtained was as favorable as expected. A summary of the major results of the demonstration are summarized as follows:

The maximum production rate achieved was 2.3 assemblies in a 20 hour day, with 2 assemblies per day common. While this is less than the desired rate of 4 assemblies per day, it is felt that this level is achievable via the use of 2 elevator stations instead of one, along with several other simple time saving modifications.

The personnel exposure received was as expected before the demonstration started. The major source of exposure was attributed to the normal background radiation levels in the spent fuel pool, with no major increases due to the consolidated activities. Because of this, the major avenue for keeping exposure as low as possible is by maintaining a production rate that is as fast as possible.

The amount of debris generated during the process was one of the most disappointing aspects of the demonstration. Debris was generated during two aspects of the operation, removal of the end fittings, and removal of the rods from the assembly. The equipment used during the demonstration was not prepared for the magnitude of the debris generated during the process. The main concern that arose was the potential for this debris to migrate from the water to the work area and cause a personnel overexposure. Future equipment will require much more extensive debris collection and disposal systems in order to prevent this from occurring. An additional concern with the debris generated is it's potential for damaging the equipment. During the demonstration one fuel rod became stuck while being pushed out of the assembly and was damaged to the point that while it did not break, it was damaged to the point that it couldn't be consolidated. This rod was placed in a separate storage can and the rest of the assembly was successfully consolidated.

A contributing factor to the damage of this rod was the fact that the rods were pushed from the assembly using a manual operated air driven push tool. This method was chosen by Westinghouse due to its simple design and ease of repair. The compromise made with this type of simple design is that it relies heavily on the skill of the operator for both the speed and safety of the operation. If future designs are to use this method of rod extraction, modifications must be made to reduce it's dependence on a skilled operator, especially in a large campaign of 200-250 fuel assemblies.

The results of the waste classification performed showed that all the Zircaloy and stainless steel components were burnable as Class C waste or below, but that grid strap material made of Inconel was beyond Class C waste, mainly due to the concentration of Nb-94. From crushing the 9 cages using the WasteChem equipment, it was determined that the Inconel grid pieces could not be segregated from the zirc guide tubes. Because of this, the grid and guide tube material was packaged into storage liners designed to be kept in the fuel racks. The results of this packaging yielded a compaction ration of 5:1, including the end fittings. This is a lower compaction ratio than was hoped for, and while some improvements are possible, it leads to the conclusion that more assemblies must be consolidated in order to gain a specific number of storage spaces than was originally thought.

FUTURE PLANS AT PRAIRIE ISLAND

Following the completion of the consolidation demonstration, NSP, performed an evaluation of the various options available for increasing Prairie Island's spent fuel storage capacity. Besides gathering together all of the cost and engineering data selected to the consolidation demonstration, all of the current dry storage technologies were studied as well. Information concerning the dry technologies was obtained by discussions with vendors and also by visits to existing dry storage installations. The results of this evaluation were documented in a formal report authored jointly by the General Office Special Nuclear Programs Department and the Prairie Island Nuclear Engineering Department. This evaluation report recommended a strategy which was then presented to and endorsed by NSP management. The report contained the following conclusions and recommendations:

1. Rod Consolidation is a viable, relatively low cost method of fuel storage expansion.
2. Consolidation has large negative impacts on the operation of the plant, mainly in the area of scheduling of work load, the potential for plant contamination and worker overexposure, and the potential for damaging of fuel rods.
3. Due to the negative impacts rod consolidation would have on plant operations, NSP has decided

to pursue dry spent fuel storage as a primary means of increasing Prairie Island's spent fuel storage capacity.

4. NSP wants to have the option of consolidating fuel if for some reason it becomes necessary at a future date, and therefore plans to pursue a license amendment with the NRC to allow this.

NSP is currently in the process of soliciting bids from dry storage vendors for a system at Prairie Island. The current schedule calls for selection of a vendor by June 1989, with license applications submitted by December 1989. Besides an application to the NRC for a license under 10CFR72, Minnesota state law requires that the state Public Utilities Commission grant a Certificate of Need for expansion of an existing spent fuel storage installation by more than 20%. The application to the state for this permit will be submitted in the same time frame as the NRC application. The hearing process for the state permit is expected to be at least as lengthy as the NRC process.

Jon Kapitz completed his B.S. and M.S. degrees in nuclear engineering in 1980 and 1981 at the University of Wisconsin-Madison. He is presently working in the Nuclear Engineering Department at the Prairie Island plant with responsibility for site activities related to increasing spent fuel storage capacity. From 1981 to 1987 he worked in the Nuclear Analysis Department of Northern State Power during which time he worked in the safety analysis area performing thermal-hydraulic and probabilistic safety analyses for the Prairie Island and Monticello plants. Kapitz is a registered Professional Engineer in the state of Minnesota.

Laura McCarten received a B.S. in nuclear engineering from the University of Wisconsin-Madison and has worked for the Northern States Power Company since 1979. From 1979 to 1984, McCarten performed reactor analysis and fuel-cycle management. In 1984, she became responsible for strategic planning to meet the interim spent fuel storage needs of NSP's Monticello and Prairie Island nuclear plants. She was the project manager for the Monticello Spent Fuel Shipping Project and the Prairie Island Spent Fuel Consolidation Demonstration Project.

Status of Consideration of Consolidation at Calvert Cliffs

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ABSTRACT

The Baltimore Gas and Electric Company, operator of the Calvert Cliffs Nuclear Power Plant, has studied the status of the rod consolidation and dry spent fuel storage techniques as they currently exist. Issues to be considered are discussed and tentative conclusions as to the relative advantages are presented.

When Calvert Cliffs began operations in the mid 1970's, long term storage of spent fuel was not a part of the plan. The pool's racks were capable of storing 400 fuel assemblies, and it was envisioned that, after a short cooling period, the fuel would be shipped offsite for reprocessing. Government policy regarding commercial reprocessing helped to change that vision, and during the late 1970's and early 1980's the pool was reracked several times to allow for storage of more fuel. Our most recent rerack effort, completed in 1983, provided us with 1830 spaces, allowing maintenance of full core reserve until our spring 1992 refueling.

The Nuclear Waste Policy Act provided us with a 1998 repository date—a target which we could use in planning our spent fuel storage strategy. We began to assess alternative methods of expanding our storage capacity from 1992 out to the 1998-2000 time frame. The potential feasibility of fuel consolidation—reconfiguration of the fuel rod array to allow more efficient use of available storage space—was of high interest because it could expand our storage capacity past 1998, perhaps as far as 2002, and because it could allow that capacity expansion to remain in-pool. This minimized capital cost and public visibility.

Because of our interest in consolidation technology, BG&E became involved in the EPRI demonstration program along with Northeast Utilities, Combustion Engineering and later, DOE. The program schedule called for a hot demonstration well before BG&E would need to begin capacity expansion activity. Because of the desired end product, the system became very sophisticated, and the hot demonstration did not occur until the fall of 1987. At the same time, DOE was officially moving the repository date back to 2003, and the Nuclear Waste Policy Amendments Act was being enacted.

Both the slippage in the repository date and the results of the several consolidation demonstrations that were occurring made BG&E take a very critical look at the role consolidation might play in meeting our storage needs. With respect to the DOE slippage, a potential repository date of 2003 or beyond meant that consolidation alone could not provide sufficent capacity to permit continued generation of spent fuel, but would have to be combined with another capacity expansion method. It could be performed up front to defer the need for out-of-pool storage, or it could be performed in tandem with out-of-pool storage. The benefit of performing consolidation up front was deferment of the capital cost associated with commissioning an out-of-pool storage facility. Consolidation vendors always played that point up, saying that it allowed utilities time to permit dry storage technologies to mature. But after seeing the results of the consolidation demonstrations that occurred, we felt that the reverse was true—consolidation technology was what needed to mature.

We reconsidered our commitment to consolidation, and decided to look at all storage options. Focusing purely on technical issues, four main considerations weighed heavily in our evaluation of all options:

- Licensability (issue-free, on-schedule)
- Operating Experience (position on learning curve)
- Impact on Plant (fuel handling, modifications)
- Compatibility with final disposal (package format, accessibility)

With respect to consolidation, we had the following thoughts regarding these considerations.

Licensability

There are two licensing considerations associated with fuel consolidation. One involves the increase in storage capacity, and the other involves the actual consolidation process itself. Technically, the increase in storage capacity requires treatment of the same issues as would be required in a rerack submittal—criticality, seismic, T-H, etc., although some areas, such as SNM accountability, may be more complicated. The amendment to Millstone's license, allowing for storage of consolidated fuel, demonstrated that this aspect of consolidation licensing has also been proven and relatively straightforward. The approach taken by NUSCO with respect to the process itself was to perform an internal 50.59 evaluation, determining that no unreviewed safety question existed. NRC concurrence with that determination demonstrated that this aspect of consolidation licensing has also been proven. This determination, however, is not only site specific, but system specific as well. With several different systems available, some substantially different, some may raise more difficult questions than others during the process evaluation.

Overall, from a licensing perspective, consolidation looked acceptable to us.

Operating Experience

Consolidation systems "available" in the near term were in many cases first generation, or in some cases, not even that far along. Questions raised at hot demonstrations had yet to be answered, and vendors with systems which were still untested had yet to find out what questions would be raised. Although hot demonstrations had shown that 2:1 consolidation is achievable, there was little if any data in such areas as waste generation and grid cage volume reduction—areas that were important to know about before making commitments. Also, we felt that the equipment which had been demonstrated was not adequately production oriented (discussed under Impact on Plant), and that overall, the technology was not mature.

From this perspective, consolidation as an immediate activity did not look acceptable to us.

Impact on Plant

There were two major considerations under this heading. One involved physical modifications required, meaning both new construction and modifications to existing structures. The other involved the impact on fuel handling operations, including ALARA concerns and labor requirements. Fuel consolidation clearly involved less physical modification than did dry storage options, with the added benefit that the physical work/modification could be done out of the public eye. It was with respect to fuel handling impact that consolidation looked weakest. Proven systems yielded production rates of about one assembly per shift. Since 2-2.5 assemblies must be consolidated for every storage space gained (depending on how scrap is treated), that type of production rate means the investment of a substantial amount of time in consolidation activity. For example, at Calvert Cliffs, where we discharge 96 fuel assemblies per year, 192-240 shifts of consolidation would be required just to keep up with annual discharge (compared with 30 to 40 shifts required to achieve the same end with dry storage systems). Also, since a utility typically does not begin *any* capacity expansion activity until almost all available space is used, there is usually a desire to not only keep up with annual discharge, but to regain additional space to provide access to divers without the need for substantial fuel movement. So with current (proven) production rates, even to gain just 100 spaces a year could require 400-500 shifts of consolidation activity (compared to 60-80 shifts for dry storage)—*and* all of that work is performed at poolside (compared with 18-24 shifts of poolside work associated with dry storage). This scenario was unacceptable to BG&E for several reasons.

1. Logistically, it would require full time and often around the clock work during non-refueling times, prohibiting other activities from taking place in the pool.
2. If BG&E personnel performed the work, additional labor forces would be required. Given the repetitive, tedious nature of the work, it would be expected that heavy attrition would occur, requiring retraining of new personnel without affecting work schedule. If this burden were to be passed on to a vendor by contracting out the consolidation work, the cost of the work would increase significantly.
3. Given the drive to reduce worker exposure and minimize personnel contamination, we felt that best achieved by avoiding the need to perform 15 to 30 times as much poolside work and 5 to 10 times as much work in a radiologically controlled area.

From this perspective, we felt that the state of proven consolidation technology was unacceptable for near term implementation.

In must be noted that second (and later) generation systems are being developed with the goal of addressing these concerns. Higher production rates are being sought, and the move by some toward the use of automated manipulators to perform consolidation tasks could alleviate many plant concerns.

Compatibility with Final Disposal

Our concern here, as it related to consolidation, was the possible effect of consolidation activities on the scheduled acceptance at the repository. The lack of clear policy on the acceptability of consolidated canisters and scrap containers with no schedule penalty could only have a negative effect as a decision was made with regard to near term consolidation. There is no justification for this to remain as an issue, but it does.

To summarize our thoughts as we made our decision earlier this year, while we believed consolidation to be a licensable and potentially economical expansion method, we saw that the complete cycle of consolidation related activities had not been sufficiently demonstrated, that the production rates achieved by proven systems had an unacceptable impact on our plant operations, and that the uncertainty associated with DOE acceptance of consolidated fuel and scrap made the process appear less appealing to us as a near term part of our storage strategy. We chose to pursue NUTECH's NUHOMS 24P system to meet our needs, and plan to submit a license application to the NRC in midyear for life of plant storage with fuel loading targeted to begin in mid 1991. We continue to maintain our interest in the development of consolidation, and should sufficient strides be made in the technology, we will evaluate its potential use in conjunction with dry storage.

Penney File received a B.S. in mathematics from the University of the South at Sewanee and a B.S. in Nuclear Engineering from Georgia Tech. Currently, he is a senior engineer in the Nuclear Engineering Unit at the Baltimore Gas & Electric Calvert Cliffs Nuclear Power Plant where his areas of responsibility include spent fuel storage and disposal, reactor refueling and fuel performance. He is a member of ANS, EPRI Core Materials Subcommittee, ANS 57.10 and ANS 5.3 working groups.

Metal Cask Storage at Virginia Power

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ABSTRACT

By the end of 1990, Virginia Power will have loaded five different dry storage cask designs for the Surry Power Station Independent Spent Fuel Storage Installation. This should provide a valuable base of information when making future cask purchase decisions.

In 1989, Virginia Power will be working with the Electric Power Research Institute (EPRI), the Sandia and Oak Ridge National Laboratories and other nuclear utilities to validate computer codes used to predict storage cask reactivity. This effort will involve reactivity measurements of Surry spent fuel storage racks and a loaded CASTOR V/21 storage cask.

Increased burnup of discharged fuel means that cask designs for burnups up to 35,000 MWD/MTU will be inadequate. New designs for fuel burnups up to 45,000 MWD/MTU will be needed in 2 to 3 years.

INTRODUCTION

As part of a Cooperative Agreement Program with the U.S. DOE and EPRI, Virginia Power has been conducting a metal cask storage demonstration program. The status of this program is as follows.

1. Testing of three storage casks at the Idaho National Engineering Laboratory with intact and consolidated fuel has been completed. These three casks are the GNSI CASTOR V/21, Transnuclear TN-24P, and Westinghouse MC-10.
2. Eight CASTOR V/21 casks have been loaded at the Surry Power Station. An NAC-28 cask has been delivered to Surry, but won't be loaded until late 1989.
3. Three casks have been ordered for delivery in late 1989 and early 1990. These casks are the MC-10, GNSI CASTOR X/33 and NAC-31.

CASTOR V/21 Casks

Last summer we became concerned that enough storage capacity might not be available for the Surry outages then scheduled in the fall of 1989 and spring of 1990. As a result, we ordered four CASTOR V/21 casks for delivery in March through June, 1989. These four casks will be loaded right after their delivery.

NAC-28 Cask

This cask, designed by the Nuclear Assurance Corporation (NAC), was delivered to Surry in February 1988. The Topical Safety Analysis Report (TSAR) for storage of consolidated fuel in this cask was approved by the U.S. NRC on September 29, 1988. We plan to store intact fuel in this cask, so a TSAR amendment for this purpose was submitted to the U.S. NRC on December 14, 1988. Current plans call for loading this cask in late 1989.

MC-10 Cask

This cask, designed by Westinghouse for 24 fuel assemblies, is scheduled for delivery to Surry in September 1989. The TSAR for this cask was approved by the U.S. NRC on September 30, 1987. Current plans call for loading this cask in late 1989.

CASTOR X/33 Cask

This cask, designed by GNSI for 33 fuel assemblies in a burnup credit design basket, is scheduled for delivery to Surry in January 1990. The TSAR for this cask was submitted to the U.S. NRC on July 1, 1988. Loading this cask is currently scheduled for the last half of 1990.

NAC-31 Cask

This cask, designed by NAC for storage of 31 fuel assemblies in a burnup credit design basket, is scheduled for delivery to Surry in February 1990. The TSAR for this cask was submitted on November 20, 1988. Loading is currently scheduled for the last half of 1990.

Table 1 provides a summary of the casks loaded to date and those under contract for delivery in 1989 and 1990. Figure 1 shows the effect on full core reserve of cask loadings and refuelings of Surry Units 1 and 2 from 1988 through 1991.

Burnup Credit

TSAR submittal for two storage casks (CASTOR X/33 and NAC-31) employing burnup credit for criticality con-

Table 1
Surry Power Station Spent Fuel Dry Storage
Present and Future

Cask	*Location*	*Capacity*	*Date Loaded*
V/21.1	INEL	21	08/85
TN-24P	INEL	24	12/85
MC-10.1	INEL	24	06/86
V/21.5	SPS	21	10/86
V/21.3	SPS	21	11/86
V/21.4	SPS	21	06/87
V/21.2	SPS	21	07/87
V/21.6	SPS	21	12/87
V/21.7	SPS	21	03/88
V/21.8	SPS	21	08/88
V/21.9	SPS	21	09/88
		237	
V/21.10	SPS	21	04/89
V/21.11	SPS	21	05/89
V/21.12	SPS	21	06/89
V/21.13	SPS	21	08/89
NAC-28	SPS	28	10/89
MC-10.2	SPS	24	12/89
X/33.1	SPS	33	10/90
NAC-31	SPS	31	12/90
		200	

trol provides clear evidence that the development stage of this technology is ending and the licensing and demonstration phases are starting. At Surry, we have submitted two generic ISFSI Technical Specification changes that provide for certain controls during loading of such casks. These controls include loading casks with fuel from a certain area of the spent fuel pool already using burnup credit for criticality control. In addition, water used during loading and unloading of casks will contain at least 2,000 ppm of soluble boron.

Once the TSARs for the two casks are approved, we will submit cask specific criticality analyses, as ISFSI SAR amendments, showing the minimum burnups needed for each fuel enrichment.

We are working to complete a report for EPRI on our investigations of burnup credit, and this report should be available in several months.

Criticality Measurement

As a supplement to our previous burnup credit investigations, Virginia Power is working with EPRI, Sandia National Laboratory, Oak Ridge National Laboratory, Duke Power and Yankee Atomic on a "proof of principle" demonstration of burnup credit. This demonstration will involve validation of computer analyses with measurements of Surry spent fuel racks and a loaded CASTOR V/21 cask. The CASTOR V/21 test is scheduled for August 1989 using Cf-252 sources and fission chamber detectors.

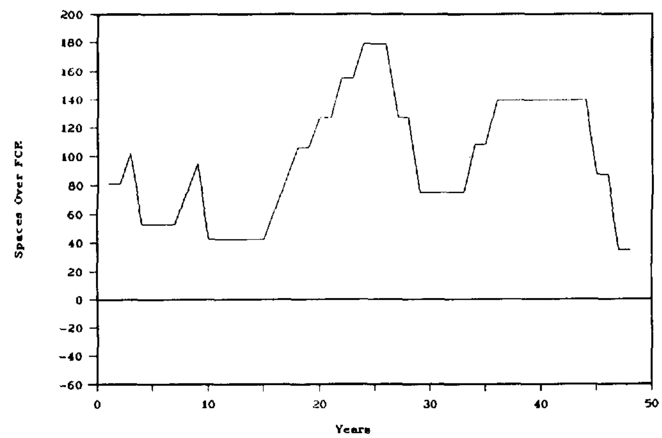

Figure 1. Storage Over Full Core Reserve

ISFSI License Changes

In the near term, planned Surry ISFSI license changes will include the MC-10 cask as an approved storage cask and will convert the SAR/ER to "generic" license documents. This change involves moving specific cask references from the body of the document to appendices, thus eliminating the need to revise the entire document each time a new cask design is approved. As previously mentioned, Technical Specification changes have been proposed for burnup credit cask designs.

For the longer term, additional cask designs (NAC-28, NAC-31 and CASTOR X/33) will be added as each cask TSAR is approved.

Design and Installation Report

In November 1988, Virginia Power submitted its Surry ISFSI design and installation report to EPRI for publishing. Designated as report NP-6032, copies should be available in the next two months.

Increased Burnup

Table 2 provides a summary of current cask designs and their enrichment and burnup capabilities. Generally, current cask enrichment and burnup limits are 3.5 to 3.7 w/o U235 and 35,000 MWD/MTU, respectively.

Table 2
Metal Storage Cask Enrichment/Burnup Capabilities

Cask	*NRC Status*	*Maximum Enrichment (w/o U235)*	*Maximum Burnup (MWD/MTU)*	*Minimum Cooling (Years)*
CASTOR V/21	Approved	3.5	35,000	5
CASTOR X/28	Under Review	3.5	35,000	10
CASTOR X/33	Under Review	3.5	35,000	10
MC-10	Approved	3.7	35,000	10
NAC-26	Approved	3.3	35,000	5
NAC-28C	Approved	3.5	35,000	10
NAC-28U	Under Review	3.7	45,000	10
NAC-33	Under Review	3.7	35,000	10
TN-24P	Under Review	3.7	35,000	5

To date, 984 fuel assemblies have been discharged from Surry Units 1 and 2. A breakdown on the current disposition of these fuel assemblies is shown below.

Discharged	984
In casks at INEL	-69
In casks at Surry	-168
To go in casks in '89 and '90	-200
	547

Figure 2 shows a graph of the burnup distribution of the 984 discharged fuel assemblies. Additional information on fuel assemblies with burnup less than 35,000 MWD/MTU is shown below.

	Total	<35,000
Discharged	984	794
In casks at INEL	- 69	- 54
In casks at Surry	-168	-168
To go in casks in 89 and 90	-200	-200
	547	372

As shown above and in Figure 2, about 50% of the discharged fuel with burnup less than 35,000 MWD/MTU has been committed to dry storage. Of the 372 remaining lower burnup assemblies, 88 would need detailed examination and evaluation prior to use in dry storage, leaving 284 fuel assemblies to meet future dry storage needs. This inventory is not being added to as all current discharged fuel exceeds 35,000 MWD/MTU and two thirds exceeds 40,000 MWD/MTU. At present storage rates, the 284 remaining assemblies will be exhausted in 1993.

This analysis shows that during the next two to three years, cask designs for higher burnup fuel need to be developed and licensed. We recognize that higher burnup means a higher source term, which means either more shielding is needed (with a decrease in cask capacities) or cooling times need to be longer.

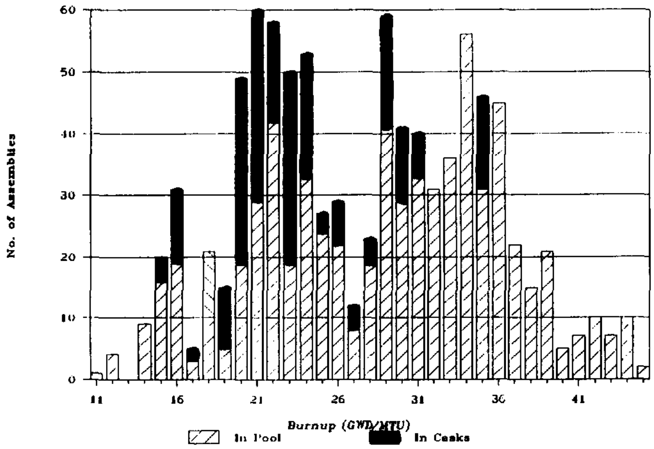

Figure 2. Spent Fuel Burnup Distribution

Conclusion

In the next two years, we will gain valuable experience loading five different cask designs. Validation of burnup credit analyses should be realized from the "proof-of-principle" test on a CASTOR V/21 in mid-1989. Over the next several years, we expect cask vendors to be working on transport/storage cask designs with burnup credit baskets and burnup limits of 45,000 MWD/MTU.

Brian H. Wakeman received his B.S. in nuclear engineering from Pennsylvania State University. He has been the senior engineer at the Virginia Power/Nuclear Engineering for shipment of spent fuel from Surry to the Idaho National Engineering Laboratory since 1985. During this time, he conducted time/motion/dose analyses for loading spent fuel casks under contracts to Battelle-PNL. Prior to his current work, Wakeman was an engineer at Virginia Power/Nuclear Fuel Procurement where he developed and administrated lease contracts for spent fuel shipping casks.

Disposition of Skeleton Hardware from Spent Fuel Consolidation

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ABSTRACT

Rod consolidation has the potential for significantly increasing the on-site storage capacity of reactor's sent fuel storage pools. The cost, operational practicality, and overall consolidation efficiency, however, are likely to be highly dependent on the manner in which the scrap non fuel bearing components (NFBC) are treated, packaged and stored. Materials contained in typical fuel skeletons were surveyed and analyzed to determine the critical parameters (radioisotopes, mass, activation levels, decay characteristics) effecting storage, handling and disposal alternatives. Alternative processing, packaging and at-plant storage schemes (wet and dry) are being studied and compared. A full report on this study by Rochester Gas and Electric, sponsored by EPRI and ESEERCO, will be issued by EPRI later this year.

INTRODUCTION

Background

Rod consolidation has the potential for significantly increasing the on-site storage capacity of reactor's spent fuel storage pools. The cost, operational practicality, and overall consolidation efficiency, however, are likely to be highly dependent on the manner in which the scrap non-fuel bearing components (NFBC) are treated, packaged and stored. To date, the range of alternative technologies that may be applied to these tasks has not been well investigated and the challenges and opportunities involved have not been well quantified.

It does seem clear, however, that utilities are highly likely to retain these NFBC on-site until they are removed by DOE as part of DOE's commitments under the terms of the Nuclear Waste Policy Act (NWPA). DOE has indicated that they plan to accept this responsibility at no added cost to the utility. On the other hand, any removal or disposal actions done prior to DOE removal would likely be at utility expense unless other arrangements were negotiated with DOE.

Given this background, it is the belief of many utilities that the decision of whether or not to pursue rod consolidation may largely depend on the degree to which technology has been developed and demonstrated to efficiently treat, package and store the NFBC.

Sponsors

As part of Rochester Gas and Electric's Consolidation program at West Valley in late 1985 and early 1986 (sponsored primarily by ESEERCO, a research organization supported by the electric utilities of New York state and supported by DOE and EPRI) the technology considered to be the conventional industry approach to hardware disposition at that time was utilized i.e. crushing/shearing into fuel-cell-sized canisters at 10:1 compaction. Although considerable information was gained, the attempt was disappointingly unsuccessful. As part of a following consolidation demonstration at Battelle Columbus, a different approach was investigated. Hardware components were simply segregated for packaging, and samples were subjected to radiochemical analysis to characterize the activated hardware for waste disposal and/or shielded storage. The results indicated that much of the activated hardware qualified for low level waste disposal, and that some of the analytical work that had been done previously may have over-predicted the activity levels of fuel hardware.

The results of this program, and the unsatisfactory results of hardware compaction in other consolidation demonstrations, sparked new interest in the investigation of hardware disposition alternatives whose economics are less dependent on compaction factor achieved. Both EPRI and ESEERCO expressed interest in funding R&D. In September, 1987 EPRI issued a request for proposal for a study on the "Treatment, Packaging and Storage of Bundle Scrap Hardware" resulting from consolidation operations. Rochester Gas and Electric was awarded the contract based on its proposed scope of work to be conducted by itself and its main subcontractors Chem-Nuclear Systems and WMG Inc. and the co-funding of 45% of the cost through RG&E's contract with ESEERCO.

SPECIFIC TASKS AND DISCUSSION OF PRELIMINARY RESULTS

Task 1: Identification of Critical Isotopes

The volume, mass and material composition of the NFBCs have been identified for several of the most currently used BWR and PWR fuels. This information is being used as input in the studies on processing and storage options. Optimum storage/disposal options largely depend on the type of material and radioisotope concentrations, as well as the anticipated dose rates of the components.

The results of characterization of activated materials using typical operating reactor parameters indicate that different isotopes are the controlling materials for each different class of options.

Cobalt-60 is a strong gamma emitter and is the controlling isotope for shielding. Therefore it is the isotope controlling the design of dry storage containers. The highest gamma emitters per unit of weight are stainless steel guide tubes residing in the active fuel region of cores where they are subjected to high neutron flux levels. The other strong gamma emitters are stainless end fittings with the bottom end fitting as much as an order of magnitude greater than the top end fitting. These are out of active core region with the top fitting further away than the bottom. Stainless guide tubes were used in early Westinghouse and B&W assemblies and represent about one third of the total stainless in these skeletons. These early assemblies have the longest decay times and are the assemblies on the low end of the fuel burnup spectrum. Both of these conditions help to lower activity levels. But despite this they tend to be the worst case condition for hardware dry storage. However, virtually all fuel assemblies have stainless end fittings. The newer high burnup assemblies will have the highest activation levels for stainless end fittings as well as the shortest decay times, and could result in worst case conditions.

On the other hand, stainless steel is well within limits for low level burial, except that it must be transported in heavily shielded casks with relatively low payloads; increasing transport and handling costs. It also has relatively high levels of Iron-55, which does not present a shielding or handling problem because it is a low energy Gamma emitter, but it contributes a very large percentage of the total curie content of the waste; and disposal fees are based on curie levels. Iron-55 has a relatively short half life of 2.7 years, so with time curie levels reduce significantly. Spent fuel has generally undergone several half lives by the time it is consolidated. Interim storage following consolidation may also be utilized to reduce curie levels.

Niobium-94 is the controlling isotope for low level waste disposal. Although very little niobium exists in metals, the Class C limits for burial are so low (0.2 ci/m^3) that very little is needed to exceed the limit. High Nickel alloys such as Inconel-718 is the villain when it comes to Niobium (generally$<$0.1% of total activity). Therefore Inconel grids are clearly out of limits for burial, and even grids containing small Inconel springs are questionable.

Inconel is not a particular problem when it comes to dry storage. It is primarily a Beta emitter with some low energy gamma which does not present a difficult shielding problem.

The least bothersome material from all aspects is Zirconium. It does not present a particularly difficult situation for either disposal or storage.

The storage of hardware in the spent fuel pool is the easiest and most economical from a packaging and shielding viewpoint. However, the availability and value of space is the determining factor here. A spent fuel pool is much more valuable as a fuel storage facility than it is for hardware storage.

The above discussion clearly illustrates that there is no single answer for all NFBC. There are so many interactions that each case must be examined individually. The attempt in this study will be to place generic boxes around typical situations and to provide some insight and tools for utilities to study and plan their individual situations.

B. Task 2: At Reactor Storage Alternatives

At reactor storage alternatives have been studied with respect to various concepts of processing, packaging and storage. The basic concepts of on-site storage fall into two categories; wet and dry. Trade-offs involving the value of space occupied, processing and packaging costs, and compatibility with transportation casks for future shipment to the DOE Waste Repository, have been and are being studied. In each case, various degrees of compaction and/or segregation are being studied. Results will be presented in a format usable by individual utilities for customization to their particular needs.

The studies to date have evolved into two basic configurations for packaging; the 55-gallon drum and the square fuel geometry. These two basic configurations are a result of the standards and facilities developed through common use to date and therefore appear to be the most sensible to continue into the future. The packages to be recommended for use in this study will be multiples of these two basic configurations. In general liners and cask cavities will be designed to accommodate the outside dimensions of these packages.

1. In-Pool Storage—Several in-pool storage concepts have been studied, but attention is centering around the following:

- NFBCs compacted to 10:1 or better stored separately within the fuel racks in canisters similar to the consolidated fuel canister in both full height and half height configurations.
- Compacted NFBCs stored in a short canister approximately 30-40 in. tall mounted on top of the consolidated fuel canister totaling a combined height of less than 207 inches. The NFBC for two assemblies can fit in such a package with a 7 or 8:1 compaction ratio. DOE must be able to accommodate a 207" package to accept the long fuel assembly utilized in some Westinghouse designs. An EPRI study[1] on Waste Acceptance Criteria indicated that such a package may have negligible impact on the Waste System.
- On-rack storage; i.e storage of NFBC in containers placed on a platform placed over the fuel racks containing consolidated fuel canisters. It is anticipated that these containers will approximate the size of two 55-gallon drums in height (24"OD × 72" high). Hardware compacted to 5:1 into such containers will occupy a platform area slightly less than the cross-sectional area of the rack space occupied by the consolidated fuel.

2. Dry Storage—On-site storage in shielded metal and/or concrete casks and shielded structures is being studied.

Standard package concepts have been investigated considering currently available licensed packages and possible enhancements for storage containers. Design criteria for these packages are cavity dimensions in size multiples of 55-gallon drums. Storage containers can be designed with varying wall thickness depending on material to be shielded and shielding material used. In general, stainless components will require heavy shielding (on the order of 40" of concrete of 8" lead equivalent). Zirconium components and low burnup fuel that has decayed for some time may need as little as 25" of concrete or 4" of lead. Tradeoffs between concrete, lead, steel and combinations thereof are being studied for various sizes of containers and payloads.

Shielded containers may be mass produced off site and delivered as required or they may be fabricated on site. Mass produced containers fabricated off site are assumed to have a maximum dimension of 102" OD such that they can be delivered by truck. Additionally, storage in metal casks which are licensed for transport will be addressed.

Another type of storage container having greater storage capacity and consisting mainly of concrete is a storage module to be fabricated on site. By fabricating on site one is not limited by dimensions. We are avoiding utilization of a permanent structure because of uncertainty in licensing and cost. It is our opinion that a modular cask of standard design used at many sites is more predictable from a cost standpoint and adequate for the scope of this study.

C. Task 3: NFBC Processing

The fuel hardware skeleton remaining after all fuel has been removed during the process of fuel consolidation occupies a volume essentially equal to that of the intact fuel assembly. Therefore at this point one has a total volume of fuel and activated hardware which is 150 percent of that occupied by the intact fuel. The fuel consolidation concept to date has been to process this hardware using various means to cut, shear, chop and/or compress this skeleton to a smaller volume. The industry objective has been a 10:1 compaction factor, which results in one hardware canister for every 10 fuel assemblies. The fuel consolidation operation would then result in 5 canisters of 2:1 compacted fuel and 1 canister of hardware. Therefore for every 10 assemblies consolidated one gains 4 additional storage cells for a 40% increase in fuel storage space. The demonstration fuel consolidation programs to date have demonstrated that a 2:1 compaction in fuel can be achieved. None of the programs have demonstrated a 10:1 compaction in hardware. Experience to date can only demonstrate reliable compaction ratios for commercial fuel skeletons somewhere in the range of 5-6:1. Although it is expected that these results will improve with experience, a utility cannot justify fuel consolidation with hardware storage in fuel racks with compaction ratios that low. A gain of 3 fuel storage spaces for the consolidation of 10 assemblies is a questionable return for the effort.

This study is being done to identify the economics of storage alternatives which do not consume fuel storage space and which may not depend to such a high degree on compaction ratios achieved. In fact some attributes have been identified where high compaction contributes a negative impact on hardware disposition. In such cases the space savings gained through increased compaction may not justify the processing cost.

This study is being done to identify the economics of storage alternatives which do not consume fuel storage space and which may not depend to such a high degree on compaction ratios achieve. In fact some attributes have been identified where high compaction contributes a negative impact on hardware disposition. In such cases the space savings gained through increased compaction may not justify the processing cost.

This study examines the various storage alternatives and the many interacting attributes over a range of compaction ratios starting at 4:1 and going to 23:1. The 4:1 was selected as the lower range because this compaction ratio can be achieved simply by cutting up the skeletons with no compaction. The upper range of 23:1 was selected because it appears that it can be achieved with equipment that can be utilized in a utility pool. One of the approaches used in this study is segregation of classes of hardware where differences in characteristics can offer optimization of disposition alternatives.

Hardware processing can be carried out in parallel with fuel consolidation or in series depending upon the space available at the utility for skeleton storage and the size of the consolidation campaign. If hardware processing is carried out following fuel consolidation the cask lay-down area of the fuel pool can be used. It is assumed that if hardware processing is carried out in parallel with fuel consolidation that the cask lay-down area is occupied by fuel consolidation equipment. In such a situation it is more appropriate to process hardware on a platform constructed on top of empty fuel racks in the core off-load area.

D. Task 4: Disposal Options

The quantities and characteristics of the NFBCs identified in Task 1 are being used to determine disposal options. The options being investigated are:

- Immediate disposal
- Disposal after decay
- Disposal of selected components

Based on study results to date only stainless end fittings and Zirconium hardware will qualify for disposal. The economics of disposal versus storage have not been completed. Disposal has the advantage of moving material off site and eliminating the cost of storage facilities. Although DOE is by contract responsible for accepting NFBC, one cannot at this time be certain that all or a portion of the cost for disposal at a commercial site will be recoverable.

D. Task 4: Waste Characterization Program

A simplified interactive personal computer program developed by one of our subcontractors was used to obtain estimates of radioisotope concentrations for waste classification which is necessary to evaluate options available for NFBCs. The program is specifically designed for engineering use to assist companies in this endeavor, and has been used for all material classification in this study. The program results have been compared to the limited data available from radiochemical analysis of hardware samples of NFBC and reasonable agreement with this data has been achieved. The program disk will be made available as part of the final report for this study. It is intended that the data base utilized in the program will be updated as more experimental data becomes available.

DISCUSSION OF PRELIMINARY RESULTS

The study has not been completed and the interrelationship and costs of the many factors being evaluated have not been finalized. The final report is scheduled to be completed in the first quarter of this year and will be published as an EPRI report co-funded by ESEERCO.

Some preliminary economic comparisons of alternatives has been done, and these preliminary results indicate that alternatives are available for the disposition of activated hardware where compaction ratios as low as 4:1 are acceptable and can compare favorably with the value of fuel storage space consumed by hardware where compaction ratios as low as 4:1 are acceptable and can compare favorably with the value of fuel storage space consumed by hardware compacted at a 10:1 ratio and stored in fuel racks. These alternatives have the added advantage of freeing up all available fuel rack space for the storage of fuel such that a full doubling of fuel storage through consolidation can be achieved. If one of the alternatives which removes fuel from the pool is utilized, then one has the added advantage of decreasing floor load in the pool.

Confidence that disposition of fuel hardware at reasonable cost can be accomplished, enhances considerably the fuel consolidation alternative for increasing spent fuel storage at reactor sites.

[1]EPRI NP-6001, "Waste Acceptance Criteria Study", E.R. Johnson Associates, Inc., September, 1988.

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On the Use of Automatic Pipets in Volume Calibration of Accountability Tanks

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ABSTRACT

The applicability of automatic pipets for dispensing known volumes of water in the volume calibration of accountability tanks of tens-of-liters capacity is demonstrated. Six pipet of capacity in the approximate range 495 to 716 milliliters were calibrated using gravimetric techniques. The relative standard deviation ranged from 0.0015% to 0.0074%. The pipets were used successfully in the volume calibration of accountability tanks of tens-of-liters capacity in the Savannah River Plant.

INTRODUCTION

Volumetric test measures of nominal capacities 139 liters (50 gallons), 378 liters (100 gallons), and other capacities have been used in the calibration of accountability tanks (Jones, 1979; Jones, Schoonover, and Houser, 1980; Jones, 1984). The test measures dispense known volumes of water into the accountability tanks. For tanks of tens-of-liters capacity, automatic pipets of smaller capacity have been used. It is the purpose of this paper to illustrate the precision with which known volume of water can be dispensed into accountability tanks from automatic pipets. A typical automatic pipet is pictured in Figure 1.

CALIBRATION OF AUTOMATED PIPETS

Six automatic pipets with volumes in the approximate range 495 to 716 milliliters (mL) were calibrated using gravimetric techniques (Lembeck, 1974). One of the pipets, designated No. 31, was supplied by a commercial glassware supplier; the over five, designated Nos. 1, 2,12,22, and 32, were fabricated by Savannah River Plant personnel.

The gravimetric techniques involve two weighings of a weighing flask, one when empty and when it contains the quantity of water delivered by the pipet. The weighing flask is first weighed empty; the pipet is filled from the bottom through the stopcock until excess water passes through the automatic overflow tip, the temperature of the water is measured during the filling. After the pipet has overflowed, the stopcock is closed and rotated 90° to start the flow from the pipet to the weighing flask. After a 10-second drain time following the cessation of the main flow from the pipet, the stopcock is closed. The weighing flask, containing the water delivered by the pipet, is then reweighed. The mass difference between the two weighings is then divided by the density of water (Wagnebreth and Blanke, 1975) at its temperature in the flask to calculate the volume delivered by the pipet. This volume determination is then adjusted to 20°C using the temperature coefficient of expansion of the material from which the automatic pipet is constructed.

RESULTS AND CONCLUSIONS

Six or seven determinations of the volume, V_{20}, delivered by each of the six pipets at 20°C were made. The results are listed in Table 1, along with the mean volume for each

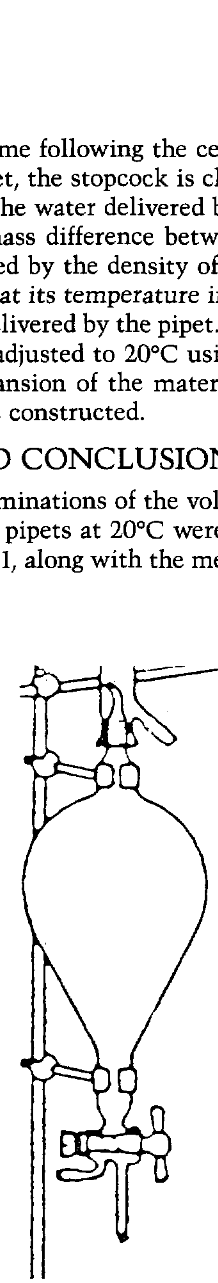

Figure 1. Sketch of pipet and inset of automatic tip.

Table 1
Determinations of Volume Delivered at 20°C, V_{20}, by Automatic Pipets

	V_{20}, mL					
	Pipet No.					
	31	1	2	12	22	32
	495.326	653.069	667.022	512.407	578.098	716.315
	495.378	653.008	666.971	512.337	578.081	716.337
	495.264	653.061	666.959	512.353	578.097	716.268
	495.339	653.039	667.016	512.336	578.087	716.321
	495.333	653.041	666.942	512.335	578.077	716.298
	495.328	653.060	667.016	512.317	578.083	716.318
					578.096	
$\overline{V_{20}}$	495.328	653.046	666.988	512.348	578.088	716.310
SD	0.037	0.022	0.035	0.031	0.0086	0.024
RSD	7.4×10^{-5}	3.4×10^{-5}	5.2×10^{-5}	6.1×10^{-5}	1.5×10^{-5}	3.3×10^{-5}

pipet, $\overline{V_{20}}$, the estimate of standard deviation, SD, and the relative standard deviation,

$$RSD = SD\sqrt{V_{20}}.$$

These results demonstrate that the automatic pipets are capable of being used to deliver volumes of water in the approximate range 497 to 716 mL with a relative precision, RSD, ranging from 1.5×10^{-5} to 7.4×10^{-5}, that is, from 0.0015% to 0.0074%.

The automatic pipet is very simple in operation and easy to use very precisely. The quantity delivered by the pipet is very reproducible. The only measurement required is the temperature of the water. These automatic pipets have been used successfully in the volume calibration of accountability tanks of tens-of-liters capacity in the Savannah River Plant.

ACKNOWLEDGMENT

The author is pleased to acknowledge the contributions of John F. Houser who made the measurements reported here, and the helpful discussions with him.

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Frank E. Jones is a physicist and independent consultant, having retired from the National Bureau of Standards (now the National Institute of Standards and Technology) in 1987. He has been actively engaged in tank volume calibration for more than 10 years. He designed, directed and participated in the first definitive tank calibration at the Savannah River Plant as well as many other tank calibrations. He also performed a definitive in-tank measurement of solution density. He served as deputy office chief in the NBS Nuclear Safeguards Program. He has authored more than 50 technical papers and holds two patents. Jones holds a master's degree in physics from the University of Maryland and has done doctoral work in meteorology at the same university. He was a consultant to the writing group for American National Standard ANSI N15.19, "Volume Calibration Techniques for Nuclear Material Control."

A Uranium Enrichment Monitor for Surveillance of a Small Centrifuge Cascade

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ABSTRACT

An automatic, remotely controlled enrichment monitor based on a quadrupole mass spectrometer is described. This system was used to measure the uranium isotopic ratios in the product and feed streams of a small gas centrifuge cascade. The performance of the system under design conditions is illustrated with typical results. Several difficulties encountered during the study are discussed in the light of subsequent operating experience.

1. INTRODUCTION

In 1984, the Australian Nuclear Science and Technology Organization (ANSTO)*, at the behest of the Australian Safeguards Office (ASO), undertook the development of a surveillance system for safeguarding nuclear materials. This work formed part of Australia's contribution to a bilateral program of assistance to the International Atomic Energy Agency (BAAP-2). A history of the project and an overview of the complete system have been presented by Evans *et al*[1].

One feature of the system was an on-line enrichment monitor. The design guidelines called for a unit capable of automatic, unattended operation for extended periods. The latter was defined as limited inspector access for a period of at least three months; access within this interval only occurs in the event of a failure. In addition, the monitor was to be amenable to remote control from a distant location.

Quadrupole mass spectrometry was considered to be the most suitable method for the determination of uranium isotope ratios in gaseous UF_6. Even though early reports on the use of this technique were largely negative [2,3], several groups [4-9] subsequently described instruments capable of measuring such ratios with acceptable accuracy. The most detailed of these studies is the work of Rettinghaus and co-workers [4-6]. A quadrupole mass spectrometer (QMS) system derived from their research is now manufactured and marketed by Balzers Aktiengesellschaft of Liechtenstien.

Two groups have used QMS based enrichment monitors similar to that developed by Rettinghaus for nuclear safeguards applications. Guzzi and Federico [10-12] constructed a transportable instrument and described its performance in field trials. The aim of their work was to examine the feasibility of transporting this type of system in a vehicle to enrichment plants where safeguards inspectors could use it to perform on-site analysis of collected UF_6 samples.

Nuclear materials accountancy was also given as the reason for the work of Kusahara and Rodrigues [8,9]. As these authors have only published a brief description of their enrichment monitor, we do not know the full extent of their work. From the available information the performance of their system appears to be comparable with that achieved by others [4-6].

The studies cited above are all concerned with the isotopic analysis of gaseous UF_6 samples by quadrupole mass spectrometry. This technique has also been used for the isotopic analysis of solid samples containing uranium and plutonium [13-15]. Several groups have performed such analyses with a thermal ionization quadrupole mass spectrometer developed by Finnigan MAT and reports of their findings have been published [14,15]. In several cases, safeguards was cited as the primary reason for the work.

In selecting quadrupole mass spectrometry for the present work, we were influenced by the success of the afore mentioned studies [4-12]. The latter had shown that this technique could measure the uranium isotope ratios in UF_6 gas samples both reliably and to reasonable accuracy (i.e. 1%). Furthermore, the sample consumption rate was low and fully automatic computer-controlled operation was possible. All these features served to meet the design criteria applied to our system. In addition, a tight schedule prevented us from investigating less well established or more speculative methods for uranium isotope analysis.

* The Australian Nuclear Science and Technology Organization replaced the Australian Atomic Energy Commission on 27 April 1987.

2. DESCRIPTION OF ENRICHMENT MONITOR

The design of the ANSTO system is shown schematically in Figure 1. The main components of this system are the quadrupole mass spectrometer (Balzers model QMG 511), main vacuum chamber (purchased from Balzers), gas inlet manifold, two independent pumping systems, control computer (Digital Equipment Corp. LSI 11/03) and several control modules.

The above monitor is similar in many respects to the one described by Rettinghaus and co-workers [4-6]. However, their design had to be modified to suit our particular application. Also, unattended operation of the monitor with computer access from a remote location necessitated the addition of several features specific to these functions and these are discussed below.

The quadrupole mass analyzer (Balzers model QMA 150) used in the ANSTO monitor was equipped with a cross beam ion source. The source was surrounded by a liquid nitrogen cooled baffle which was incorporated into the main vacuum chamber. The QMS control unit was supplied with a remote control interface (buffer BF 511 and line transceiver LT 511) for connection to the QMS computer. The standard 6 m cable normally used for linking the control unit to the computer was replaced by a 22 m cable to suit the plant layout.

The gas manifold depicted in Figure 1 incorporated two sampling ports connected to the feed and product lines of the gas centrifuge plant. This was adequate for the planned surveillance exercise which did not involve the routine analysis of off-line samples. However, it was possible to introduce samples of known composition into the mass spectrometer through ports located in the lines connecting the manifold to the enrichment plant. These samples were used to test and optimize the system during initial trials and following modifications.

The enrichment monitor contained three liquid nitrogen cooled traps or baffles. These collected the condensable gas introduced into the QMS system during and following each measurement cycle. The baffle surrounding the QMS ion source reduced the background level of UF_6 in this region. It was an important feature of the QMS selected for the present study as it minimized the UF_6 memory effect. The multi-coolant baffle at the inlet to the pumping system on the main chamber can be operated with several different coolants. In initial tests, an acceptable vacuum was achieved when this baffle was water cooled. However, liquid nitrogen was deemed to be the better coolant for unattended operation as it offered the possibility of a back-up trap should the one in the QMS chamber fail through a loss of coolant. The third trap was incorporated in the manifold pumping system.

Unattended operation of the enrichment monitor required long-term automatic replenishment of the above traps with coolant. Since a liquid nitrogen supply capable of achieving this was not available at the site of the enrichment monitor, a compromise was adopted. By using two 50-liter liquid nitrogen Dewar flasks, a maximum supply time of 45 hours was achieved, an operator being required to fill the Dewars every second day.

High and low-level sensors located in each trap controlled the supply of liquid nitrogen. Transfer of coolant was facilitated by pressurizing the Dewars with dry nitrogen supplied from cylinders, rather than by electric centrifugal pump which had proved less reliable and wasteful of coolant in early trials.

A control module associated with the level sensors operated solenoid valves in the liquid nitrogen supply lines. In addition, the liquid nitrogen controller was interfaced to the QMS computer to enable a continuous low-level signal in any trap or power failure to the controller to be signalled to the computer.

Other items of equipment included a computer power fail relay and a control unit for the manifold valves. The relay controlled the power supply to both the QMS and the valve control unit. Switches on the latter permitted manual operation of the manifold valves when necessary. Electronic interface circuits between the computer and the enrichment monitor incorporated an extensive system of failure detection. If an alarm condition was registered with the computer, appropriate parts of the QMS were shut down and the manifold valves closed. If the computer failed, the regular pulses transmitted to the relay ceased, disconnecting power to the QMS and closing the manifold valves.

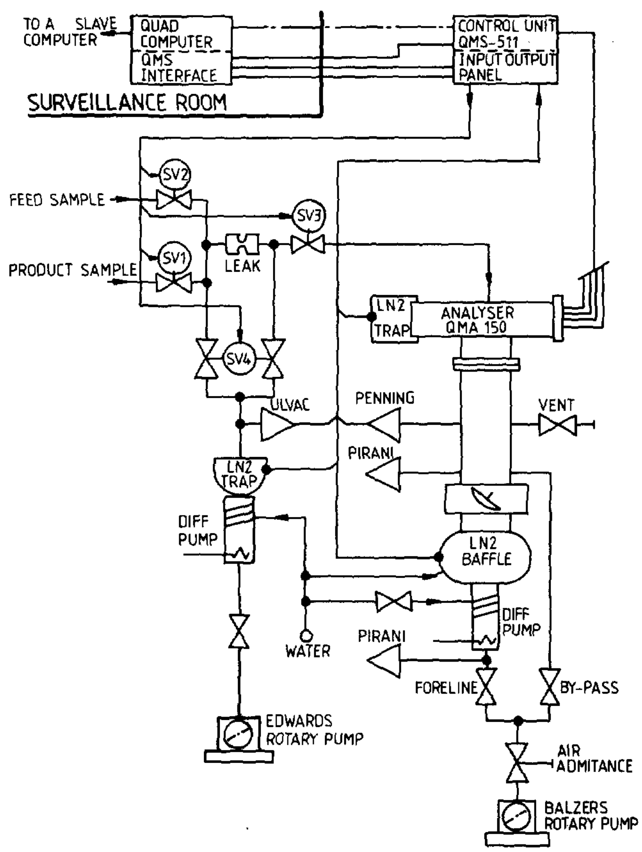

Figure 1. Schematic diagram of enrichment monitor.

3. COMPUTER SYSTEM AND PROGRAMMING

The computer system and programming architecture for the complete surveillance project have been described previously[1]. Briefly, the QMS was operated by a dedicated computer which, in turn, was controlled by a slave computer (DEC LSI 11/23). Both computers were located at the Lucas Heights Research Laboratories. The slave unit was connected by means of modem via the public telephone network to a control computer installed at the ASO office in Sydney (30 km away). All requests for enrichment values or QMS status originated from the control computer and were transmitted to the QMS computer through the slave unit.

All programs for the above computers were written by ANSTO personnel. The option of purchasing software for the QMS computer from Balzers was rejected in favor of an in-house code tailored to our need.

Programming for the QMS computer comprised a number of individual routines:

- Fortran program QUAD which monitored and controlled QMS operation.
- Fortran subroutine MEASUR which performed mass spectral measurements and analyzed the data to yield enrichment ratios.
- a series of assembly language routines which set and read the QMS operating parameters.
- a group of Fortran programs which tested the hardware and the computer controlled procedures.

The program QUAD, which ran on the DEC RT-11 FB operating system, controlled the overall operation of the QMS. It accepted instructions from the slave computer and responded with the QMS status information and the most recent enrichment ratio. Calls to other QMS programs and subroutines were made from QUAD. Its basic functions are summarized in Figure 2.

The measurement procedure for determining the relative concentrations of ^{235}U and ^{238}U in sampled UF_6 is shown in Figure 3. This scheme formed the basis for subprogram MEASUR, a flow diagram of which is shown in Figure 4. At the end of each call to MEASUR, an enrichment value, as defined by

$$\%\ \text{enrichment} \times {}^{235}U \times 100/({}^{235}U + {}^{238}U),$$

was returned to the program QUAD.

The routines for communicating with the QMS console were written in assembly language using the DEC MACRO assembler. They were called from Fortran programs and their higher operating speed was necessary for fast data acquisition. Many of them performed the same functions as routines described in Balzers QMG-BASIC literature (Balzers Publication BK800 093 BE, 1981).

The hardware and the computer-controlled procedures were tested by a group of stand-alone Fortran programs, run from the computer terminal. These were only used when the enrichment monitor was being commissioned.

4. RESULTS

The complete enrichment monitor system including software was initially tested with UF_6 gas samples of known composition. These were contained in sample cylinders

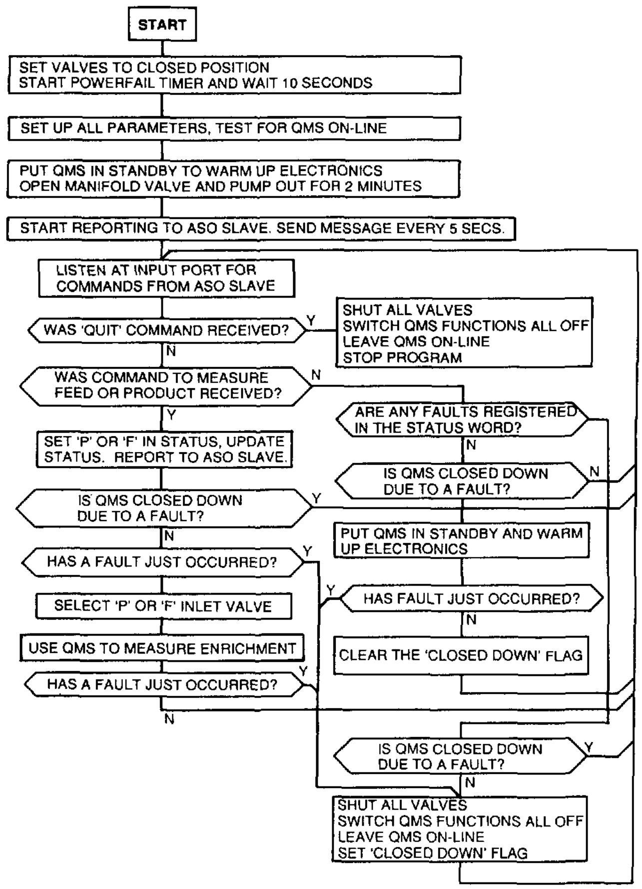

Figure 2. Flow chart of program QUAD.

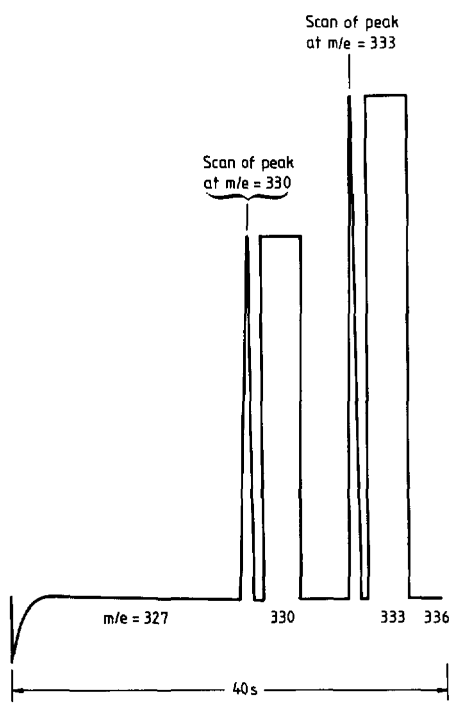

Figure 3. Measurement scheme for the determination of uranium isotope ratios.

connected to ports installed for this purpose (refer Figure 1). The results of two such tests are as follows.

Sample 1: Natural UF_6 (0.71% ^{235}U)
Run 1 — % enrichment = 0.724 + 0.006 Run 2 — % enrichment = 0.725 + 0.008
Sample 2: Enriched UF_6 (8.35% ^{235}U)
Run 1 — % enrichment = 8.43 + 0.06 Run 2 — % enrichment = 8.45 + 0.03

These results are the averages of five separate determinations of percentage enrichment and the errors are one standard deviation. The compositions of the reference samples were confirmed by high precision mass spectrometry. In all cases, the differences between the true values and those measured with the enrichment monitor correspond to percentage errors of 2% or less. The observed trend of slightly higher values from uncalibrated QMS measurements of UF_6 is similar to Balzers' experience under these conditions[16].

Following the satisfactory results obtained in the above off-line analyses, the enrichment monitor was used mostly in the automatic on-line mode until the end of the test period (three months). During this time, the centrifuge enrichment plant was operated over a wide range of conditions. The enrichment monitor performed on-line analyses while these tests were in progress. The results of a typical sequence of measurements from this phase of the work are presented in Figure 5. Also shown in this figure are the enrichment values of collected samples which were measured on a magnetic sector mass spectrometer (O). In all cases, the agreement is very good.

5. DISCUSSION

The work described in this paper was carried out over a six- month period during which time the system was commissioned and tested. The final two months were largely devoted to testing the enrichment monitor by using it to measure the performance of the ANSTO centrifuge enrichment plant. For the final phase of the project, the system was usually controlled remotely from the ASO office. On several occasions, control was returned to ANSTO personnel at Lucas Heights to allow them to adjust the overall surveillance procedure. Some modifications to the QMS software led to small improvements in the accuracy of the enrichment measurements.

The enrichment values shown in Figure 5 are representative of the large number of measurements made with the system in its final form. The accuracy of these measurements is certainly higher than that required for surveillance purposes where the primary concern is with gross changes in plant enrichment. However, at the beginning of the present project it was decided to aim for the highest possible accuracy from the system without resorting to calibration standards. Once this has been achieved, the system can be readily configured to operate as a Go-NoGo monitor of the type proposed for other diagnostic methods [17]. Thus, discrimination between say UF_6 gas enriched to

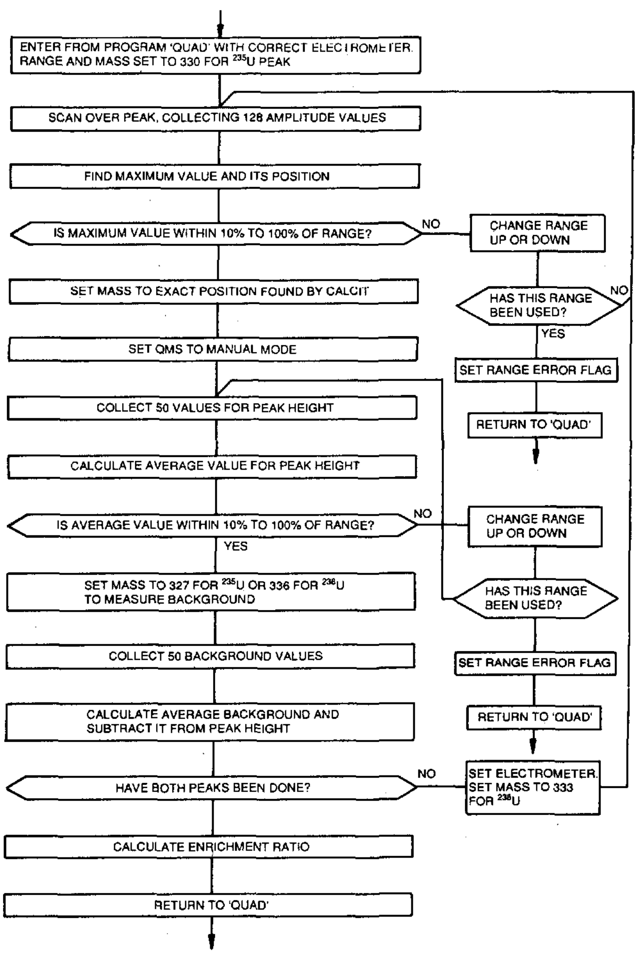

Figure 4. Flow chart of sub-program MEASUR.

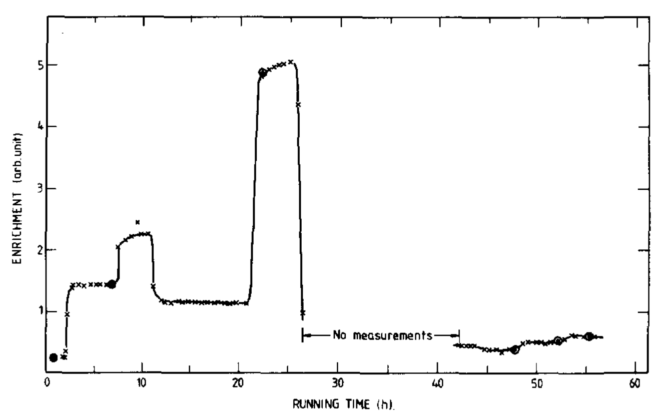

Figure 5. Record of product stream enrichment for a 60-hour period.

4% in ^{235}U and that enriched to 20% can easily be realized on an enrichment monitor capable of yielding results with percentage errors of 2% or less.

As indicated earlier, unattended operation of the monitor was not possible because of the need to replenish the liquid nitrogen storage tanks. If sufficient funds had been available, alternatives, such as closed-cycle refrigeration, would have been investigated. Despite this compromise, it did not significantly detract from the viability of the present system.

A more serious shortcoming of the project was the inability to test the enrichment monitor under continuous remote control conditions for a period of three months or longer. This was considered to be an adequate time for establishing the reliability of the system in a long-term surveillance exercise. The long-term test was not carried out due to the firm completion date set for all experimental work (28 February 1986) combined with delays in commissioning both the enrichment monitor and the centrifuge test facility. Even though an extended test of three months was not carried out, a largely successful test of six weeks duration was completed before the end of the project. This, together with the fact that almost fault-free operation of the system prevailed during the four-month commissioning period suggests that the three-month target is well within the capability of the ANSTO monitor.

Complex instruments of the type described here are always susceptible to component failure. In the present study, a voltage converter in the line transceiver failed when the system was being commissioned, as did several solid-state relays in the manifold-valve controller. Following these failures, all the relays in the controller were replaced by an electro-mechanical equivalent which then provided trouble-free operation.

Had the above failures occurred during remote control of the system, they would have been detected immediately at the remote terminal. This possibility does not exist with a safeguards surveillance system which stores data for some extended period; these data are then retrieved and the condition of the system examined during occasional visits by a safeguards inspector. Subject to prevailing safeguards agreements, a remotely controlled monitor offers the prospect of repairing a malfunctioning system and returning it to operation with a minimum of delay. In addition, such a system provides immediate information on enrichment plant performance. These are the strengths of the system developed for the present study.

6. CONCLUSIONS

A QMS-based uranium enrichment monitor has been developed and subsequently tested on a research scale gas centrifuge plant. Several factors prevented us from meeting the goal of remote, unattended operation for a minimum period of three months. These factors have been considered when assessing the viability of the present approach.

From the experience gained during the course of this project, we conclude there are no major technical obstacles to the use of this type of monitoring system for safeguards related surveillance of enrichment facilities. Ultimately, the adoption of the approach described here will depend upon its acceptance by the operators of enrichment facilities, their national governments and international regulators.

7. ACKNOWLEDGEMENTS

The authors wish to thank D.J. Mercer for his contribution to the computer programming and J.T. Noorman and B.J. Reed for their efforts in the construction and initial testing of the system. The editorial assistance of P.J.F. Newton is also gratefully acknowledged.

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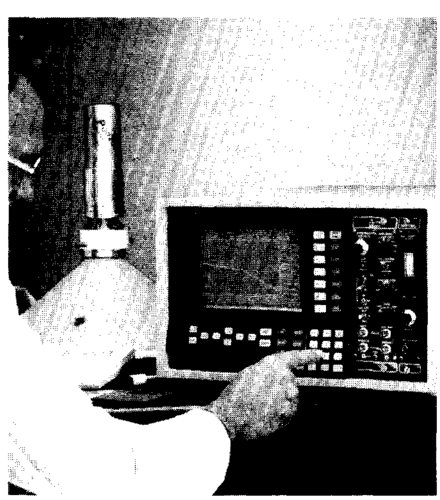

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