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On the Cover:
A mobile crane places the Concrete Integrated Container (CIC) cask on a timber crib at the outside storage area at Ontario Hydro in Toronto.

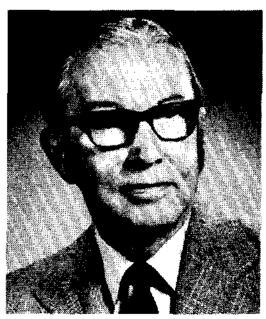
Although our annual meetings attract more participants and the areas of interest have been expanded, our membership has remained at about 800 for the past several years. Your officers need your advice and assistance regarding membership.

Membership provides a number of benefits including:

- Four regular and one proceedings issues of the *Journal*
- Announcements of future meetings and other events of interest
- Reduced fees for attending the meetings and workshops
- Membership in local chapters
- The opportunity to vote for officers and to influence the policies and programs of the Institute.
- Membership has value on a resume

It is, of course, true that many of those who attend the meetings and seminars are only interested in one or a few of the several areas discussed there. Some of them are active members of other professional societies. It is not reasonable to assume that everyone who participates in our meetings should be a member. However, many of them and others should be but are not presently members. Some of those in the newer areas of waste management and transportation may not consider a largely safeguards-oriented society as interested in them. In the few cases when representatives of these areas have been persuaded to run for office, they received few votes. Some thought is being given as to how all of the important interests of our membership might be better represented on the Executive Committee. They are all represented on the working committees which are always represented at meetings of the Executive Committee.

It is not evident that the policy of charging reduced fees for attending our meetings is persuasive to potential members. Many other professional organizations have the same problem. Attendance at a meeting involves paying for transportation and hotels as well as for the attendance fee. Frequently, the employer will pay all of the expenses if it considers that to be important to the company or agency, or if the individual is to present a paper. This stimulates contributions for the meetings, which is constructive; but there is little incentive to save the sponsor a few dollars by becoming a member and paying the dues.



Progress in science and technology relies heavily on the professional societies for communication between those involved in the development, management and implementation of the various disciplines. The societies are the members. The societies organize meetings, publish journals, and stimulate communication in other ways. These activities cost money. More importantly, they involve the voluntary participation of many dedicated individuals, often generously supported by their employers, in developing and implementing the programs of the societies. At the base there must be members who pay dues and vote. On this base the INMM and others arrange for and pay for the necessary full-time staff, persuade the dedicated volunteers to run for office or work on the committees, and raise additional funds in various ways to supplement the income from membership dues.

The point is that membership is basic and that membership dues, although only a small part of our total income, are essential. It does not cost much to be a member. Membership is important for the Institute. The income is important. Advertisers look at our membership and at the number of those who subscribe to the *Journal*. The volunteers work for the members and for their causes. They need the support and encouragement of the members.

Scientists and engineers have a number of responsibilities, one of which is to support their societies.

Some of you should have friends who should become members. Show them this editorial and give them an application form. Some of you may have some good ideas as to how we might increase our membership. We need your suggestions.

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

It Sounds Like the Weather

Each year since 1949 the Institute of Nuclear Materials Management has held an Annual Meeting. We get together and talk about a lot of things. We meet old friends and make new ones. We hear about the latest developments in our field. And, inevitably, someone will say that because of the political climate or the economic forecast or some other national or international barometer, that this will be the last great meeting for the INMM.

Then we turn around and have another great one the following year. Maybe it's in our nature to forecast conservatively. We are a group that assumes all possibilities. And we like to be precise; stay out of heavy weather.

Despite the gloomy forecasts, we must be doing something right. Every year our meeting grows: more people, more papers, more exhibitors. In fact, we have consistently posted record highs with every meeting since 1979.

Here is a prediction based in solid research: We are going to do it again. This year's INMM Annual Meeting will take place in Los Angeles, July 15-18, 1990. We will have 220 papers spread out over 29 sessions. That's another record.

Of course, the point of this meeting isn't simply to set records. The point is to gather together and exchange information and concerns, formally and informally. There are many exciting things happening in nuclear materials management, and they will be covered in depth at our meeting. In addition to the many aspects to be covered in international safeguards, waste management, physical protection, materials control and accounting, and transportation, we are adding a new session on treaty verification.

We will do it again, that is, with your participation. People are what makes it a valuable meeting. And more than 500 safeguards professionals from all over the world will be there to make sure it is. I sincerely hope you're one of them.

Sure, we will again plan conservatively for next year. Since when have weather forecasters been right? See you in Los Angeles.

John Lemming
EG&G Mound Applied Technologies
Miamisburg, Ohio

Progress in Spent Fuel Storage Technology, Delay in Disposal

Several papers are included in this issue of the *Journal* from the INMM Spent Fuel Storage Seminar VII, held January 17-19, 1990 at the L'Enfant Plaza Hotel in Washington, D.C. The meeting provided another in a series of snapshots of the status of the nation's high level waste management and disposal program, and the various technologies that support the storage and ultimately, removal of nuclear power plants' spent fuel. Nuclear power plants now produce about 20% of the nation's electricity, which makes nuclear fuel the second most important source of electricity, after coal.

The general sense of the meeting was determined by the paradoxical contrast between the additional seven-year delay in repository startup schedule, and the technical progress that was evident in spent fuel storage technology. The seven-year repository startup delay, acknowledged by the Secretary of Energy just one-and-a-half months before this seminar, implies a need for an even greater capacity for storing spent fuel prior to disposal, and thus intensifies the issue of at-reactor storage versus centralized Monitored Retrievable Storage (MRS). However, the context in which the repository delay was announced has distinctly positive overtones; the acknowledgement of the over-optimism in prior schedules indicates that DOE is going to be much more articulate about distinguishing between what it can reasonably control and be accountable for, and what it cannot.

The major programmatic benefit of a clear articulation of the limits of DOE schedule accountability would be to better identify and publicize the real sources of delay—the political and judicial processes—in order that the central political and constitutional issues in siting the first repository can be addressed and ultimately resolved by the legislative branch. Clear evidence of realism within at least a part of the legislative branch was demonstrated in an excellent and provocative assessment of the Program by Mary Louise Wagner of the Senate Committee on Energy and Natural Resources staff.

A direct quote from her paper characterizes its theme and context: "But this (Nevada politics) is not an issue that DOE can solve. The Department needs to concentrate on solving the problems that it can solve. And learn to cope with those it can't. Coping with the political problem in Nevada—and the day-after-day rhetoric by Nevada politicians—will mean that DOE will have to get tough. Hard ball will be the name of the game if the Department is going to succeed with the program." Thus, the announced repository delay, though directly negative, indicates possible new approaches within DOE and improved longer-term Program prospects.

The status of the MRS facility and the MRS Commission Report of November 1989 were also the subject of several papers. The Wagner paper, mentioned above, indicated that the MRS Commission Report made it close to impossible to pass 1990 MRS legislation to de-link the MRS schedule from the repository schedule because "The body of the Commission's report made a strong case for a de-linked MRS, but the conclusions rejected it."

In general, both the utilities and DOE agreed with the substance of the MRS Commission's conclusions, but faulted the recommendation of two small facilities. In view of this situation, the principal hope for the near-term progress toward an MRS facility seems to be the appointment of the Congressionally-authorized Nuclear Waste Negotiator, and his subsequent identification of candidate sites and related terms and conditions that are acceptable to both the site host and Congress.

In contrast to the continued uncertainties in the repository and MRS areas, clearly identifiable progress is being made in at-reactor storage technology. Several years ago, the INMM Spent Fuel seminars tended to emphasize spent fuel consolidation as the technology that would probably be providing the next major increment of at-reactor spent fuel storage capacity. This general consensus was based on cost projections that indicated considerably lower costs for consolidation than for the various out-of-pool dry storage methods.

In subsequent years, a number of consolidation demonstrations have been reported. On balance, these demonstrations have probably had a neutral effect on the development of operational consolidation equipment and practice. Resolution of concerns and uncertainties regarding costs and impacts of in-pool consolidation operations, and the management of fuel assembly structurals still remains largely in the future. Nonetheless, consolidation still has several strong utility advocates, in many cases because in-pool storage has major advantages over out-of-pool storage at particular sites.

The past year has also seen the entry of an additional vendor, with some innovations as to approach. It thus remains likely that consolidation will prove to be an economic and viable alternative for storage capacity extensions.

The most favorable storage technology development in the past year is the strong confirmation of the viability of concrete dry storage casks. This trend had been evident in the past two or three INMM seminars. It was particularly notable in the past year because of the relatively large number of new commitments to concrete casks, the growing numbers of NRC-approved designs, and the successful operation experience with the earlier commitments. The fixed horizontal concrete cask (NUHOMS) continues to be popular, but the movable vertical concrete cask has also become a viable contender. Concrete casks have been chosen in preference to consolidation because the latter is still viewed as operationally more complex and uncertain. Thus, the potential cost advantages of consolidation can be viewed as uncertain. Although concrete casks are somewhat less convenient to load and unload, they are now generally favored over metal casks because they are cheaper, and can now be licensed on a comparable schedule. The original dry storage facility, which employs metal casks, continues to function well.

In the largest sense, the most significant characteristics of U.S. spent fuel storage technology is the diversity of acceptable alternatives and vendors that are available. This diversity permits the matching of storage technology to the individual circumstances at each site. It also leads to significant inter-technology and inter-vendor competition, which continues to drive innovation and decrease costs.

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

Correction

The upper right hand photo on page 5 of the *Journal of Nuclear Materials Management* [February 1990] is Mr. Kenichi Murakami, Nuclear Safety Bureau/Science & Technology Agency, speaking at the 10th Annual Meeting of the Japan Chapter in Tokyo, Japan. Murakami was misidentified as Mr. Yousuke Nakae.

Radioactive Waste Management

The Technical Working Group for Radioactive Waste Management conducted the INMM Spent Fuel Management Seminar VII, January 17-20, 1990 at Loew's L'Enfant Plaza Hotel in Washington, D.C. A total of 29 speakers covered the subjects of spent fuel storage technology, fuel failures, behavior of spent fuel in storage, monitored retrievable storage, DOE programs, and public policy issues. The luncheon speaker was Leo Duffy, Special Assistant to the Secretary of Energy for waste management affairs. Approximately 135 persons attended the meeting. Attendance included personnel from industrial, government, research activities—both domestic and foreign. Three media personnel also attended. Foreign representation totaled about 15 persons.

Plans were initiated for a waste management program for the INMM Annual Meeting in Los Angeles in July. To date, four sessions are planned covering Greater-than-Class C wastes, spent fuel and high level waste disposal, low level waste disposal, and the safeguards aspects of waste disposal. Speakers have been invited for most of the program, with the remaining program to be filled out by responses to the Call for Papers. Abstracts of a portion of the invited papers have been received and others are expected shortly.

The TWG was successful in arranging for John W. Bartlett, Ph.D. to be the keynote speaker at the INMM Annual Meeting. In January, Mr. Bartlett was nominated by President George Bush to be the Director of the Office of Civilian Radioactive Waste Management (DOE). His confirmation hearing was held February 5, 1990 and his nomination was reported out of committee February 21, 1990; full Senate vote on the nomination is expected in the near future. In the unlikely event he should not be confirmed, arrangements will be made for a substitute speaker (possibly Mr. Duffy).

Recommendations were sent to Mr. William Higinbotham, Ph.D. regarding papers that were presented at the January 1990 spent fuel seminar that are considered as candidates for publication in the spent fuel management edition of the *Journal of Nuclear Materials Management.*

Plans also have been initiated for the INMM Spent Fuel Management Seminar VIII to be held in Washington, D.C. in January 1991.

E.R. Johnson, Chairman
INMM Technical Working Group on Radioactive Waste Management
Oakton, Virginia U.S.A.

Physical Protection

The Technical Working Group on Physical Protection is currently organizing the INMM 31st Annual Meeting to be held July 15-18, 1990, at the Biltmore Hotel, Los Angeles, California. More than 50 papers covering a broad range of physical protection and security related issues will be presented.

Consideration is being given to a possible "Package Search Techniques" workshop to concentrate on better and more effective methods of searching packages which enter restricted areas. The workshop has not yet been scheduled. If you are interested in attending 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.

Security Personnel Training Workshop

The Security Personnel Training Workshop was held at the Clarion Four Seasons Hotel in Albuquerque, New Mexico, April 8-11, 1990. Many excellent sessions including a tour of the Department of Energy's Central Training Academy were held. Harry Leith, ERCE, was the workshop chair.

Plenary session speakers were:

Jerry C. Howell
DOE Central Training Academy

Richard J. Lavernier
DOE Office of Security Evaluation

Arthur B. Flynn
DOE Central Training Academy

James Hallihan
Mason and Hanger

Bernard W. Stapleton
ERCE

The luncheon speaker was James Jacobs, Sandia National Laboratories. Speakers during the visit to the Central Training Academy were Ted Leamons, Joe Lewis, Russell Showers, and Jack Jones.

The following workshop sessions were conducted:

- Security Force Exercises/Readiness Assessment
- Facility Modeling and Vulnerability Assessment
- Special Response Teams, Issues and Insights
- Human Factors/Physical Fitness/ Fitness for Duty
- Protection Program Planning and Administration
- Training Techniques, Equipment, and Facilities
- Implementing and Maintaining a Viable Quality Assurance Program
- The Expanding Role of Computers in Security Management
- Career Paths and Professional Development
- Administrative Factors, Legal Issues, and Financial Considerations
- Techniques for Improving Liaison With Support Agencies
- Security Force Standards, Criteria, Inspections and Evaluation
- Security Force Weapons, Tactics and Deployment Strategies

Special thanks go to the following workshop moderators:
David A. Axt, American Protective Service; Robert Batten, Harry Leith,

and Bernard W. Stapleton, ERCE; Kenneth M. Jurjevich, CRC, Inc.; and Drew Pretzello, Boeing Petroleum Service. The next workshop on this topic is tentatively planned for spring 1992.

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

Japan Chapter Report

Executive Committee Meetings were held at NMCC Headquarters in Tokyo in October and December 1989, and March 12, 1990. Chapter officer elections were held at the October meeting. A revision to the Japan Chapter bylaws was proposed by the Japan Chapter and was approved by the INMM Executive Committee, November 23, 1989.

Mr. Hiroto Kurihara was re-elected as Program Chair of the 11th Annual Meeting of the Japan Chapter.

The Annual Meeting will be held in Tokyo, June 7, 1990. The following program is planned:

Plenary Session

Four guest speakers are invited and the following lectures will be given: "Nuclear Regulatory Issue—Low Level Radioactive Waste Management," Director General of Nuclear Safety Bureau; "Nuclear Safeguards and Non-Proliferation," R. W. Getzinger, Counselor for Scientific and Technological Affairs, U.S. Embassy; "Requirements For Nuclear Scientists From the Public," conducted by a journalist; and "Current Status and Future on Economical and Social Affairs of USSR," conducted by a trade expert of Sumitomo Corp.

Technical Session

There will be five technical sessions on subjects including International Safeguards, Measurement Technology For Accounting, Containment and Surveillance, Verification

Technology, and others. Twenty papers and four exhibits will be presented.

Business Meeting

Ambassador Yousuke Nakae will be invited as a guest speaker at the Business Meeting. He will discuss nuclear safeguards issues in the People's Republic of China.

A banquet will be held following the Meeting.

Mitsuho Hirata, Chairman
INMM Japan Chapter
Nuclear Material Control Center
Tokyo, Japan

N14 Committee

The 1990 Annual N14 Committee Meeting will be held in the Washington, D.C. area the week of Oct. 14, 1990. The exact date and location will be determined by June 1.

A chair for the Writing Group is urgently needed for a proposed standard N14.26 "Inspection and Preventative Maintenance of Packaging for Radioactive Materials." Anyone interested should contact INMM headquarters or John Arendt, N14 Committee Chair, Oak Ridge Laboratories, P.O. Box 117, Oak Ridge, TN 37831-0117.

Specific N14 Standards Highlights:

ANSI N14.1—Balloting on the suggested changes has been completed. Forty-two ballots were returned with 29 yes, 7 yes with comments and 6 not voting. The Chair has incorporated comments where appropriate and the documents will be sent to ANSI for approval by May 15, 1990.

ANSI N14.2 (in process)—A draft has been completed for review within the Writing Group.

ANSI N14.6 and N14.19—These two standards must be revised or reaffirmed in 1991. Planning on these two activities will start by September 1, 1990.

ANSI N14.7 (in process)—Work on this draft is continuing.

ANSI N14.23—(Work on this draft is continuing within the Writing Group.)

ANSI N14.24—1985—Plans for reaffirmation or revision of this standard have started with a standard completion date in 1991.

ANSI N14.30 (in process)—Balloting has been completed. Forty-nine ballots were returned with 24 yes, 15 yes with comments, 9 no, and 1 not voting. The Writing Group will now consider comments and make changes where appropriate.

The ANSI N14 Subcommittee work for development of a numerical model for thermal evaluation of UF6 cylinders is in process. A draft report is scheduled to be available about July 1, 1991.

Safeguards

The Safeguards Committee met February 14, 1990 at the Nuclear Regulatory Commission's (NRC) Headquarters in Rockville, Maryland. Representatives from industry, the NRC, and the Department of Energy (DOE) attended.

The meeting covered the following topics: 1) the proposed Physical Fitness Performance Standards and Training Rule, 2) Uranium Enrichment Safeguards, 3) Access Authorization Policy, 4) Changes in Safeguards Inspection Program, 5) Regulatory Effectiveness Reviews, 6) DOE Safeguards, and 7) International Safeguards.

Robert F. Burnett, Director, Division of Safeguards and Transportation, Nuclear Regulatory Commission, provided an overview of activities for the INMM Safeguards Committee meeting. Mr. Burnett discussed issues related to transportation which included security, safety, ports in and out of the U.S., and American materials being transported among various countries. The Waste Isolation Pilot Plant, as well as transportation of Category I and II materials, was discussed. There is currently no U.S. commercial transporter for Category I material in the U.S. This material has been transported by the DOE safe secure trailers (SST). Current issues included concerns over tritium and regulation changes, the open communications issues within facilities, and the possibility of two new uranium enrichment facilities applying for licenses.

There is activity at the NRC for a proposed amendment to 10 CFR Part 73 for Category I Fuel Facilities. These changes will involve a standardized day firing qualifications and annual requalifications course using all assigned weapons, and an ongoing physical fitness training program with performance standards. These changes will ensure uniformity with current night firing qualifications, provide current requirements for tactical response exercises, and comparability with DOE. A proposed rule will be published in July 1990 with the intent to publish a final rule in April 1991.

Discussions on Uranium Enrichment centered on the possible new licensing activities for two facilities: 1) Louisiana Energy Services centrifuge plant and 2) AVLIS plant. NRC will be required to write new rules to cover the licensing of these facilities.

An update of the ongoing access authorization program for nuclear power plants was presented. A general rule is being developed along with a Regulatory Guide. These may be out as early as June 1990.

The NRC Material Control and Accounting (MC&A) Inspections program has been centralized at Headquarters. There will be 35-40 individual visits per year ranging from 2-5 days. NRC is trying to hire qualified persons at the GS13-14 level for inspectors. The physical security inspections are being performed by the Regions. Inspectors will check and test specific equipment during these inspections.

The Safeguards Committee reviewed a videotape of the ongoing Regulatory Effectiveness Reviews which are performed at power plants. The video provided a very enlightening view of some of the ways of defeating certain types of physical protection hardware.

Rod Martin, Chief, International Safeguards Branch, DOE, discussed the new organization within the Office of Safeguards and Security (OSS). The new organization will consist of three main divisions: 1) a Policy/Standards/Analysis Division, 2) a Headquarters Operations and International Safeguards Division, and 3) a Field Operations Division.

The DOE is in the process of reviewing their survey procedures, DOE orders, and comparability of field office inspections, as well as training and inspection courses. The DOE is reviewing performance requirements and the relationship between performance numbers and safeguards systems.

The new directions for research and development involve ways to design safeguards, security, safety, environmental, and operations into a totally integrated system. The use of automation, computers, robotics, and other automated equipment will play a major role in new systems. A major area of research involves system quality assurance concepts which include portal monitors, metal detectors, and other equipment.

DOE/OSS also has major activities in the International Safeguards area. They support research and development, bilateral exchanges of technology, IAEA support, international training courses, physical protection, nuclear material management, U.S. IAEA safeguards agreements, and technical/general leadership.

The International Safeguards Committee, a subcommittee to the Safeguards Committee, met and discussed the roles and direction the Committee should be pursuing. There was an excellent exchange of information among the NRC, the DOE, and industry. Cecil Sonnier, Sandia National Laboratories, is chair of this subcommittee and is in the process of obtaining selected international participation. Please contact him directly at (505) 844-8478, if you are interested in international safeguards issues and are willing to participate in this subcommittee.

The next meeting will be held at the INMM Annual Meeting in July 1990.

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

Selected Concrete Spent Fuel Storage Cask Concepts and the DOE/PSN Cooperative Cask Testing Program

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Mikal A. McKinnon
Pacific Northwest Laboratory
Richland, Washington U.S.A.

Cesar E. Collantes
U.S. Department of Energy
Richland Operations
Richland, Washington U.S.A.

ABSTRACT

To date, water pools, metal casks, horizontal concrete modules, and modular vaults have been used to store the major quantity of commercial light water reactor spent nuclear fuel. Recently, vertical concrete dry storage casks have received consideration for storage of spent nuclear fuel. This paper reviews the evolution of the development of selected vertical concrete dry storage casks and outlines a cooperative cask testing (heat transfer and shielding) program involving the U.S. Department of Energy and Pacific Sierra Nuclear Associates. Others participating in the cooperative program are Pacific Northwest Laboratory; EG&G Idaho, Inc.; Wisconsin Electric Power Company; and the Electric Power Research Institute.*

INTRODUCTION

To date, water pools, metal casks, horizontal concrete modules and modular vaults have been the primary methods used to store commercial light water reactor (LWR) spent nuclear fuel. Recently, vertical concrete casks have received consideration for storage of commercial LWR spent nuclear fuel. Conceptual designs indicate that vertical concrete dry storage casks are economically and technically competitive with metal casks, horizontal concrete modules, and modular vaults.

In July 1987, the U.S. Department of Energy (DOE) released a Solicitation for Cooperative Agreement Proposal (SCAP) to performance test (heat transfer and shielding) dry storage modules that could be used at reactors or at regional/central facilities associated with interim storage or final disposal systems.[1] The modules/casks were not to be similar to metal casks previously tested at GE-Morris by DOE and the Electric Power Research Institute (EPRI);[2,3] at the Idaho National Engineering Laboratory (INEL) by DOE and Virginia Power;[4,5,6] at INEL by DOE and EPRI;[7] or horizontal concrete modules demonstrated at Carolina Power and Light Company's (CP&L) H. B. Robinson reactor site by CP&L, DOE, EPRI, and NUTEC, Inc.[8] SCAP proposals were received and evaluated by DOE and a Cooperative Agreement was established in September 1988 with Pacific Sierra Nuclear Associates (PSN).

Through the DOE/PSN Cooperative Agreement, PSN will provide one ventilated concrete storage cask to DOE and DOE will conduct a heat transfer and shielding performance test at INEL.

The purpose of this paper is to review the evolution of the development of selected vertical concrete storage casks and to outline the DOE/PSN cooperative cask testing program.

Selected vertical concrete dry storage cask concepts are described in this section. In addition, estimated costs of spent fuel storage in vertical concrete casks are presented and compared with costs of other storage methods.

CONCRETE DRY STORAGE CASK CONCEPTS

A few selected conceptual designs of vertical concrete dry storage casks are presented. The cask concepts are discussed in order of chronological development.

DOE Retrievable Ventilated Concrete Storage Cask

In the early 1970s, DOE developed a retrievable ventilated dry storage cask shown conceptually in Figure 1.[9,10] The cask was approximately 17 ft long, 9 ft in diameter, and weighed between 70 and 130 tons depending on the waste type (spent fuel or vitrified high level) and quantity to be stored. The concrete cask contained waste containers in a sealed carbon steel canister. Natural convection caused air to enter inlets at the bottom of the cask, flow upward through an annulus between the cask and sealed carbon steel canister, and flow out the discharge vents near the top of the cask. The advantage of the "ventilated" concept is that the air transfers a large fraction of the heat from the waste spent fuel canister and thereby reduces the temperature of the concrete cask wall.

In 1976,[11] a proof-of-concept test was performed to evaluate the heat transfer, thermal stress, and shielding performance of the ventilated concrete cask. The cask performance in all three areas was satisfactory and indicated comfortable margins of conservatism in heat transfer and thermal stress areas.[11]

Canadian Unventilated Concrete Storage Casks

Canada is the leader in the implementation of vertical concrete storage casks. Their program has progressed from demonstrations of storage-only concrete casks to feasibility testing and planned demonstrations of a storage/transport/disposal concrete integrated cask (CIC).

In 1974, Atomic Energy Canada Limited (AECL) began a development and demonstration program of concrete dry storage casks similar to that shown in Figure 2 at its Whiteshell Nuclear Research Establishment (WNRE).[12] Analyses and experimental tests have provided conclusive evidence that concrete casks have wide margins of safety in shielding effectiveness, integrity after repeated freeze/thaw cycles, man-rem dose requirements, potential accident situations, and safeguards requirements. In addition, implementation of a transfer cask-to-storage cask dry loading technique eliminates the need to load concrete casks in reactor pools. Because of their excellent performance, AECL has opted to use concrete casks to store spent fuel from the WR-1, Gentilly-1, and Douglas Point reactors.

More recently, Ontario Hydro has undertaken a program to assess the feasibility of using concrete casks to store, transport, and possibly dispose of CANDU spent fuel.[13] A concrete integrated cask (CIC) was designed (Figure 3) and is approximately 8 ft in diameter, 12 ft long, and weighs ~75 tons loaded with spent fuel. The cask walls and bottom are fabricated from high-density concrete with rebar reinforcement. The reinforced concrete wall is lined on both the interior and exterior surfaces with carbon steel. Trunnions are attached to the cask body for lifting. A lid, also fabricated from reinforced concrete, is bolted to the cask body and sealed with conventional elastomer and metallic seals.

Two half-scale cask models were fabricated and they were tested for a 30-ft corner drop, a 3-ft drop onto a steel pin, and a 961°C fire lasting 7.5 minutes. The results from these tests indicated that concrete is a good material for a transport cask and only minor refinements in the design and fabrication method are required. It is believed that the CIC is feasible and could become an important alternative method for spent fuel storage, transport, and possibly disposal of CANDU spent fuel.[13]

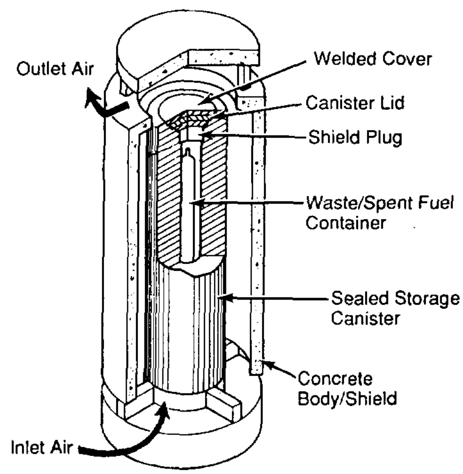

Figure 1. DOE Retrievable Ventilated Concrete Storage Cask[10]

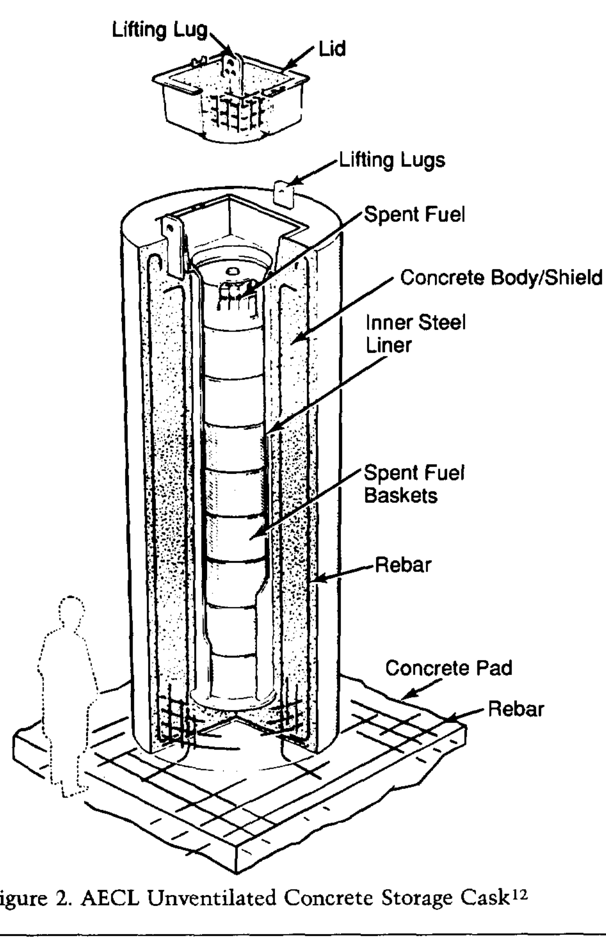

Figure 2. AECL Unventilated Concrete Storage Cask[12]

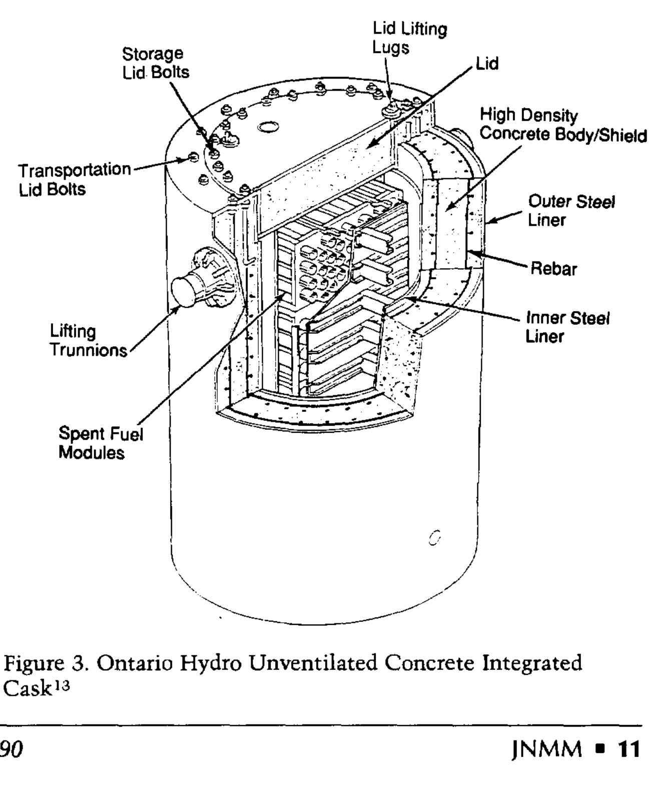

Figure 3. Ontario Hydro Unventilated Concrete Integrated Cask[13]

DOE Unventilated Concrete Storage Silo

During the period from 1978 through 1982, a large concrete silo (Figure 4) was designed, fabricated, and tested by DOE at the Nevada Test Site.[14] The concrete silo consisted of

1. a carbon steel liner encased in a locally transportable reinforced concrete silo, 9 ft in diameter by 21 ft high,
2. a 16 ft by 16 ft by 9 ft thick reinforced concrete foundation pad,
3. a canister assembly consisting of a canister body, closure lid, and a concrete-filled shield plug, and
4. a pressurized water reactor (PWR) spent fuel assembly.[14]

A heat transfer performance test was conducted. The measured temperatures in the PWR spent fuel assembly, canister, and silo were satisfactory and were less than pretest computer code predictions.

NUPAC CP-9 Unventilated Concrete Storage Cask

In 1987, Nuclear Packaging, Inc. (NUPAC) submitted a topical report to the U.S. Nuclear Regulatory Commission (NRC) on its CP-9 unventilated PWR storage cask.[15,16] The cask (Figure 5) is approximately 8-1/2 ft in diameter, 19 ft long, and weighs approximately 90 tons loaded with 9 PWR spent fuel assemblies. A steel basket is contained in the steel-lined, reinforced concrete cask body. The cask closure includes an inner steel shield plug, a steel lid welded to the top of the cask, and a solid RX-277 borated neutron absorber plate. Three lift lug embedment assemblies are provided for handling the cask. A water drain is provided near the bottom of the cask to permit loading spent fuel in a pool. Dry loading of the cask can also be accomplished using a transfer cask and collar mechanism.[16]

NUPAC submitted the CP-9 topical report to NRC for review and approval in November 1987. Pacific Nuclear Systems, Inc., who owns NUPAC, purchased a share of Pacific Sierra Nuclear Associates (PSN) in 1988 or 1989, and responsibility for developing and marketing vertical concrete casks was transferred to PSN. PSN opted to postpone further review of the CP-9 unventilated cask and pursue approval of a ventilated concrete cask as discussed below.

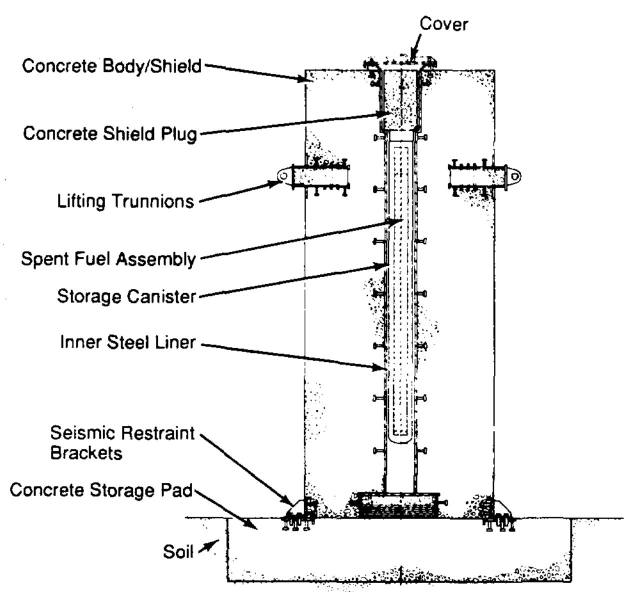

Figure 4. DOE Unventilated Concrete Storage Silo[14]

PSN Ventilated Concrete Storage Cask

In February 1989, PSN submitted a topical report to NRC for its ventilated storage cask (VSC) system.[17] The cask (Figure 6) consists of a steel lined reinforced concrete body; a multiassembly shielded, sealed basket (MSB); an air flow channel between the cask inner liner and MSB; and optional snow skirts and lifting lugs/trunnions (not shown) if desired.

VSC casks can be sized to store 4 to 24 PWR assemblies or 9 to 56 BWR assemblies.[17] The present MSB/VSC design is loaded dry via a metal transfer cask and collar mechanism. The metal transfer cask containing the MSB is loaded with spent fuel in a pool; the transfer cask and

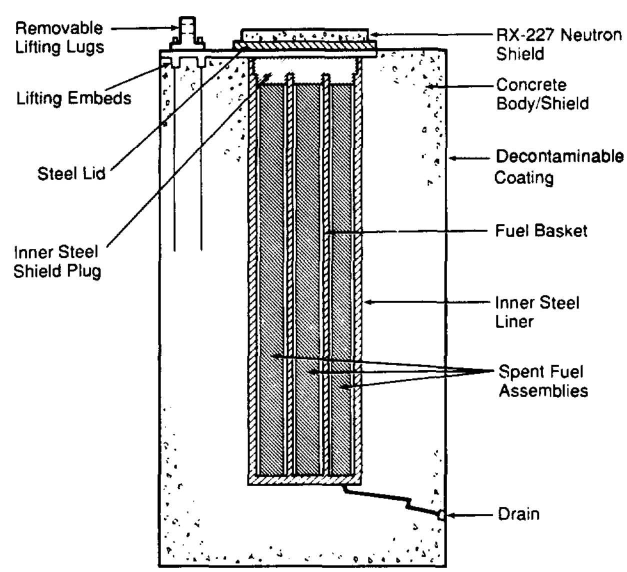

Figure 5. NUPAC CP-9 Unventilated Concrete Storage Cask[16]

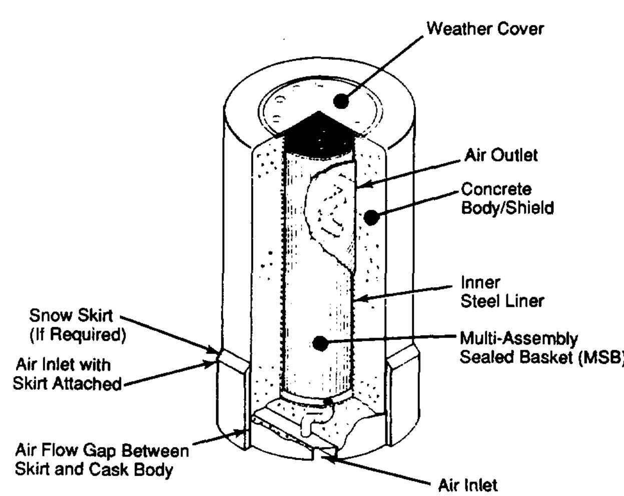

Figure 6. PSN Ventilated Concrete Storage Cask[17]

MSB are drained and the MSB lid is welded on; the MSB is vacuum dried, backfilled with helium, and moved to the VSC via the metal transfer cask; and the MSB containing spent fuel is transferred into the VSC using a collar mechanism mounted on top of the VSC.[17]

PSN's topical report is presently being reviewed by the NRC. Wisconsin Electric Power Company[18] and Consumers Power Company[19] have indicated intentions to submit license applications for use of VSCs at reactor independent spent fuel storage installations (ISFSI).

B&W CONSTAR Unventilated Concrete Storage Cask

In 1988 and 1989, Babcock and Wilcox Company (B&W) presented its CONSTAR (Concrete Storage at Reactor/Repository) dry storage cask system.[20,21] The CONSTAR-28P storage cask (Figure 7) is approximately 11 ft in diameter, 20 ft long, and weighs 165 tons fully loaded.[21] The cask consists of a steel-lined, high-density, steel-fiber reinforced concrete body; heat pipes attached to the inner steel liner with a radiator section located near the top of the cask; a spent fuel storage basket; a lid system consisting of an inner concrete shield plug, a primary steel lid, and a secondary steel weather cover; and lifting lugs.[20,21]

Enhanced heat transfer using heat pipes is unique to the CONSTAR design. Heat generated in the spent fuel assemblies is transferred to the inner steel liner by conduction, convection, and radiation. Heat pipes efficiently transfer heat to a radiator located at the top of the cask to maintain the concrete cask wall below allowable temperature limits.

The CONSTAR cask is loaded dry using a shielded transfer cask that interfaces with the top of the CONSTAR via a shielded transfer platform.[21] This loading/unloading technique provides compatibility with the steps in storage, transport, and ultimate disposal of spent fuel. The spent fuel may never have to be returned to the pool because the shielded transfer cask/platform technique could be used to transfer spent fuel from a CONSTAR to a transport cask in the future.[21]

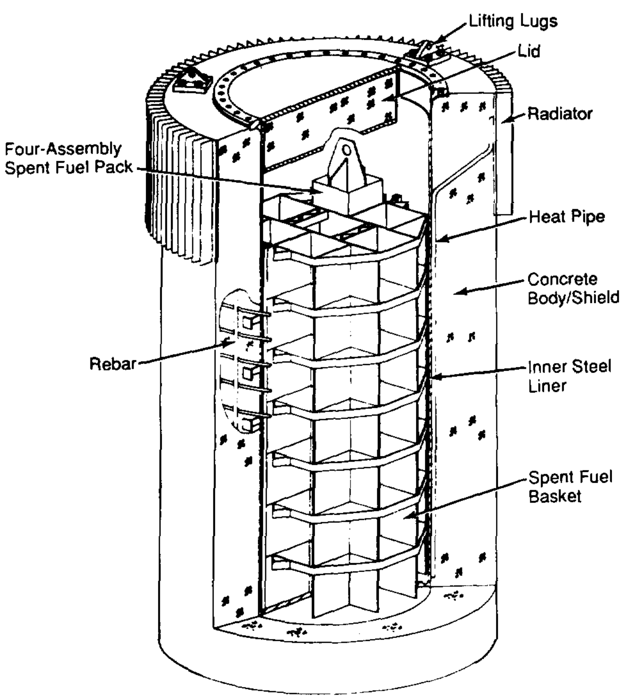

Figure 7. B&W CONSTAR Unventilated Concrete Storage Cask[21]

ESTIMATED COSTS OF CONCRETE CASK STORAGE

In 1988, DOE performed a study for Congress on at-reactor dry storage methods.[22] As part of that study, costs for various at-reactor storage expansion options were estimated. The estimated costs presented in Table 1 were extracted from estimates provided by vendors, utilities, and other DOE studies.[22] The cost estimates are associated with large uncertainties resulting from different accounting methods, site-specific differences, and unknown cost reductions due to technology advancements or improvements in production methods. The cost estimates were intended to estimate a waste system-wide cost resulting from at-reactor storage and were not intended to be used by a utility to select a storage technology.[22] Direct quotes from vendors should be used to accurately determine spent fuel storage costs.

It is clear from Table 1 that vertical concrete casks and horizontal concrete modules were estimated to cost less than metal casks or modular vaults for quantities of 300 MTHM or less. As the quantity approaches 1000 MTHM, metal casks and modular vaults become economically competitive with concrete modules/casks. It should be emphasized that selection of a storage type will depend on more than economics. Considerations such as operability and transportability will also significantly influence final selection of a storage system.

Table 1
Estimated Storage Expansion Cost Ranges, $/kg Heavy Metal[22]

	Capacity Increase		
Storage Technology	*100 MTHM**	*300 MTHM*	*1000 MTHM*
In-pool consolidated fuel	40-75	30-50	NA**
Metal casks	60-115	55-105	55-100
Concrete casks	50-110	45-95	45-85
Horizontal concrete modules	60-80	45-60	40-55
Modular concrete vaults	105-155	70-105	45-70

* Metric tons of heavy metal.

** A 1000 MTHM increase is not practical for in-pool rod consolidation because at typical reactors not more than approximately 350 MTHM of additional storage space can be gained through consolidation.[22]

DOE/PSN COOPERATIVE CONCRETE STORAGE CASK TEST

This section outlines/describes the objectives; organization; scope and responsibilities; ventilated concrete storage test cask; cask instrumentation; test matrix; and status and future activities of the DOE/PSN Cooperative Concrete Cask Test.

The primary objectives of the concrete cask test are to obtain heat transfer and shielding data, and to gain fabrication and handling experience with a ventilated concrete storage cask. This data/experience will be available to future applicants to assist them in obtaining approvals/licenses of similar storage modules and ISFSIs. In addition, the information generated in this test will be provided to the NRC to aid in obtaining generic approvals for the use of concrete casks to store LWR spent fuel at reactors and other storage facilities.

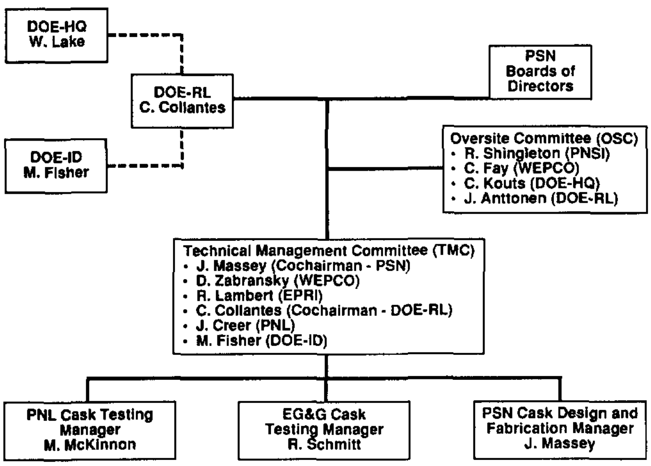

Figure 8. DOE/PSN Organization

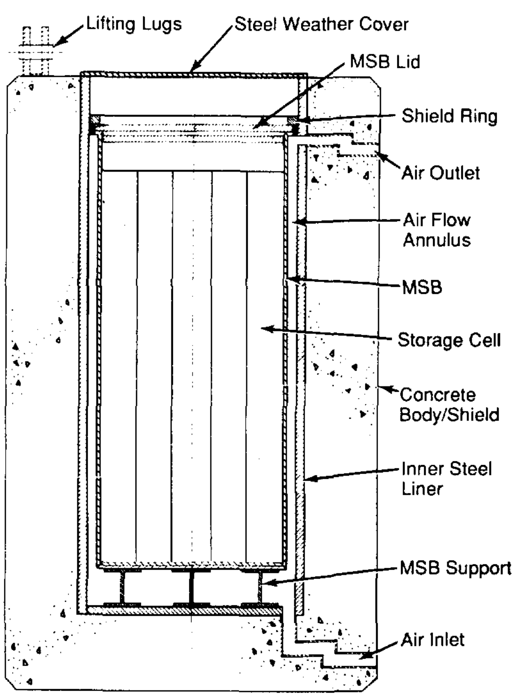

Figure 9. PSN VSC-17 Ventilated Concrete Test Cask

COOPERATIVE AGREEMENT ORGANIZATION

The DOE/PSN Cooperative Agreement involves three DOE organizations, two DOE prime contractors, PSN, a utility, and EPRI. The organization is shown in Figure 8 and consists of a cooperative agreement between DOE and PSN, an Oversight Committee (OSC), a Technical Management Committee (TMC), and technical projects at Pacific Northwest Laboratory (PNL), EG&G Idaho, Inc., and PSN.

The overall management of the DOE/PSN Cooperative Concrete Cask Test is vested in the TMC. The TMC consists of members from DOE-Richland (RL), PNL, DOE-Idaho (ID), PSN, WEPCO, and EPRI. Issues that cannot be resolved by the TMC are resolved by the OSC. The OSC consists of members from DOE-RL; DOE-Headquarters (HQ); Pacific Nuclear Systems, Inc. (a part owner of PSN), and WEPCO.

The DOE/PSN Cooperative Agreement is the contractual document controlling concrete cask testing activities. An overall Program Management Plan and DOE and PSN Technical Project Plans are used to plan, integrate, coordinate, and control the technical work needed to conduct and complete the concrete ventilated storage cask (VSC) performance test.

VSC-17 CONCRETE TEST CASK

The VSC-17 concrete test cask is of the ventilated design (Figures 1, 6, and 9) and stores 17 PWR unconsolidated/consolidated spent fuel assemblies/canisters. The VSC-17 is approximately 9 ft in diameter, 20 ft long, and weighs ~100 tons loaded with consolidated PWR spent fuel canisters.

The cask consists of a reinforced concrete body containing an inner steel liner, two lifting lugs, and a steel weather cover. A multiassembly, sealed and shielded bas-

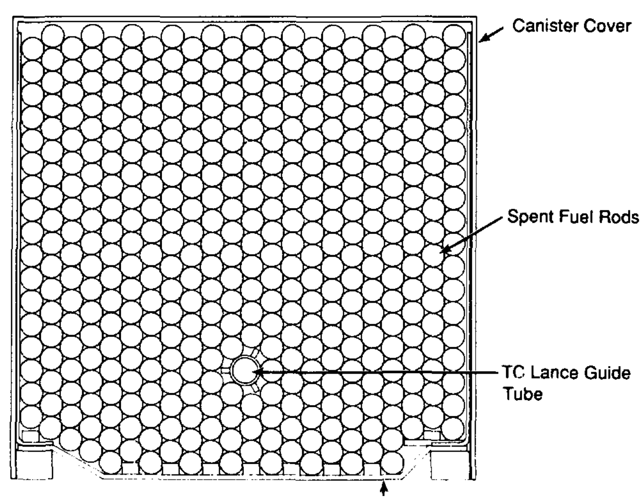

Figure 10. Cross Section of a Loaded INEL Consolidated Spent Fuel Canister[7,27]

ket (MSB) made of carbon steel is located inside the concrete cask body. The PWR MSB is supported on three wide flanges joined by a common circular plate. The PWR MSB support can be removed and a longer, BWR MSB can be inserted in the concrete cask body.

Heat is transferred to the walls of the MSB via conduction, convection, and radiation. As air adjacent to the outer surface of the MSB heats up, it becomes less dense and flows, by natural circulation, upward adjacent to the MSB, and exits through four outlets located near the top of the cask. Air enters the cooling annulus between the MSB and cask inner liner through four inlets located near the bottom of the cask. Because most of the heat is transferred from the MSB by axial natural air convection, the heat passing through the concrete body is significantly reduced, which results in lower concrete temperatures. Pretest heat transfer predictions using the COBRA-SFS code[23, 24, 25, and 26] will be performed to accurately calculate cask and spent fuel temperatures prior to performing actual test runs.

CONSOLIDATED SPENT FUEL

Spent fuel rods to be used in the test are from Surry and Turkey Point PWR spent fuel assemblies having square cross sections, nominally 8.4 in. on a side, and having a total length of 159.8 in. The rod active fuel column is 144 in. long.

The fuel rods in a fuel assembly are arranged in a square array with 15 rods per side and a nominal rod-to-rod centerline pitch of 0.563 in. In addition to 204 fuel rods, a fuel assembly also includes a top nozzle, a bottom nozzle, seven grid spacers, twenty guide tubes, and one central incore instrumentation thimble.

During the consolidation process,[27] spent fuel rods were removed from Surry and Turkey Point fuel assemblies and placed into consolidation canisters. A two-to-one rod consolidation ratio was consistently achieved because each canister could contain 410 spent fuel rods, and two fuel assemblies provide only 408 rods. This resulted in two extra fuel rod storage locations per canister.

A simulated guide tube (Figure 10) with a funnel-shaped top was placed in each of seven canisters to provide locations for inserting thermocouple (TC) lances (Figure 11) during cask performance testing. The simulated guide tubes were designed to occupy three fuel rod locations. The overflow fuel rods caused by inserting guide tubes in canisters were placed in the next canister of fuel consolidated.

A stainless steel fuel consolidation canister (Figure 10) consists of a base and top-locking cover. A series of spacers, support bars, and ties is attached to the base of the canister to align and hold the fuel rods during consolidation. Once all the fuel rods have been placed on the base, the top cover is placed over the rods and locked into place.[27] The top cover, the sliding fit between the top cover and base, and the canister locking mechanism do not seal the canister, but do limit gas flow in and out of the canister. The loaded canister is 8.5 in. square by 159.6 in. long.

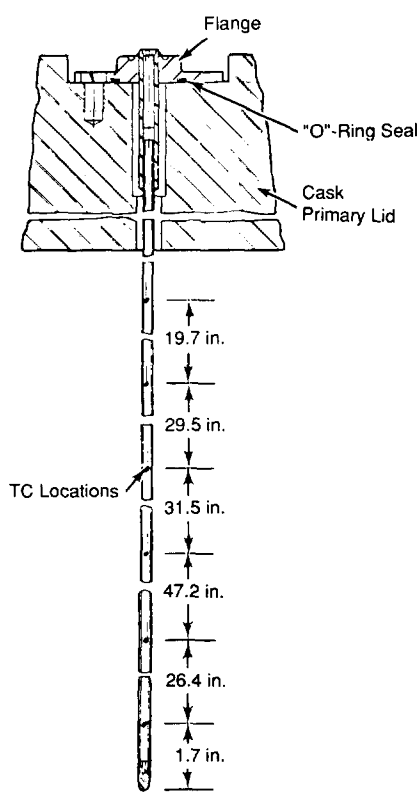

Figure 11. Thermocouple Lance[4,5,6,7]

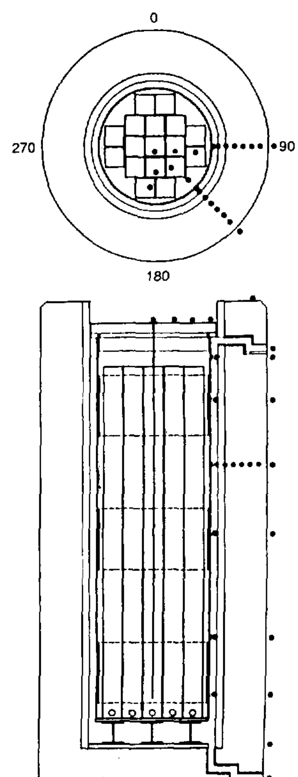

Figure 12. Selected VSC-17 Cask and Consolidated Spent Fuel Thermocouple Locations

INSTRUMENTATION

The VSC-17 cask will be instrumented with approximately 50 TCs and the consolidated spent fuel and basket will be instrumented with 42 TCs. The locations of the cask and spent fuel TCs are shown in Figure 12. Note that 10 TCs are to be imbedded in the concrete cask wall to detect short-and long-term changes in concrete thermal conductivity which will indicate changes in wall structure. Consolidation canisters will be instrumented as shown in Figure 11 and explained in detail in References 4, 5, 6, 7, and 27.

Gamma and neutron dose rates will be taken at approximately 70 locations, as indicated in Figure 13.

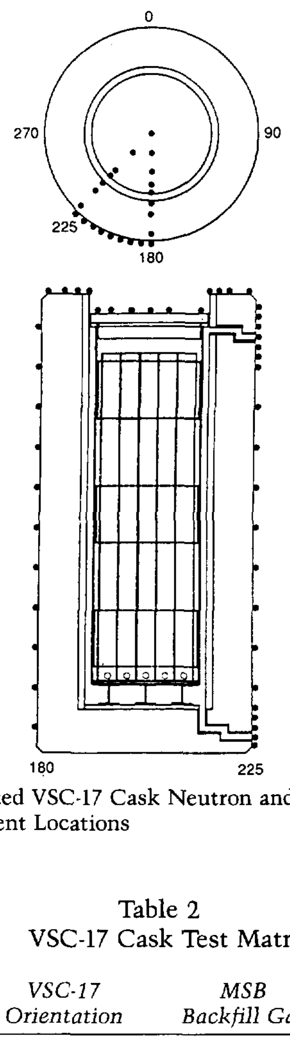

Figure 13. Selected VSC-17 Cask Neutron and Gamma Dose Rate Measurement Locations

Table 2
VSC-17 Cask Test Matrix

Run No.	*VSC-17 Orientation*	*MSB Backfill Gas*	*Inlet Vent Condition*
1	Vertical	Helium	Unplugged
2	Vertical	Helium	1/2 plugged
3	Vertical	Helium	All plugged
4	Vertical	Nitrogen	Unplugged
5	Vertical	Vacuum	Unplugged

TEST MATRIX

The test runs planned for the VSC-17 cask are indicated in Table 2. All cask test runs will be performed in a vertical orientation because a horizontal storage orientation is not practical for vertical concrete casks. The complexities associated with lifting lug/trunnion designs for horizontal handling/storing are the major reasons concrete casks are not normally oriented in a horizontal position.

The test run with a helium backfill gas in the MSB and unplugged inlet vents is prototypical of normal cask operation. Plugging the inlet vents simulates clogging due to blowing debris or shallow flooding. The nitrogen run is performed because nitrogen is also an acceptable storage gas.[28] A vacuum run will be performed to determine temperatures during vacuum drying of the MSB.

STATUS AND FUTURE ACTIVITIES

The VSC-17 cask design and analyses have been completed. The cask is to be fabricated in late spring and delivered to INEL in early summer of 1990. Testing is planned for late summer 1990 with data reduction, analysis, and reporting to follow.

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Cesar E. Collantes is program engineer at the Department of Energy, Richland Operations Office, responsible for the Commercial Spent Fuel Management Program, Concrete Cask Testing Program, and Systems Integration Program. Prior to his work at Richland, Collantes worked in the construction division of the Department of Energy, Las Vegas Operations Office. He also worked for Rockwell International, Canoga Park Office. Collantes earned a B.S. in Civil Engineering from the California State University of Los Angeles (1970) and M.S. in Organization Development from the Central Washington University (1985). He is a registered civil engineer in the state of Nevada.

The Ontario Hydro Dry Irradiated Fuel Storage Program and Concrete Integrated Container Demonstration

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ABSTRACT

For the first time anywhere the practicality of loading irradiated fuel into a concrete cask underwater in an existing pool facility has been successfully demonstrated. The cask holds about 7.7 metric-tons-uranium. Special design features allow the cask to be used for dry storage, for transportation, and for disposal without re-handling the fuel. The cask, called the Concrete Integrated Container, or CIC, has been developed by Ontario Hydro. These notes accompanied a talk describing the loading, monitoring, and IAEA-based transportation certification of testing of the CIC. The talk was given at the INMM Spent Fuel Management Seminar VII in Washington, D.C. Ontario Hydro now has a pool storage capacity of close to 32,000 metric-tons-uranium. The need for storage will more than double before 2025, the year a repository is expected to open in Canada. Some of the new capacity will be in CIC-style dry storage.

INTRODUCTION

Duck hunters tell us that the way to hit a target is to aim at where it is going to be, not where it has been. It's the same in the fuel storage business. We must aim our planning at where the moving targets are likely to be.

Like many of you here today, one of the moving targets we've been aiming for at Ontario Hydro is the future storage of irradiated fuel. The fuel arising from the sixteen operating reactors and the four we have under construction will almost equal that from all of the power reactors in the United States. Considering that a repository for this fuel won't be available until about the year 2025, you can see how we will need a major storage program to get there.

STORAGE NEEDS

Figure 1 illustrates the now typical hyperbolic production curve for irradiated fuel from our utility. Faced with this future need we investigated many wet and dry storage options. Currently, all of our fuel is stored underwater in nine pools with a combined capacity of about 32,000 metric-tons-uranium (MTU) at the reactor sites. So, with this investment in pools, why should we look at dry storage?

One reason is the shift in emphasis from short to long term storage. Most of the pools in our generating stations were sized to allow off-site shipping after five to ten years. This hasn't happened and is not likely to happen until a repository is opened. With this in mind, we looked at dry storage as a way to minimize the caretaking costs for the additional 42,000 MTU of used fuel to be produced. Assessments are continuing to determine whether we will adopt wet or dry storage systems or combinations of both until a repository is available.

Here are the options we have looked at:

- Concrete Integrated Container, CIC
- Convection Cooled Vault
- Cast Iron Casks
- Fixed Canister
- 1/2 Sized CIC
- CIC Underground

The first of these is the one which was chosen for demonstration.

CIC STORAGE DEMONSTRATION PROGRAM

The following figures show the first concrete integrated container CIC and the operations involved in transporting it and loading it with used fuel. Figure 2 shows one of two

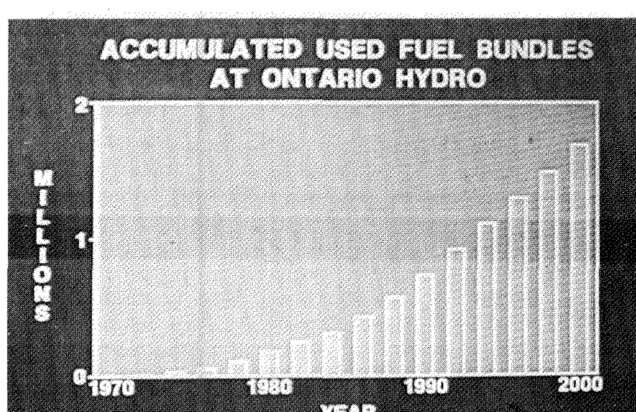

Figure 1. Accumulated used fuel bundles at Ontario Hydro

containers that were built and loaded during the past two years at our Pickering Generating Station. It is about 8 feet in diameter, 12 feet high, and weighs just under 80 tons. It holds 384 Candu fuel bundles containing 7.7 MTU.

The cut-away drawing shows the inner and outer liners, the fuel location, the bolted lid, and the drain port (Figure 3).

This type of construction gives us several advantages. To make the CIC "operator friendly," it has been designed to be loaded in the reactor pool using the existing equipment. No other concrete cask proposed so far has this feature. In keeping with our philosophy we also designed the CIC to be transportable and suitable for disposal.

As shown in the illustrations: The CIC was loaded onto a trailer and transported to the reactor pool transfer station (Figure 4). There the lid was removed and the cask transferred and lowered into the pool using the pool crane (Figure 5). Water was circulated to minimize contamination. The pool crane picked-up four fuel modules, each containing 96 fuel bundles, and lowered them into the container (Figure 6). The lid was installed on the CIC before the pool crane lifted it out of the pool. Water was sprayed over the container and the water inside of the CIC was drained (Figure 7). After draining, the cavity of the cask was vacuum-dried and helium leak-tested. The surface of the cask was inspected for loose contamination and the external radiation field was measured. On this first CIC, with 10-year cooled fuel, the maximum dose rate was 10 mrem on contact. The pool crane then transported the container and loaded it onto a trailer which moved it to the outside storage area (Figure 8). There a mobile crane placed the cask on a timber crib (Figure 9). After two years of storage, the CIC will be taken back into the storage pool and unloaded for examination.

The second similar CIC was loaded this fall with 6-year-cooled fuel.

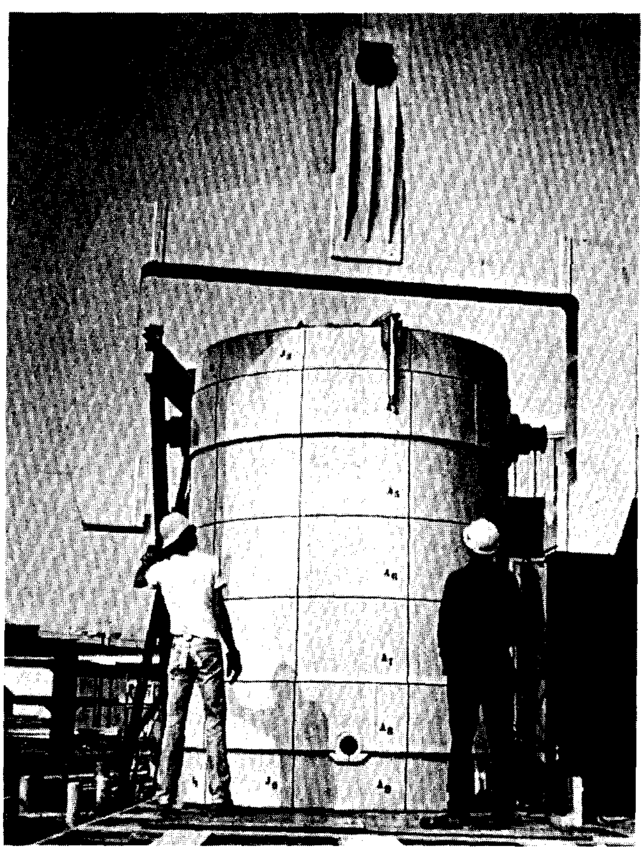

Figure 2. Concrete Integrated Container

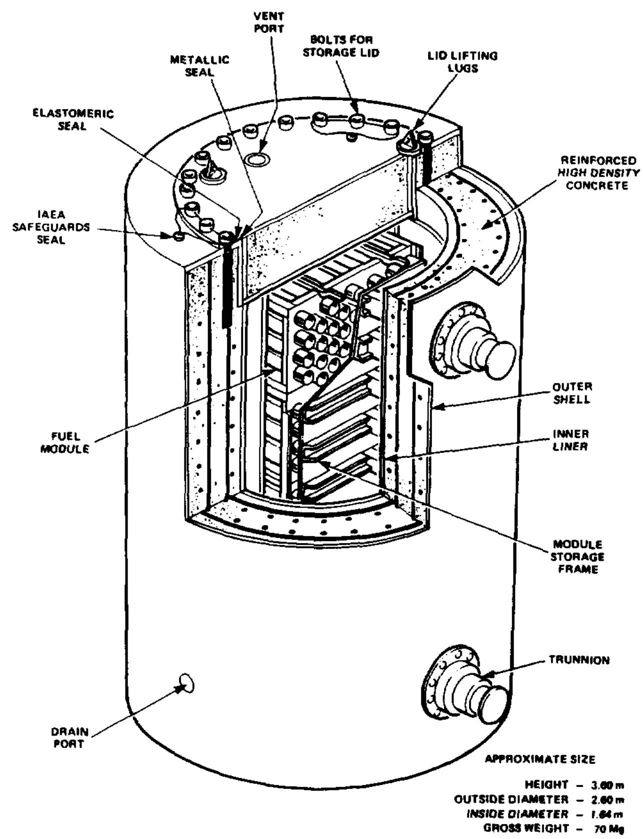

Figure 3. Cut-away drawing of the CIC showing design features

Figure 4. Transport of CIC to reactor building

CIC TRANSPORTATION DEMONSTRATION PROGRAM

In parallel with our storage demonstration program we are also funding a program to demonstrate compliance with transportation requirements. Analytical modelling and scale model drop tests are being conducted to confirm that the CIC will survive the accidents postulated by the IAEA transportation regulations familiar to you.

One of the predictions generated by the DYNA-3D program for a 29-foot drop of the CIC on a corner of the lid side is shown as an example (Figure 10). Also shown is a picture of the scale model of the CIC (with impact limiters) which was tested (Figure 11).

In addition to photographing these activities, they were recorded on video tape.

The authors gratefully acknowledge the many contributors within Ontario Hydro who made the dry storage system demonstration a success. This includes those in the engineering, construction, operations, and research divisions.

Information on this and related long-range programs of Ontario Hydro was presented at the two most recent INMM Annual Meetings: Frost, C.R., "Proposed Safeguards and Security During Future Decommissioning of Ontario Hydro's Reactors," *Journal of the INMM*, vol. XVII, proceedings issue, p. 351, 1988; Frost, C.R., "Safeguards and Security Requirements for a Canadian Conceptual Used Fuel Disposal Center," *Journal of the INMM*, vol. XVIII, proceedings issue, p. 559, 1989.

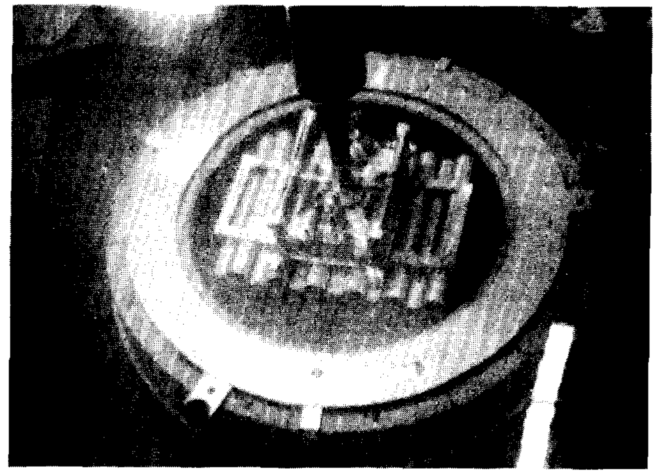

Figure 6. Loading spent fuel bundles into the CIC

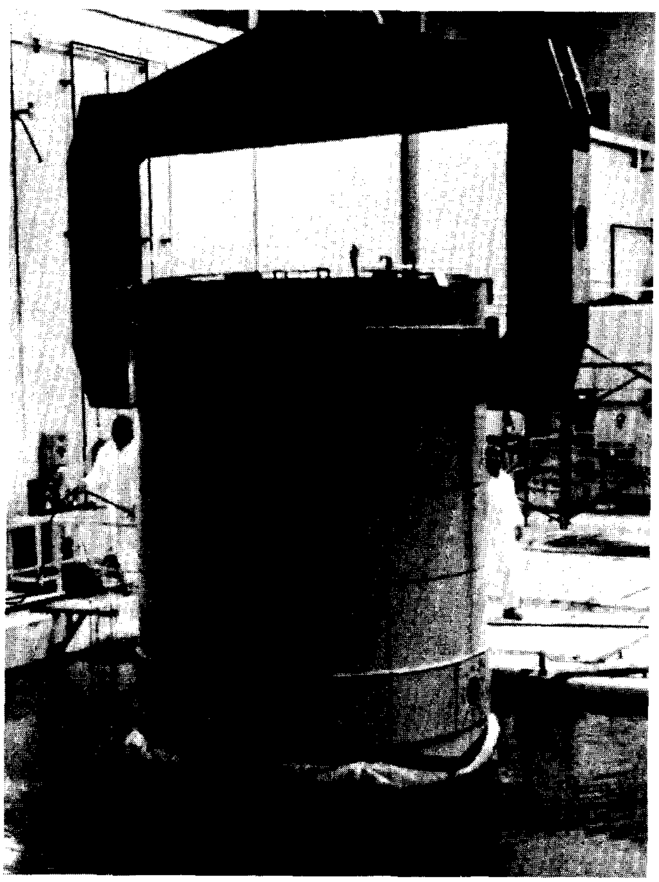

Figure 5. Lowering cask into the reactor pool

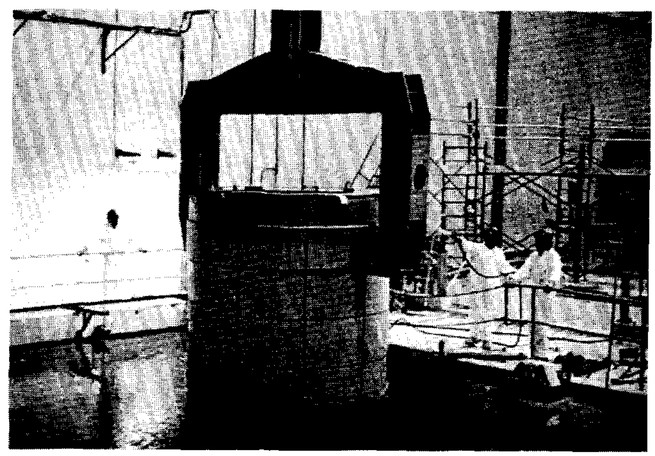

Figure 7. Raising CIC from the pool after installing the lid

Figure 8. Transporting from reactor to storage site

Phillip J. Armstrong earned a B.Sc. in Civil Engineering from the University of Leeds, England (1967). Since 1981 he has been working as supervising design engineer in the nuclear materials management department at Ontario Hydro where he is responsible for the concept and preliminary engineering of a tritium immobilization system. Armstrong was also responsible for supervising the completion of the engineering concept for a facility to volume reduce and store irradiated fuel channels as well as preparing nuclear station decommissioning plans and the development of reactor waste storage and disposal concepts. Prior to his employment at Ontario Hydro in 1971, he was a civil engineer at Falconbridge Nickel Mines Ltd. and a trainee engineer at the United Kingdom Department of Highways. Armstrong is a member of the Canadian Nuclear Society and became a registered professional engineer in the Province of Ontario in 1970. He is the author of "Evaluation of Immediate Dismantlement of CANDU Reactors at the End of Service Life" which was presented at the 1982 International Decommissioning Symposium.

Lou Grande earned a BASc in Mechanical Engineering, from the University of Toronto in 1972. Grande has worked at the Radioactive Materials Storage Unit of Ontario Hydro since 1975. He has been Design Engineer-Specialist since 1981, and is responsible for planning and coordinating work programs related to the disposal of low- and medium-level reactor waste. Prior to 1975 Grande served as design engineer and engineer in training in the Electrical Engineering in Department at the Steel Company of Canada Ltd. In recent years, Grande has worked on the development of improvements to the CIC making it wet-loadable and competitive with other fuel storage alternatives. A patent naming L. Grande and P.J. Armstrong as inventors, is in its final stages. The CIC concept provides protection in Canada and the United States. Grande is a member of the Canadian Nuclear Society and is a registered professional engineer in the province of Ontario since 1974.

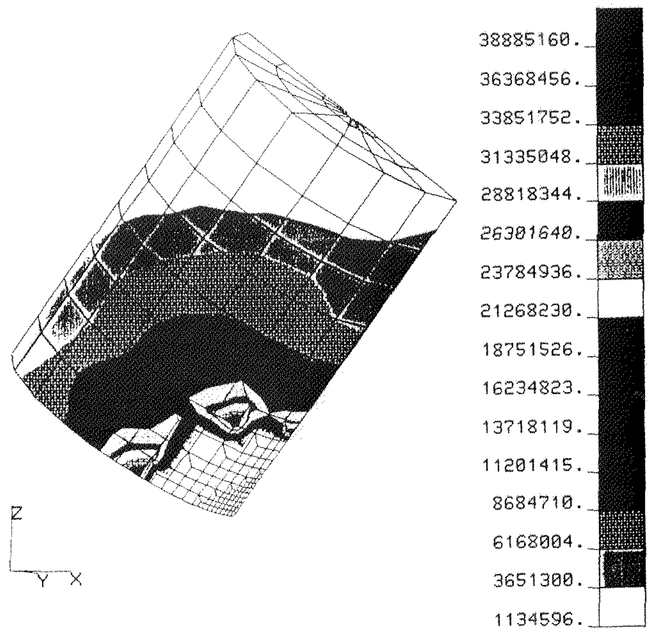

Figure 10. DYNA-3D Program Analysis of 29-foot corner drop of CIC without impact limiters

Figure 9. Placing the loaded CIC on the timber crib for storage

Figure 11. Scale model of CIC with impact limiters before drop test

Application of Modular Vault Dry Storage to Public Service of Colorado—Fort St. Vrain

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ABSTRACT

In March of 1988 NRC gave Topical Report approval for the first dry vault storage system to be submitted for non-site specific design approval. This paper describes the first site specific application of the FW/GEC Modular Vault Dry Store (MVDS) for the Public Service of Colorado—Fort St. Vrain site.

The decision to permanently shut down the Fort St. Vrain high temperature gas-cooled reactor in August 1989 provides the need to store on site 1,482 irradiated fuel blocks and a quantity of other reactor core components categorised as greater than Class C wastes. The MVDS system has been selected for the safe storage of these items in a facility designed to receive, store and discharge fuel to the repository totally independent from the original reactor facilities.

1. INTRODUCTION

In August 1989 the Public Service Company of Colorado (PSC) board of directors voted to shut down the Fort St. Vrain High Temperature Gas Cooled Reactor (FSV-HTGR). The need to defuel the reactor following closure of the plant was anticipated and extensive negotiations to transfer the fuel inventory to the DOE Idaho Falls Plant were undertaken, and are still ongoing. Prudency dictated a fallback position in the event these fuel movements were not possible. Therefore PSC have contracted with Foster Wheeler Energy Applications Inc. (FWEA) to design, construct and commission a Modular Vault Dry Store (MVDS) for storage of the complete FSV fuel inventory. The new storage facility will be located adjacent to the reactor site.

Application of vault dry storage options in the USA has previously been limited to storage of experimental, military and HTGR fuels largely on DOE sites. In Europe and particularly in the United Kingdom, vault type dry storage systems have accumulated commercial scale operating experience since 1971[1] and several major future applications are either in construction or in the advanced planning stages.[2,3] The experience, gained using vault dry stores, has been totally successful and the concept has now also been applied to the storage of high level vitrified blocks generated from the European reprocessing industry.

The dry vault storage system experience has been applied to the USA Independent Spent Fuel Storage Installation (ISFSI) market by FWEA in collaboration with the Anglo-French Company of GEC ALSTHOM Engineering Systems Ltd (GEC-A). The Modular Vault Dry Store (MVDS) concept has been submitted to the US NRC as a generic Topical Report for irradiated LWR fuels and was approved in March 1988.

This generic design embraced wide range of site and fuel assembly parameters, sufficient to fully develop the unique features of the FSV fuel. Flexibility of the MVDS concept also allows PSC to store reactor core components classified as greater than Class C waste. Independence of operation of the MVDS concept will allow storage and eventual shipment to the MRS or Repository of the stored fuel and waste even after decommissioning of the reactor site.

2. FORT ST. VRAIN FUEL & STORAGE REQUIREMENTS

The FSV-HTGR building is illustrated in Figures 1 & 2. The reactor core comprises an assembly of hexagonal graphite blocks. The graphite fuel blocks contain Fuel Rods in closed ended channels. Coolant channels pass through the fuel blocks allowing the vertical flow of helium reactor coolant as illustrated in Figure 3. The Fuel Rods are a blend of coated fuel particles and a coke filler. The coated particle fuel, called the TRISO type, is a dominant feature of the fuel element. Two types of TRISO particles are used in the FSV reactor. Fissile particles contain a mixture of thorium and uranium carbide in the kernel, while larger fertile particles contain only thorium carbide. The coatings surrounding the kernel are designed to retain the fission products over the life time of the fuel. Operating experience with the FSV fuel indicates that the fraction of fuel particles with failed coatings is less than 1%.

The temperature limits of the fuel block materials are very high (average gas outlet temperature is typically 1440°F). In an oxidising environment (air) the safe storage temperature limit of 750°F is established by the onset of graphite oxidation.

The core is surrounded by graphite side reflector blocks of various shapes, the majority of which will be disposed of during decomissioning as Class C waste. A number of metal clad reflector blocks from the top layer of the core are classified as greater than Class C waste because of the irradiated stainless steel components included in these assemblies.

The FSV ISFSI has a requirement to store 1,482 fuel blocks and the 37 metal clad reflector blocks. In addition 6 (six) fuel blocks each containing a Californium neutron source have to be stored. The neutron sources increase the normal fuel block neutron source term very considerably.

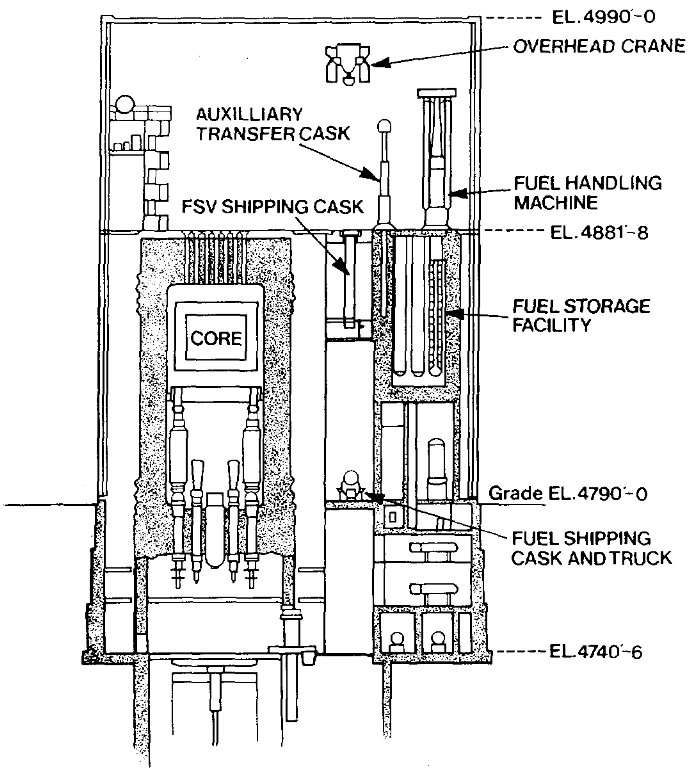

Figure 1. Reactor building cross section

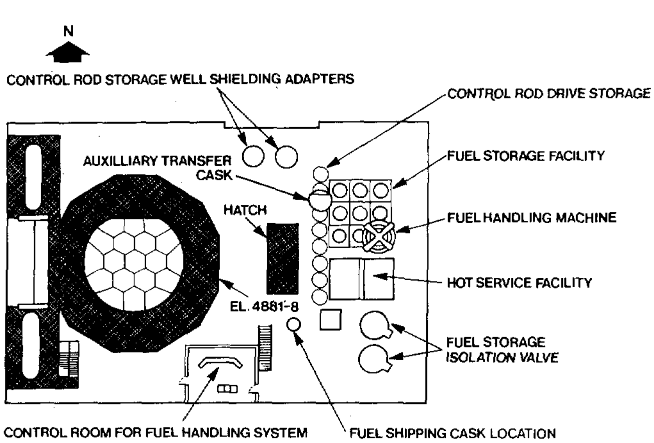

Figure 2. Reactor refuelling floor

The site specific storage parameters for FSV fuel are summarised in Table 1 and compared with the parameters used in the MVDS Topical Safety Analysis Report (TSAR). The comparison shows that the TSAR bounds the important design parameters except the neutron flux level associated with the neutron source fuel blocks.

3. FORT ST. VRAIN FUEL HANDLING ROUTE

FSV refuelling operations utilise an Overhead Crane and a Fuel Handling Machine, operated in conjunction with Isolation Valves and an Auxiliary Transfer Cask to transfer irradiated fuel blocks to a helium gas cooled Fuel Storage Facility integral with the reactor building (see Figures 1 & 2). After a minimum decay cooling period of 100 days the fuel blocks can be transferred using the refuelling equipment to a Fuel Shipping Cask unique to the FSV facility. This operation is carried out at the reactor charge face level (elevation 4881'-8") by raising the Shipping Cask to that level using the building Overhead Crane. With the cask positioned in the load/unload position the Fuel Handling Machine can load fuel blocks directly into the Shipping Cask cavity. The Isolation Valve is a sliding gate valve used to provide a sealing function for the contained helium, and a shielding interface between the machine and the cask.

All charge face equipment is positioned using the Overhead Crane.

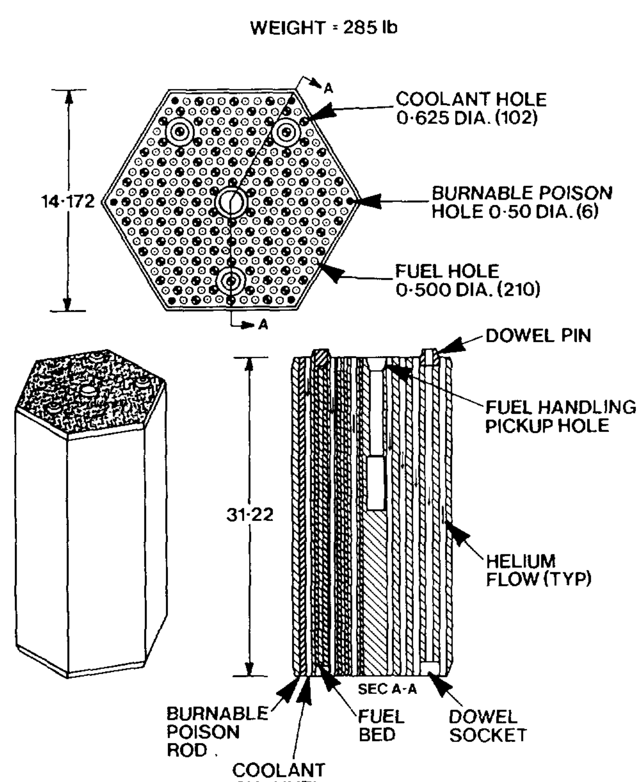

Figure 3. Fuel element

The FSV Shipping Cask, Figure 4, holds six fuel blocks, stacked one upon another, within an Inner Container designed to provide a containment for the fuel in addition to that offered by the cask body and closure. The Inner Container is closed by a bolted lid containing depleted uranium shielding material. The lid utilises elastomeric seals for the relatively short transit time between FSV and the Idaho Falls site. The Idaho site hosts the Irradiated Fuels Storage Facility[4] originally designed to receive the Shipping Cask and to store the FSV fuel (with other fuels) in a forced convection, air cooled vault dry storage facility. The storage environment for the fuel in the facility is atmospheric air. The Shipping Cask Inner Container is not used to contain the fuel during storage at Idaho but is returned to FSV with the Shipping Cask.

In earlier years of FSV operation approximately 500 fuel blocks have been successfully shipped to Idaho and are in storage. Therefore PSC reactor operators are very familiar with the fuel handling route equipment and operational procedures. This experience will be extended in the forthcoming reactor defuelling operations. The ISFSI selected by PSC for FSV will fully utilise the existing equipment and procedures without significant modification.

4. THE MVDS CONCEPT

The MVDS concept described by the Topical Report has been adapted in certain detail features to suit the storage needs of FSV. To allow comparison the basic concept as reviewed by the US NRC is briefly described.

The MVDS design holds irradiated LWR fuel in individual vertical storage tubes retained within a concrete structure forming the storage vault. The vault provides the necessary shielding for the stored fuel and integral air flow ducts for a natural thermal buoyancy cooling system. The warmed air within the vault moves upwards and passes to the environment via the outlet ducts. Cooler air is drawn into the vault and over the outside of the storage tubes to maintain an air flow consistent with the heat load (number of fuel assemblies) within the vault. A typical vault system and enclosing weatherproof building is illustrated in Figure 5.

The storage tube provides the containment boundary for each fuel assembly. A removable shield plug is used to close the top of each store tube allowing fuel to easily be loaded, accessed or removed. The storage tube can be reused if desired. One diameter and length of storage tube provides a universal containment for all unconsolidated or consolidated PWR fuels. A second smaller diameter provides a universal containment for all unconsolidated or consolidated BWR fuels. The MVDS design provides fully shielded access to the top of each storage tube at all times. Accessibility of the individual fuel containments allows the contained gas to be simply and safely monitored or changed if required.

The number of storage tubes (number of fuel assemblies) installed within a vault can be selected at the site specific design stage to suit the Utility storage need. Typically for at-reactor site applications a vault will be selected to hold approximately 100 or alternatively 200 PWR storage locations or significantly more for the smaller BWR assembly.

Uncontainerised fuel is moved to the MVDS installation using any dry transport cask type or capacity to suit the existing reactor building craneage. A special bottom loading cask system (the Fuel Handling Machine) operates above the storage vaults on a seismically designed gantry system. The machine is designed to raise individual fuel assemblies from the transport cask, transfer to the storage position and lower into the selected storage tube. The fuel handling route can be reversed for ultimate transfer of fuel from the store to the MRS or Repository using the transport cask or a DOE supplied shipping cask.

Table 1
Design Parameters

Parameter	FSV	TSAR
Heat Load/Storage Location	0.7 kw (average)	1kw
Decay Period	400 day	5 year
Fuel Source — Normal Assembly		
Gamma/Storage Location	1.52×10^{15} MeV/sec	9.32×10^{15} MeV/sec
Neutron/Storage Location	2.08×10^{6} N/sec	5.17×10^{8} N/sec
Fuel Source — Neutron Source Assembly		
Gamma/Storage Location	1.52×10^{15} MeV/sec	
Neutron/Storage Location	5.61×10^{10} N/sec	
Ambient Temperatures	-32°F to 102°F	-20° to 100°F
Flood Levels	6ft	Site specific
Seismic		
Ground Acceleration	0.1	0.25
Spectrum	Reg. Guide 1.61	Reg. Guide 1.61
Tornado Missile	Site specific	NUREG 0800 II
Tornado Maximum Velocity	300 mph	Reg. Guide 1.76 (360 mph)
Snow Loading	30 psf	100 psf

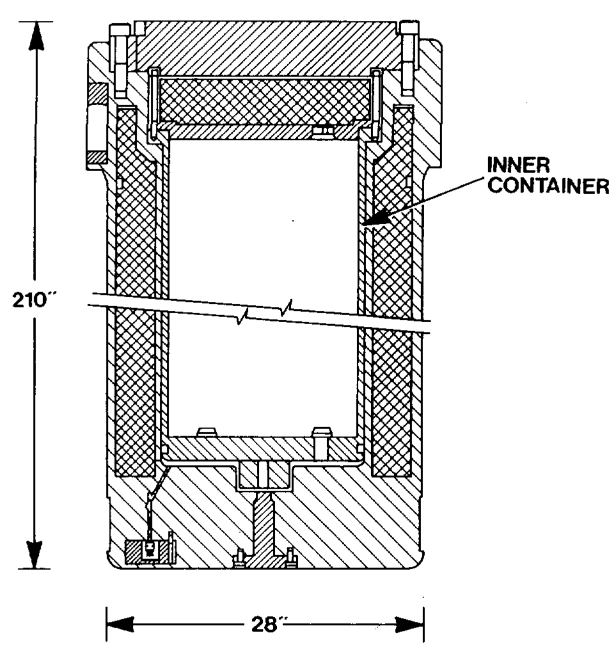

Figure 4. Fort St. Vrain HTGR spent fuel shipping cask

5. MVDS FOR FORT ST. VRAIN— SPECIAL REQUIREMENTS

Application at the MVDS design to FSV required adaptation of certain features to meet the storage need specified by PSC, at the lowest capital cost. The factors influencing these changes were as follows:

5.1 HTR Fuel & Reflector Blocks

The nature of the HTR fuel previously described is significantly different from the LWR types originally considered in the TSAR. Despite the disparity of material, form and enrichment to LWR fuel, the HTR fuel parameters have the least influence on the MVDS design given the decision to containerise the fuel blocks at the reactor (see 5.2). The reflector blocks are also easily accommodated because they have the same cross section as fuel blocks, same handling features and half the physical length. Two reflector blocks stacked are therefore dimensionally equal to one fuel block.

5.2 Fuel Containerisation at Reactor

As described, the standard fuel route equipment at FSV results in six fuel blocks being loaded into the Inner Container of the existing and licensed FSV Shipping Cask. The Inner Container dimensions are not dissimilar to the standard TSAR fuel storage tube for PWR fuel and also provides a high integrity sealed containment boundary with a shielded closure. These similarities and the practical desire to utilise the existing and proven reactor building refuelling route resulted in the decision to containerise the fuel blocks at the reactor by placing them into a containment boundary very similar to the existing Inner Container. The proposed use of the existing Shipping Casks for site transfer of fuel to the ISFSI and potentially for subsequent movement of fuel to MRS or the Repository, completed the concept modification logic.

Utilisation of the containerisation procedures at the reactor building has the added advantage of providing an uncontained fuel route to the ISFSI and for all normal

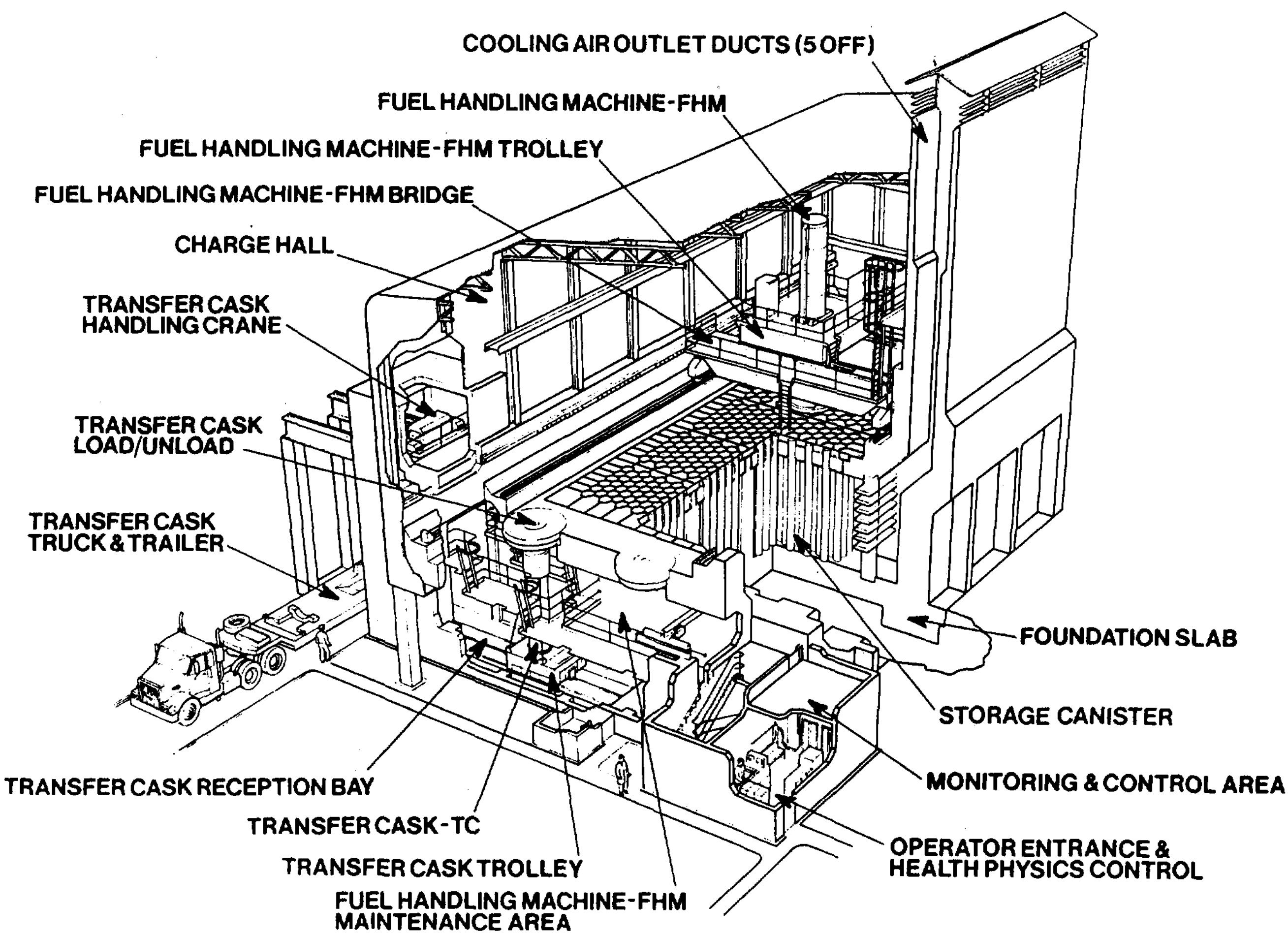

Figure 5. Modular Vault Dry Store—MVDS typical 200 MTU PWR spent fuel facility

operations within the MVDS, thus eliminating the need for contamination control and monitoring systems that were incorporated in the TSAR design.

5.3 Storage Capacity

The minimum specified PSC storage need was as little as 25 fuel blocks or total storage of only 45 Inner Containers. The option to increase the storage capacity to 1,482 fuel blocks (i.e. six fold) only arose post contract, therefore the competitiveness of the concept at the minimum storage capacity had to be addressed. This required a review and simplification of detail features not important to safety but with a significant capital cost implication.

5.4 Total Independence from On-site Reactor

The early decommissioning of the FSV nuclear reactor system and possible repower of the present turbine generator with replacement steam supplies, as considered by PSC, would leave the MVDS totally independent of an adjacent 10 CFR Part 50 licensed nuclear site. This possibility was a major consideration of PSC in selecting an ISFSI system.

The 10 CFR Part 72 license requirements for an ISFSI requires independence from reactor site services. However, many of the available storage systems suitable for ISFSI application vary in the level of independence from reactor facilities in the event of unforeseen fuel confinement system failures and for the eventual shipment of fuel to the MRS or Repository. All storage systems can overcome these deficiencies by the licensing and construction of additional facilities to allow fuel transfer to a new confinement boundary or to an off-site shipping cask. A claim of the MVDS concept is that these provisions are fundamentally incorporated into the design without significant additional cost. This was satisfactorily demonstrated to PSC and the method by which this is possible using a minor addition to the MVDS concept will be clear from the following description of the MVDS design for FSV.

6. MVDS FOR FORT ST. VRAIN—DESCRIPTION

The MVDS facility arrangement is shown in Figures 6 and 7. The Fort St. Vrain MVDS comprises six vault modules each containing forty-five individual storage containers.

6.1 Principles of Operation

The fuel blocks or other items to be stored are loaded into Fuel Storage Containers (FSCs) at the reactor building using the existing fuel handling equipment. The FSCs are transported from the reactor building to the MVDS building in the FSV Shipping Cask..In the Cask Reception Bay, the Shipping Cask is lifted from the site trailer and parked in a Cask Load/Unload Port using the MVDS overhead crane. In the Load/Unload Port the FSV Shipping Cask closure lid can be removed and parked. Use of a Fuel Handling Machine (FHM) and an independent shielding isolating valve mounted above the Shipping Cask allows the FSC to be safely raised into the FHM. The building crane moves the FHM to the selected storage location which has been previously prepared by removal of a shield plug and installation of a second shielding isolating valve. The FHM lowers the FSC into the vault storage position, the FHM is removed, the shield plug replaced and the

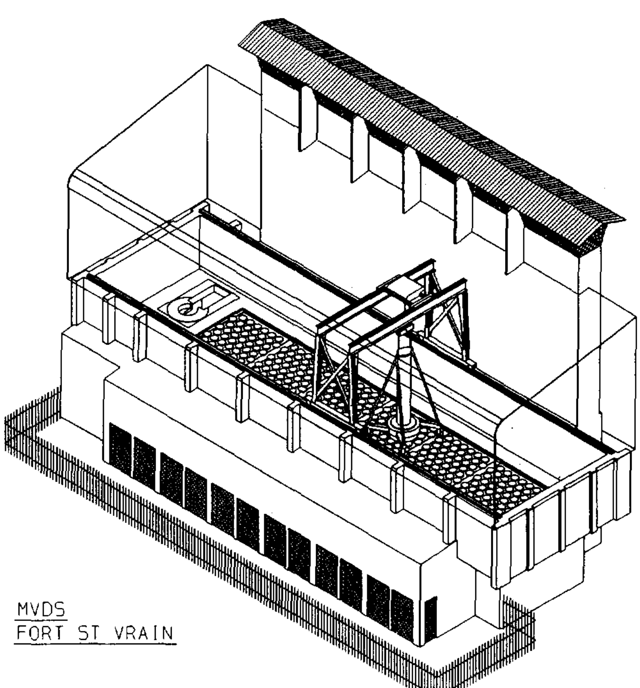

Figure 6. Cutaway view of modular storage vault for Fort St. 'rain

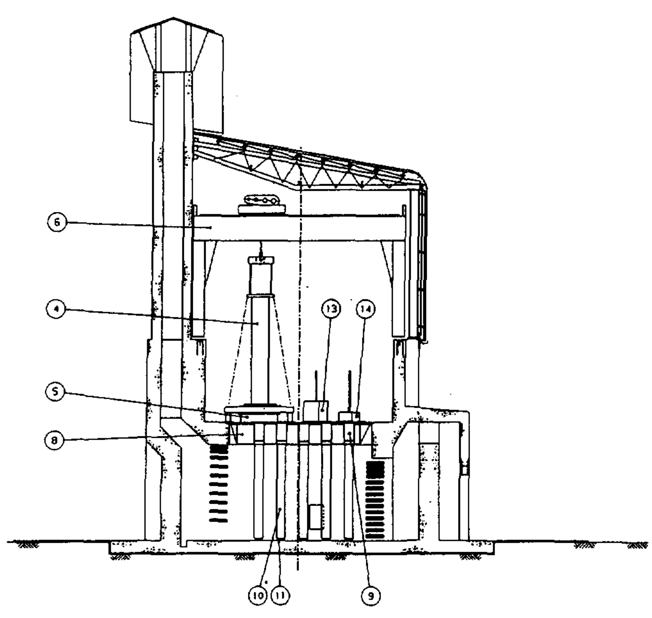

Figure 7. End of Fort St. Vrain storage vault: (4) Container handling machine; (5) Isolation valve; (6) Crane; (8) Charge face structure; (9) Shield plug; (10) Fuel storage container; (11) Container support; (13) Shield plug handling device; (14) Uranium shield handling device

valve moved to the next storage location to be filled. From this position the equipment is used to remove an empty FSC from the vault and transfer this to the FSV Shipping Cask in the Load/Unload Port.

Return of the FSV Shipping Cask to the reactor building with the empty FSC allows the cycle to be repeated. Two FSV Shipping Casks are available, therefore loading/unloading will normally be conducted simultaneously at the reactor building and the MVDS building.

6.2 Cask Reception Bay

The bay is an enclosed reception area which is used for the receipt of the Shipping Cask trailer and for transferring the FSC between the Shipping Cask and the FHM. The bay location is indicated in the illustrating figures. The bay has a concrete floor, walls and ceiling with an access roller shutter door at one end for the cask trailer to be backed into. A specially shaped aperture in the ceiling allows the Shipping Cask to be raised to the vertical and parked in the Load/Unload Port using the building crane. The arrangement limits the lift height of the Cask above the floor of the Bay. A Shipping Cask lower restraint is used to stabilise the Cask whilst in the Load/Unload Port in the event of a seismic disturbance.

6.3 Storage Vault

The storage vault is shown as part of the building section view in Figure 7. The vault has a concrete floor and walls with inlet and outlet air cooling ducts. The roof of the vault is a composite steel and concrete structure called the Charge Face Structure which has holes to receive the FSCs. The Charge Face Structure combined with the fixed support stools in the vault floor maintain the fixed array of FSCs optimised for sub-criticality and efficient cooling.

Cooling air enters through a protective bird mesh, flows through a labyrinth which prevents radiation streaming and enters the vault through the distribution louvres. The heated air from the vault flows through the exit louvres, up the concrete duct and exits to atmosphere via the bird mesh and weather protected duct top.

The cooling air only contacts the outer surface of the FSCs and does not become contaminated. The cooling system is passive, self regulatory and driven by the buoyancy of warm air. The cooling system maintains a wide margin on acceptable fuel and structural concrete temperatures for all site specific environmental conditions. The cooling system is insensitive to partial blockage of the inlet or outlet ducts and the design provides several days to recover from the very unlikely event of total blockage.

The massive concrete structure forming the vault and the cooling ducts prevents radiation from exceeding acceptable dose levels outside the MVDS building.

The triangular spacing of the FSCs within the vault is such that the stored fuel cannot become critical even in the unlikely event of the vault becoming fully flooded. No credit for burn-up is necessary to guarantee acceptable values for K effective in the MVDS design.

The civil structure is designed to the ACI 349-85N standard and constructed to ACI 318-83 requirements.

6.4 Charge Hall

The Charge Hall is the volume above the Charge Face Structure and the Cask Reception Bay providing a weatherproof enclosure for the loading/unloading operations. PSC have specified that the Charge Hall is provided with limited space heating and simple ventilation equipment to provide operator comfort during extremes of weather conditions.

The concrete structure extends up to the 30 ft elevation for protection of the Charge Hall equipment from the heavier regulatory tornado missiles. The balance of the enclosure is a steel frame structure, clad with insulated steel sheeting.

The building crane operates along the length of the Charge Hall and provides all lifting operations for equipment handling. Design limits and operational procedures control the potential drop heights of the crane load onto the Charge Face Structure and into the Cask Reception Bay. No bounding impact fault case will result in a release of radioactivity.

6.5 Storage Wells

Three supplementary storage locations are provided at the end of the building. The Storage Wells provide a sealable, shielded and natural thermosyphon cooled containment for a defective FSC in the unlikely event of a failure in service. They can be used to isolate a defective FSC after transfer from the vault cooling system, to allow individual leak testing of a suspect FSC, to act as a temporary storage location whilst changing fuel blocks from a defec-

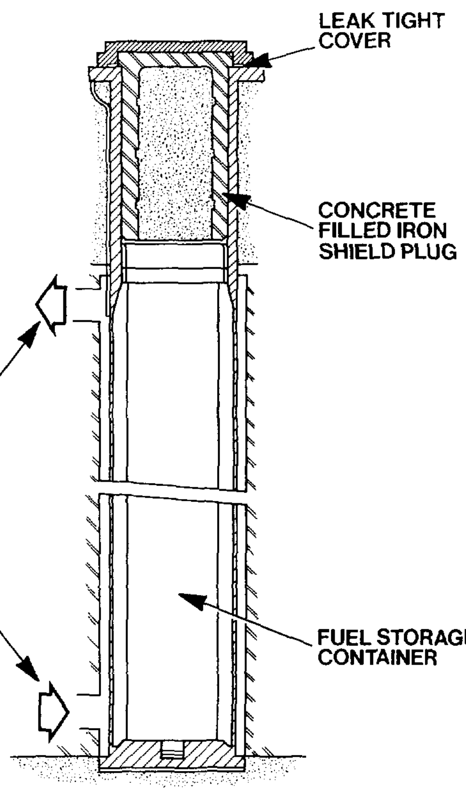

Figure 8. Storage well

tive storage cantainer to a new container or as part of the transfer process of fuel blocks to a DOE shipping cask. The Storage Wells provide the MVDS with an additional degree of independence from supporting site facilities during off normal events and during future MVDS defuelling operations. See Figure 8.

A supplementary function of a Storage Well is to house the single FSC holding the six neutron source containing fuel blocks. This storage position, independent from the vaults allows additional neutron shield material to be incorporated into the charge face shield plug.

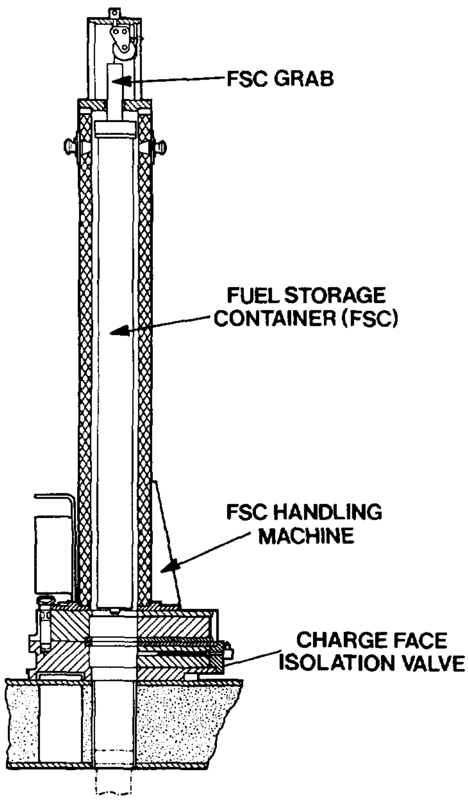

Figure 9. FSC loading equipment

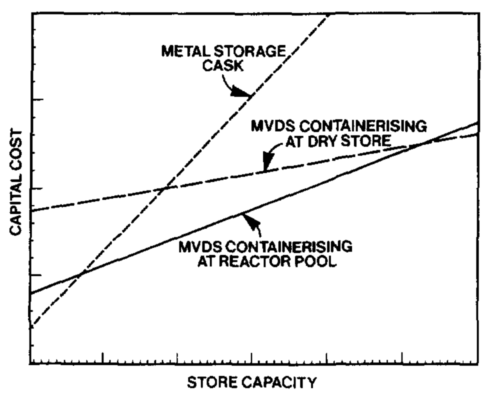

Figure 10. Modular vault and metal cask dry storage capital cost vs. capacity

6.6 Fuel Storage Container (FSC)

The FSC replaces the Inner Container of the existing FSV Shipping Cask for fuel transfers to the MVDS and provides fuel confinement whilst in storage. The FSC could also be used with the FSV Shipping Cask for the future fuel movements to the MRS or Repository.

Carbon steel is used for the FSC and it is designed to the code requirements of ASME 111—Class ND. The exterior surfaces are protected from atmospheric corrosion by a flame sprayed aluminium coating. The bolted lid to the FSC contains a removable depleted uranium insert used during lid bolting and leak checking procedures at the reactor building. Twin metal 'O' ring seals provide a high integrity seal between the container body and lid. The lid design incorporates features to allow seal interspace and container body leak checking. A spigot boss on the base of the FSC serves to positively locate the FSC in the floor of the storage vault. The safety analysis for the FSV-MVDS will assume that the storage environment within the FSC is atmospheric air.

6.7 Fuel Handling Machine (FHM) & Associated Equipment

The FHM provides the means of raising/lowering a FSC either from the FSV Shipping Cask or the vault storage location. The FSC is fully shielded by the FHM during handling operations and the decay heat dissipated from the exterior surfaces of the machine. The FHM is moved by the building crane using a special lifting yoke designed to limit the maximum possible lift height of the machine above the charge face level. A sliding gate valve at the base of the machine provides necessary gamma shielding. The valve cannot be operated unless it is coupled to the similar isolation valve on the Cask Load/Unload Port or the valve on the Charge Face. The base of the FHM has four outriggers designed to prevent toppling of the machine in the event of a drop incident. A hoist and FSC grab system are designed to meet single failure proof standards for nuclear lifting applications. They provide the vertical motion for FSC transfer operations. The FHM is illustrated in Figure 9 positioned on the Charge Face isolating valve.

The FHM can be adapted on a non-routine basis to safely handle single FSV fuel blocks.

7. FORT ST. VRAIN LICENSING & CONSTRUCTION SCHEDULE

Design and licensing of the FSV MVDS commenced in October 1989. The schedule requires a start of site construction by February 1991 and commencement of fuel loading in December 1991. It is planned to complete the defuelling of FSV by August 1992.

8. FUTURE MVDS APPLICATIONS

The MVDS concept for LWR fuels submitted and approved by US NRC in 1988 was based upon the premise that containerisation of the irradiated fuel at the ISFSI would be advantageous to a Utility because it provided the option of three unique features:

a. Ability to have rapid direct access to stored fuel for inspection and for nuclear safeguard procedures.

b. Ability to discharge uncontainerised fuel to the standard DOE Shipping Cask at the ISFSI without licensing and construction of additional facilities.
c. Minimal modification of the reactor fuel building craneage.

Handling of uncontainerised fuel at the ISFSI does however have a capital cost impact because of the need, albeit limited, for control of radioactive particulate arising from the fuel assembly surfaces (crud). Utility selection of the competing ISFSI designs is strongly affected by the capital cost of the installation. The capital cost of an ISFSI is a function of the storage capacity and typical relationships are illustrated in Figure 10. For MVDS the initial capital cost for zero storage is relatively high compared with, say, metal storage casks but the incremental cost (cost per additional unit stored) is low. This is clearly illustrated in Figure 10.

The opportunity to consider MVDS for FSV where containerisation at the reactor was obviously most advantageous has resulted in an improvement in the capital cost relationship by lowering the cost level with only a modest increase in the incremental rate. This charge has therefore a beneficial effect on the competitiveness of the MVDS concept where a Utility requires to purchase limited storage capacity.

FSV has a dry fuel handling route (no storage pond) and the fuel is a HTGR type assembly, however the MVDS design for this application can be applied to the LWR fuel storage needs of the Utility industry. The fuel storage containers, each containing one or more LWR fuel assemblies, can be sealed into a special transfer cask designed for site fuel movement to the MVDS. The transfer cask complete with empty storage containers can be loaded in the reactor pool and fuel loaded using the existing fuel handling equipment. After insertion of a shield plug in the top of each container the loaded transfer cask is removed from the pool, drained, the internals dried and the storage container sealing covers fitted. Reassembly of the transfer cask outer closure and inspection allows transfer of the cask to the MVDS. At the MVDS the storage containers are removed using a fuel handling machine and loaded into the storage vault.

Design and development of this type of transfer cask and MVDS storage container for LWR fuels is now being carried out by FWEA/GEC-A. This MVDS variant will shortly be available for Utility application, giving further advantages to this most flexible of the storage systems.

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Experience With Failed or Damaged Spent Fuel and Its Impacts on Handling

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ABSTRACT

Spent fuel management planning needs to include consideration of failed or damaged spent light-water reactor fuel. Described in this paper are the following: the importance of fuel integrity and the behavior of failed fuel, the quantity and burnup of failed or damaged fuel in storage, types of defects, difficulties in evaluating data on failed or damaged fuel, experience with wet storage, experience with dry storage, handling of failed or damaged fuel, transporting of fuel, and experience with higher burnup fuel.

Spent fuel management planning needs to include consideration of failed or damaged spent light-water reactor (LWR)** fuel. Described in this paper, which was prepared under the Commercial Spent Fuel Management (CSFM) Program that is sponsored by the U.S. Department of Energy (DOE), are the following: the importance of fuel integrity and the behavior of failed fuel, the quantity and burnup of failed or damaged fuel in storage, types of defects, difficulties in evaluating data on failed or damaged fuel, experience with wet storage, experience with dry storage, handling of failed or damaged fuel, transporting of fuel, experience with higher burnup fuel, and conclusions.

IMPORTANCE OF FUEL INTEGRITY AND THE BEHAVIOR OF FAILED FUEL

Release of fission gases and particulates of fuel from irradiated fuel rods is prevented by the primary barrier, the fuel cladding. Hence, consideration of the integrity of the cladding on spent fuel is an important factor in spent fuel management planning. Regulation of spent fuel storage at facilities away from reactors is addressed in 10 CFR 72,† which requires that gross degradation of spent fuel and release of radioactive particulates be prevented. Inspection techniques are available to detect most fuel assemblies containing reactor-induced defects, although elimination of every failed or nearly failed fuel rod cannot be assured. As a result, the condition of spent LWR fuel is an important aspect to be considered in planning activities such as storing, handling (including examination, fuel assembly reconstitution, and rod consolidation), and shipping such fuel.

QUANTITY AND BURNUP OF FAILED AND DAMAGED FUEL IN STORAGE

At the end of 1988, the inventory of spent fuel in storage included over 62,700 LWR fuel assemblies (59% BWR and 41% PWR type).[1] Nearly all of the fuel assemblies are stored at reactor sites; only about 5% are stored at other facilities such as Independent Spent Fuel Storage Installations (ISFSIs). Most (>97%) of the fuel assemblies contain Zircaloy-clad fuel rods; <3% contain stainless-steel-clad fuel rods. The utilities indicate that among the >62,700 fuel assemblies, there are >3,200 failed or damaged fuel assemblies. It should be noted that at the end of 1986, the utilities had reported that there were approximately 5,000 failed or damaged spent fuel assemblies in storage;[2] however, the next year the utilities significantly reduced their estimates and indicated that there were nearly 2,400 failed or damaged fuel assemblies in storage at the end of 1987. The burnup distribution of the >3,200 failed or damaged fuel assemblies is shown below:

Burnup Range, MWd/MTU
0 to ≤5,000
>5,000 to ≤10,000
>10,000 to ≤15,000
>15,000 to ≤20,000
>20,000 to ≤25,000
>25,000 to ≤30,000
>30,000 to ≤35,000
>35,000 to ≤40,000
>40,000 to ≤45,000
>45,000 to ≤50,000
>50,000 to ≤55,000

Number of Failed or Damaged Fuel Assemblies	
BWR	*PWR*
125	0
592	23
971	46
552	95
334	75
129	102
40	94
0	40
0	4
0	0
0	0
2,743	479

Most of the failed and damaged fuel is old; over 83% of it was discharged between 1969 and 1980. Of the failed and damaged fuel, about 11% of it from BWRs and about 53% of it from PWRs has been discharged and placed in storage between 1981 and 1988.

TYPES OF DEFECTS

Most of the cladding defects are small, and few are large. The utilities indicated in 1987 that of the failed or damaged fuel assemblies, only 35 required special handling and only one required encapsulation.

Failure types on fuel rods can vary over a wide range: from pinholes, small holes, and cracks to large defect areas (rare) in which fuel pellets are visually observable or are missing. Only a small fraction of the fuel rod cladding breaches are readily visible when fuel assemblies are visually inspected because the defects are generally too small for easy detection or they are located on interior rods. If breached fuel rods are allowed to continue operation in LWRs, they can sustain substantial secondary hydriding that can dominate the visual appearance. Such hydriding has led (especially in the early days) to misinterpretation of the primary cause of failure. Longitudinal splitting of the cladding, if noted under normal operating conditions, is now typically viewed as a secondary defect.

DIFFICULTIES IN EVALUATING DATA ON FAILED AND DAMAGED FUEL

It is difficult to evaluate data on failed and damaged fuel reported by industry for several reasons:

1. the definition of abnormal degradation is not uniform throughout the industry;
2. in many cases the number of fuel failures is inferred from indirect evidence;
3. in some cases only directly observed failures are counted;
4. in other cases, a group of fuel assemblies suspected or known to contain failed fuel rods is discharged but not inspected; and
5. whether the fuel is classed as failed can depend on how closely the fuel is inspected and on the capability of the inspection technique being used (in the case of leak testing, also called sipping, it can depend on when it takes place).[2]

Development and adoption of more uniform definitions of failed fuel and abnormal degradation would be helpful because it would make it easier to more accurately compare data from different reactors, utilities, and fuel vendors.

EXPERIENCE WITH WET STORAGE

Experience to date indicates that failed fuel has had a minimal impact on storage of spent fuel in water.[2,3,4] A world survey showed that most pools store fuel assemblies containing failed fuel rods on the same basis as intact assemblies; some pools ($\sim 30\%$) store fuel assemblies containing failed rods in canisters.[4] Further degradation of cladding defects during storage does not appear to be occurring. An irradiated PWR fuel rod that had a large hole (nearly as wide as the rod diameter and about two rod diameters in length), and an associated 8-cm (3-in.) long section where fuel was missing, was examined after seven years in wet storage; the cladding breach and the missing section were determined to be no larger than before.[5]

EXPERIENCE WITH DRY STORAGE

The successful experience to date with Zircaloy-clad fuel over a large range of dry storage conditions suggests that none of the potential failure mechanisms (e.g., stress rupture, stress corrosion cracking, hydrogen redistribution) is likely to have a significant influence on the dry storage of spent LWR fuel in inert gas or nitrogen. Current experience indicates that the incidence of cladding failures during dry storage will be low; however, some fuel rod failures cannot be ruled out (see comment above regarding inspection techniques).[6,7] Even if fuel with cladding defects were placed in dry storage, or if defects develop during storage, such defects would not propagate if an inert or nitrogen cover gas is used.[6,7] In general, the impact of a cladding defect that develops during dry storage is anticipated to involve release of fission gas to the sealed cask or canister, but essentially no release of fuel particles.

HANDLING OF FAILED OR DAMAGED FUEL

Tens of thousands of LWR fuel assemblies have been satisfactorily moved during normal handling operations at commercial power reactors and independent spent fuel storage facilities in the U.S.[8] Only a few fuel assemblies have suffered major mechanical damage during handling (~ 250 in the U.S.). Fuel rods (a total of slightly more than the complement of one BWR fuel assembly) have fallen out of spent fuel assemblies during handling. There have been five cases in the U.S. where spent fuel assemblies have come apart during handling [the latest event[9] was at Oyster Creek in 1987 and involved a previously damaged fuel assembly].

The likelihood of breaking rods is potentially higher with fuel rods with large cladding defects. Domestic experience to date indicates that only about seven rods—out of a biased sample of fuel assemblies (ones that were known to contain or suspected of containing failed or damaged fuel rods and were selected for examination, reconstitution, or rod consolidation) containing over 54,000 irradiated fuel rods—broke during handling, examination, fuel assembly reconstitution, or rod consolidation activities. Among the >54,000 rods were nearly 2,000 that were intentionally involved in rod consolidation operations. Among those 2,000 fuel rods were some that were known to have collapsed cladding (a result of in-reactor fuel densification and the coolant pressure). During rod consolidation operations, one rod with collapsed cladding released fission gas, but operators were able to resume work shortly after evacuating the work platform and taking necessary precautions. Another rod with collapsed cladding broke unexpectedly during handling operations subsequent to rod consolidation; the broken rod created no operations problems. One rod consolidation demonstration intentionally included an irradiated PWR fuel assembly with severely bowed fuel rods and showed that such rods can be accommodated.[10] In another study, it was found that even after bowing up to three feet (or more), irradiated PWR fuel rods did not break.[11]

TRANSPORTING OF FUEL

More than 5,100 fuel assemblies have been transported in the U.S. Very few fuel assemblies (<37 in the U.S.) have been damaged during normal transporting operations.[8] There have been two events, one in France and one in the U.S., in which substantial radioactive releases have occurred when uncanned PWR fuel assemblies containing stainless-steel-clad fuel rods (known in the U.S. case to include some failed rods) were shipped in a dry but oxidizing atmosphere. Because of the event in the U.S.,[12] which resulted in release of airborne contamination during the underwater unloading of a failed PWR spent fuel assembly, no more shipments of casks containing uncanned failed fuel with more than small cladding breaches (i.e., with cladding defects larger than pinholes or hairline cracks) are permitted in the U.S.

EXPERIENCE WITH HIGHER BURNUP FUEL

Average burnup levels have been increasing yearly, but fuel rod failure rates have not exhibited a similar trend.[13] To date, the data indicate that extending burnup has not been detrimental to fuel performance. In 1989, five PWR fuel assemblies with burnups >58,000 MWd/MTU were discharged from U.S. PWRs. The only known occurrence in the U.S. of failure of fuel operating in the extended burnup range took place in a core having many debris-induced failures of fuel of traditional design.[14] It will be important for spent fuel management planning to continue to maintain surveillance of the behavior of intact and failed or damaged spent fuel that is placed in wet and dry storage to gather additional evidence, in particular with higher burnup fuel, to assure that fuel integrity is being appropriately maintained.

CONCLUSIONS

The overall domestic nuclear fuel operating experience continues to be excellent: current fuel rod reliabilities are typically >99.99%, which corresponds to fuel rod failure rates of <0.01%.[2] Fuel failure rates in the early days were higher, but the rates decreased as fuel failure mechanisms were identified and eliminated. Infrequent events have occurred in which fuel failed or was damaged during reactor service or in subsequent operations (e.g., handling, storage, rod consolidation, and shipping).[15]

It is concluded that failed fuel has had minimal impact on the wet storage—even for an extended period—of spent LWR fuel; additional degradation of defective fuel does not appear to be occurring. Even if the spent fuel is inspected before being placed in dry storage, the presence of some failed fuel rods cannot be ruled out. However, even if fuel assemblies with failed rods are put in dry storage, or if cladding breaches develop during storage, the breaches are not expected to propagate if an inert or nitrogen cover gas is employed. Fuel handling experience to date at spent fuel storage pools indicates that failed and damaged fuel assemblies, failed fuel rods, and the inadvertent breaking of fuel rods (including prepressurized rods) can be accommodated. In general, when fuel has been damaged as a result of handling, there has been only minor degradation of the fuel assembly components, no breaching of the fuel cladding, and no release of radioactive fission gases or fuel particulates. Little information is available on damage that has been sustained by intact irradiated fuel during shipping, but the general indications are that the damage appears to have been minor. Problems did occur in two cases where uncanned fuel assemblies, known or suspected to contain failed fuel rods, were shipped in a dry, but oxidizing atmosphere.

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** LWRs in the U.S. include boiling water reactors (BWRs) and pressurized water reactors (PWRs).
† Code of Federal Regulations (CFR).

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Description of From-Reactor Transportation Cask Designs

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ABSTRACT

As the Department of Energy's (DOE's) from-reactor cask development program contracts reach the end of preliminary design the program will proceed with two fully funded contracts. They are General Atomic's (GA's) contract for legal weight truck cask designs and Babcock and Wilcox's (B&W's) contract for a rail/barge cask design. The designs are described in the paper. In addition, DOE will pursue specific design features with two of its cask contractors Westinghouse Electric Company (WEC) and Nuclear Assurance Corporation (NAC). The paper also presents several general considerations affecting the cask development program. Two of these which are covered in some detail are the technical topics of burnup credit and source term evaluation.

INTRODUCTION

Under the Nuclear Waste Policy Act (NWPA) of 1982, and its 1987 amendment, the DOE has been assigned the responsibility to develop a waste management system for permanent disposal of spent fuel and high level nuclear waste; the Office of Civilian Radioactive Waste Management (OCRWM) was established within DOE for that purpose. Transportation is an important component of the waste management system. A transportation system is necessary to move spent fuel and high level nuclear waste from existing storage sites (e.g., utility facilities) to a final disposal site or a temporary storage facility such as a Monitored Retrievable Storage (MRS) facility.

To meet the transportation requirements of the NWPA, as amended, OCRWM has established a transportation program comprised of four major elements: (1) cask design and development, (2) the development of support facilities and operational planning, (3) economic and systems analysis, and (4) institutional interactions.

The subject of this paper pertains to cask design and development which is divided into four initiatives. The first initiative covers development of spent fuel casks for shipment from reactors (hence, From-Reactor Casks). The second initiative is development of a From-MRS cask if an MRS is established. The third initiative covers development of specialty casks for nonstandard fuel and certain irradiated hardware and components. The fourth initiative is development of a defense high level waste cask.

Currently cask development activities are concentrating on the first initiative. That is, development of from-reactor casks. The activities for this initiative started when the DOE's Idaho Operations Office (DOE-ID) issued RFP No. DE-RP07-86ID12625 in July 1986, inviting bidders to offer proposals to develop from-reactor Casks. Seventeen organizations submitted a total of forty-six proposals to design, obtain NRC certification, build, and test spent fuel casks from four different cask categories. The categories included legal weight truck (LWT), overweight truck (OWT), rail/barge (R/B), and dual purpose (DP) storage/transport casks. Selection was based on a set of criteria that included technical, management, and cost considerations. In June 1987, six proposers were identified by DOE for contract negotiations. Eventually, five cask contracts were awarded. Two of the awarded contracts were for LWT casks, and three were for R/B casks.

STATUS OF FROM-REACTOR CASK DESIGNS

All five cask contractors are in the process of completing their preliminary design reports. Because of budgetary constraints and to satisfy programmatic needs DOE has chosen to pursue only one LWT cask design and one R/B cask design at full funding for the remainder of FY 90. The second LWT cask design and R/B cask design will continue at reduced funding, concentrating on specific design features identified by DOE. The third R/B cask design contract has been terminated.

The two fully funded cask contractors are GA, and B&W. GA is developing two dedicated LWT cask designs, one for shipment of pressurized water reactor (PWR) spent fuel, the other for boiling water reactor (BWR) spent fuel. B&W is developing a single common use R/B cask design with interchangeable fuel baskets for shipment of PWR or BWR spent fuel. We are confident that these cask designs will be certified in sufficient time to assure transport capability by 1998 which, along with developing a safe and efficient transportation system are primary goals of the program.

The cask contractors who will continue at reduced funding are WEC, and NAC. WEC is developing a LWT cask design which uses a titanium alloy as a structural material. WEC will concentrate its efforts on development of materials specifications and qualification of the titanium alloy as an ASME code material. NAC is developing a R/B cask design which uses an innovative wedge-loc closure design. NAC will concentrate its efforts on development of the wedge-loc closure design. Both contractors will address other design features as they are identified by DOE.

THE GA LEGAL WEIGHT TRUCK CASK DESIGN

GA is pursuing two dedicated LWT cask designs (i.e., separate cask bodies for shipment of PWR and BWR fuel). The GA-4 is being designed to carry four PWR spent fuel assemblies. The GA-9 is being designed to carry nine BWR spent fuel assemblies. Both casks have square rather than the more common circular cross-section, and both are stainless steel structures with depleted uranium for gamma shielding. Both have external solid neutron shields made of a borated polyethylene. The designs both have fully recessed closure heads. Burnup credit is used for the GA-4 (PWR cask), but not for the GA-9 (BWR cask). The fuel support structure for both casks is removable, and is made of stainless steel with boron carbide rod inserts for criticality control. Both the GA-4 and the GA-9 cask designs have the capability of achieving leaktight containment. The GA cask designs use aluminum honeycomb impact limiters.

The GA cask designs represent a significant payload increase over currently certified LWT casks which have capacities of one PWR or two BWR spent fuel assemblies. The 4/9 capacity is a four and four and one-half increase over current cask capacities for PWR and BWR spent fuel. A significant contributor to this increase is the fact that the casks are designed for older and cooler spent fuel; however, several innovative design features have also contributed to the high payloads expected for these designs.

Current generation spent fuel casks are designed for spent fuel that is cooled (out of reactor) for about 150 days. The OCRWM from-reactor cask designs are being optimized for ten-year cooled spent fuel. The additional cooling time represents a reduction of internal heat generation by about a factor of twenty, with a corresponding decrease in gamma radiation from the spent fuel. For LWT casks heat dissipation is no longer an important consideration, and gamma shielding provided by dense metal can be reduced significantly. Both factors contribute to higher payload.

The significant innovative features of the GA cask designs that have contributed to higher payload are the use of dedicated cask bodies, the square cross-section, and the use of burnup credit which will be addressed later in this paper. Because of differences in PWR and BWR spent fuel dedicated designs allow for optimization on each fuel type. PWR spent fuel is shorter but hotter (thermally and radioactively) than BWR spent fuel. A PWR cask can be shorter, but requires thicker heavy metal gamma shielding than a BWR cask. Conversely, a common use cask is penalized by carrying extra length for PWR shipments, and extra shielding for BWR shipments. The square cross-section of the GA designs closely follow the envelope of the square fuel assemblies reducing excess material inherent in placing a square peg into a round hole. Additionally, the thick steel containment shell is external to the depleted uranium gamma shield rather than internal. The depleted uranium is a more effective shield than the steel and having it inboard is more efficient.

THE B&W RAIL/BARGE CASK DESIGN

B&W is developing the BR-100, a common use R/B cask (i.e., one cask with interchangeable baskets for PWR or BWR fuel assemblies). The cask design capacity is 21 PWR assemblies or 52 BWR assemblies. The cask design has a circular cross-section and has a stainless steel structure with a lead gamma shield. Neutron shielding is provided by an internal neutron/thermal shield made of borated concrete. The cask design uses a fully recessed closure head. The design uses burnup credit. In addition, an aluminum basket with cermet plates and flux traps is used for criticality control. The cask is a leaktight design. The impact limiters are made of balsa and redwood, and are encased in Kevlar with a thin outer shell of steel.

The B&W cask design represents a significant payload increase over currently certified R/B casks which have capacities of up to 10 PWR or 24 BWR spent fuel assemblies. The 21/52 capacity is more than a factor of two increase over current cask capacities for PWR and BWR spent fuel. As in the case of the LWT designs a significant contributor to this increase is the fact that the casks are designed for older and cooler spent fuel; and again there are several innovative design features being developed by B&W, some of these have contributed to the high payloads expected for these designs.

The benefits associated with casks designed for ten-year cooled fuel instead of 150-day cooled fuel are similar to those for LWT casks. However, for the R/B casks heat dissipation is more of a challenge than for LWT cask designs due to the higher payload and resulting higher internal heat.

The innovations in the R/B cask design do not impact cask capacities as dramatically as those associated with the LWT cask design, but they do help. The wood and Kevlar impact limiter is a weight saver, and the aluminum basket is light weight and provides a good heat transfer path for the internal heat of the spent fuel. Again, burnup credit is used in the B&W design, its contribution will be discussed later. A significant innovation of the B&W design is the neutron/thermal shield. Although it does not increase capacity it provides an effective neutron shield with the added benefit of thermal protection to the lead gamma shield.

BENEFICIAL DESIGN FEATURES

WEC and NAC will both be performing their OCRWM cask contracts under reduced funding for the last threequarters of FY 90. During that period they will pursue specific design features identified by DOE. Each contractor has a feature that could prove very beneficial to the OCRWM program. WEC's proposed titanium alloy could result in significant weight savings because of its high strength to weight ratio. However, titanium alloys have not been approved as structural material for transport casks in the U.S. Therefore, WEC's task will be to demonstrate the acceptability of titanium to the NRC. The first step in this process is to gain approval as an ASME code material. NAC's wedge-loc closure could significantly reduce cask turnaround time. The wedge-loc closure is a hydraulically actuated device that offers an alternative to the flanged-bolted closure common to spent fuel casks. The benefit of this innovation is in the time to open or close a cask. The process of opening or closing a 100-ton R/B cask is estimated to take about two hours, the opening or closing process for the wedge-loc closure is estimated to take 10 to 15 minutes. The result would be reduced worker exposure and reduced operation costs.

TRANSPORTER DEVELOPMENT

The from-reactor cask contracts call for development of a total transport system, that is, a cask and a transporter. For the LWT cask a trailer is required, for the R/B cask a railcar. In each case it gives the cask designer the latitude to distribute system weights between the cask and its transporter. For the LWT system the constraint is the 80,000 pound gross vehicle weight (GVW). For the R/B cask the constraint is the cask hook weight of 100-tons and the GVW of 263,000 pounds. In each case the push to higher cask capacities results in the need to develop lighter transporters. The goals are about 9,000 and 25,000 pounds, respectively, for the truck trailer and railcar. Based on current efforts of the cask contractors these goals appear to be achievable. For example, GA has been working with their subcontractor, Foster Wheeler, to use state-of-the-art methods for a lightweight trailer design. They have used a finite element structural model to design a trailer dedicated to the GA cask configurations.

DESIGN REQUIREMENTS

The from-reactor casks are being designed to transport spent fuel from commercial reactor facilities to federal waste facilities. The casks must be designed to interface with numerous reactor facilities that are already built and operating as well as with one or more federal waste facilities that are still to be designed and built. Other interface design requirements that must be considered include highway and rail limits, and spent fuel characteristics to name a few. From an efficiency stand point design standardization will be a beneficial pursuit.

An excellent source of information for utility design interface is the ongoing Facility Interface Capability Assessment (FICA). This has been used by, and will continue to be used by, the cask contractors and DOE in monitoring the cask development program. For interface with the Federal facilities we have established contacts within the OCRWM program to assure compatibility between cask designs and facility designs and concepts as they evolve. OCRWM has established institutional and technical interfaces to assure knowledge of highway and rail requirements. We continue to perform systems studies to address impact of fuel characteristics (e.g., age and burnup) on transportation capabilities. As we proceed beyond preliminary design we will work on improved handling capabilities of the cask designs, both manually and robotically.

TECHNICAL ISSUES

There are two technical issues that will be addressed in this paper: (1) burnup credit, and (2) source term evaluation. Both have the potential to significantly affect the efficiency of the OCRWM transportation system. Because both affect more than one cask design, they are addressed under OCRWM's transportation technology development program.

The use of burnup credit is expected to significantly enhance the ability to increase cask capacities. Burnup credit is simply accounting for reduced reactivity of spent fuel that results from power production. That is, as the fuel is used or "burned" in the reactor there is a net reduction in the fissile content of the fuel, and an accompanying buildup of neutron absorbing fission products or "poisons." Traditionally, a "fresh fuel" assumption is made in analysis used to demonstrate criticality safety. The use of burnup credit in demonstrating criticality safety would be more consistent with the actual conditions of spent fuel. Its use would allow closer spacing of fuel, less need for flux traps (empty spaces in fuel baskets), less need for neutron absorbing poisons within fuel baskets, and generally simplified basket designs. All of which results in higher capacities, and fewer shipments to move a fixed inventory of spent fuel. Fewer shipments reduce public and worker exposure, reduce radiological and nonradiological risks associated with transport of spent fuel, and reduce transport costs.

Because of the long history behind the use of the fresh fuel assumption for criticality safety analysis we must establish a sound technical base if we wish to use burnup credit. A number of factors have been identified that must be addressed in order to confidently use burnup credit for the OCRWM cask designs. These include such things as: characterization of fissile and poison content of spent fuel, establishing the adequacy of analysis methods, developing benchmark data, identifying practical measurement techniques for verification of implementation procedures for burnup credit, and demonstrating the effectiveness and reliability of implementation approaches. Efforts are currently underway to enable OCRWM to address and resolve these factors to satisfy ourselves, the NRC, the criticality safety community, and the interested public.

Although the from-reactor casks are being designed as "leaktight," we anticipate using source term evaluation to demonstrate containment adequacy in compliance with NRC's transportation regulations (10 CFR 71). Designing and building leaktight casks is not considered difficult, but testing to demonstrate leaktightness could be an operational burden, leading to longer turnaround times and increased worker exposure.

The source term evaluation approach to containment requires an understanding of the characteristics and dispersibility of a cask's radioactive contents, the behavior of the contents under various transport environments, and potential release mechanisms for the material contained. The alternative is to use a leaktight container. Leaktightness is demonstrated if a containment system is tested without showing leakage using a method capable of detecting a 1 x E-7 leakage of dry air at 25 C and 1 atm pressure potentially leaking to a 0.01 atm pressure environment.

NRC's transportation regulations (10 CFR 71), its regulatory guidance (Reg Guide 7.4), and the national standards (ANSI N14.5-1987) all support the source term evaluation approach to containment of radioactive materials. Furthermore, the source term evaluation approach has strong precedent in NRC's cask certification practice.

OCRWM is in the process of establishing a technically sound and consistent approach to demonstrating the containment adequacy of its casks through source term evaluation. The ongoing efforts in this area are addressing the technical issues that have been identified. Preliminary work has been completed and reports will be issued in the near future.

CONCLUSION

As the from-reactor cask contracts reach completion of their preliminary design reports they show significant increases in cask capacities over current spent fuel cask designs. Much of this increased capacity is due to designing casks for older and cooler fuel, but innovation has also played a part in this achievement. Although the from-reactor cask development effort has been reduced in scope, OCRWM remains confident that the program goals of providing a safe and efficient transport capability by 1998 will be accomplished.

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An Improved Technique For Passive Neutron Assay Through the Use of Extendable Dead Time and Higher Moments Analysis

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ABSTRACT

The well known High Level Neutron Coincidence Counting technique for the non-destructive determination of plutonium in sealed packages has for long been plagued by complications arising from self-multiplication effects. We propose here an extension of the techniques developed recently by us for the measurement of the degree of subcriticality of highly subcritical systems in the field of reactor noise analysis, to the passive neutron assay problem. It is shown that deployment of a variable dead time filter of the extendable type rather than of the commonly adopted non-extendable type leads to a simple and practical method for measuring the mean, variance and higher moments of the multiplicity distribution of the neutron emitting source and deducing therefrom the spontaneous fission rate, the self-multiplication probability and other parameters. The applicability of the technique is demonstrated using a Cf-252 neutron source.

INTRODUCTION

The non-destructive estimation of plutonium contained in sealed packages is of interest from the safeguards point of view. The passive neutron assay techniques[1,2] developed for this purpose exploit the fact that Pu-240, which is always present to some extent in plutonium samples, decays by spontaneous fission, emitting neutrons in bursts of two or more. If the Pu sample is surrounded by a neutron moderator-detector system having a sufficiently high detection efficiency (>10%), the detected neutron pulses are correlated with one another over time durations comparable to the neutron die-away time in the moderator-detector assembly. Statistical analysis of these pulses is used to determine the spontaneous fission rate after distinguishing the background counts due to (α,n) reactions.

The earliest instrument employed for passive neutron assay was the variable dead time counter (VDC) of Birkhoff et al.[3] This technique exploits the fact that the introduction of a dead time in the path of a correlated pulse train results in a larger count rate loss than what may be expected for uncorrelated pulses. A similar technique had earlier been developed by Jacquesson[4] for estimating Pu-240 concentration in Pu samples and by Srinivasan[5] for measuring α, the prompt neutron decay constant of a nuclear reactor. The VDC technique however had a notable drawback viz that no exact relation between the deviation from Poisson behaviour and the spontaneous fission rate was available.

The VDC method has since been superceded by the Shift Register (SR) technique of Bohnel.[6] In this technique, every incoming pulse triggers a time interval of length t during which the number of real plus accidental coincidences (with the trigger pulse) is measured and another interval of the same duration but after a delay which is large compared to the neutron die-away time which gives only the accidental coincidences. The results are accumulated on two registers viz the R + A and A registers and by subtracting the latter from the former a true measure of the magnitude of correlations is derived. The SR method thus measures two quantities from which two parameters can be derived. For large oxide samples, in addition to spontaneous fissions and (α, n) reactions, self multiplication is important and it is necessary to measure an additional parameter viz the self multiplication probability. The SR technique has been extended[7] for such samples if the latter quantity can be obtained by some independent method.

A number of experimental techniques have been proposed in recent years for deriving this quantity by analyzing the information contained in the pulse train itself.[8-11] Most of these techniques either attempt to measure the multiplicity distribution of source neutrons or its third factorial moment. The simple extendable dead time method proposed here is of interest in this context since the count rate (or the first moment) alone contains the information on all moments of the multiplicity distribution of source neutrons as we shall see later. In this paper we report the results of an experiment conducted at Trombay to demonstrate the validity of our method, using a Cf-252 source. The objective of the experiment was twofold: Firstly, to examine the validity of a number of dead time formulae published in the areas of both reactor noise analysis and nuclear safeguards. Secondly, to compare the extendable dead time method with some other method commonly used for the passive neutron assay of plutonium. The moments method with randomly triggered intervals[11] was chosen for this purpose.

REVIEW OF DEAD TIME FORMULAE

The introduction of a dead time filter in the path of a pulse train besides reducing the number of counts in any channel significantly alters the statistics of the counts. Two kinds of dead times filters referred to as extendable dead time and non-extendable dead time (also called paralysable and non-paralysable respectively) are conceivable. In the extendable type of dead time unit, the duration of the dead time gets extended if a new pulse arrives even during the time that the counting system is dead. In the non-extendable dead time case no extension of dead time occurs if a pulse arrives when the counting system is already dead. These are illustrated schematically in Fig.1.

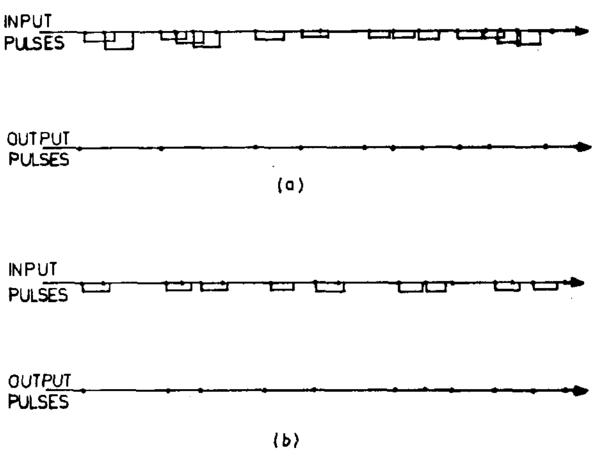

Figure 1. Effect of introducing a dead time gate in the path of a pulse train: (a) extendable dead time; (b) non-extendable dead time.

Uncorrelated Pulses

If the incoming pulses are purely random i.e. uncorrelated, then the following relations[12] are applicable for the filtered count rate:

$$N_e(d) = N(o)\exp\{-N(o)d\} \tag{1}$$

$$N_n(d) = \frac{N(o)}{1+N(o)d} \tag{2}$$

where $N_e(d)$ and $N_n(d)$ are the count rates in the presence of extendable and non-extendable dead times respectively and d is the dead time introduced. N(O) is the count rate in the absence of any dead time gate. Similarly we have the following relations[12] for the variance V(d) of counts in a (randomly triggered) measuring interval t:

$$V_e(d) = m_e(d)\left[1 - m_e(d)\left\{1 - \left(\frac{t-d}{t}\right)^2\right\}\right] \tag{3}$$

$$V_n(d) = \frac{m_n(d)}{[1+N(o)d]^2} \tag{4}$$

where m(d) is the mean number of counts in the interval in the presence of a dead time d and the subscripts e and n stand for extendable and non-extendable as before.

Correlated Pulses: Reactor Noise

If the pulses are correlated as in the case of reactor noise studies, the derivation of relations is much more involved. The earliest formulae for the count rate (First Moment) and the variance (Second Moment) in the presence of a non-extendable dead time filter were published by Srinivasan and Sahni[5,13] in the context of experimental determination of the prompt neutron decay constant of reactor assemblies.

$$N_n(d) = \frac{N(o)}{1+N(o)d+Q(1-e^{-\alpha d})} \tag{5}$$

$$\frac{V_n(d)}{m_n(d)} + 1 + A(\alpha,d) + 2QB(\alpha,d)\left\{1 - \frac{1-e^{-\alpha t}}{\alpha t}\right\}e^{-\alpha d} \tag{6}$$

where, $Q = \frac{\epsilon\,\overline{\nu(\nu-1)}}{2\,\bar{\nu}^2}\left(\frac{K_p}{1-K_p}\right)^2$

α is the prompt neutron decay constant, Kp is the prompt neutron multiplication factor and t is the length of the observation interval assumed to be randomly-triggered. ϵ is the detection-efficiency defined as the number of counts per fission and $\bar{\nu}$ and $\overline{\nu(\nu\text{-}1)}$ are the mean and second factorial moment of the multiplicity distribution of fission neutrons. A and B are complicated functions of α and d.

Eq. (6) has been verified by Edelman et al over a limited range of t and d.[14] However, the problem of deriving the statistics of correlated counts in the presence of a non-extendable dead time filter is extremely complicated and the simple formulae (Eqs. 5 and 6) are expected to have limited validity only. An attempt was made by Teodosik[15] to modify Eq. (5) so that it could be applied to extendable dead times. However his modified expression for the count rate is applicable to neither extendable nor non-extendable dead times.

If the correlated pulse train is due to a spontaneously fissioning source rather than due to a chain multiplying assembly, then it is not difficult to show that Eq. (5) and (6) remain unaltered except for a change in Q which now becomes:

$$Q = \frac{\epsilon\,\overline{\nu(\nu-1)}}{2\,\bar{\nu}}$$

where $\bar{\nu}$ and $\overline{\nu(\nu\text{-}1)}$ refer to the spontaneously fissioning nuclide and the efficiency ϵ is now defined as the number of counts per neutron produced.

The following formulae for the count rate and the variance to mean ratio in a randomly triggered time interval of length t in the presence of an extendable dead time filter were derived by Babala.[16]

$$N_e(d) = P_{\gamma o}(d) \tag{7}$$

$$\frac{V_e(d)}{m_e(d)} = 1 - m_e(d)\left\{1 - \left(\frac{t-d}{t}\right)^2\right\} +$$

$$2QP_{co}(d)Be^{-\alpha d}\left(\frac{t-d}{t}\right)\left[1 - \frac{1-e^{-\alpha(t-d)}}{\alpha(t-d)}\right] \tag{8}$$

where $P_{\gamma o}$ is the random origin time interval distribution function and P_{co} is the probability of zero counts in a time interval of duration d following a count. P_{ro} and P_{co} are related to each other as follows

$$P_{co} = \frac{1}{N(o)} P_{\gamma o}$$

B is a complicated function of the system parameters and the other symbols have the same meaning as in Eqs. (1) to (6).

Eq. (7) is exact provided it is possible to derive a closed form expression for the zero count probability in a time interval. The same however is not true of Eq. (8) which has been derived by integration of the probability of obtaining a count between t_1 and $t_1 + dt_1$ and another between t_2 and $t_2 + dt_2$. In deriving the latter, it is assumed that the conditional probability of getting an output count at t_2 given an output count at t_1 is the same as the conditional probability of getting an output count at t_2 given an input count at t_1.

Correlated Pulses: Nuclear Safeguards

Interest in the dead time method was revived in the context of nuclear safeguards following the development of the variable dead time counter technique. This instrument also used the non-extendable kind of dead time for which a simple extension of Eq. (2) was proposed by Stanners.[17] In the Stanners formalism, two quantities X and Y are defined as follows:

$$X = N(o) - \frac{N(d)}{1 - N(d)d} \tag{9a}$$

$$\gamma = \sum_{i=1}^{m} (n_i - 1) \tag{9b}$$

where n_i is the number of pulses in the i th pulse group and m is the average number of pulse groups per second. Y is clearly proportional to the number of spontaneous fissions per second and X is the degree of deviation from Poisson behaviour. Y is then related to X as follows:

$$\gamma = \frac{X}{1 - E_1 - \dfrac{N(d)d}{1-N(d)d} E_2} \tag{10}$$

where E_1 and E_2 are functions of d, the count rate N(d) and a parameter q. Interest in the VDC method subsequently waned as the Stanners formalism could not adequately give the correct Pu content in all circumstances.

The problem of extendable dead time has been recently studied by Mathes and Haas[18] for obtaining corrections due to neutron detector dead time to the count rate and the coincidence count rate in the shift register method. Their formulae include only second order correlations and are applicable to dead times which are short compared to the neutron die away time in the moderator-detector system. The following exact expressions for the count rate N (d) and the function f_2 defined such that $f_2 dt_1 dt_2$ gives the probability of detecting one neutron between t_1 and $t_1 + dt_1$ and another between t_2 and $t_2 + dt_2$, in the presence of an extendable dead time filter have been derived by Degweker:[19]

$$N(d) = \left\{S \sum_{\gamma=0}^{\infty} \frac{(-1)^\gamma \epsilon^{\gamma+1}}{(\gamma+1)!} m_{\gamma+1}(1-e^{-\lambda d})^\gamma\right\}$$

$$x \exp\left[S \sum_{\gamma=1}^{\infty} \frac{(-1)^\gamma \epsilon^\gamma}{\lambda \gamma!} m_\gamma\right.$$

$$\left.\left[\left\{\sum_{l=1}^{\gamma} (-1)^l \left(\frac{\gamma}{l}\right)\left\{\left(\frac{1}{\gamma} - \frac{1}{l}\right)e^{-l\lambda d} + \frac{1}{l}\right\} + \lambda d + \frac{1}{\gamma}\right]\right] \tag{11}$$

$$f_2 = \left[\lambda s \sum_{\gamma=0}^{\infty} \frac{(-1)^\gamma \epsilon^{\gamma+2}}{(\gamma+2)!} m_{\gamma+2} A_\gamma(\lambda, d, t_1 - t_2)\right.$$

$$+ S^2 \left\{\sum_{\gamma=0}^{\infty} \frac{(-1)^\gamma \epsilon^{\gamma+1}}{(\gamma+1)!} m_{\gamma+1} B_\gamma(\lambda, d, t_1 - t_2)\right\}$$

$$\left. x \left\{\sum_{\gamma=0}^{\infty} \frac{(-1)^\gamma \epsilon^{\gamma+1}}{(\gamma+1)!} m_{\gamma+1} C_\gamma(\lambda, d, t_1 - t_2)\right\}\right]$$

$$\exp\left\{S \sum_{\gamma=1}^{\infty} \frac{(-1)^\gamma \epsilon^\gamma}{\gamma!} m_\gamma D_\gamma(\lambda, d, t_1 - t_2)\right\} \tag{12}$$

where s is the source event rate, ϵ is the detection efficiency, $1/\lambda$ is the neutron lifetime and m_γ are the factorial moments of the source multiplicity distribution. A, B, C, and D are complicated functions of their arguments and are explained in Ref. 19. The variance of counts in a randomly triggered interval of length t can be derived from the function f2 using the following relations:

$$\frac{1}{2} M_2 = \iint f_2(t_1, t_2) dt_1 dt_2 \tag{13}$$

$$V = M_2 + M_1 - M_1^2 \tag{14}$$

where V is variance and M_1 and M_2 are the first and second factorial moments of the counts distribution in the interval. Eqs. (11) and (12) are exact in contrast to the formulae presented earlier by other authors. It is clear from these equations that even the average count rate following the introduction of an extendable dead time gate in the pulse path contains the information of all factorial moments of the source multiplicity distribution. Thus by fitting the measured count rate as a function of the dead time introduced to the above equations it should be possible to deduce various parameters including the self multiplication probability as is shown later in this paper. While Eq. (12) is too complicated for this purpose, Eq. (11) is fairly tractable and forms the basis of the extendable dead time technique described in this paper.

THE MOMENTS METHOD

In this method the system parameters are derived by measuring the probability distribution of the number of counts in either count triggered or randomly triggered time intervals from which factorial moments up to various orders may be calculated. It is possible to derive the system parameters directly from the probability distribution using the following relation[9] between the 'real' multiplicity distribution of detected neutrons in a count triggered interval, t, and the source multiplicity distribution, ρ

$$\gamma = f\Lambda WD\rho \tag{15}$$

where D, w and Λ are the detection, weighting and the SR response matrices respectively and f is a normalisation factor. However it is advantageous[10,11] to use factorial moments because these are much more simply related to the system parameters. In its simplest version the moments method reduces either to the shift register method or to the variance/mean method of Feynman[20] depending upon whether count triggered or randomly triggered intervals are employed.

Expressions for the factorial moments of the probability distribution of detected neutrons in count and randomly triggered intervals have been derived by Hage and Cifarelli.[8] These expressions can also be easily obtained from the n interval probability generating function (pgf) of detected neutrons. The following expression for the n interval pgf was derived in:[19]

$$F_n(z_1, z_2 \ldots z_n)$$

$$= \exp\left[S\int_{-\infty}^{t_{n_2}}\left\{\sum_\nu \rho(\nu)\left(P_o(\tau) + \sum_{i=1}^{n} P_i(\tau)z_i\right)^\nu - 1\right\}d\tau\right] \tag{16}$$

where, $\quad P_o(\tau) = 1 - \sum_{i=1}^{n} P_i(\tau)$

and

$$P_i(\tau) = \epsilon\left\{\exp - \lambda(t_{i_1} - \tau)\text{-}\exp\text{-}\lambda(t_{i_2}\text{-}\tau)\right\} \qquad \tau < t_{i_1}$$

$$= \epsilon\left\{1 - \exp - \lambda(t_{i_2} - \tau)\right\} \qquad t_{i_1} < \tau < t_{i_2}$$

$$= 0 \qquad \tau > t_{i_2}$$

are the probabilities of detection of a neutron born at time τ in the i th time interval (t_{i_1}, t_{i_2}), $F\eta$ is the n interval pgf of detected neutrons, ϵ is the detection efficiency, s is the source event rate, $\rho(\nu)$ is the multiplicity distribution of the source and $1/\lambda$ is as before the neutron die away time.

For randomly triggered intervals we consider the one interval pgf for an interval extending from 0 to t. The factorial moments are then obtained by successively differentiating F_1 w.r.t. z and setting $z=1$. The following formulae are obtained for the first three factorial moments on carrying out the integrations over τ.

$$M_1 = s\epsilon m_1 t \tag{17}$$

$$M_2 = s\epsilon^2 m_2\left(t - \frac{1}{\lambda} + \frac{1}{\lambda}e^{-\lambda t}\right) + M_1^2 \tag{18}$$

$$M_3 = s\epsilon^3 m_3\left(t - \frac{3}{2\lambda} + \frac{2}{\lambda}e^{-\lambda t} - \frac{1}{2\lambda}e^{-2\lambda t}\right) + 3M_1M_2 - M_1^3 \tag{19}$$

where M_1 M_2 and M_3 are the first three factorial moments of the distribution of the number of detected neutrons in the time interval t while m_1 m_2 and m_3 stand for the corresponding moments of the multiplicity distribution of the source. ϵ and $1/\lambda$ as before are the detection efficiency and the die away time respectively.

For deriving expressions for the factorial moments of counts in count triggered intervals we consider the two interval pgf of detected neutrons in which the first interval is infinitesimal and extends from 0 to δt and the other is finite from δt to t. The conditional pgf of counting in the second interval given one count in the first interval is then given by

$$\frac{1}{s\epsilon m_1 \delta t}\frac{\partial F_2}{\partial z_1}\bigg|_{Z_1=0} \tag{20}$$

The conditional factorial moments of the counts distribution in the second interval are obtained by successively differentiating (20) w.r.t z_2 and setting $z_2=0$. On carrying out the integrations over τ we obtain the following expressions for these moments.

$$\mu_1 = s\epsilon m_1 t + \frac{\epsilon m_2}{2m_1}(1 - e^{-\lambda t}) \tag{21}$$

$$\mu_2 = s^2\epsilon^2 m_1^2 t^2 + s\epsilon^2 m_2\left\{t(1 - e^{-\lambda t}) + t - \frac{1}{\lambda} + \frac{1}{\lambda}e^{-\lambda t}\right\}$$

$$+ \frac{\epsilon^2 m_3}{3m_1}(1 - e^{-\lambda t})^2 \tag{22}$$

In Eqs. (21) and (22) the symbols μ_1 and μ_2 have been used for the first two factorial moments of the distribution of counts in count triggered intervals to distinguish them from the corresponding moments in randomly triggered intervals [Eqs. (14-16)].

Eq. (21) gives the mean number of counts in a count triggered interval. The real plus accidental coincidence count rate (R + A) of the SR method, can be obtained by simply multiplying Eq. (21) by the total count rate given by the product $s\epsilon m_1$ since in this method every count is used to trigger an interval. On the other hand it is easy to recognize the first term on the right hand side of Eq. (18) as the excess variance of counts in a time interval of duration t.

BOHNELS FORMULAE

It is to be noted that both in the moments methods as well as in the extendable dead time method proposed here, the observed quantities viz N(d) [Eq. (11)], M_γ [Eqs. (17-20)], and μ_γ [Eqs. (21, 22)] are related to the system parameters (s, ϵ and λ) and the factorial moments m_γ of the source multiplicity distribution. The latter can be related to basic nuclear data of the spontaneously fissioning nuclide and to

$$m_1 = \frac{\bar{\nu}^{(s)}(1-P)}{1-\bar{\nu}p} \tag{23}$$

$$m_2 = \left(\frac{1-P}{1-\bar{\nu}P}\right)^2\left[\overline{\nu(\nu-1)}^{(s)} + \frac{P}{1-\bar{\nu}p}\bar{\nu}^{(s)}\overline{\nu(\nu-1)}\right] \tag{24}$$

$$m_3 = \left(\frac{1-P}{1-\bar{\nu}P}\right)^3\left[\overline{\nu(\nu-1)(\nu-2)}^{(s)} + \frac{P}{1-\bar{\nu}p}\left\{3\overline{\nu(\nu-1)}^{(s)}\overline{\nu(\nu-1)} + \overline{\nu(\nu-1)(\nu-2)}\,\bar{\nu}^{(s)}\right\} + 3\left(\frac{P}{1-\bar{\nu}p}\right)^2\bar{\nu}^{(s)}\overline{\nu(\nu-1)}^2\right] \tag{25}$$

the self multiplication probability within the sample as follows:[7,21]

where p is the self multiplication probability ν,$\nu(\nu$-1) etc., have their usual meaning and the superscript (s) indicates spontaneous fission. Thus the number of parameters that can be deduced from the measurements is identical to the number of factorial moments m. that can be inferred from the measured data.

Table I summarizes various features of the above techniques that may be used in the field of passive neutron assay of plutonium. Among the methods listed in the table the SR is the most commonly used technique for two parameter estimation for small and large samples. For large samples involving simultaneous estimation of three parameters viz the spontaneous fission rate s_f, the (α, n) rate and the self multiplication probability p, the probability distribution and the moments methods have been reported with some degree of success. The extendable dead time (1 st moment) method was proposed[19] as a possible alternative to these methods.

In what follows, we describe the experiments conducted at Trombay to demonstrate the extendable dead time method for the passive neutron assay of plutonium. For the purpose of comparison, the moments method with randomly triggered intervals was selected. Cf-252 was used as the neutron source in these experiments.

Table I

Name of Technique	*Quantities Measured*	*Equipment Required*	*Formula Used*	*Parameters Derived*	*Remarks*
VDC (Birkhoff)	Count rate as function of dead time	Dead time filter unit (non-ext.)	Eq. (10)	Sp. Fiss rate and (α,n) rate	No exact relation; Applicable for small masses
SR method (Bohnel)	R + A, A	SR unit	Eq. (21)	-do-	Good for small and large samples; No third parameter
Probability Distribution Method		Modified SR unit	Eq. (15)	Sp. Fiss. rate (α,n) and self mult.	Good for small and large samples; Large errors possible in p_2, the prob. of doubles.
Moments Methods (Hage)	Factorial moments of counts in count or randomly triggered int.	Time int. between successive counts recorder		-do-	Good for small and large samples; Large errors possible in 3rd moment (m_3)
i) Count triggered intervals			Eqs. (21-22)		
ii) Randomly triggered intervals			Eqs. (17-19)		
Dead time (1st moment method)	Count rate as function of dead time	Dead time filter unit (extendable)	Eq. (11)	-do-	Good for samples upto 1-2Kg mass. Large errors possible in 3rd moment (m_3)
Dead time (2nd moment method)	PDA as function of dead time	PDA and dead time unit	Eq. (12-14)		Difficult to estimate parameters

EXPERIMENTAL ASPECTS

Description of the Neutron Source-Detector Arrangement

The experiment was conducted in a large 60 cm side rectangular tank filled with water (Fig. 2) at the centre of the which is a hollow tube housing the Cf-252 source. Surrounding the tube is a ring of six BF_3 neutron detectors (45 cm long and 2.5 cm in diameter). The distance of these detectors from the central tube was adjusted to achieve maximum neutron detection efficiency. The detectors are connected in parallel to form a bank and the combined output is fed to an 8085 microprocessor based Probability Distribution Analyser (PDA)[22] after passing through a variable dead time unit.

Dead Time Unit

This is fabricated from a single monostable IC 71423 having two mono-shots. One mono-shot is used either in the retriggerable or non-retriggerable mode corresponding

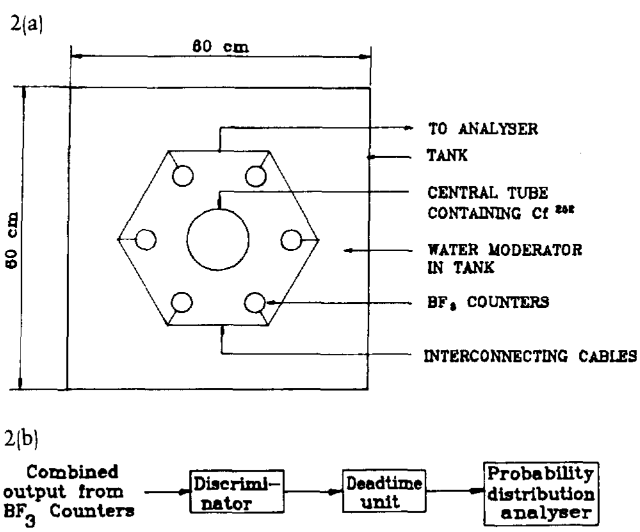

Figure 2. Experimental setup used: (a) source and detector arrangement; (b) electronics.

to extendable or non-extendable dead time modes of operation. The desired dead time value is determined by the output pulse width and can be varied by adjusting the RC time constant. The second mono-shot is used to give an output pulse of a fixed width (0.5 μs) at the end of each timing pulse. This is fed to a scaler for counting. The input is from a discriminator scaled down to suit TTL logic.

Probability Distribution Analyser

This is made from a 8085 microprocessor and three 16 bit counters contained in the 8253 chip. Two of these counters serve as a timer when fed with the microprocessor clock pulses while the third is used as a scaler. The microprocessor is programmed to periodically open a gate for a time interval (which can be varied from 10 μs to 10 s) while the scaler records the number of incoming pulses. Depending on the number of counts recorded the contents of the corresponding location in the microprocessor memory are incremented by one. The period of the entire operation is only slightly (about 10%) more than the counting interval chosen. Thus a frequency distribution of the number of counts in the selected time interval is generated which is ultimately transferred to a PC for further processing.

Experimental Results

The experiment was performed on a Cf-252 source containing a few nanograms of Cf-252. The counting interval was kept fixed at 10 ms throughout the experiments i.e. at a value much larger than the neutron die away time in the moderator detector assembly so as to be in the asymptotic prompt variance region. The probability distribution analysis was carried out following the introduction of either an extendable or a non-extendable dead time in the range of 0 to 450 μs. The zero dead time readings were obtained by bypassing the dead time unit. The inherent dead time of the BF_3 detectors which was in the range of 1 to 2 μs was ignored in the analysis. Since the source was rather weak, it was necessary to accumulate data for about one hour in order to obtain good statistics.

Table II
Variation of Mean and Variance
with Extendable Dead Time
(Counting Interval Fixed at 10 ms)

Dead time (μs)	*Mean (M_1 or m)*	*N(d)/N(O)*	*Variance (V)*	*V/m*	*M_3*
0	24.608±0.010	1.0	30.478±0.075	1.238	15336.65
13	23.500±0.009	0.955	26.910±0.067	1.145	
95	18.286±0.006	0.743	12.777±0.032	0.699	
185	14.543±0.005	0.591	6.836±0.017	0.470	
275	11.746±0.004	0.477	4.150±0.010	0.353	
360	9.607±0.003	0.390	2.904±0.007	0.288	
450	7.912±0.003	0.322	2.280±0.007	0.288	

Table III
Variation of Mean and Variance
with Non-extendable Dead Time
(Counting Interval Fixed at 10 ms)

Dead time (μs)	*Mean (M_1 or m)*	*N(d)/N(O)*	*Variance (V)*	*V/m*	*M_3*
0	24.426±0.010	1.0	30.179±0.074	1.236	14996.09
13	23.360±0.009	0.956	26.936±0.067	1.153	
95	18.937±0.007	0.775	14.301±0.035	0.755	
185	15.958±0.005	0.653	8.691±0.022	0.545	
275	13.964±0.004	0.592	5.854±0.015	0.419	
360	12.397±0.004	0.507	4.119±0.010	0.332	
450	11.094±0.003	0.454	3.052±0.007	0.275	

The mean and variance of counts obtained with extendable and non-extendable dead times are summarized in Tables II and III respectively. Also shown is the third factorial moment of detected neutrons when the dead time unit is completely bypassed. It may be noted that the detector efficiencies in the two cases are slightly different since the source may not have been placed in exactly the same positions in the two experiments. The error limits on the mean and variance are theoretical estimates for the number of observation intervals used.

The count rate and the variance can be seen to fall with dead time of both varieties; the fall being more pronounced for the extendable dead time case. Fig. 3 shows the variation of count rate (normalized to the zero dead time count rate) for both extendable and non-extendable dead times cases. The curve for extendable dead times lies below that for non-extendable dead times as expected. The correlated nature of the pulses is evident from the fact that with no dead time introduced, the variance to mean ratio is greater than unity. The V/m ratio decreases with increasing dead time. For the non-extendable dead time case the ratio goes to zero at large dead times showing that with increasing dead time the pulses become more and more like clock pulses. For the extendable dead time case the ratio approaches a constant.

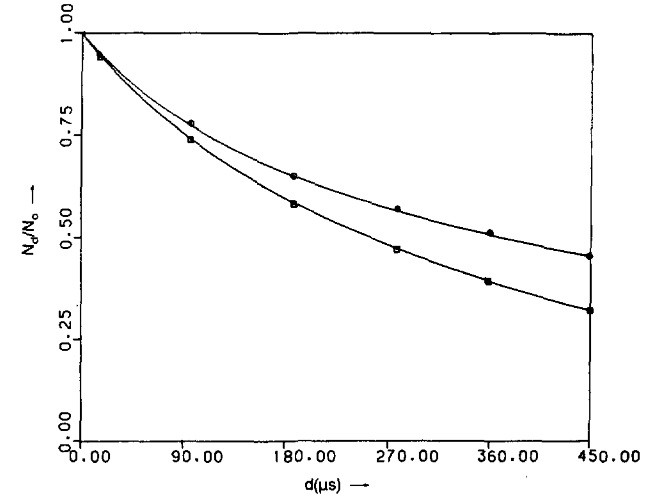

Figure 3. Variation of count rate with dead time introduced. The squares indicate the points obtained with extendable dead time while the circles indicate the points obtained with non-extendable dead time.

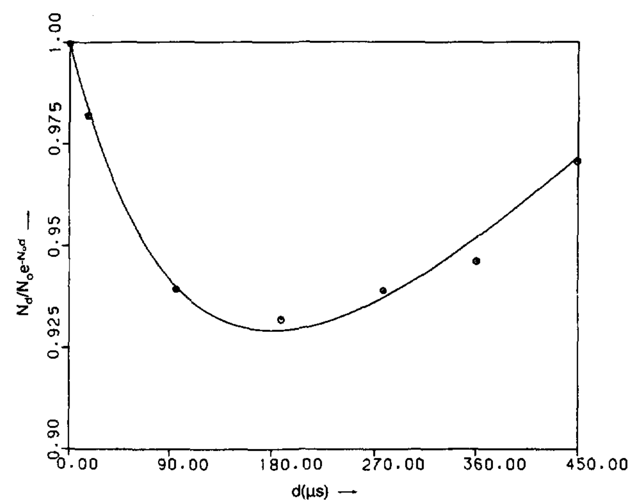

Figure 4. Variation of N_d/N_o exp($-N_o d$) with dead time of the extendable type. The curve is obtained by fitting the experimental points to Eq.(11).

Fig. 4 shows a plot of N(d) / {N(O)exp(-N(d))} against d for the extendable dead time case. (For uncorrelated pulses this graph is expected to be a horizontal line passing through (0,1)). The circles represent experimental data while the curve is the best fit to Eq. (11) with three terms of the expansion in moments. The neutron multiplicity data used as input in these computations are taken from Ref. 23 and are presented in Table IV. The parameters deduced from the fit are the spontaneous fission rate, the detection efficiency and the neutron die away time. The fit is quite good even with two terms of the expansion in moments and further improves on including the third term. This clearly suggests that we may estimate an additional parameter from the data viz the third factorial moment (m_3) of the source multiplicity distribution by treating it as a free parameter rather than as an input. The results are summarized in Table V.

Fig. 5 shows a similar plot for the non-extendable dead times. Here N(d) / {N(O) (1-N(d)d)} is plotted against d. As for the previous case the curve represents the best fit to the experimental data (circles) using Eq. (5). The parameters deduced viz the spontaneous fission rate the effi-

Table IV
Multiplicity Data for CF-252

$\bar{\nu}$	(m_1)	3.757
$\overline{\nu(\nu-1)}$	(m_2)	11.95
$\overline{\nu(\nu-1)(\nu-2)}$	(m_3)	31.6

Table V
Parameters Derived by Various Methods

S_f(sec^{-1})	ϵ	$1/\lambda$(μs)	m3
Experimental Method			
Extendable Deadtime			
8760±85	0.0746±0.0007	109±7	42±21
Non-extendable Dead Time (1st moment)			
10320±500	0.0627±0.003	114±17	—
Moments Method (Random Trigger)			
8740±119	0.0757±0.0010	—	49.2
8791±119	0.0747±0.0010	—	34.6

ciency and the neutron die away time are presented in Table V. The last two rows of Table V show the parameters derived using the moments method with randomly triggered intervals and without any dead time. While it was not possible to get the neutron die away time, (since the counting interval was kept fixed at 10ms) the third factorial moment of the multiplicity distribution of fission neutrons of Cf-252 was derived.

The error estimates for the parameters by the two dead time methods have been derived from the goodness of fit. For the moments method, the error estimate for s and ϵ is derived from the errors in the mean and variance shown in Tables III and IV. A similar estimate for M_3 (and hence for m_3) is not easy to obtain and the difference in the two values of m_3 computed using the data in Tables III and IV can be taken as a rough measure of the error in this quantity.

Figs. 6 and 7 show the variation of V/m with dead time of the extendable and the non-extendable varieties respectively. Here again the experimental data is shown by circles and the curves are obtained from Eqs. (12-14) and Eq. (6) (for the extendable and non-extendable cases respectively) using the parameters given in the first row of Table V. The V/m in the extendable case was obtained by numerical integration of Eq. (13) since analytical integration does not appear to be feasible.

DISCUSSION

It is seen from the results presented above that Eq. (11) which gives the variation of count rate in the presence of an extendable dead time filter fits the count rate data well and also gives s and ϵ in agreement with those obtained by the V/m (moments) method.

However, Eq. (5) does not fit the data for non-extendable dead times very well (as may be seen from the larger uncertainties in the parameters), and the deduced parameters also do not agree with those obtained by the moments method. This is not surprising since the simple formula (5) cannot be expected to be valid for large dead times. The neutron die away time however does agree with that obtained by the extendable dead time method.

Regarding the determination of the third factorial moment (m_3) of the multiplicity distribution of Cf-252, we see that large errors are possible with both the moments method as well as the extendable dead time method. Nevertheless, the estimates for m_3 may be said to agree with the value given in Table IV by both the methods within experimental error. The magnitude of this error is comparable to that obtained by Hage and Ciffarelli[11] with the moments method in an experiment on a Pu sample.

The techniques involving measurement of the second moment as a function of dead time are rather complicated from the point of view of deriving the system parameters

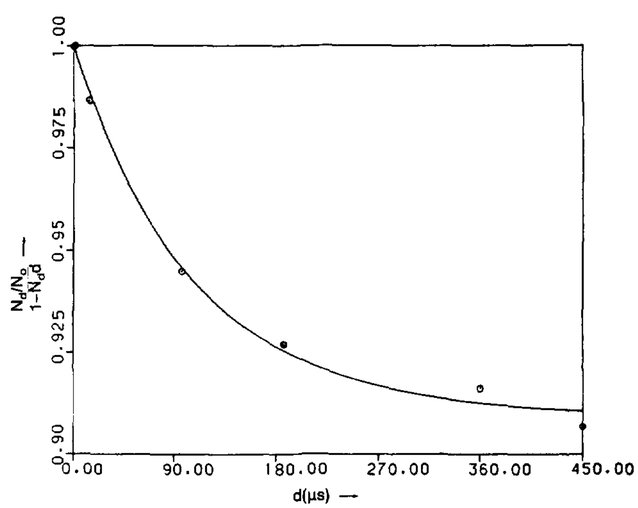

Figure 5. Variation of $(N_d/N_o)/(1\text{-}N_d d)$ with dead time of the non-extendable type. The curve is obtained by fitting the experimental points to Eq.(5).

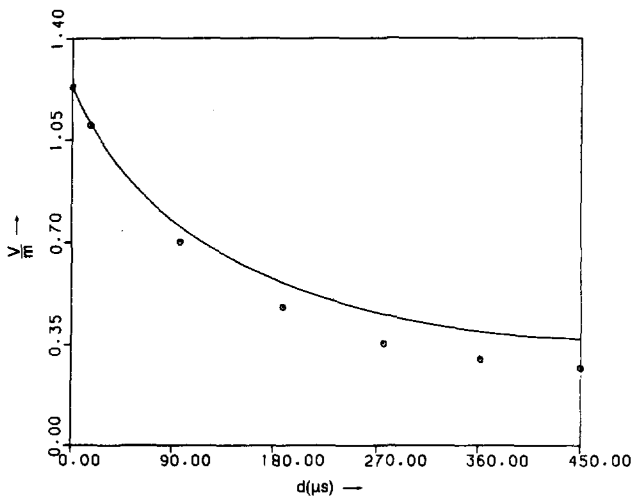

Figure 6. Variation of V/m with dead time of the extendable type. The curve shows the variation expected on the basis of Eq.(12) and the parameters s, ϵ, and λ obtained by the extendable dead time (first moment) method.

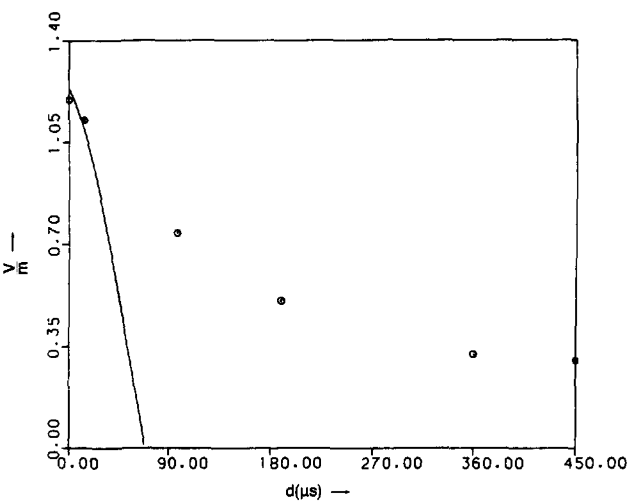

Figure 7. Variation of V/m with dead time of the non-extendable type. The curve shows the variation expected on the basis of Eq.(6) and the parameters s, ϵ, and λ obtained by the extendable dead time (first moment) method.

from the measured data. Moreover, these do not appear to be as accurate as the first moment method. While Eq. (12) agrees reasonably well with the measured data for all values of the dead time, Eq. (6) deviates considerably from the measured data for d>20 μs and the V/m even becomes negative beyond d>100 μs. This decides the limits to the validity of Eq. (6).

CONCLUSION

The applicability of the extendable dead time (1st moment method) to study neutron correlations and to obtain system parameters therefrom has been demonstrated. It has the potential of measuring higher moments of the multiplicity distribution, from which it is possible to derive the self multiplication probability in the case of large Pu samples using Bohnel's formulae (Eqs. 16-18). An advantage of this technique is that it requires very simple electronic hardware viz a dead time filter and a scaler. Online processing of data is clearly possible.

A possible objection to the utility of this method in practice is that the extendable dead time counter would recover rather infrequently at high count rates characteristic of large samples, leading to large statistical uncertainties. Thus the method would appear to be suitable only for small samples in which self multiplication is not very important. We examine this with the help of a numerical example.

The largest dead time gate that must be introduced in order that the third moment affect the count rate must be greater than (say two or three times) the die away time. At the same time the product N(O)d should not be much greater than unity since the count rate falls off roughly as exp(-N (0) d). This means that at high count rates a shorter die away time should be employed. This requirement is true for all other techniques as well. Now 1Kg of PuO_2 (20% Pu-240) gives a count rate N(O) of about 40,000 counts per sec in a detector having an efficiency of 10%. Employing a die away time of 20 μs and the dead times in the range of 0 to 60μs the largest value of N(O)d is about 2.4. This corresponds to a count rate N(d) of about 3000 cps in the slowest scaler and it should be possible to obtain accurate estimates of N(d) in a reasonable amount of time. Further since the variance (which is related to the statistical errors in the count rate) falls off faster than the mean as the dead time is increased these errors would tend to be somewhat lower. Thus the technique should be applicable up to about 1-2 Kg of plutonium but perhaps not for higher masses.

The variable dead time technique of the earlier days which employed a non-extendable dead time presents an extremely complicated problem from a theoretical point of view and it appears almost impossible to obtain any simple formula relating the count rate to the system parameters.

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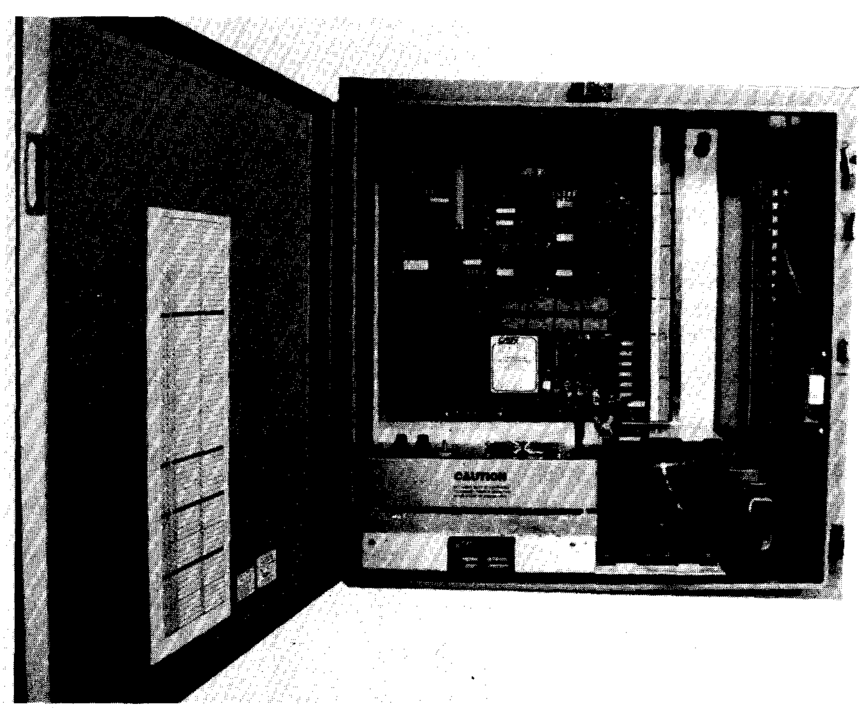

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