JNMM

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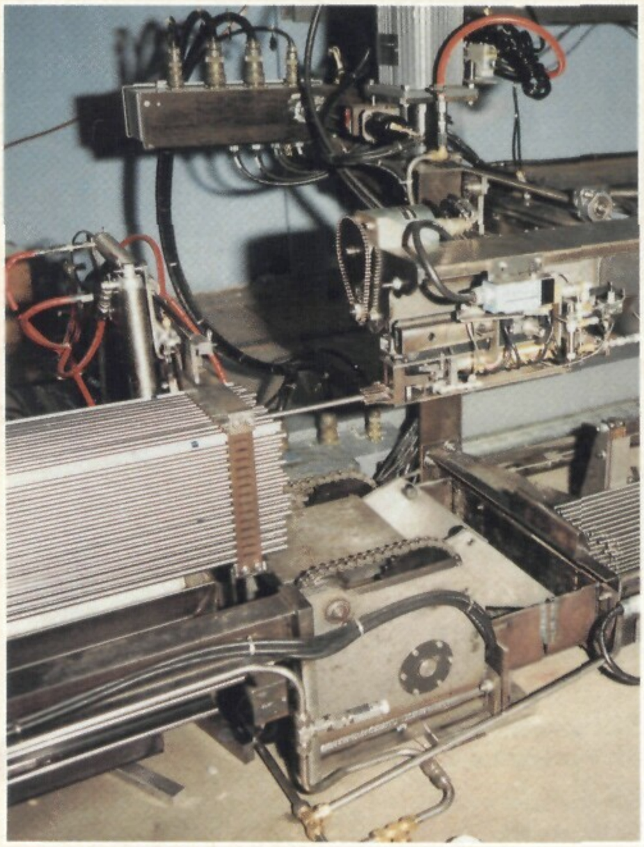

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On the Cover:
Dry rod consolidation hardware. Individual rods are pulled from the fuel assembly and loaded into consolidation canister achieving 2:1 volume reduction. Photo courtesy Idaho National Engineering Laboratory, U.S. Department of Energy.

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Filling in the Gaps

In 1982 the Institute decided to add two areas of interest, transportation and spent fuel management, to the four more directly related to safeguards; International Material Control and Accounting, International Containment-Surveillance, State's Material Control and Accounting, and State's Physical Protection. It is now the time to review this decision.

1982 was the year that the United States Congress passed the Nuclear Waste Policy Act which directed the U.S. Department of Energy to receive and to safely dispose of all high-level radioactive wastes, including spent reactor fuels, starting in 1998. This would obviously have a significant impact on the nuclear industry in the United States, on the transportation of radioactive materials, and on the management of spent fuel.

The new law required the electric utilities to pay one mil (0.1 cents) per kilowatt hour of nuclear generated electricity into a fund to pay the government for the proposed services. Within a few months the government initiated many system studies and R&D programs, some of which are described in the following articles. Many organizations became involved and waste-disposal became a popular subject for government sponsored and professional society meetings.

The Institute has not attempted to play a key role in this area, but rather to sponsor activities which other organizations have neglected. One example is the series of seminars on spent fuel management of which the seminar this January is the fifth. With its background in safeguards, the Institute has been able to bring together representatives from the government, the electric utilities, the organizations developing the equipment which the utilities will need, and those engaged in R&D for the Department of Energy, to discuss their mutual problems.

Spent fuel management and radioactive waste disposal are of concern

to every country with nuclear power reactors. Other nations have adopted similar or different policies. Some representatives of the European community, Japan, Taiwan, and China attended the recent symposium. Hopefully, such participation will increase since the experience gained in any one country should be of interest to others and many of the procedures under development will be employed, whatever the local policies may be.

The sessions on transportation and management of spent fuel at the INMM annual meetings have attracted wide attention and participation. They obviously serve a useful function and have increased the Institute's influence and membership. Also, as has been reported in the *Journal*, the Institute has assumed the responsibility for developing U.S. standards for the transportation of radioactive materials, which other organizations have neglected.

Thus far, the activities of the Transportation and Spent Fuel Management committees have concentrated primarily on U.S. domestic developments such as how to store the spent fuel accumulating at the U.S. nuclear power plants until the government will be prepared to receive it, and encouraging cooperation among the several interested parties, governmental and private. Except for France, temporary and longer-term storage of spent fuel is a universal problem.

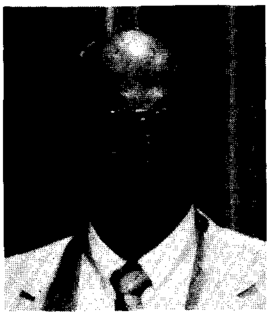
However, safeguards are relevant for the transportation of nuclear materials and for spent fuel management. Domestically there must be some acceptable means of accounting for the uranium and the plutonium and for protecting the highly radioactive materials in store or in transit from sabotage by sub-national adversaries. Internationally, means must be developed so that the International Atomic Energy Agency can verify that the sensitive contents of spent fuel assemblies are accounted for in the storage pools at reactors, if the fuel assemblies are to be consolidated for storage there, or if the assemblies are to be loaded into dry storage casks, or if spent fuel is to be permanently disposed of by burial deep underground.

Safeguards are an important feature of nuclear energy, as are safety, environmental impacts, economic factors, and public acceptance. Too often these are treated as separable functions. They are not separable. Those involved in each of these areas must learn to work with each other so that the elements, when combined, will effectively compliment each other.

William A. Higinbotham
Brookhaven National Laboratory
Upton, New York

In Search of Input

Greetings fellow INMM members and supporters. It seems that only a couple of months ago we were having our Annual Meeting. However, a whole year has gone by and the next Annual Meeting is nearly upon us. Las Vegas is going to be a good meeting. The agenda is outstanding and there should be plenty of everything for everyone. The meeting appears to continue to grow.



The INMM-sponsored workshops and training have been popular this year. I believe this signifies true need in the nuclear community and speaks well for the quality of our programs. At times it seems we do not respond quickly enough; however, it takes time and a lot of good volunteer effort to make these sessions a success. Members, continue to support us and also let us know what you need to make your job easier or make you a better professional.

We have been working very hard over the years to implement and use long range planning for the INMM. The effort has been successful and worthwhile. Given the dynamics and velocity of change in today's environment, I believe that our long range planning can be given a shot-in-the-arm, so to speak, with more member input. What are the mechanics for such input? The most straightforward is a call or note to the Long Range Planning Committee Chairman. When this does not seem to be most convenient, a note or phone call or even a chat in the hall with an officer or Member-at-Large. Suggestion boxes at the Annual Meeting provide easy input. The message is that we need continuous input from our members.

Again, let me say that the Annual Meeting this year at Bally's in Las Vegas will be the best ever. I am looking forward to seeing all of you there. Make your plans now to attend.

Charles M. Vaughan
GE Nuclear Energy
Wilmington, North Carolina

Waste Management Update

In 1982 the U.S. Congress passed the Nuclear Waste Policy Act, which instructed the U.S. Depart of Energy (DOE) to develop one or more permanent geological disposal sites for highly radioactive nuclear wastes and to start taking such wastes, including spent fuel, from the nuclear power reactors in 1998. The required DOE activities of transportation, temporary storage, repackaging, and waste burial were to be subject to review and licensing by the Nuclear Regulatory Commission (NRC). To pay for this, the nuclear electric facilities were required to pay 0.1 cents per kilowatt hour into a government fund.

The DOE promptly established an Office of Civilian Radioactive Waste Management (OCRWM) to identify and investigate suitable locations for permanent geological radioactive waste disposal facilities and to develop the transportation, spent fuel and waste treatment, and other techniques and systems it would need. The Nuclear Waste Policy Act specified that the DOE should make use of U.S. industry, as far as possible. Consequently, many of the DOE projects involve contracts with private industries and other cooperative undertakings.

Even before 1982, some of the storage pools at U.S. nuclear power plants were running out of storage space. As a temporary solution, the racks used to hold the spent fuel assemblies in the pool were redesigned to hold more assemblies closer together. It was obvious that long before 1998 additional measures would be needed at almost all of the operating nuclear power plants. Consequently, utilities individually, utilities in cooperation with other utilities and contractors, and utilities in cooperation with the DOE began exploring the possibilities: further compaction for storage in the pools, dry storage in a variety of ways at the reactor sites, and combinations thereof.

This series of INMM seminars on

spent fuel management was initiated in 1984 to provide a forum where the nuclear utilities, the companies which were developing the fuel compaction techniques and the dry fuel storage systems and those in the DOE who were developing the transportation, interim storage and fuel packaging systems could report on their developments and discuss mutual problems. Topics and papers selected by the INMM waste management committee have continued to be useful to the participants, as have been reported each year in JNMM. Last year, several of the papers were reprinted in JNMM as well as a paper which described the Canadian development with the International Atomic Energy Agency (IAEA) of techniques to provide assurance that Canadian spent fuel assemblies transferred to and stored dry in concrete containers continued to be accounted for.

This year's meeting began with a description of how and why the U.S. Congress amended the Nuclear Waste Policy Act of 1982. For a number of reasons the DOE was falling behind the schedule for selecting sites for geological repositories and a monitored retrievable storage facility. The Congress decided that selection of one site for a repository would be adequate for the near future and considerably cheaper than to attempt to reach agreement on and to evaluate three such sites. So it decided that the first site should be located in the state of Nevada, and that Nevada should receive financial compensation. It instructed the Administration to appoint a "negotiator" who would seek to negotiate for a monitored retrievable storage site with any potentially interested states or Indian Tribes. Any such agreement would be subject to the approval of Congress. The response of the nuclear industry to these amendments has been favorable.

The DOE program involves contracts with industrial firms and cooperative arrangements, as well as activities conducted at several of the government contractor sites. The current status of the DOE program to develop the spent fuel casks and transportation system and the program to develop the equipment for repackaging the spent fuel for burial in a permanent repository is reported in two of the following papers.

This year substantial progress was reported by the utilities in the dry storage and rod-compaction techniques. Several firms have developed casks to store spent fuel, with the ultimate goal of using such casks for the transport of spent fuel assemblies. Several of these designs have been licensed by the NRC for on-site storage, though not yet for transportation. One nuclear power station is already storing spent fuel in these containers and others are considering this option. Another approach is to transfer the spent fuel, in the pool, to a multi-element sealed container, which is then transferred into individual concrete cells, located nearby. Two nuclear utilities are beginning to use this system developed by Nutech Engineers, Inc.

A somewhat different approach has been developed by Foster Wheeler Energy Applications and GEC Energy Systems Ltd. The design is derived from a dry magnox fuel storage facility which has been in operation in the U.K. for 20 years. A very solid concrete structure protects the storage modules. Spent fuel would be removed from a reactor pool using an available transport cask, which is moved to the receiving/shipping station at the storage facility. There the assemblies would be removed and transferred into individual canisters in a storage module, all dry and by remote control. Any number of storage modules could be added.

During the past year several companies have been able to demonstrate 2:1 rod consolidation at reactor pools. Two utilities are now beginning to use this method. It may be possible in this manner to store the lifetime output in the existing pools. The possibility of storing the compacted fuel containers in dry storage casks is also being explored.

A considerable amount of information on the costs and radiation exposures which will be involved in the different dry storage and consolidation methods was presented at the meeting. However, the actual costs will depend on decisions yet to be made by the DOE and the regulatory agencies. For example, if the dual purpose storage and transportation casks should not be accepted for transportation, it would be necessary at some future date to retransfer the assemblies from the dry storage casks into DOE transport casks. Ultimately, the DOE may compact the rods for permanent geological disposal. If the compaction procedure employed by the utilities now should eliminate the need for subsequent repackaging, the DOE system could also benefit from this operation. Several papers discussed how future government decisions may ultimately affect the cost of the decisions that the utilities must make today.

Finally, the subject of accountability was discussed — accountability measures which the Nuclear Regulatory Commission and DOE may impose, and the accountability measures which the United

States should require to support international (IAEA) safeguards. To a considerable extent it appears that the objective will be to maintain continuity of knowledge of the information which exists at a reactor on the content and burnup of the individual spent fuel assemblies as they are transferred from a pool to dry storage, or the rods from two or more assemblies are consolidated into fewer containers and sealed. Special emphasis will have to be placed on containment and surveillance measures and remote monitoring, taking advantage of the details of each transfer system which must also be employed for safety and effective operations.

William A. Higinbotham
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Separating Fact from "Faction"

Chernobyl
Frederick Pohl
Bantam Books,
New York, 1987
(hardcover, $18.95)

It is not customary to review fiction in this space. However, the present work, a novel by the science fiction writer Frederick Pohl, lays claim to our attention by taking as its subject the Chernobyl disaster. In the two years since that unparalleled nuclear accident, reams of accounts have appeared — journalistic, technical, and political. To my knowledge, this is the first fictional treatment.

The literary genre that adopts a fictional approach to recent events has been described by the coined term "faction" (which will be used here with quotation marks to distinguish it from the real word meaning something altogether different). It differs from traditional historical fiction in that the latter deals with events whose impact has been softened by time. The supreme example, of course, is Tolstoy's *War and Peace*, published some 50 years after Napoleon's invasion of Russia. "Faction", on the other hand, describes events so recent they are still news. A writer who undertakes such an approach risks involving himself in current controversy, and exposes himself to the double jeopardy of having his work judged not only on its literary merits but also on its fidelity to the facts.

The story of *Chernobyl* begins on Friday, April 25, 1986, the day on which the ill-fated experiment that caused the accident began, and ends on Friday, May 23, four weeks later, with the outbreak and extinguishing of a new fire in the entombed plant. The scheme adopted by Pohl is to present the drama of the accident and its aftermath through the stories of three workers at the plant: Simyon Smin, the Deputy Director of the Chernobyl Nuclear Power Station, Leonid Seranchuk, the chief "hydrologist engineer," and Bohdan Kalychenko, a reactor operator. Since the Director of the plant is merely a political figurehead whose main functions are to entertain visiting dignitaries and represent the power station at formal meetings (at the time of the accident he is in Moscow, cultivating his higher-ups in the Ministry of Nuclear Energy), Smin is the *de facto* manager of the plant.

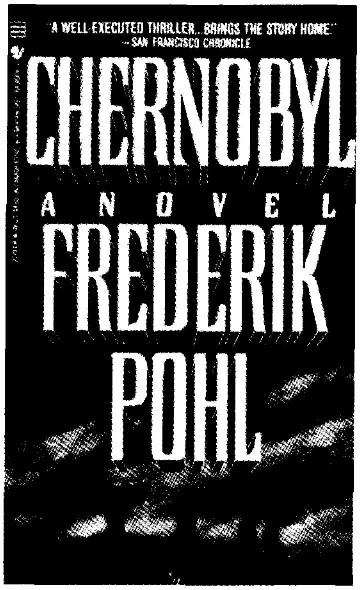
Smin and Seranchuk are Stakhanovite heroes of the Gorbachev era. Brave beyond the call of duty (a tank commander during World War II, Smin was grievously wounded and decorated for valor), modest, hardworking and competent, they are aware of the shortcomings of the Communist system, but, like Gorbachev, are loyal to the basic tenets of Marxism-Leninism. Thus, to get acceptable materials for the key areas of his plant, Smin has had to accommodate the managers of cement works and steel plants by also accepting the substandard products manufactured hastily at the end of the month in order to meet production quotas. As a result, he unjustly becomes the target of an expose in the Soviet magazine *Literatura Ukraina*. He accepts this scapegoat status stoically, as a necessary condition for getting his job done.

Seranchuk, whose previous experience had been in a peat-burning power plant, is a rough-hewn type who worships his boss, Smin. Off-duty the night of the accident, he is unable to sleep because of uneasiness about the coast-down experiment, and returns to the plant just in time to see it erupt in a ball of orange-white flame. Smin, too, has been off duty entertaining his visiting American cousin and his wife in his mother's flat in Kiev, but races back to the plant in response to an emergency call. Both he and Seranchuk perform heroically in trying to contain the disaster, but are overcome by massive radiation exposures. Along with roughly 300 others, they are rushed to a specially equipped hospital in Moscow. There, the desperately ill Smin is visited first by members of the KGB, who are supposedly conducting an objective investigation but seem to be more interested in finding a scapegoat, and later, mysteriously, by two members of the all-powerful Central Committee, obviously Gorbachev allies, who want to get at the truth but are forced to move with extreme caution.

Kalychenko, the operator on duty when the disaster strikes, escapes the ensuing holocaust another way — he runs away to the nearby town of Pripyat. Feigning injury, he joins the inhabitants of that town when they are finally evacuated.

There are a number of subplots and minor characters designed to provide a background of current events. Smin's eldest son returns demoralized from duty as a helicopter pilot in Afghanistan, where he has picked up a drug habit (shades of Vietnam!), and runs afoul of the police. Seranchuk's wife, obsessively jealous of her husband's attractiveness to other women, is found to have her own deep, dark secret. A mysterious anonymous document calling for drastic reform of the Soviet system circulates surreptitiously among the highest echelons of officialdom. And lurking everywhere are the watchful agents of the KGB

The book is at its best in its vivid depiction of the accident itself, the chaos and confusion of the fire-fighting efforts, and the massive evacuation of the 30-kilometer zone surrounding the plant. As he explains in an afterword, Pohl visited the Soviet Union and interviewed scores of participants and eye-witnesses to get the information on which these descriptions are based. In this, as an expression of the new policy of *glasnost*, he had the help and cooperation of the leadership of the Union of Soviet Writers, who, he claims, imposed no restrictions on whom he might see or on what he might write.



From the frank criticism of the events leading up to the accident and of certain aspects of the Soviet system one can believe him.

Nevertheless, this is a book that Gorbachev could give to his daughter (if he has one) without worrying about its effect on her ideological purity. Through his characters' discussions and arguments about nuclear power and East-West differences, Pohl manages to distribute credit and blame so evenhandly between the U.S. and U.S.S.R. that there appears to be no compelling reason to choose between them. If anything, Americans come off worse. Smin's cousin, a TV producer, is crude, boastful, and pushy, and his actress-wife is spoiled and self-centered. The Georgia-born science attache at the U.S. Embassy is a cynical lecher who is always making advances to the black female cultural attache, the only sympathetically-portrayed American in the book.

The tendency of the main characters and some of the subordinate ones to represent stereotypes is the greatest weakness of the novel. Smin and Seranchuk are so noble and self-sacrificing as to be unbelievable. Smin, in particular, borders on the saintly. Ordered to the hospital for treatment of acute radiation sickness (earlier, he had concealed the severity of his exposure by surreptitiously exchanging his discharge dosimeter for a fresh one) he protests, "There is work that I must do here". At the hospital he sees the bright side of his total loss of appetite, a common symptom of acute radiation sickness. "I have always been too fat," he says to Seranchuk, who has been trying unsuccessfully to feed him. "To lose a few kilos is no bad thing". One half expects to hear him say, as in the old joke about the stiff-upper-lip British colonial officer pinned to a tree by native spears, "It only hurts when I laugh".

Seranchuk, a younger version of Smin, is another paragon of self-denial. During the frenzied efforts to secure the burning reactor he demands to be allowed to stand double duty, and when he is temporarily evacuated to a town 100 kilometers away, he frets to his wife, "I should be at the plant." His response to praise for his courage is to blush and shrug it off with the modest disclaimer, "Someone had to do it," and he objects to being kept at the crowded hospital in Moscow on the grounds that others need the space more.

Some of the supporting characters are portrayed more vividly, particularly Smin's aged mother, Aftasia, a prominent Old Bolshevik who is disillusioned with the regime. In one of the most effective passages in the book, during a visit with her American counsins to the ravine at Babi Yar she conveys grippingly the slaughter she witnessed by the Nazis of virtually the entire Jewish population of Kiev, 100,000 men, women, and children in a single day.

On the technical side, Pohl has obviously done a great deal of background research, and presents the sequence of major events reasonably accurately. However, there are enough errors to make one wonder how much of what he read he really understood. For example, he first describes the purpose of the coast-down experiment as being "to see if useful power could be generated from the heat usually wasted when a nuclear reactor was down for maintenance" — that is, as an exercise in energy conservation. In fact, as he recognizes later, the objective was to determine whether, following a loss of off-station power and a scram, the residual kinetic energy of the spinning turbines could be used to keep the coolant circulating until the emergency diesels could be started.

More serious, Pohl implies that the Chernobyl reactor, like Western light-water reactors, had a strong and continuous containment. Actually, it differed crucially, in that its containment did not include the top of the reactor, which was covered by a thermal and biological shield penetrated by the fuel channel ducts (to allow access to the fuel by the overhead refueling machine) and held down essentially by its own weight. Since it also supported the weight of the fuel channels and the control-rod drives, when it was blown aside it lifted up some of the control rods and ruptured the fuel channels, exposing the disintegrating fuel to the atmosphere.

Many less serious errors occur as well. Moderation is explained as a means of making the reactor more controllable by slowing down the neutrons and thus reducing the speed of the fission reactions. If that were so, fast reactors would be uncontrollable. In actuality, it is the emission of delayed neutrons by short-lived fission products that makes control possible in both moderated (thermal) and unmoderated (fast) reactors.

Elsewhere, it is stated that it takes "weeks" for a reactor to recover from xenon poisoning (hours or, at most, days would be more like it), that half of the activity of the cesium and strontium fission products from Chernobyl would decay in a few months (the cesium half lives are measured in years), that the half-life of plutonium is 240,000 years (that of PU-239 is 24,000 years), and that uranium's radioactivity is due to spontaneous fission, which goes on "all the time" (spontaneous fission in uranium is much rarer than alpha decay, which accounts for most of its radioactivity).

Many of his errors are in such a direction as to make the accident seem even worse than it was, and he is given to vague but melodramatic hints of greater horrors yet to come. Thus, the encapsulated core "would stay there -still hot, still deadly — forever". All of us, he warns, will be breathing in some of the "Chernobyl molecules — for the rest of our lives!" (The consequences are not stated here, but implied; in fact, they are completely negligible.) Seranchuk's wife — a doctor — fears the possibility of radiation damage to her fetus, conceived *after* she and her husband had left the area. (It is worthwhile noting in this connection that, as of a year after the accident, no health effects were observed among 300 infants born to mothers exposed to Chernobyl's fall-out.)

The prevalence of these generally one-sided errors and exaggerations makes one wonder to what extent they reflect an underlying attitude toward nuclear power. The answer may be provided by a jarring passage toward the end of the book. During an interview on an American TV news program a spokesman for an organization called "The American Association of Nuclear Engineers" makes a statement so arrogant, truculent, abusive to anti-nukes and flagrantly derisive of the Russians as to be completely unbelievable to any knowledgable person. No such organization as the American Association of Nuclear Engineers exists (I checked) nor, as far as I know, was any such statement, or one even remotely resembling it, ever issued on behalf of any of the professional nuclear societies, either here or abroad, but since Pohl also cites statements actually made by others in the wake of the accident, how is the average reader to know this? All of which underscores the basic problem with "faction" — you never know where fact leaves off and fiction begins.

Judged, then, by the dual standards which Mr. Pohl implicitly accepts by writing in this genre, *Chernobyl* gets mixed grades. As a factual account of the causes and consequences of the accident, it earns the author plaudits for attempting to master a complex and difficult subject and present it in understandable terms to a lay audience, but it is marred by many errors and a tendency to cater to that audience's prejudices — prejudices which one suspects the author shares. As a novel, it has gripping action and vivid scenes, but with a few exceptions its characters, like those in Soviet tractor epics, are more symbolic than real. To sum up, although it's a fast and easy read, *War and Peace* it's not.

Reviewed by
Eugene V. Weinstock
Brookhaven National Laboratory
Upton, New York

Vienna

The membership of the Vienna Chapter of the INMM stands at 78 and still growing. The 1987-88 Executive Committee consists of Ed Kerr, chairman; Winston Alston, vice-chairman, Juri Janov, treasurer; Neil Harms, secretary; Reza Abedin-Zadeh and Tom Shea, members-at-large; and Samir Morsy, past chairman.

INMM activities started the season with the traditional get-acquainted Heuriger in October. The new wine and friendly atmosphere of Weingut Diem was enjoyed by new and old members alike.

November was the start of the regular luncheon meetings at the Vienna International Centre, and the first topic was "ISPO — Lessons from the 1st Decade — Goals for the 2nd." The program was presented jointly by Chris Kessler, Department of State, and Leon Green, Director, ISPO.

Chris discussed the evolution of the POTAS program from the initial five year — $5 million proposal to the current stable expenditure of $6 — 6.5 million per year in its 11th year of operation. The focus is now on implementation tasks, e.g. providing training in maintenance and use of equipment, and the appropriate documentation to accompany this training. He described POTAS success in strengthening the area of equipment procurement permitting the Agency to substantially expand its inventory. The present need involves developing an infrastructure to handle the equipment and to relate to the needs of the inspector in the field.

Leon continued with some specific examples of the organized approach to transfer technology and equipment to the Agency. He listed conditions such as reliability, ruggedness, ease of use, reasonable cost, and ease of maintenance as important criteria for supplying new equipment to the Agency. He cited the example of the MIVS TV system developed for the Agency by Sandia National Laboratory as a good example of high quality equipment.

Chris concluded with a brief glimpse into the future. The zeroreal-growth budget is going to be with us for the near future. He emphasized the need for the Agency to establish priorities, for the 12+ support programs to increase coordination of efforts to be of optimal use to the Agency, and finally to refine the approach for training to make it more selective and beneficial to the Agency.

The remarks of Messrs. Kessler and Green were enthusiastically received by an overflow crowd.

The January luncheon speaker was Jon Jennekens, Deputy Director General for the Department of Safeguards since July 1986. Jon spoke to an audience of more than 70 members and provided a perspective of his day to day activities — from dealing with the role of Agency safeguards in relation to various articles in the international press, to the intensive preparation of budgetary proposals and attempting to estimate the uncertainties that are a part of any predictions. He described the Department of Safeguards as having 113 different customers, i.e. Member States, and that his personal goal was to communicate to this group the high standard and cost-effectiveness of safeguards performed by the staff of the Agency.

Jon concluded his remarks by forecasting a modest growth of about six percent in the 1989-90 time frame necessary to maintain the high standard of Agency safeguards.

As a final note, the Vienna chapter has proposed an amendment to the Constitution and Bylaws of the INMM to request representation for the Vienna and Japan chapters on the Executive Committee. This issue is currently under consideration by the Executive Committee.

Neil L. Harms
Secretary
INMM Vienna Chapter
International Atomic Energy Agency
Vienna, Austria

Japan

The Japan Chapter's 9th Annual Meeting will be held June 2, 1988, preceded by a full day of technical presentations.

We are pleased to report that future issues of JNMM sent to the Japan Chapter will include abstracts translated into Japanese.

The following individuals were elected to the FY1988 Executive Board:

Mitsuho Hirata
Chairman

Tohru Haginoya
Vice Chairman

Yoko Iwamatsu
Secretary

Reisosuke Hara
Treasurer

Kazuhisa Mori
Executive Board

Yoshinobu Seki
Executive Board

Takeshi Osabe
Executive Board

Hiromasa Nakano
Executive Board

The Chapter currently has 120 members.

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

Central

The INMM Central Chapter's Annual Meeting is tentatively scheduled for October, 1988 in Lexington, Ky.

Roger R. Miller, Chairman
INMM Central Chapter
Martin Marietta Energy Systems, Inc.
Piketon, Ohio

Pacific Northwest

The Pacific Northwest Chapter held its spring dinner meeting on March 23. M. Preston Billings of the Security Applications Center at Westinghouse Hanford gave a presentation on "MSSA Preparation."

Plans are currently underway for a Safeguards Symposium to be held May 11 at Richland, Wash. This daylong event will be followed by a dinner meeting that evening.

Two major goals for the Chapter this year are the publication of an updated local membership directory and a membership campaign to seek new members. Working groups have been established to address these issues.

Debbie A. Dickman
Secretary-Treasurer
INMM Pacific Northwest Chapter
Pacific Northwest Laboratory
Richland, Washington

Waste Management

Spent Fuel Management Seminar

Much of the effort of the TWG during the period October 1987 through January 1988 was devoted to development of the Spent Fuel Management Seminar, which was held at Loew's L'Enfant Plaza Hotel in Washington, D.C., Jan. 20-22, 1988. Speakers were arranged, papers were received and the technical portion of the program was organized.

The seminar was a great success. We have managed to establish this seminar as the premier meeting on the subject of spent fuel management, as evidenced by comments received from the management of DOE, Office of Civilian Radioactive Waste Management, and industrial attendees — even in the face of several competitive meetings dealing with the same subject. The 120 attendees at the meeting included 14 from utility companies, 13 foreign representatives from 7 countries (including 2 from China, 1 from Taiwan, 4 from Japan, 2 from France, 2 from Spain, 1 from Canada, and 1 from Germany), 2 state/local governments (Nev. and Tenn.), and 7 press. The remainder were made up principally of suppliers, research firms, Federal Government, law firms, and architect engineers.

An example of the usefulness of the seminar was recently observed in reviewing the results of a key planning study regarding the estimated costs of spent fuel management alternatives at both reactors and DOE facilities, in which 19 out of the 43 total references cited were to papers presented at INMM spent fuel storage seminars.

The next seminar is scheduled for Jan. 10-13, 1989 at the same location.

TWG Meeting

A meeting of the TWG was held on Jan. 21, 1988 in Washington and was attended by T. Wood (PNL), P. Childress (B&W), K. Klein (DOE), and E. R. Johnson, J. A. McBride and N. B. McLeod (JAI). The meeting covered plans for the 1989 Spent Fuel Management Seminar, and discussions of additional waste-related programs that the TWG should initiate, including a monograph on spent fuel storage technology and development of positions on various aspects of safeguards in connection with the disposal of spent nuclear fuel.

Annual Meeting
The TWG has been developing interest in the preparation of papers on waste management for the forthcoming INMM Annual Meeting in Las Vegas. Several waste management sessions are planned for this conference.

E. R. Johnson, Chairman
INMM Technical Working Group on Waste Management
E.R. Johnson Associates, Inc.
Oakton, Virginia

Physical Protection

The Technical Working Group on Physical Protection held the "Security Forces Workshop" April 11-14 in Albuquerque, N.M. The Workshop was co-chaired by Frederick Crane, International Energy Associates, Ltd., and Dennis C.W. Wilson, DOE Central Training Academy. The Workshop focused on the qualification, training, operations and evaluation of security forces for federal and sensitive commercial sector facilities.

The Physical Protection and Materials Control and Accountability Technical Working Groups and the Safeguards Committee are planning an integrated workshop for 1989.

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

Materials Control and Accountability

The Technical Working Group on MC&A held the successful "Process Holdup of Special Nuclear Materials Workshop" in Rockville, Md., co-chaired by K.K.S. Pillay, Los Alamos National Laboratory, and Tom L. Brumfield, Martin Marietta Energy Systems, Inc.

Workshop session topics included: Holdup and Inventory Differences of SNM in Nuclear Facilities; Plant Experiences in Holdup Measurements; Unresolved Issues regarding SNM Holdup; Holdup Assay Techniques; and Use of Statistical Sampling and Modeling in Holdup Estimation.

Summaries of the 26 papers presented were collected in a Workshop Summaries Book. The Summaries books are available from INMM headquarters (312/480-9573). Complete information on how to obtain the Workshop Summaries is included in this issue of JNMM.

Selected papers from the workshop will also be published in the July issue of JNMM.

James W. Tape, Chairman
INMM Technical Working Group on Materials Control and Accountability
Los Alamos National Laboratory
Los Alamos, New Mexico

N14 Standards

N14 Procedure Notebooks have been assembled and will be distributed prior to the INMM Annual Meeting, June 26-29, 1988. The Committee is working to promote and publicize the Institute's development of standards in the areas of packaging and transportation.

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

N15 Standards

N15 highlights for the last quarter include: approval of a revised standard; standards are nearing approval as a result of maintenance efforts; writing groups have actively pursued the completion of proposed projects; DOE has concurred with implementation of a standard; and writing group volunteers have continued to surface.

Approved
N15.20, "Guide to Calibrating Nondestructive Assay Systems," was approved Dec. 17, 1987.

Maintenance of Standards
N15.18, "Mass Calibration Techniques for Nuclear Materials," was submitted to ANSI for approval in March.

N15.19, "Volume Calibration Techniques for Nuclear Material Control," was issued for revision balloting and submission for public review during February.

Proposed Projects
P/N15.51, "Guide to Measurement Control in an Analytical Laboratory," received approval for development from ANSI Nuclear Standards Board (NSB) after diligent effort from writing group chairman Charles Pietri.

P/N15.54, "Guide to Measurement Control for Radiometric Calorimeters," was submitted to ANSI for approval in February.

DOE Implementation

N15.10, "Classification of Unirradiated Plutoniun Scrap," was approved in January 1987, but it has not yet been implemented within the DOE Nuclear Materials Management and Safeguards System (NMMSS). Implementation was pursued through the Material Control and Accounting Branch, Office of Safeguards and Security, DOE-HQ. After field offices were queried and NMMSS personnel were questioned regarding implementation impacts, the decision has been made to implement the new codes at the start of FY 1989. To aid the implementation effort, N15 provided copies of the Standard to each of the DOE Field Offices.

Volunteers

Individuals have continued to call and express an interest in working on Standards development and maintenance. Writing groups are always looking for additional support. If you are interested in standards development, please contact N15 Vice Chairman Ken Byers (509-376-0311) for more details.

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

Safeguards

The Safeguards Committee met on Feb. 17, 1988 in the new offices of the Nuclear Regulatory Commission (NRC) at the Rockville, Md. location. The purpose of the meeting was to provide a forum for interchange of information between the NRC, the Department of Energy (DOE), and members of the INMM.

R. Burnett, Director, Division of Safeguards and Transportation, provided an introduction and a general review of current NRC activities. Transportation issues and the interfaces with industry were the active topics at NRC in the safeguards area. The selection of Yucca Mountain as the primary waste repository in the U.S. is focusing energy and efforts.

Discussions were held on the following topics: 1) NRC/DOE Comparability Rule, 2) Category II-3 Amendments/Transportation Rule, 3) Reporting Requirements Rule, 4) Access Authorization Policy Statement, 4) Regulatory Effectiveness Rules, and 5) DOE Safeguards.

C. Nelsen from the NRC provided an update on the NRC/DOE Comparability Rule. The Category I facilities are in an upgrade mode based upon six recommendations from a joint NRC/DOE review committee in February 1987.

C. Sawyer, NRC, presented a status of activity in the protection of spent fuel shipments. The NRC is reviewing 10 CFR Part 73 which is on the protection of spent fuel storage. A rule in this area has been deferred till 1989.

P. Dwyer described the activity on the Reporting Requirements Rule. The NRC has tried to simplify the rule by having only two (2) categories of reporting — 1) less than one (1) hour and 2) quarterly event logs. There is new guidance (Regulatory Guide 5.62, rev. 1) and NUREG 1304 in this area.

NRC's S. Frattali overviewed the status of the Access Authorization Policy Statement which only applies to power reactors. A proposed policy statement will be published within two (2) weeks. It includes information on background investigation, psychological evaluation, and behavior observation requirements.

R. Dube from the NRC/NRR described the current Regulatory Effectiveness Reviews (RER). This involves a site visit by a team of NRC Headquarters people to primary assess the vulnerabilities to an outsider threat.

D. Emon from the DOE presented an overview of current DOE safeguards activities. His presentation covered DOE orders, Material Control & Accounting (MC&A) Guides, and DOE MC&A meetings. There are currently 25 DOE orders under development and review. The new orders are performance oriented to include detection probability, quantities, times, diversion performance levels, and defense-in-depth requirements. Specific definitions have been provided for nuclear material safeguards categories. Three guides are being developed to be published for review in mid-May to become final in October 1988. These guides include the MC&A system, facility evaluation, and field offices. Meetings of the Safeguards and Security Directors of all the field offices will be held on Feb. 23, 1988 to explain the new DOE orders. A DOE only workshop will be held on May 19-20, 1988 to describe the DOE orders in detail. A series of training meetings will be held by the DOE to explain the new guides.

The Safeguards Committee's Low Enriched Uranium subcommittee was asked to review and document what happens during a new rule, its problems, troubles, and lessons learned. This may serve as valuable insight for future rule activities by the NRC. There appears to be an increase in free form assessment activities in reactors, safety, MC&A, and I&E areas. This may indicate a changing role in regulation of the industry. The next meeting of the Safeguards Committee will be at the annual INMM meeting in June, 1988.

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

Annual Meeting

The 29th Annual Meeting of the Institute of Nuclear Materials Management will be held June 26-29, 1988, at Bally's Hotel (formerly the MGM Grand), Las Vegas, Nev. U.S.A.

The Meeting will be the largest yet, with 201 technical papers presented in 28 sessions. The Meeting will also feature an exhibit of the latest equipment and services in safeguards and nuclear materials management.

The Committee Chairmen are:

Technical Program
C.E. Pietri, U.S. Department of Energy

Arrangements
M.T. Olascoaga
Sandia National Laboratories

Local Arrangements
B.E. Meurrens, EG&G Energy Measurements

Exhibits
J.C. Hamilton
Martin Marietta Energy Systems, Inc.

Registration
G.J. Carnival
Rockwell International

For more information see the notice in this issue of JNMM or call Beth Perry at INMM Headquarters, 312/480-9573.

John F. Lemming
INMM Vice Chairman
INMM Annual Meeting Committee Chairman
Monsanto Research Corp.
Miamisburg, Ohio

Dry Rod Consolidation Advancements in the OCRWM Programs

Margaret W. Fisher
U.S. Department of Energy
Idaho Operations Office
Idaho Falls, Idaho U.S.A.

ABSTRACT

Repackaging techniques for geological disposal of spent fuel are being investigated by the Idaho Operations Office of the U.S. Department of Energy. To start this program, the Idaho National Engineering Laboratory constructed remotely operated equipment to pull rods from assemblies and stack them horizontally in new containers. Forty-eight PWR assemblies were compacted in 1987. The data on rod pulling forces, external contamination, etc. are summarized. Five companies have submitted preliminary designs for prototype production systems. Three have been selected to complete these designs. One or two of the latter will be constructed and thoroughly tested in the next few years.

INTRODUCTION

As part of the Office of Civilian Radioactive Waste Management (OCRWM) initiatives, the Idaho Operations Office is conducting research and development activities on dry rod consolidation technologies. There are two programs at Idaho sponsored by OCRWM that are advancing our understanding of dry rod consolidation.

The first program that has been conducted is the Dry Rod Consolidation Technology (DRCT) Project. Using equipment designed by EG&G, Idaho, 48 Westinghouse 15 x 15 fuel assemblies were consolidated during the summer of 1987. This was the first successful horizontal dry consolidation effort in the United States.

The second program, the Prototypical Rod Consolidation Demonstration Program (PCDP), involves the development of prototypical dry rod consolidation equipment that will be able to consolidate the majority of the spent fuels in the United States. This equipment will be built and demonstrated so that the DOE can have operations data available to use in the DOE waste management plans. If the DOE decides to conduct rod consolidation as part of the storage and disposal activities, this prototype equipment will be the basis of future designs for that function.

This paper presents some preliminary results from the DRCT program, and provides a status report of the PCDP program.

DRCT PRELIMINARY RESULTS AND OBSERVATIONS

A photograph of the consolidation equipment is provided as Figure 1. A physical description of the equipment and a discussion of the functions can be obtained from Reference 1. One of the primary functions of this project was to gather data on the consolidation operations and equipment performance. This information is being supplied to other consolidation equipment designers for their use in optimizing their designs.

Following is a description of some of the more essential types of data collections and measurements that were taken during consolidation activities.

1. Fuel Rod Behavior

Visual observations of the fuel rods were documented using still color photography and color video equipment. Features such as crud deposition, scratches, bowing and twisting, bending and discoloration were recorded.

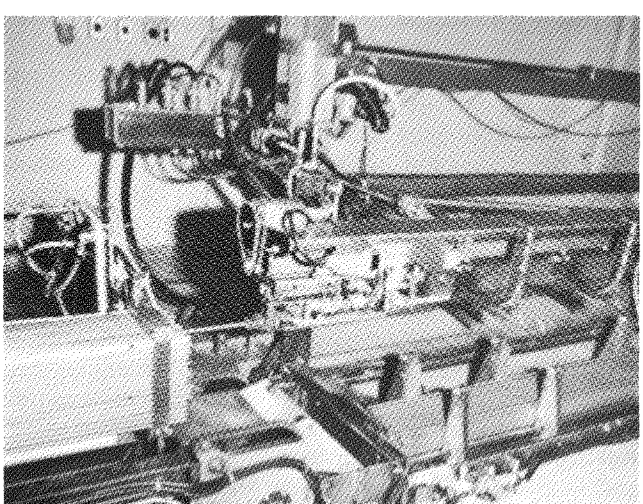


Figure 1. DRCT equipment pulling a rod.

2. Rod Pulling Forces

Rod pulling forces were measured for select fuel assemblies as the rods were pulled from the fuel assemblies. The measured forces included the initial break-away and continuous sliding friction force (as function of time/position) until the rod was free of the fuel assembly. The data were generated as a continuous analog signal proportional to the pulling force.

Additional data were recorded in conjunction with the rod pulling force data. Data identifying the fuel assembly, the rod x and y location in the fuel assembly, and the rod pulled position (z axis) were recorded and correlated with the pulling force measurements.

3. Crud and Zircaloy Collection and Characterization

A tray and vacuum system were incorporated into the consolidation equipment to capture crud during rod pulling. In addition to crud collection, zircaloy fines generated from cutting guide tubes and from rod pulling were collected. Gross radiation level surveys, and volume and weight measurements are being performed for each sample.

4. Airborne Particulate and Fission Gases

Five air sampling systems were employed in the hot cell to measure particulate airbone radionuclide concentration at different locations and distances from the consolidation equipment. The sample devices measured the airborne radioactivity in terms of activity per unit volume of air.

5. Fuel Assembly and Skeleton Activation, Product Content, and Contamination Level Measurements

The objective of these measurements was to determine the quantity of activation products generated in a fuel assembly skeleton as a function of neutron irradiation and to determine the amount of surface contamination produced from rod consolidation. Gross and isotopic gamma scans of intact fuel assemblies were performed in the Hot Shop using the existing gamma scanning equipment before the fuel assemblies were placed in the hot cell for consolidation. After these assemblies had been consolidated, the skeletons from the assemblies were again gamma scanned to measure the total activation product content, isotopic identification and distribution as functions of axial position in curies/kg of skeleton material. Swabs and metal samples were taken from several different locations on each assembly and measured to determine the loose contamination levels left on the skeleton and activation levels of the fuel skeletons.

6. Observations

In general, the equipment performed all its functions successfully. There were a number of minor equipment failures during the five month demonstration, most of which can be attributed to vibrational failures and radiation exposure problems. Since this equipment was to be used for a short demonstration, the various pieces of equipment were bolted and spot welded together. The vibrations due to the motors and the rod pulling forces caused misalignment problems to occur. The precise alignment of the various components was crucial for computer controlled movements and operations. Recovery from these problems was slow due to the high radiation environment and the need to perform remote repairs, but was accomplished without jeopardizing the demonstration.

The 48 fuel assemblies used were hand picked for this demonstration from fuel acquired from the Virginia Power Surry reactor and the Florida Power Turkey Point reactor, so all the assemblies were in good condition. The average burnup was 29,330 Mwd, and the fuels were between seven and fourteen years old.

The fuel rods were very well behaved during the rod pulling placement activities. While still in the fuel assembly, individual rods exhibited bowing along the entire length of the fuel assembly, or between several grid spacers. However, after each rod was individually pulled from the assembly, the observed bowing disappeared. The rods laid flat in the stacking array. There were several occasions when the rod gripper was not accurately aligned as it attempted to grasp a rod. On these occasions, the gripper head pushed the rod back into the assembly. The rod was already up against the lower end fitting, so the pushing force caused the rod to buckle. The operators were watching for these types of events to occur, so they were able to activate the emergency stop mechanisms. However, enough pushing force was exerted to cause a lateral rod displacement of about 16cm without apparent rod damage. This confirmed that the rods were very flexible, and could accommodate a certain amount of mishandling.

The upper end fitting removal was accomplished using an internal tube cutting device (see Figure 2). Observations were that the blades of the cutters wore out faster than expected (approximately 20% faster when cutting irradiated material than when cutting unirradiated tubes). Incomplete cuts of the 21 guide tubes in each assembly were also frequently noted. There was no detection device designed to indicate whether there was an incomplete cut made. So the first opportunity that the operators had to know if there was incomplete cutting was when the end box was trying to be pulled away from the assembly after cutting. A redesign effort for this type of cutter is needed. This problem was overcome during the demonstration by re-entering each tube and performing second cuts using the same cutter.

Rod pulling forces were less than anticipated. A rod pulling force of 50-100 lbf. was expected, however the typical rod pulling force was 27 lbf. The maximum pulling force required was 60 lbf; the minimum force was 16 lbf. The project expected to see some stuck rods with high break away forces, but there was no incident of stuck rods in the more than 9700 rods pulled. In all cases there was an initial break away force experienced followed by a constant force until the rod had cleared the last (top) grid spacer. This implies that out of the seven grid spacers, the top grid spacer exerts the majority of the friction force on the rods.

The drilling fines, cutting chips, and crud were collected using a filtered vacuum system after every two fuel assemblies had been consolidated. Figure 3 is a photograph of a typical filter pad with chips and fines from two fuel assemblies. These samples are currently being analyzed. Preliminary results show the following:

Sample Type	*Average Weight*	*Gross Radiation*
drilling/cutting chips	11 grams/cycle	4 R/hour
assembly crud-debris	1.5 grams/cycle	2 R/hour

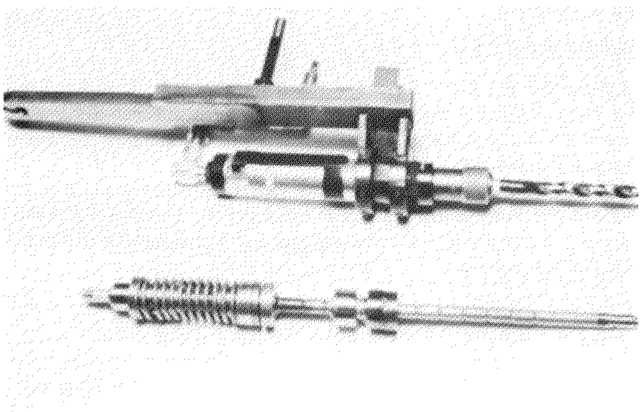


Figure 2. The top tool is a drill used to open the center guide tube. The lower tool is the internal tube cutting device.

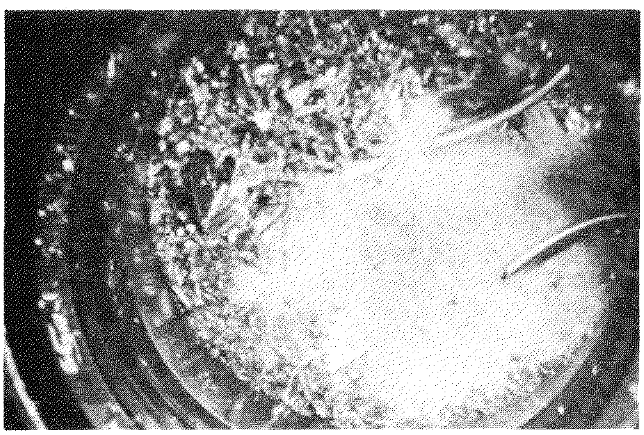


Figure 3. Drilling chips and ID tube cutting fines from two fuel assemblies. Filter pad is 2½" in diameter.

Contrary to expectations, the crud that was generated from consolidation did not migrate throughout the cell as an airborne contaminant. The crud that was generated remained on the clamping station where the fuel assembly was anchored during consolidation. The in-cell contamination and general background radiations readings were as follows: General fields were 150 mR/hour, and surface contamination typically ranged from 5 to 30 mR/hour on smears of 100 cm^2. Smears taken adjacent to the end fitting removal station were generally higher, with readings of up to 600 mR/hour. These represented the highest contamination readings in the cell.

The consolidation canister which was used for the fuel rods has cross-sectional dimensions equal to those of a Westinghouse fuel assembly. It was designed to hold 410 Westinghouse 15 x 15 fuel rods. This is equivalent to the amount of rods in two fuel assemblies plus two rods, i.e., slightly greater than 2:1 volume reduction capability. All 48 fuel assemblies were successfully packaged into 25 consolidation canisters, demonstrating for the first time a 2:1 volume reduction.

All of the observations and evaluations to date indicate that horizontal dry rod consolidation is a viable concept for volume reduction of fuel assemblies. For additional information on the results of this program, a formal EG&G, Idaho report will be issued by May 1988.

PCDP STATUS REPORT

As stated in the introduction, the Prototypical Rod Consolidation Program is another technology development activity on-going at the DOE Idaho Operations Office. This program is structured as a competitive design and procurement. Preliminary designs were developed by five U.S. companies from which three were selected to go on to final design.

Based on these final designs, one or possibly two designs will be selected to be fabricated and tested in a facility that will simulate a radioactive environment. The equipment will be tested in all normal and off-normal conditions using mock up fuel assemblies of five fuel types that are commonly used in the United States. At the conclusion of the testing programs, each of the two hardware designs will undergo a final evaluation. One set of equipment will be selected for a hot demonstration at the Idaho National Engineering Laboratory. The hot demonstration will utilize approximately 200 spent fuel assemblies that are representative of the PWR and BWR industry.

The status of the program is as follows: the three companies that were selected for the final design phase are NUS, Nutech Engineers, Inc., and Westinghouse Electric Corporation. All three companies have spent nine months developing their final designs for the rod consolidation equipment. Aspects of each design that had high risk factors associated with them were bench tested during this final design phase. The final design reports from each company were submitted to DOE-ID on December 22, 1987 for evaluation.

Each design was based on the same design requirements. The adequacy of these designs in meeting these functional requirements will be evaluated and the two best designs will be selected by May 1988. The fabrication phase (Phase III) for each of these two selected contractors is expected to continue for 18 months. Each of the selected contractors must procure, fabricate and install all the equipment into their own facilities for a cold testing program. The DOE will be providing mock up fuel assemblies to each contractor to put through the equipment to verify that the design works properly. A rigorous testing program will be followed during this phase so that validation of the performance of each contractor's equipment can be assured.

Due to the sensitivity of the competitive selection process that DOE is doing at the time of the writing of this paper, subjective statements about the pros and cons of each design cannot be presented. However, there are some general comments that can be made about the three designs.

Because DOE desires to have one set of equipment that can disassemble nearly all the fuel in the spent fuel inventory, the contractors have had to design equipment that can be adaptable to wide-ranging differences in the fuel characteristics. The end result is that all three designs attempt to use the simplest approach to various functions such as clamping, cutting, gripping and packaging of fuel.

The throughput rate required of this equipment is 750 MTHM/year. Each of the contractors has concluded that no matter which of the PWR or BWR types that are being processed, they must process between 14 and 16 PWR's per day and between 26 and 32 BWR's per day to achieve the throughput rate.

Each of the design contractors feels strongly about the need to physically restrain the fuel rods once they have been removed from the fuel assembly. Because of the large numbers of rods that will be in process, they want to assure that they lay flat and do not buckle or bend during the processing.

The functional design requirements stress that any off-normal event must be accommodated in the process cell. All time spent on the treatment of off-normal events considered so that the 750 MTHM/year throughput is still achieved. This requirement has caused the designers to think through all of the conceivable off-normal events and either design away the probability of occurrence, or design tools and develop procedures that will fix the problems. Likewise, any equipment malfunctions or repairs must be considered in the overall design and recovery plan. The emphasis must be considered in the overall design and recovery plan. The emphasis of highly-reliable, remotely-maintainable equipment that will be operated semi-automatically is the end point that all of these designs are trying to achieve.

REFERENCES

Fisher, M. W., 1987, *Dry Rod Consolidation Technology Project, Idaho National Engineering Laboratory,* Institute of Nuclear Materials Management Proceedings from Spent Fuel Storage Seminar IV, January 21-23, 1987.

Fisher, M. W., 1987, *Prototypical Rod Consolidation Demonstration Project, Idaho National Engineering Laboratory,* Institute of Nuclear Materials Management Proceedings from Spent Fuel Storage Seminar IV, January 21-23, 1987.

Margaret Fisher has been involved with nuclear waste management for the past eight years. As Manager, Spent Fuel R&D Programs with the U.S. Department of Energy-Idaho, her responsibilities include dry rod consolidation, secondary waste handling and packaging, and dry cask storage. As a registered professional geologist, she is actively involved with earth science issues for the DOE, in addition to her responsibilities in waste management.

Characterization of Activated Metals in Spent Fuel Hardware

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Pacific Northwest Laboratory
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ABSTRACT

Pacific Northwest Laboratory (PNL), for the U.S. Department of Energy (DOE), has been investigating the activation of spent fuel hardware in order to properly account for it in the federal waste management system. The paper presents a description and status of the program and tentative conclusions.

Pacific Northwest Laboratory, for the System Integration Program of the U.S. DOE's Office of Civilian Radioactive Waste Management, has been investigating the components that make up the structural portion of spent fuel assemblies. This investigation has been ongoing for the past few years and has included identifying the materials used, the quantities involved, and options for their disposal. The subject of this paper is to describe work that is currently being done at PNL to measure radionuclide activities in these activated metals.

In order to properly determine and evaluate the packaging and treatment options available for processing and disposing of activated metals, they must first be characterized. Part of this characterization is a determination of the radionuclides and their respective concentrations present in the activated metals. To date, this work has been primarily based on calculations using computer codes such as ANISN and ORIGEN2. The results of these calculations within the active (fueled) region of a reactor core are considered to be acceptable. However, much of the material that is of concern is just outside the active core region. The top and bottom end fittings are outside the active region, and there is also considerable support structure in these areas as well as other non-fuel bearing components that may require geologic disposal. The magnitude of the flux and the neutron spectrum is changing rapidly in these areas from active core values, and there is a large uncertainty on the accuracy of the calculations in these outer regions.

In order to validate the methodology being used to estimate radionuclide concentrations in these out of core regions, a sampling program was set up to compare calculations against laboratory measurements of activated metals. A number of samples have been obtained of activated metals from spent fuel and concentrations of radionuclides of interest are being determined. Calculations were made and the result of these estimates will be compared to the results of the lab analyses.

SAMPLES

During 1987, 38 samples of activated metals were obtained from 3 spent fuel assemblies. The fuel assemblies are described in Table 1. These fuel assemblies are representative of the type of spent fuel that must be accommodated by the federal waste management system and many utilities.

Table 1

Fuel Assembly Type	*Burnup (MWD/MTU)*	*Discharge*
Westinghouse 14 x 14	33,000	Oct. '81
Combustion Engineering 14 x 14	42,300	Apr. '82
General Electric 8 x 8	27,500	Apr. '81

Samples were obtained from each grid spacer in each of the fuel assemblies, and from both the bottom and top end fittings. The bottom end fitting is generally a single casting made of stainless steel. The top end fitting however, consists of several different pieces that are composed of Inconel and stainless steel. Samples were obtained that are representative of each of the materials of construction, as well as each of the main parts.

Each of the samples will be analyzed for a number of radionuclides. Of current interest to many people are the radionuclides that affect the waste classification of this hardware with respect to 10 CFR Part 61. The nuclides of primary interest are C-14, Ni-59, Ni-63, & Nb-94. Another nuclide of interest is Co-60, due to the handling problems that it can present. In addition to these radionuclides, there are number of other isotopes that will be measured.

These include Mn-54, Fe-55, Zn-65, Sb-125, Cs-134, Cs-137, and many of the transuranics.

In addition to each sample being analyzed for its radionuclide concentrations, an elemental analysis will be done to ensure that the specific amount of parent material that exists is known for each radionuclide. The importance of this cannot be underestimated if the results of these analyses are to be useful. As an example, the amount of Co-60 in a sample of steel can be readily measured. However the amount of Co-60 produced scales linearly with the initial amount of cobalt prior to irradiation. [This ignores the complication that Co-60 is also produced from nickel by the Ni-60 (n,p) reaction.] In reactor grade steel, specifications generally require the cobalt impurity level to be less than 0.10%. The cobalt level in the steel may actually run from about 0.02% to over the specification limit. In older steel, the cobalt content was not as closely watched, if at all, and significantly higher levels may be found. Therefore, in order to take the specific activity of Co-60 in one piece of steel and infer the specific activity in another piece, even with the same irradiation history, the relative values of the cobalt impurity levels must be known. Then a relationship can be developed between the amount of Co-60 present after irradiation and the initial amount of cobalt. This relation can then be used for any other material containing cobalt in whatever quantity.

This problem of relating the activation product to the parent product holds true for all radionuclides. The degree of the problem is small for activation of iron, nickel, and chrome in stainless steel and Inconel alloys, since the variation of these major constituents is relatively small. However, for minor constituents and impurities, this variation can be quite large.

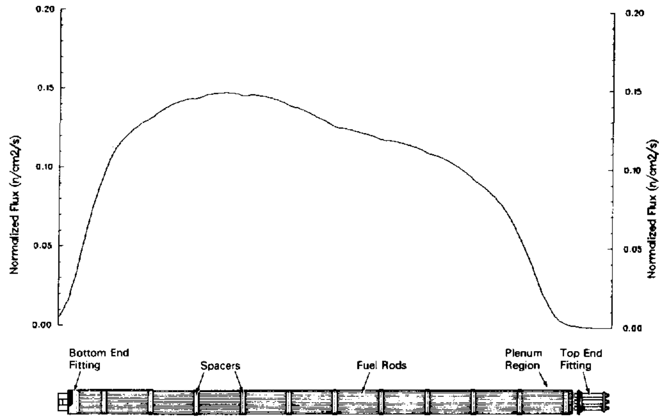

Figure 1. Normalized Flux for a Combustion Engineering Fuel Assembly.

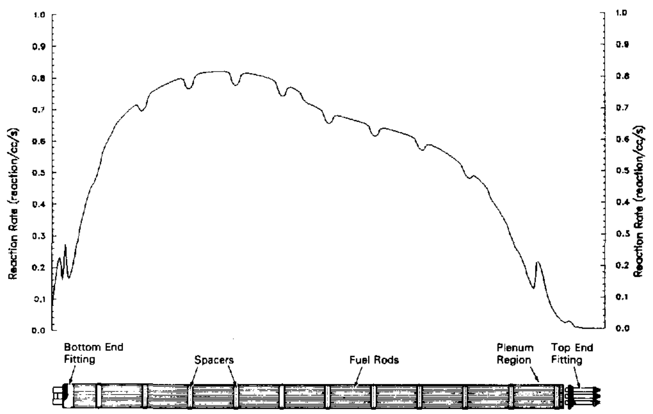

Figure 2. Co-59 (n,gamma) Normalized Reaction Rate for a Combustion Engineering Fuel Assembly

CALCULATIONS

To estimate the activation level in the active core region, a useful and commonly used tool is the ORIGEN2 computer code. The code, as part of the input, requires the mass of the material in which the activation is to be estimated and the elemental composition of the material, including any impurities. If the proper element composition is not specified, the code has no database to fall back on for default values. The internal cross-section libraries that the code uses have been developed for a variety of reactor types and burnups. They are all however only applicable to calculations over the average of the active core. If an estimate is required of a sample in a specific location either in the core or outside of the core, adjustments have to be made.

Within the active core, these are not very significant adjustments. However, outside the active core region, going axially up or down to the end fittings, or radially outward toward the core barrel, these can be significant. The magnitude of the total flux begins to drop off sharply upon leaving the fuel, and the neutron spectrum changes rapidly, strongly influencing reaction rates. This flux reduction is not dependent on the reaction rates of interest. For many of the reactions of interest, this will have the effect of increasing the spectrum averaged neutron cross-section. The manner in which the average neutron cross-section changes is *not* constant for all the reactions. This is because the energy dependent cross-section for each reaction is unique. Combined these two effects result in a reduction of the overall reaction rate that is different for each radionuclide.

In order to estimate the relative change in the reaction rates of interest, the ANISN computer code was used. ANISN is a one-dimensional neutron transport code. Models were set up for each of the reactor types that were being investigated, and calculations made of the relative flux and reaction rates for each of the radionuclides of interest. These were normalized to the core average, as calculated by ANISN. This normalization provided factors that could be applied to the results of ORIGEN2 calculations, which can then provide predictive values for areas outside the active core region. To illustrate this process, figure 1 shows the normalized flux profile for the Combustion Engineering model. Figure 2 shows the Co-59 (n,γ) reaction rate in the same model. Figure 3 is the result of dividing the values in figure 2 by the values in figure 1. This is the one-group cross-section (the total reaction rate divided by the total flux). Note that at the end fittings, the one-group cross-section shown in figure 3 increases. This is due to softening of the neutron flux and the resultant increase in

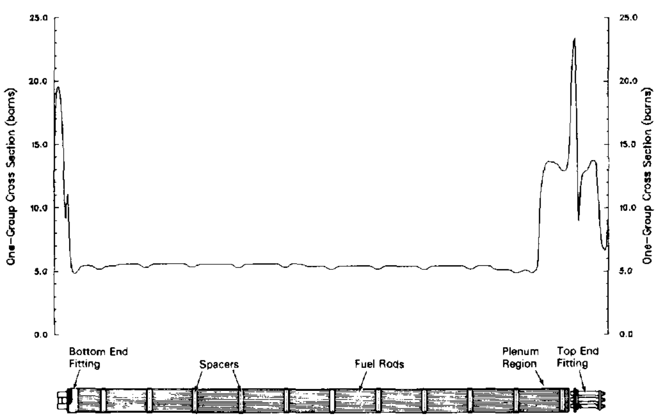

Figure 3. Co-59 (n,gamma) One-Group Cross Section for a Combustion Engineering Fuel Assembly.

the probability of capture. Given this information, and an irradiation history with which to make an ORIGEN2 calculation, one can estimate the activity of Co-60 in any part of the fuel assembly hardware. Similar data was generated for most of the radionuclides of interest. Unfortunately, not all the reactions of interest have well known cross-sections. An example of this is the (n,γ) cross-section for Ni-58 and Ni-62. There are a number of compilations for the cross-section in natural nickel, but not for the individual isotopes. This brings in considerable uncertainty for those reaction rates.

RESULTS

All the samples have been obtained and are currently being processed through the radio-chem labs at PNL. To date we have preliminary results for the first samples. These results indicate that a mixture of the end fittings (top & bottom) is greater than Class C. This is primarily due to Ni-63 in both the stainless steel and Inconel components and Nb-94 in the Inconel components. It was not expected that the Ni-63 would be as high as it has been measured.

Mr. Luksic received his Bachelor's degree in Mathematics from the State University of New York at Cortland in 1973, and his Master's degree in Nuclear Engineering from Brooklyn Polytechnic Institute in 1976. Since then he has worked in the nuclear field, for Burns & Roe Engineering and Westinghouse Hanford. For the past five years, Mr. Luksic has worked at the Pacific Northwest Laboratory, operated by Battelle for the U.S. Department of Energy. For the past four years, he has been investigating the activation of hardware in light water reactors, and the effect that this hardware will have on the Federal waste management system.

DOE Procurement Activities for Spent Fuel Shipping Casks

Eugene F. Callaghan
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ABSTRACT

This paper discusses the DOE cask development program established to satisfy the requirements of the NWPA. The program is designed to provide safe efficient casks on a timely schedule. The casks will be certified by the NRC in compliance with the 1987 amendment to NWPA. Private industry will be used to the maximum extent. DOE will encourage use of present cask technology, but will not hesitate to advance the state-of-the-art to improve efficiency in transport operations, provided that safety is not compromised. DOE will support the contractor's efforts to advance the state-of-the-art by maintaining a technical development effort that will respond to the common needs of all the contractors. DOE and the cask contractors will develop comprehensive and well integrated programs of test and analysis for cask certification. Finally, the DOE will monitor the cask development program within a system that fosters early identification of improvement opportunities as well as potential problems, and is sufficiently flexible to respond quickly yet rationally to assure a fully successful program.

INTRODUCTION

The Department of Energy (DOE) has been authorized under the Nuclear Waste Policy Act (NWPA) of 1982, and its 1987 amendment, to develop a national program for the disposal of spent fuel and high level nuclear waste. One of the responsibilities derived from NWPA is to provide a means of transporting the waste material from reactors to a final disposal site. It is also possible that some type of temporary disposal facility could be used in the program. Two significant requirements of NWPA are that private industry be used to the maximum extent possible, and that transport of spent fuel be subject to licensing and regulation of the Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT).

The Office of Civilian Radioactive Waste Management (OCRWM) will carry out the DOE transportation responsibility under the NWPA. The technical development of the transportation system or the "Transportation System Acquisition Task" is described in the report, "Transportation Business Plan" (DOE/RW-0046, January 1986). The report identifies two phases of the task. Phase I covers cask development, and Phase II covers operation of the transportation system. The present paper will concentrate on a segment of the Phase I aspects of the transportation task. Another important aspect of transportation involves institutional issues. The "Institutional Task" is concerned with effective participation of interested parties in the overall transportation program. The Institutional Task is not covered in this paper. The Institutional Task is described in the report, "Transportation, Institutional Plan" (DOE/RW-0094, August 1986), and more briefly in the Executive Summary (DOE/RW-0143, April 1987).

The Transportation Business Plan which forms the basis for the cask development strategy will be summarized before discussing where we are and where we are going. A brief look at present generation casks, and new generation casks to be developed under our program, forms the basis for discussing the OCRWM cask development strategy.

THE BUSINESS PLAN SUMMARIZED

The transportation system acquisition activities are divided into two phases. Phase I, which is the topic of this paper, covers the cask development program. Phase II covers transport operations. Phase I is further divided into four initiatives. Initiative 1 covers development of spent fuel casks for shipment from reactors. The casks developed under initiative 1 will handle most of the radioactive waste to be shipped. Initiative 2 will be completed if a monitored retrievable storage (MRS) facility is approved. A request for proposal (RFP) would then be issued to develop specialized casks for shipment of waste from an MRS to a repository. A third initiative covers the development and/or the procurement of specialty casks for shipment of non-standard fuel and radioactive hardware destined for repository disposal. The fourth initiative addresses development of defense high level waste casks.

Initiative 1, Phase I of the business plan calls for development of at least two casks for each of the primary transport modes (truck and rail). Because rail casks can also double as barge casks the two primary transport modes are referred to as truck and rail/barge. The business plan calls for a diversity of cask suppliers as well as a variety of cask designs. Variety in design and modal capability along with supplier diversity is intended to assure sufficient flexibility in the program to avoid adverse impact if any designs do not become available when needed.

WHERE WE ARE

We are now well into the first initiative of Phase 1 of the business plan. The DOE's Idaho Operations Office (DOE-ID) issued RFP No. 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. Following selection, but prior to signing any contracts as revised cask development strategy was established (November 1987).

The revised strategy is based on new storage technology developments, further study of life cycle costs, changing utility demand for DP casks, and budget constraints. The revised strategy calls for immediate development of the LWT and R/B casks. The OWT cask development is delayed, and the DP cask development is deferred. The same contractors were selected for both the LWT and OWT casks. This makes it possible to include OWT casks as options on the LWT cask contracts. The revised strategy assures at least two casks in each of the two primary transport modes and a diversity of suppliers.

WHERE WE ARE GOING

The revised cask development strategy has reduced the number of contractors for cask development to five. The contract negotiation process is now underway, and is expected to be completed in the Spring of 1988. Work will begin on each of the proposed designs as their individual contracts are signed.

Table 1
Typical Cask Development Schedule

Milestone	*Time (yr)*	*Times from Start (yr)*	*Contract Approval Complete (yr)*
Preliminary Design	1.0	0.0	1.0
Final Design	1.0	1.0	2.0
Prepare Safety Analysis Report	1.0	1.5	2.5
Certification by NRC	2.0	2.5	4.5
Frabricate Test Models	0.5	1.0	1.5
Design Verification Tests	0.5	1.5	2.0
Procure Long-Lead Items	1.5	3.0	4.5
Prototype Fabrication	2.25	4.5	6.75
Acceptance Tests	0.25	6.75	7.0

Although schedules are likely to vary with contractors, designs, and numerous unforseen situations we do have a fair idea of reasonable From-Reactor Cask Development schedule goals. The general development schedule goals are presented in Table 1. Schedules for specific cask designs will depend on contract approval dates.

Based on the expected contract approval schedules, we would see the cask contractors applying to the NRC for certification between the third quarter of 1990 and the second quarter of 1991. Ideally we can expect ten cask prototypes, two of each design, certified, acceptance tested, and ready to go at the end of 1995.

THE PRESENT GENERATION CASKS

Of the six light water reactor (LWR) spent fuel casks presently certified by the NRC, two are rail casks and four are truck casks. Three of the truck casks are quite similar in design. The rail casks range in weight from seventy to one-hundred tons, they carry between seven and ten pressurized water reactor (PWR) fuel assemblies, they carry between eighteen and twenty-four boiling water reactor (BWR) spent fuel assemblies. Only one of the truck casks is within legal weight limits, it weighs twenty-five tons and can carry either one PWR fuel assembly or two BWR fuel assemblies. The three remaining designs are variations of essentially the same design, each weighing about forty tons, and carrying three PWR fuel assemblies or seven BWR fuel assemblies. Table 2 presents some details of these cask designs.

Table 2
Present Generation LWR Spent Fuel Casks

Cask Model	*Mode*	*WT (tons)*	*Capacity PWR's*	*(assemblies) BWR's*
IF-300	rail	70	7	18
NLI-10/24	rail	100	10	24
NLI-1/2	truck	25	1	2
TN-8	truck	40	3	—
TN-8L	truck	40	3	—
TN-9	truck	40	—	7

Another cask that deserves mention, but which is not strictly speaking a spent fuel cask, is the Model 125-B cask. It is a 91-ton rail cask used to transport canistered dewatered core debris from the damaged TMI-2 commercial LWR. Although the Model 125-B was designed quite recently, and certified by the NRC in April 1986, it is quite similar to the older present generation cask designs.

The six spent fuel casks in Table 2 are designed for transport of spent fuel that has been cooled, or out of reactor, for one-hundred and fifty days or less. These casks are designed to dissipate high internal heat and shield against high gamma radiation. The high internal heat associated with short cooling time contributes to limitations on the number of fuel assemblies that a cask can accommodate. The high gamma radiation, also associated with short cooling time, requires the use of thick high density metal shielding which drives the cask weight up.

All of the present generation spent fuel casks identified in Table 2, and the Model 125-B waste cask have been certified by the NRC as meeting the requirements of 10 CFR 71. The NRC regulations are performance requirements which are similar to the transport regulations of the International Atomic Energy Agency (IAEA). Both the NRC and the IAEA publish guidance on various aspects of their regulations.

The transportation regulations (NRC's and IAEA's) identify nuclear performance requirements. The nuclear performance requirements address containment of radioactive material, protection from external radiation (shielding), and assurance of subcriticality. Cask designs must satisfy these nuclear performance requirements under conditions and/or tests that are intended to represent normal and accident conditions of transport. Demonstrations of compliance with the performance requirements can be based on actually performing prototype or scale model tests, but are more often done by analysis at present.

A NEW GENERATION OF SPENT FUEL CASKS

DOE's task under NWPA involves the transport of about three-thousand metric tons of spent fuel each year in From-Reactor casks. The spent fuel that is to be shipped is of the long cooled variety, probably being cooled (or out of reactor) between at least five and ten years before shipping. This fuel is much cooler than the one-hundred fifty day or less cooled fuel used as a design basis for present generation casks. The fuel will be both thermally and radioactively cooler than one-hundred and fifty day cooled fuel. The problem of heat dissipation becomes less severe. The gamma shielding requirements are reduced significantly. These factors contribute to higher payload to cask weight ratios.

Because this program involves movement of large amounts of spent fuel at high cost, it offers an incentive to look for and take advantage of all opportunities to improve safety and efficiency. The operating costs for the program will be much greater than the development costs, making it an ideal candidate for investment in technical development that will pay off later.

The means of improving efficiency of the transportation system is simple. We need to increase cask capacities. We feel that this can be done without sacrificing safety. As a matter of fact, if we develop a transportation system with individual casks that are as safe as present generation casks, but with larger capacity, we can expect fewer shipments, thereby, reducing environmental impacts, both radiological and non-radiological. OCRWM's original goals for increased cask capacities are shown in Table 3 along with the even higher capacities that are being proposed by the selected contractors. As the program develops we will continue to look for additional opportunities to increase cask capacities.

Table 3
Cask Capacities

	Present	*OCRWM (ref)*	*Proposed*
LWT with PWR	1	2	2 - 4
LWT with BWR	2	5	6 - 9
OWT with PWR	3	4	4 - 6
OWT with BWR	7	10	14
R/B with PWR	10	14	16 - 26
R/B with BWR	24	36	40 - 52

OCRWM's strategy for providing safe and efficient casks is to work from the existing technology base, and to identify promising areas where technical development may pay off in advancing the state-of-the-art. We will pursue those opportunities identified and incorporate the clear "winners" into the cask designs being developed. We hope that most of the design process will be based on present cask technology which may be supported by NRC guidance through Regulatory Guides or NUREG documents. In some cases we may have to look for guidance from other sources such as the IAEA, national standards, and technical literature. Some of these items that will advance the state-of-the-art will be new and unfamiliar to the NRC. The NRC will expect a complete and thorough demonstration and proof of the soundness and applicability of any advances that we propose. We will provide the demonstrations and proofs as needed. We will document our development procedures. We will use the technical development option extensively and appropriately in our certification efforts, and we will use a fully integrated program of test and analysis in this program.

Eugene Callaghan is a graduate of the United States Military Academy at West Point (1944), and the Harvard Business School (1953). He has recently retired after 39 years with the Federal government. His federal career includes both military and civilian service. He was with the Department of Energy's Office of Civilian Radioactive Waste Management for the past three years, and was a principal contributor in establishing the transportation business strategy now being followed.

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

Results of Studies on the Behavior of Spent Fuel in Storage

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ABSTRACT

Spent fuel integrity is a major consideration in licensing actions for spent fuel storage technologies. While wet storage continues to be the predominant U.S. spent fuel management technology, dry storage has been licensed and implemented in the U.S. Rod consolidation technology is being developed and demonstrated. This paper summarizes the status of technology trends and understanding regarding fuel behavior for the three technologies.

I. INTRODUCTION

This paper provides an overview of the pre-storage integrity of spent fuel cladding, effects of wet storage and dry storage on spent fuel integrity, and the status of fuel integrity during and after rod consolidation. The implications of the industry trend for increasing fuel burnups on spent fuel integrity is reviewed.

Investigations of the behavior of spent fuel in dry storage have led to identification of licensable storage conditions for inert atmospheres. The investigations consisted of laboratory tests, demonstrations with spent fuel in storage casks, and modeling of degradation phenomena. Licensed dry storage of spent fuel has been achieved in the U.S. by Virginia Electric Power Company for dry storage at the Surry nuclear power plant and by Carolina Power and Light for dry storage at the H.B. Robinson nuclear power plant. Laboratory testing and modeling to identify acceptable conditions for storing spent fuel in air are continuing. This report summarizes the activities of these investigations.

II. PRE-STORAGE INTEGRITY OF SPENT FUEL

The five fuel vendors that provide nuclear fuel for U.S. commercial plants indicate that as of the end of 1986 over 11.6 million fuel rods in over 97,800 fuel assemblies are in operation, or have completed operation, in commercial nuclear power reactors. Six fuel assemblies have attained burnups of 52,000 MWd/MTU or higher. The overall domestic fuel operation experience continues to be excellent: current fuel rod reliabilities are typically 99.99%, which corresponds to fuel rod failure rates of <0.01% (Bailey, November 1987). Average burnup levels have been increasing yearly, but fuel rod failure rates have not exhibited a similar trend; so far, the data indicate that extending burnup has not been detrimental to fuel performance.

Very few fuel assemblies have sustained major mechanical damage during handling (~250 in the U.S.) or been damaged during normal transporting operations (<23 in the U.S.). Only about seven fuel rods — out of a biased sample of fuel assemblies (those known to contain or suspected to contain defective fuel rods and were selected for examination, reconstitution, or rod consolidation) containing over 54,000 fuel rods — broke during handling, examination, fuel assembly reconstitution, or rod consolidation activities. Over 3,100 of those rods have been involved in rod consolidation operations. The likelihood of rod breakage was potentially higher with fuel rods having large cladding defects.

At the end of 1986, over 50,920 fuel assemblies (60% BWR type, 40% PWR type) were stored in U.S. facilities (U.S. DOE 1987). Nearly all of the assemblies are in wet storage. Among the more than 50,920 assemblies are 101 assemblies that are involved in R&D programs and some of the 101 assemblies are in dry storage. It appears that there are nearly 5,000 defective (i.e., failed or damaged) fuel assemblies among the more than 50,920 fuel assemblies in storage; however, of the 5,000 defective assemblies, only 35 require special handling and 1 required encapsulation (Bailey, November 1987).

III. WET STORAGE

The American Nuclear Society issued a policy statement (ANS 1986) in 1986 regarding the storage of spent nuclear fuel. The statement indicates that continued wet storage of spent fuel at nuclear power plant sites until the federal government accepts it under existing contracts with the utilities is safe, economical, and environmentally acceptable.

A. International Implementation

Wet storage remains the primary mode of storing LWR spent fuel in the U.S. Several countries rely solely on wet storage for fuel management. Finland, France, Sweden, the United Kingdom, and East European countries have recently commissioned wet storage facilities. The principal continuing study of cladding integrity in wet storage for Zircaloy-clad fuel is in Canada, although earlier studies were conducted in the FRG, USA, and UK. Canadian Zircaloy-clad fuel bundles stored up to 27 years will be examined in 1988. The last examination was in 1978 (Hunt et al. 1979).

There continues to be no evidence that LWR spent fuel with Zircaloy or stainless steel cladding degrades significantly during wet storage (EPRI 1986; IAEA 1982). The NRC has concluded (NRC 1978) that the storage of LWR spent fuels in water pools has an insignificant impact on the environment.

B. Failed Fuel in Pool Storage

Experience to date indicates that failed fuel has had a minimal impact on storage of spent fuel in water. A world survey showed that approximately 70% of storage pools store defective fuel assemblies on the same basis as intact assemblies; the remaining pools store fuel assemblies with failed rods in canisters. Further degradation of cladding defects during storage does not appear to be occurring (Bailey, November 1987).

C. Storage Optimization

U.S. utilities have moved aggressively to implement optimized utilization of wet storage technology, assisted in some areas by federal programs. Wet storage technology developments include: dense racking, double tiering, credit for burnup in rack designs, transshipment, impacts of extended burnup, rod consolidation, and pool decommissioning. Of these technology developments, only extended burnup and rod consolidation are anticipated to have significant effects on spent fuel integrity. Trends of integrity with extended burnup fuel were discussed in the earlier section on Pre-storage Integrity of Spent Fuel.

The expected impact of increasing burnup trends on storage requirements for U.S. LWR fuel are summarized in Table 1 (Johnson, Bailey, and Klein 1987). An advantage of extending fuel burnup is that it reduces the storage requirements for spent fuel. The impacts of extended burnup on fuel integrity, as presented in Table 1, may be realized in the additional cooling time required to maintain the fuel within allowable temperature limits during dry storage.

The impacts of rod consolidation on fuel storage are expected to be related to potential degradation of the spent fuel rods by the mechanical handling operations. These operations problems are manageable, if they are dealt with properly. No insoluble problems have been encountered.

D. Effects of Crud

Crud may be dislodged from spent fuel during water storage pool manipulations, but it rapidly settles and can be removed by system cleanup equipment to satisfactory levels. After long-term water-pool storage, once-tenacious crud has been observed to loosen from spent fuel and to have a tendency to peel. During one demonstration of PWR rod consolidation, crud dislodged from the fuel during rod-pulling formed a "cloud" in the storage pool water. Although annoying and causing delay in operations, the dispersion had little effect on pool radioactivity or personnel exposure. In a case involving shipping of BWR spent fuel, crud was dislodged and deposited into the shipping cask during shipping or subsequent operations, and, in spite of extensive flushing, a portion was retained in the cask. Later, during draining of the cask, crud was transferred to the drain hose, and because connecting the hose is a hands-on operation, radiation dose to the staff increased slightly. In another program involving fuel shipping, small amounts of crud were deposited in shipping casks during transportation and handling of PWR spent fuel used for tests of the CASTOR-V/21 dry storage cask. Crud was not a major problem and was restricted to the casks and easily handled. During long-term, low-temperature (229°C), single-rod dry storage tests, using PWR and BWR spent fuel rods, only a small amount of crud was dislodged. As a result, the researchers concluded that crud spallation during dry storage should be a manageable problem. From experience with wet storage, rod consolidation, transportation, and dry storage, it appears crud spallation can be managed effectively. (Hazelton 1987)

Table I
Effects of Extended Burnup
on U.S. LWR Spent Fuel Storage Requirements

Assumptions	*Additional Storage Requirements U.S. Reactor Pools, MTU (U.S. DOE 1985)* 1990	1995	2000
Utility projected discharges	759	3392	9332
Extended burnup[a]	572	2236	4742

(a) Assumes 3%/year increase in burnup to maximum of 45 GWd/MTU for PWR and 38 GWd/MTU for BWR fuel, starting in 1985.

IV. DRY STORAGE

Dry storage of spent fuel has been licensed in the USA with inert atmospheres. Dry storage has been supported internationally by spent fuel tests and demonstrations as well as by theoretical modeling. Testing and modeling of spent fuel behavior is under way to identify acceptable storage and handling conditions in air.

A. Inert Gas

The licensing of interim dry storage of LWR spent fuel requires assurance that release limits of radioactive materials are not exceeded. The extent to which Zircaloy cladding can be relied upon as a barrier for release of radioactive spent fuel and fission products, depends upon its integrity. Current experience indicates that the incidence of cladding failures during dry storage will be low; however, some fuel failures cannot be ruled out. Inspection techniques are available to detect most fuel assemblies containing reactor-induced defects, although elimination of every defective fuel rod cannot be assured. Even if fuel with cladding defects were placed in dry storage, or if defects developed during storage, such defects would not propagate if an inert cover gas is used. No cladding degradation mechanisms are known that can significantly degrade the Zircaloy cladding on LWR spent fuel in inert gases, if the initial fuel temperatures do not exceed approximately 380°C. The specific fuel temperature limits are sensitive to a number of fuel and storage parameters that are discussed in the following sections.

1. *Potential for Fuel Degradation in Inert Gas.* All known potential mechanisms that could degrade Zircaloy cladding on LWR spent fuel during inerted dry storage (IDS) were assessed (Cunningham et al. 1987). The mechanism with the fastest rate for cumulative damage that could lead to cladding failure was considered to be the controlling degradation mechanism. Because the kinetics for each mechanism were accelerated by temperature, it was concluded that degradation could be eliminated by controlling the temperature-time exposure of the fuel during dry storage.

2. *Degradation Mechanisms.* The principal potential Zircaloy cladding breach mechanisms during IDS were identified as creep rupture, stress corrosion cracking, and delayed hydride cracking. Stress rupture was found to be the primary cladding breach mechanism during IDS. Cladding breach due to stress corrosion cracking and delayed hydride cracking is not expected because the threshold stress intensity levels for these mechanisms are greater than those estimated to exist in spent fuel cladding.

3. *Stress Rupture Predictive Methodology.* The internal pressure from helium and fission gases is a source of hoop stress for creep rupture, if pressures and temperatures were sufficiently high. Consequently, it is of interest to predict the condition of spent fuel cladding during up to 40 years of interim storage. To develop this prediction, theoretical deformation and fracture theory was used to develop maps.

A typical fracture map is shown in Figure 1. The vertical axis is the natural logarithm of the ratio of the hoop stress to Young's elastic modulus. The constant hoop stress curves increase on this scale because of the temperature dependence of Young's modulus. The horizontal axis is the ratio of the absolute melting temperature to the absolute storage temperature. The shaded area includes the acceptable stresses and temperatures for storage up to 40 years.

Where available, experimental deformation and fracture data were used to test the validity of the maps. Predictive equations were developed and cumulative damage methodology was used to take credit for the declining temperature. A comparison of the spent fuel storage temperature history used for stress-rupture modeling (CSFM Model

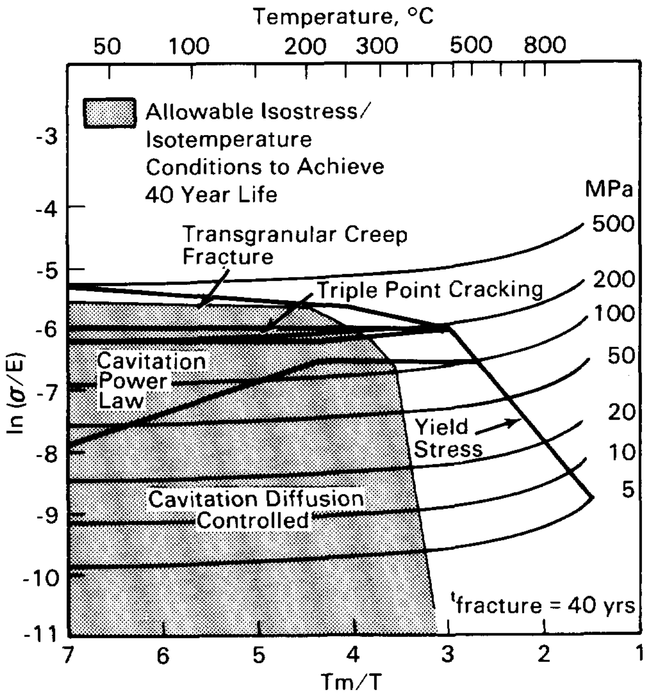

Figure 1. Fracture Map Showing Region in Which a Fuel Rod Subjected to Isothermal-Isostress Conditions Would Last 40 Years or More (Chin et al. 1986).

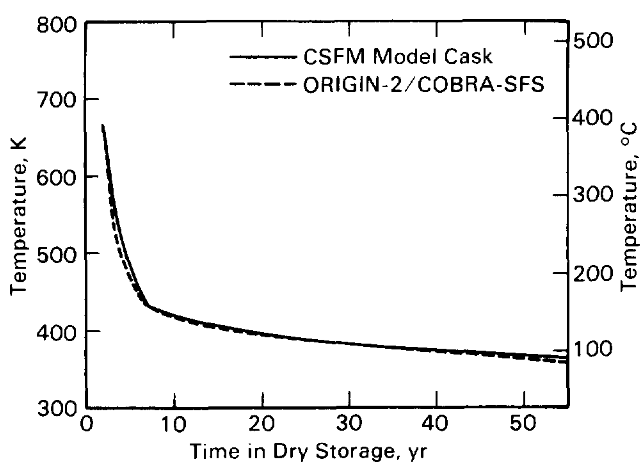

Figure 2. Comparison of Temperature Decay Predictions for the CSFM Model Cask for 1-yr-old Fuel to ORIGEN-2/COBRA-SFS Data for a Helium-Backfilled TN-24P Cask Containing 1-yr-old PWR Fuel with 30 MWd/kgM Burnup (Levy et al. 1987).

Cask) with the prediction by the ORIGIN-2/COBRA-SFS Code is illustrated in Figure 2. This methodology was then used to predict storage temperatures below which creep rupture would not be expected to occur, except in fuel rods with pre-existing flaws. Predictions were also made and compared with results from tests conducted under abnormal conditions (Stahl et al. 1985).

4. *Consequences of Cladding Failure.* Cladding temperature limits during spent fuel storage were derived on the basis that the probability of cladding breach by stress rupture should be less than 0.5% per spent fuel rod. The possibility of a few failures is considered acceptable because of the low consequence of a cladding breach. The mechanism of cladding breach by stress rupture leads to a pin hole penetration through the cladding. The pin hole breach is too small to compromise confinement of the fuel by the cladding. The driving force for continued cracking and enlargement of the breach is the internal gas pressure. The internal gas pressure is reduced in response to gas leakage through the pin hole breach until the mechanism for continued cracking is eventually deactivated. After the gas pressure is relieved, there are no further driving forces for cladding failure.

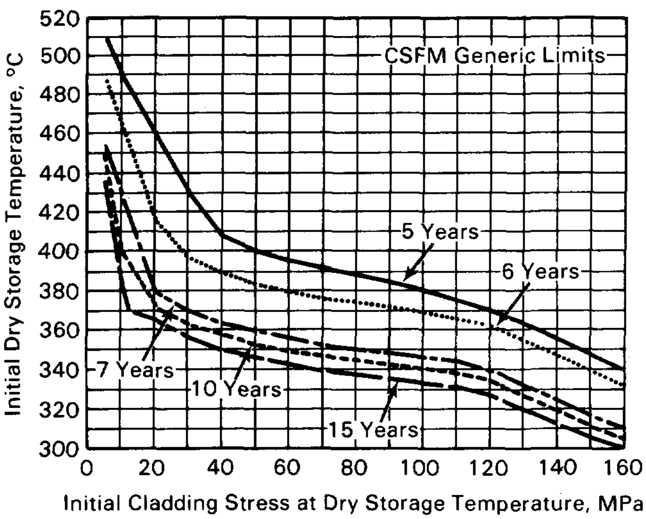

Figure 3. Comparison of CSFM Generic Limit Curves for 5-, 6-, 7-, 10-yr Fuel Age (Levy et al. 1987).

5. *Recommended storage conditions.* To accommodate the effects of variations in fuel design, burnup level, fuel age, and the geometry and makeup of dry storage casks, a model was developed and used to predict temperature limits for dry storage conditions. The model predictions are presented as a family of generic limit curves in Figure 3 for fuel that has been out of reactor for 5, 6, 7, 10, and 15 years. The effect of the internal gas pressure is reflected in the initial cladding stress at dry storage temperature.

The following is an example of how Figure 3 can be applied to select allowable initial dry storage temperatures for inerted-dry storage. If the initial cladding stress, due to internal gases at the storage temperature in fuel that had been cooled for 5 years, were 100 MPa (horizontal axis Figure 3), then the initial dry storage temperature limit would be 380° (corresponding temperature on vertical axis for curve labeled 5 years).

B. Effects of Gas Composition

During the dry storage of spent nuclear fuel, the fuel cladding must be protected against degradation and gross rupture that would release radioactive contamination in the storage environment. Current applications for the dry storage of light-water reactor spent fuel specify the use of an inert gas storage environment to prevent oxidation of cladding or exposed UO_2 fuel. However, even inert cover gases will contain residual reactive impurity gases from various sources.

A study (Knoll and Gilbert 1987) was performed to determine whether these impurities could significantly degrade cladding or exposed fuel at the site of a cladding breach during storage. The sources of impurity gases in spent fuel storage casks were identified and the expected concentrations and types of reactive impurity gases from these sources over an operating lifetime of 40 years.

Four potential sources of impurity gases in the helium cover gas in operating casks were identified and evaluated:

- Residual impurities in the inert cover gas
- Air leakage into the cask cavity through the seals between the cask body and primary cover
- Residual impurity gases remaining in the cask after evacuation to a finite pressure level
- Impurity gases introduced into the cask atmosphere after evacuation and back fill because of outgassing from the structural materials and fuel assemblies.

These sources were evaluated using design information obtained from technical and operating manuals for the TN-24P, Castor-V/21, MC-10, and MSF-IV spent fuel storage casks. Impurity contributions from each source were calculated when possible, and gas analysis data from recent cask performance tests were used. Essentially no air in leakage is expected under normal operating conditions because of seal designs and safeguard systems. Contributions from indigenous impurities in the source gas are insignificant (<0.2 mol 0_2) if the cask is back filled with high-purity (99.995%) gas. Most impurities originate from the remaining two sources, which together are estimated to produce a reactive gas inventory of <0.6 mol each of 0_2 (or equivalent oxidizing gases) and H_2.

Reactive impurity gases could potentially degrade the cladding either by reacting directly with it or by reacting with exposed $U0_2$ fuel (if cladding breaches are present). There are insufficient quantities of impurities to cause significant degradation of the Zircaloy cladding. Oxidation of exposed $U0_2$ fuel could cause fuel swelling and cladding crack-extension; but only if all of the impurities reacted with the fuel at a single breach site. Because the rates of these reactions are strongly temperature dependent, the behavior of the reactive gases and the subsequent effects on the Zircaloy cladding depend largely on two factors: 1) the cladding temperature distribution and its time dependence and 2) the number, size, and location of cladding breaches that could expose UO_2 fuel to oxidation.

C. Spent Fuel Tests and Demonstrations

Spent fuel tests and demonstrations performed in the U.S. and in several other nations have served as a key element in the development of licensed dry storage.

1. Nevada Test Site Experience. A 1987 report summarizes fuel integrity aspects of dry storage demonstrations conducted at the Nevada Test Site (NTS) (Johnson et al. 1987). The report indicates that the 17 PWR assemblies consisting of 3468 Zircaloy-clad fuel rods survived shipment, handling, packaging, storage, retrieval, inspections and reshipment to the Idaho National Engineering Laboratory. The maximum cladding temperatures are summarized in Table II. The dry storage modes included surface and deep dry wells, silo, and vault. The storage atmosphere was helium. The principal method of cladding integrity monitoring was periodic measurement of Krypton-85 concentrations in the cover gases. Assembly B43 was suspected of having a leaking rod prior to testing because of elevated Krypton-85 in its cover gas.

Table II
Dry Storage Tests with 17 PWR Spent Fuel Assemblies at the Nevada Test Site

Facilities	*Dates*	*Maximum Cladding Temperature, °C*
Silo	12/78 to 7/82	168
Surface dry well	1/79 to 10/84	233
Deep dry wells	4/80 to 3/83	233
Vault	10/79 to 6/86	278

After 3.5 years exposure in the silo at 150°C, Assembly B02 was exposed to air starting at 275°C in the Fuel Temperature Test Facility (FTT). The Krypton-85 in the cover gas abruptly increased early during the two years of testing in the (FTT), indicating that a single rod breached. Estimates based on the Poiseuille equation and the gas leakage rates through the cladding breach indicated that the cladding breach size was on the order of 1 micron. Results of gas analyses, visual inspections, and surface smear analyses indicated that significant fuel was not released by the breached rod and the remaining rods survived the operations without detectable deterioration.

2. U.S. Cask Tests. Tests and demonstrations with BWR spent fuel in the MSF-IV (REA-2023) cask (McKinnon et al. 1986), and PWR spent fuel in the Castor V/21 (Creer et al. 1986), TN-24P (Creer et al. 1987), and MC-10 (McKinnon et al. 1987) casks provided valuable thermal performance data and experience with handling and storing spent fuel. Metal cask tests and demonstrations have also provided important information regarding spent fuel behavior during dry storage (Schoonen, Jensen, and Fisher 1987). In a few cases Krypton-85 monitoring of cover gases provided evidence of one or more leaking rods. However, the major conclusion is that cladding behavior was satisfactory in the dry storage tests and demonstrations.

3. Federal Republic of Germany (FRG). Spent fuel integrity investigations in the FRG have been completed. The key results have been reported in the Proceedings of the Third International Spent Fuel Storage Technology Symposium/Workshop (CONF-860417) and in (Peehs and Fleisch 1986). Hot cell and laboratory test results are summarized in the publications. The FRG dry storage cask demonstrations are summarized in Table III. Examination of fuel from the CASTOR Ia, Ib, and Ic tests showed no degradation from the dry storage. Also, cover gas monitoring from Krypton-85 demonstrated satisfactory performance of the Zircaloy-clad fuel. The German tests and demonstrations focused principally on Zircaloy-clad fuel behavior in helium, but included some testing of PWR fuel in moist nitrogen.

4. Canada. Fuel integrity tests are continuing at the Whiteshell Laboratory in Canada; conditions and status are summarized below:

- CEX 1. Dry air, 150°C, defective rods - stored 84 months, December 1987. No exam was made in 1987. Next exam is scheduled for 1989 (tentative).
- CEX 2. Moist air, 150°C, defective rods - stored 64 months, December 1987. A high-flux bundle was examined in 1987 after 5 years. One bundle per year is scheduled for examination; the test includes both naturally and artificially defected fuel rods.
- ERB. In air at near-ambient temperatures (e.g., fuel at − 7°C, December 1986). Exposed 110 months, December 1987. No defective fuel rods. Next exam in 2 to 3 years.

The Canadian tests focus principally on behavior of Zircaloy-clad fuel with cladding defects in air. Earlier test results are reported in Wasywich et al. 1986.

Fuel storage in concrete canisters is under way at three sites in Canada:

- Whiteshell (test reactor and PHWR fuel)
- Gentilly (boiling light water reactor fuel)
- Pickering (PHWR fuel to be loaded in early 1988).

The Pickering demonstration will involve storage in two concrete canisters for two years followed by shipment in the same canisters. Whiteshell and Gentilly storage programs are described in (Patterson and Hoye 1986).

5. Summary of Cask Tests. U.S. and foreign metal cask tests and demonstrations are summarized in Table III. For completeness, metal cask demonstrations conducted in Switzerland and USSR are listed, even though no specific fuel integrity information is currently available. These demonstrations provided verification that spent fuel can be handled and stored under conditions that did not lead to gross degradation of the spent fuel. The storage conditions were maintained within environmental conditions recommended by investigations of behavior of spent fuel in storage.

Table III
Summary of Metal Cask Test/Demonstrations

Country	*Cask*	*Fuel Type/ No. of Assys.*	*Test Period, d*	*Max Clad Temp °C*	*Remarks*	*Reference*
USA	MSF-IV (REA-2023)	BWR/52	125	241 (Vac)	Cover gas monitoring	PNL-5777 Vol. 1
	CASTOR V/21	PWR/21	15 (Cont)[1]	424 (Vac)	"	EPRI-NP-4887
	TN-24P	PWR/24	25 + 450[3]	290 (Vac)	"	NP 5128
	MC-10	PWR/24	25 (Cont)[1,3]	217 (Vac)	"	NP 5268
	TN-24P	PWR/48	~30 (Cont)[1,2,3]	250 (Vac)	Consolidated Assemblies	—
FRG	CASTOR Ia	PWR/4	730	333 (He)	Cover gas monitoring	CONF-860417
	CASTOR Ib	PWR/4	730	433 (He)	"	"
	CASTOR Ic	BWR/16	730	380 (He)	"	"
	TN 1300	PWR/12	~15d	272 (He) ~15d	"	"
					monitoring cask now resides in an AFR facility	CONF-860417 "
USSR	CASTOR V	PWR	Long-Term	325 (He)	"	—

1) Cont - Fuel exposure continuing after initial test period.
2) Test period to begin ~1-10-88.
3) Some fuel from original TN 24P and MC-10 casks and Turkey Pt. fuel (from EMAD) was consolidated and placed in TN 24P.

D. Storage in Air

Because there were no data on the oxidation behavior of LWR spent fuel that could be used to provide guidance in developing acceptable storage conditions in an oxidizing atmosphere, an activity was initiated in the Commercial Spent Fuel Management Program (CSFMP) in 1982 to determine the allowable storage temperature of spent nuclear fuel in oxidizing atmospheres (air). To define the principal variables that affect $U0_2$ oxidation, the work began with nonirradiated fuel pellets and has progressed to testing BWR and PWR spent fuel pellets and short Zircaloy-clad rod segments (short sections of actual spent fuel rods) with small intentional machined defects. Pellet tests on irradiated fuel provide conservatively high oxidation rates because of unimpeded oxygen access to the spent fuel while the rod segment tests are more representative of actual defected spent fuel rods. Oxidation test temperatures range between 135°C and 230°C which is lower than the majority of the published oxidation data. The lower temperature range is more representative of expected spent fuel storage temperatures and is thought to include the allowable temperature in air.

Existing oxidation data consist of numerous test runs at elevated temperatures (>250°C) for short periods, and the initial PNL oxidation data for the temperature range 135 to 230°C (Campbell et al. 1987). Studies of $U0_2$ oxidation kinetics generally have involved short-time/high temperature tests, with results extrapolated to long-time/low-temperature conditions. The PNL test is conducted at temperatures directly applicable to dry storage in air, but the lower temperatures require longer run periods to provide adequate data.

The data being generated in the spent fuel oxidation test include: bare $U0_2$ pellet weight gain as a function of time, temperature, and spent fuel characteristics; rod segment weight gain and diametral strain as a function of time, temperature, and spent fuel characteristics; and observation/analysis of spent fuel microstructural changes as a function of oxidation. It has been estimated that a 5 to 10°C increase in allowable air storage temperature may be achieved by a 10% reduction in the weight-gain relationship used in determining allowable storage temperatures. Therefore, long-term testing is important for reducing the uncertainty band for the data and model (the upper uncertainty bound for oxidation, rather than the mean, will probably be used in determining allowable storage temperatures).

The work by the CSFMP on spent fuel oxidation led to significant increases in understanding the oxidation processes involving irradiated UO_2 at various storage temperatures (Roberts 1987). However, there is concern that the currently available data do not sufficiently define the long-term stability of the various oxides. For example, according to some phase diagrams, U_3O_8 should not form in the temperature range 100 to 150°C and U_3O_8 has not yet been observed for specimens in this temperature range[a]. Unfortunately the phase diagrams are based on short-term data, as are the existing spent fuel oxidation data. If it can be determined that the oxidation reaction does not proceed to U_3O_8 at temperatures applicable to dry storage, then cladding strain due to UO_2 oxidation may be eliminated as a concern during air dry storage[b].

The UO_2 oxidation data are/will be used in the following areas:

- limiting conditions and system designs for dry storage of spent fuel in air;[c]
- determining the period and frequency for which monitoring inerted dry storage under low-temperature/long-term conditions will be required;
- determining limiting operating conditions and system designs for fuel handling, rod consolidation, and lag storage operations in air; and
- providing a data base for modeling or determining the effects of abnormal storage conditions on fuel integrity following an incident.

V. ROD CONSOLIDATION

Rod consolidation is a leading candidate for more efficient utilization of existing space in spent fuel storage pools and also has the potential to be applied to dry storage of LWR fuel. An American National Standard for design criteria for consolidation of LWR spent fuel, ANSI/ANS-57.10-1987, was recently issued. "The standard provides design criteria for the equipment and systems comprising the rod consolidation process for commercial LWR spent fuel assemblies. The criteria are applicable to wet and dry, and horizontal and vertical consolidation concepts. The standard does not include storage of the spent fuel either prior to performing consolidation or upon completion of the process."[d]

A. U.S. Rod Consolidation Demonstrations

The current experience base for consolidation of irradiated LWR fuel in the U.S. is shown in Table IV (Johnson, Bailey, and Klein 1987; Gerstberger 1987; and *NuclearFuel* 1987). As indicated in Table IV, during 1986-1987 rod consolidation operations with irradiated LWR fuel were conducted at five sites: Western New York Nuclear Service Center (West Valley) (Bailey 1986), Battelle Columbus Laboratories (BCL) (Bailey, May 1987), Test Area North (TAN), Millstone-2, and Prairie Island.

The PWR fuel assemblies consolidated at West Valley, and BCL were known to contain some fuel rods with collapsed Zircaloy cladding, a result of inreactor fuel densification and non-uniform cladding creep down. Based on videotapes of the rods at West Valley, BCL, TAN, and Millstone-2 and visual observations at TAN, it appears in general that no substantial damage to the fuel cladding is occurring during rod consolidation that is attributable to the consolidation operation, although longitudinal burnish lines or scratches are visible on most rods. The lines appear to be the result of the crud and oxide layers being penetrated as the rod passes through the fuel assembly's spacer grids. Scratches or mars that extend into the underlying metal (Zircaloy-2 or -4) could be of concern if the associated fuel rods came into direct contact with aluminum (e.g., storage racks or canisters) during subsequent storage in impure water because of the potential for galvanic coupling that could result in accelerated hydriding of the Zircaloy under certain conditions (Johnson 1977).

It was encouraging to observe during the demonstration at BCL that the cladding on fuel rods with collapsed cladding regions had sufficient ductility so that the flattened areas could be reformed to make the rod cross sections more circular. Reforming improved the likelihood that the rod would enter the consolidation canister in normal fashion (i.e, like an intact rod). A 1985 report (Bailey 1985) provides an overview of the status of rod consolidation in the U.S.

B. International Rod Consolidation Activities

Consolidation of spent fuel rods is of interest in the United Kingdom, Sweden, and the Federal Republic of Germany (FRG).

1. *United Kingdom.* Consolidation of stainless-clad AGR fuel is under way at Sellafield in the United Kingdom. Consolidation factors are 3:1.

2. *Sweden.* A Swedish evaluation of rod consolidation is under way as an alternative to expansion of the CLAB wet storage AFR. The study is expected to be completed by March 1988.

3. *Federal Republic of Germany (FRG).* There is a continuing interest in dry rod consolidation in the Federal Republic of Germany. Cold dry rod consolidation tests are under way that involve dismantling and placing rods from nonirradiated fuel assemblies into prototypic canisters. The focus is on consolidation for disposal.

VI. SUMMARY AND CONCLUSIONS

- Wet storage continues to be the predominant fuel storage method. Canadian Zircaloy-clad fuel bundles stored up to 27 years will be examined in 1988.
- Significant progress by U.S. utilities and DOE in solving spent fuel and storage problems has been realized through a variety of innovative approaches such as: dense racking, double tiering, credit for burnup in rack designs, transshipment, extended burnup, dry storage, and rod consolidation.
- Spent fuel temperature-time conditions have been developed that are expected to limit cladding breaches

Table IV
Experience Base for Consolidation of Irradiated LWR Fuel

Demonstrations	No. of Nonirradiated Fuel Assemblies BWR	PWR	Consolidation Operation Medium	Comments
COLD DEMONSTRATIONS				
• Allied General Nuclear Services	—	4	Dry	Done in horizontal position. A total of 17 runs were made with the 4 assemblies.
• Combustion Engineering	—	1	Wet	
• Nuclear Assurance Corp.	—	6	Wet	
• U.S. Tool & Die, Inc.	1	—	Wet	
• Westinghouse Electric Corp.	—	3	Wet and Dry	There were multiple campaigns with the 3 assemblies.

	Reactor or Site	No. of Irradiated Fuel Assemblies BWR	PWR	Consolidation Ratio	Consolidation Operation Medium
HOT DEMONSTRATIONS (Date)					
• Westinghouse Electric Corp./ Duke Power Co. (Oct.-Nov. 1982)	Oconee	—	4	2:1[a]	Wet
• Maine Yankee Atomic Power Co. (Aug. 1983)	Main Yankee	—	1	1.6:1	Wet
• Nuclear Assurance Corp./ Rochester Gas and Electric (Dec. 1985-Feb. 1986)	Facilities at West Valley, [b] NY	—	6[c]	1.8:1[d]	Wet
• U.S. Tool & Die, Inc./Rochester Gas and Electric (Aug.-Oct. 1986)	Battelle Columbus Laboratories (BCL)	—	5[c]	1.88:1 to 2.0:1[a]	Wet
• INEL[e]/Virginia Power/Florida Power and Light (1987)	TAN[f]	—	48		Dry
• Combustion Engineering/ NUSCO[g] (Aug.-Sept. 1987)	Millstone-2	—	6	2:1	Wet
• Westinghouse Electric Corp./ Northern States Power (Oct.-Nov. 1987)	Prairie Island	—	36		Wet

(a) Consolidation ratio of 2:1 achieved in one canister with rods from 2 of the assemblies.
(b) Western New York Nuclear Service Center, operated by West Valley Nuclear Services, Co., Inc., for DOE. The West Valley facilities were originally built and operated by Nuclear Fuel Services (NFS).
(c) All assemblies had some rods with collapsed cladding (cause: in-reactor fuel densification).
(d) Consolidation ratio of 1.8:1 achieved in one canister.
(e) Idaho National Engineering Laboratory, Idaho Falls, Idaho (INEL).
(f) Test Area North (TAN) at INEL.
(g) Northeast Utilities Service Co. (NUSCO).

Table IV
Experience Base for Consolidation of Irradiated LWR Fuel
continued

	Reactor or Site	*No. of Irradiated Fuel Assemblies* BWR	PWR	*Probable Date*	*Consolidation Operation Medium*
UPCOMING OR PROPOSED DEMONSTRATIONS					
• Combustion Engineering/Virginia Power	Surry	—	~48	1988	Wet
• Combustion Engineering/ NUSCO/EPRI[h]/Baltimore Gas and Electric	Millstone-2	—	2000	1988	Wet
• INEL/DOE Prototypic Consolidation Demonstration Program[i]	TAN	~100	~100	1988-1989	Dry
• Maine Yankee Atomic Power Co.[j]	Maine Yankee	—	20	TBD[k]	Wet
• Nuclear Assurance Corp./ Tennessee Valley Authority	Browns Ferry	12	—	(l)	Wet

(h) Electric Power Research Institute (EPRI).
(i) Supported by the Waste Fund.
(j) Self-funded program that involves intact fuel; the utility plans to pursue rod consolidation in the 1990s.
(k) To be determined (TBD).
(l) Equipment installed in pool; however, the demonstration has been postponed indefinitely.

during dry storage in inert gas to less than 0.5% per rod. Breaches, if any, will consist of pin hole cracks that do not compromise confinement of fuel.
- Testing and modeling is under way to determine acceptable conditions under which spent fuel can be exposed to air.
- The experience gained from consolidating spent fuel during several demonstrations has revealed no problems that would compromise spent fuel integrity or consolidation operations.

VII. ACKNOWLEDGEMENTS

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VIII. NOTES

a. The formation of U_3O_8 has been observed at temperatures of greater than 180°C, while only U_4O_9 has been observed at 135 and 150°C.
b. The phase change from UO_2 to U_4O_9 involves a small volume decrease, while the phase change to U_3O_8 involves a large volume increase. Both phase changes may cause fuel break-up leading to powder formation; however, a volume decrease cannot place any strain on the cladding and therefore cannot cause an increase in the original cladding defect. Therefore, none of the powder formed from a volume decrease should be released from within the cladding.
c. For example, G.E. Company Ltd. (UK), has prepared an Air Vault SAR that depends on spent fuel oxidation data supporting allowable temperature-time conditions for air storage. (Letter from H. C. Pickering, Jr., Vice President, FW Energy Applications, Inc. to L. C. Rouse, Chief, Advanced Fuel and Spent Fuel Licensing Branch, U.S. NRC, dated September 15, 1986, with topical report. Available from U.S. NRC Public Document Room, 1717 H Street NW, Washington, D.C., docketed under Project No. M-46.)
d. Extracted from American National Standard ANSI/ANS-57.10-1987 (Design Criteria for Consolidation of LWR Spent Fuel) with permission of the publisher, the American Nuclear Society.

IX. REFERENCES

American Nuclear Society, *ANS News*, "Storage of Spent Nuclear Fuel, p.5, June 1986.

Bailey, W. J., "Status of Rod Consolidation," PNL-5122, Pacific Northwest Laboratory, Richland, Washington, 1985.

Bailey, W. J., "Rod Consolidation at the West Valley Demonstration Project," PNL-5946, Pacific Northwest Laboratory, Richland, Washington, December 1986.

Bailey, W. J., "Categorization of Failed and Damaged Spent LWR Fuel Currently in Storage," PNL-5882, Pacific Northwest Laboratory, Richland, Washington, November 1987.

Bailey, W. J., "Rod Consolidation of RG&E's Spent PWR Fuel," PNL-6162, Pacific Northwest Laboratory, Richland, Washington, May 1987.

Campbell, T. K., et al., "The Interim Results from UO_2 Fuel Oxidation Tests in Air," PNL-6201, Pacific Northwest Laboratory, Richland, Washington, August 1987.

Chin, B. A., M. A. Khan, and J. C. L. Tarn, "Deformation and Fracture Map Methodology for Predicting Cladding Behavior During Dry Storage," PNL-5998, prepared by Auburn University for Pacific Northwest Laboratory, Richland, Washington, September 1986.

Creer, J. M., et al., "The Castor-V/21 PWR Spent-Fuel Storage Cask: Testing and Analyses," EPRI NP-4887, Electric Power Research Institute, Palo Alto, California, 1986.

Cunningham, M. E., et al., "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," PNL-6364, Pacific Northwest Laboratory, Richland, Washington, October 1987.

Electric Power Research Institute, *Fuel and Pool Component Performance in Storage Pools*, EPRI NP-4651, Copyright 1986, Palo Alto, California, May 1986. Reprinted with permission.

Gerstberger, Jr., C., "Westinghouse At-Reactor Consolidation Program," *Journal of the Institute of Nuclear Materials Management* XV(3):30-31 (April 1987).

Hazelton, R. F., "Characteristics of Fuel Crud and Its Impact on Storage, Handling, and Shipment of Spent Fuel," PNL-6273, Pacific Northwest Laboratory, Richland, Washington, September, 1987.

Hunt, C. E. L., et al., "Long Term Storage of Fuel in Water," AECL-6577, Atomic Energy of Canada, Ltd., Chalk River, Ontario, Canada, 1978.

International Atomic Energy Agency, *Technical Report Series No. 218: "Storage of Water Reactor Spent Fuel in Water Pools — Survey of World Experience,"* Vienna, Austria, 1982.

Johnson, Jr., A. B., "Behavior of Spent Nuclear Fuel in Water Pool Storage," BNWL-2256, Pacific Northwest Laboratory, Richland, Washington, 1977.

Johnson, Jr., A. B., W. J. Bailey, and K. Klein, 1987, "Wet Storage in the USA: Recent Experience and Directions," *Long-Term Spent Nuclear Fuel Storage*, IAEA-TECDOC-418, International Atomic Energy Agency, Vienna, Proceedings of a Technical Committee Meeting on Behavior of Used Fuel Assemblies and Storage Equipment at Long Term Wet Storage Conditions held in Leningrad, USSR, May 26-30, 1986.

Johnson, Jr., A. B., et al., "Assessment of the Integrity of Spent Fuel Assemblies Used in Dry Storage Demonstrations at the Nevada Test Site," PNL-6207, Pacific Northwest Laboratory, Richland, Washington, 1987.

Knoll, R. W., and E. R. Gilbert, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, Pacific Northwest Laboratory, Richland, Washington, 1987.

Levy, I. S. et al., "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas," PNL-6189, Pacific Northwest Laboratory, Richland, Washington, May 1987.

McKinnon, J. A., et al., "The MC-10 PWR Spent-Fuel Storage Cask; Testing and Analysis, EPRI NP-5268, Electric Power Research Institute, Palo Alto, California, 1987.

McKinnon, M. A., et al., "BWR Spent Fuel Storage Cask Performance Test: Cask Handling Experience and "Decay Heat, Heat Transfer, and Shielding Data," PNL-5777 Vol. 1, Pacific Northwest Laboratory, Richland, Washington, 1986.

NuclearFuel, "Uncertainties Cloud Efforts of Vendors, Utilities to Develop Rod Consolidation," pp. 3-4, McGraw-Hill, Inc., New York, January 12, 1987.

Patterson, D. W., and D. S. Hoye, "Canadian Experience Storing Irradiated CANDU Fuel in Concrete Canisters," Proceedings of the Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986.

Peehs, M., and J. Fleisch, "LWR Spent Fuel Storage Behavior," J. Nucl. Mat., 137:190-202, 1986.

Roberts, J. P., "Dry Spent Fuel Storage — Licensing Update," *JNMM*, Institute of Nuclear Materials Management, Northbrook, Illinois, April 1987.

Schoonen, D. H., M. F. Jensen, and M. W. Fisher, "Spent Fuel Dry Cask Storage Program at the Idaho National Engineering Laboratory," *JNMM*, Institute of Nuclear Materials Management, Northbrook, Illinois, April 1987.

Stahl, D. et al., "Spent Fuel Behavior Under Abnormal Thermal Transients During Dry Storage," PNL-5456, Pacific Northwest Laboratory, Richland, Washington, January 1985.

U.S. Department of Energy, *Spent Fuel Storage Requirements*, DOE/RL-85-2, Richland Operations Office, October 1985.

U.S. Department of Energy, *Spent Fuel Storage Requirements 1987.* DOE/RL-87-11, Richland Operations Office, September 1987.

U.S. Nuclear Regulatory Commission, *Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel*, NUREG-0404, March 1978.

Wasywich, K. M., et al., "Examination of Intact and Defected Irradiated CANDU Fuel Bundles Stored up to 30 Months in Moist Air at 150°C," Proceedings of the Third International Spent Fuel Storage Technology Symposium/Workshop, Seattle, Washington, April 8-10, 1986.

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Summary of International Activities in Spent Fuel Storage

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ABSTRACT

In 1986, nuclear power electrical generating systems in operation worldwide numbered 397 with a combined net capacity of 273 Gigawatts electric (GWe). This net production resulted in approximately 39 thousand metric tonnes of Initial Heavy Metal being discharged as spent fuel from these systems. The storage requirements for spent fuel discharged differ in each nation and are dependent on national strategies for fuel cycle management. The design of spent fuel storage facilities is a function of the type of nuclear electrical generating system adopted, and, therefore, the spent fuel characteristics. This paper summarizes the current international status of spent-fuel storage strategies.

INTRODUCTION

Fuel cycle management strategies include either reprocessing or direct disposal of spent fuel. A few countries are already committed to a closed fuel cycle. This cycle will include an interim storage of spent fuel in spent-fuel pools until short-lived radioactive fission products have decayed, followed by reprocessing to recover Plutonium and Uranium. Plutonium and Uranium will then be recycled to an appropriate nuclear power plant with interim storage and disposal of the remaining wastes. Many countries, including the U.S. have selected the one-time fuel cycle which involves direct disposal of spent fuel following an interim storage period ranging from 5 to 100 years.

Spent-fuel storage needs have thus far been met primarily through on-site, at-reactor storage in water pools. Both wet and dry storage technologies are being considered for interim spent fuel storage. Where spent fuel has cooled for several years, dry storage in either casks or concrete modules has proven a viable alternative. Rod consolidation is also a means considered to increase storage capacity. Whichever option is chosen, there are no apparent technical problems to overcome.

U.S. DEPARTMENT OF ENERGY COOPERATIVE PROGRAMS

Since the passage of the Nuclear Waste Policy Act of 1982 (NWPA), it has been the United States' policy to exchange information and to seek benefits through cooperation with other nations and international organizations in the development of waste management and disposal technology. Developments in United States radioactive waste management programs have much to offer for exchange with the international community. Several foreign nations and organizations are developing sophisticated waste management programs which have significant benefits and are ahead of the United States programs. Thus, participation in both general and specific cooperative activities with these nations and international organizations has been and is expected to continue under the Nuclear Waste Policy Amendment Act of 1987 (NWPAA).

Section 223 of the NWPA outlines the policy under which the United States is to cooperate with and provide technical assistance to nonnuclear weapon states in the field of spent-fuel storage and disposal. Since 1983, a notice has been published annually in the *Federal Register* summarizing the resources that are available for international cooperation and assistance. By the end of 1987, 12 nations had requested assistance; these nations included Netherlands, Spain, Norway, Italy, Columbia, Argentina, Brazil, Mexico, Madagascar, Egypt, South Korea, and Kenya. The United States has also cooperated with the American Institute in Taiwan. It should be noted that many of these nations have requested background technical information only. The Department of Energy (DOE) has also provided briefings, seminars, and workshops under this program to those nations which have limited nuclear development or are planning to develop nuclear power as part of that nation's energy policy.

The DOE also cooperates with the International Atomic Energy Agency (IAEA). The IAEA activities include an exchange of scientific information on spent-fuel management. The principal effort involves the preparation of documents to exchange research and development of spent-fuel disposal occurring throughout the world. The DOE participated in "Behavior of Spent Fuel in Assemblies in Extended Storage" (Befast-I) from 1985 to 1987. This focused on materials behavior and its effect by wet and dry storage. The second Befast-II agreement has recently been initiated and will cover dry storage and consolidation operations, along with current safety and monitoring methods.

The DOE also participates in IAEA technical committees which study various aspects of spent-fuel management. The DOE has participated on topics such as surveillance, storage safety, and storage economics. These documents are important to the DOE in providing internationally-derived guidance on spent-fuel disposal. Further international assistance is through the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). The OECD/NEA promotes the development of peaceful uses of nuclear energy through cooperation of states. The NEA provides a forum for sharing common interests and experiences in disposal technology. The DOE office is currently participating with the NEA in efforts through committee/study groups to demonstrate the safe disposal of radioactive waste. The DOE considers participation in these efforts important to the spent-fuel storage disposal program. Participating nations are represented by delegates on the Radioactive Waste Management Committee (RWMC). The RWMC involves senior experts who meet at least annually to review proposed OECD/NEA radioactive waste management activities and to ensure the overall programmatic objectives are met.

In addition to multinational organizations, the DOE has specific bilateral cooperative agreements with several nations. The DOE has exchanged information on dry metal cask storage and rod consolidation with the Federal Republic of Germany (FRG). Other dry-storage agreements have been with Canada on their concrete canisters for storage and the United Kingdom with a joint study on reactivity of oxide fuels. The DOE is currently reviewing its international agreements to stay within the framework of the NWPA and to plan for future agreements.

The DOE recognizes the major benefits in continuing and enhancing international cooperation. Significant contributions to technology and experience are evolving and approaches to the institutional and regulatory issues of spent-fuel storage are being developed.

SUMMARY OF INTERNATIONAL SPENT FUEL STORAGE MANAGEMENT STRATEGIES

Spent Fuel Strategy	*Away from Reactor Storage Facilities*
NORTH AMERICA	
Canada	
Overall Spent-fuel management policy is to store for approximately 50 years, then prepare for geologic disposal with a possibility of reprocessing based on economic indications. Current policy is retrievable storage in AR wet pools and dry concrete canisters.	Four dry concrete canisters at Whiteshell and Gentilly for approximately 370 MTU.
United States	
Overall spent-fuel management policy is to store for a minimum of 5 years, then disposal in geologic repository. Current policy is AR wet pools or AFR dry casks or dry vaults.	Four Caster V/21's at Virginia Power sited at Surry Plant. Three NUMOMS dry vaults at Carolina Power & Light.
SOUTH AMERICA	
Argentina	
Overall spent-fuel management policy is AR and AFR storage of spent fuel for 10 years minimum and then reprocess with disposal of vitrified HLW in deep geologic repository.	None
Brazil	
Overall spent-fuel management policy is AR spent fuel storage followed by reprocessing with vitrified wastes placed in geologic repository.	None
EUROPE	
Belgium	
Overall spent-fuel management policy is to reprocess spent-fuel and disposal of vitrified HLW after 50-75 years of storage. Spent fuel is stored in AR pools with expansion added as needed.	None
Finland	
Overall spent-fuel management policy is to store AR for minimum of 5 years and then return to foreign suppliers (Soviet Union) or transport to other foreign country for reprocessing and possible disposal. Domestic geologic disposal has not been ruled out. Spent fuel AR site expansion using wet pools has fulfilled storage needs.	None
Federal Republic of Germany	
Overall spent-fuel management strategy is to store spent fuel in reactor pools for 5-10 years to be followed by reprocessing and disposal of vitrified HLW in geologic repository. Spent fuel is stored in AR wet pools and dry casks.	Gorleben has operating permit for dry cask spent fuel but the permit is under litigation. Guelich has 3 fuel elements stored in a metal cask inside a concrete vault.
France	
Overall spent-fuel management policy is to store at reactor for not more than one year and then transport for reprocessing. Vitrified HLW is stored for 20 years minimum and then placed in deep geologic disposal.	150-200 MTU vault facility under construction at Cadarache. Wet pool storage at La Hague.

Italy	
Overall spent-fuel management policy is to reprocess via foreign nation and then return HLW for internal geologic disposal after 50 years of storage. High density and compact racks have solved near-term storage needs with demonstration project in dry storage underway.	12 PWR assemblies in modular concrete vault sited at Trino Veriellese power plant.
Netherlands	
Overall spent-fuel management policy is to reprocess via foreign nation and then return vitrified waste for geologic disposal after 50-100 years of storage. Current needs are met with AR storage pools with plans to build AFR storage facility.	5000 MTU capacity dry vault in planning stages for all types of radioactive waste including spent-fuel.
Spain	
Overall spent-fuel policy is at-reactor storage for 10 years followed by 10-20 years of dry cask storage. Direct disposal in geologic repository is planned. Current and near term needs are met with AR storage pools.	None
Sweden	
Overall spent-fuel policy is to store in AFR facility for 40 years and then disposal in geologic repository. Current spent fuel is stored on-site for six months before transport to central storage facility (CLAB).	CLAB capacity is 3000 MTU with expansion planned to 9000. Assemblies stored in open canisters in stainless steel-lined, reinforced concrete pools approximately 25-30 meters below the surface. CLAB is located near Oskarsham power station.
Switzerland	
Overall spent-fuel policy is ship to foreign country for reprocessing and return HLW for internal disposal. Current needs met with AR high-density racks.	Castor 1C dry cask tested and licensed.
United Kingdom	
Overall spent-fuel management policy is for interim storage AR followed by reprocessing. Vitrified HLW to be placed in deep geologic repository after 50 years of storage. AR spent fuel pools have met storage needs. Dry vault storage currently used at Wylfa sites for Magnox.	3 modular dry vaults at Wylfa, Wales power station using natural convection for cooling capacity is 83 MTU.
ASIA	
India	
Overall spent-fuel management policy is for interim storage on site followed by reprocessing. Vitrified HLW to be buried in geologic repository. Current near term needs met by spent fuel pools.	None
Japan	
Overall spent-fuel management policy is for reprocessing by foreign nation until domestic reprocessing capability is achieved. AR pool storage is planned. 30-50 years of interim storage of HLW planned before geologic disposal.	Drywall vault storage tested at Tokai. Development of dry storage casks tested by CRIEPI.
South Korea	
Overall spent-fuel management policy is for AR storage followed by central dry storage facility (50 years). Near term storage needs met through high density racks.	None
Taiwan	
Overall spent-fuel policy is to store at reactor site to be followed by AR or central dry cask storage facility for 50 years. Current and near term storage capacity met through high density racks.	None

Mr. Shelor is currently the Acting Chief, Systems Development Branch, Office of Civilian Radioactive Waste Management, U.S. Department of Energy. His responsibilities include Spent Fuel Storage Technology, Federal Interim Storage Planning, Engineering Development of Prototypical Rod Consolidation Equipment and Technical Coordinator for International Activities.
Mr. Shelor had been employed by the Federal Government since 1975, first in the Energy Research and Development Agency and then the Department of Energy in 1977. He has worked in the Office of Civilian Radioactive Waste Management since 1984 and was previously a Project Engineer for High Level treatment in the Office of Nuclear Energy for about one and a half years.

William Whittington earned his B.S. in Chemical Engineering from the University of Illinois in 1984 and M.S. degree in Nuclear Engineering from the University of Maryland in 1987. From 1984-1986 he was a reactor engineer at the University of Maryland Research Reactor. Since 1986 he has been a systems engineer with Roy F. Weston/Williams Brothers Engineering, where he is involved in support activities for international programs in spent fuel storage.

Spent Fuel Seminar Proceedings Available

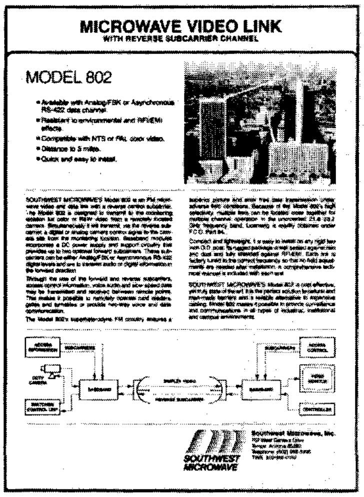
The proceedings of the fifth Institute of Nuclear Materials Management Spent Fuel Management Seminar are now available. These proceedings are a valuable reference, containing the complete text of the 23 papers presented at the seminar, held Jan. 20-22, 1988 in Washington, D.C.

The papers represent the current state of technology and regulation in Institutional Issues; Developments in Dry Storage Technology; Developments in Consolidation of Spent Fuel; Reactor Storage/DOE Transportation System Interface Considerations; and Technical Issues and Programs.

Copies are available for $200. For information contact Beth Perry, INMM, 60 Revere Dr., Suite 500, Northbrook, Ill. 60062 U.S.A. Telephone 312/480-9573.

Southwest Microwave Offers Free Data Sheet

A two-page data sheet describing a new microwave video link featuring reverse subcarrier channel transmission is now available from Southwest Microwave, Inc.



The recently developed Model 802 is designed to transmit full color or black and white video. Simultaneously it transmits, via the reverse subcarrier, a digital or analog camera control signal to the camera site for controlling pan/tilt, zoom or switching from the monitoring location.

Two optional forward subcarriers carry audio or digital information. The Model 802 thus allows remote operation of card readers, gates, and turnstiles as well as two-way voice/data communications.

The data sheet gives full specifications and ordering information. For a free copy, contact Southwest Microwave, Inc., 707 West Geneva Drive, Tempe, Ariz. 85282. Phone (602) 968-5995.

CALENDAR

May 23-25, 1988
International Conference on Transportation for the Nuclear Industry, Stratford-on-Avon, Warwickshire, U.K. *Sponsor:* Institution of Nuclear Engineers *Contact:* Mrs. S.M. Blackburn, Institution of Nuclear Engineers, Allen House, 1 Penerley Rd., London SE6 2LQ U.K. Telephone 01-698-1500.

May 24-26, 1988
Uranium Hexafluoride — Safe Handling, Processing, and Transporting, Oak Ridge, Tennessee *Sponsor:* U.S. Department of Energy and Martin Marietta Energy Systems, Inc. *Contact:* Sheila G. Thornton, Administrative Coordinator, Martin Marietta Energy Systems, Inc., Building K-1020, MS 403, P.O. Box P, Oak Ridge, Tenn. 37831, Phone 615/574-9200.

June 12-15, 1988
28th Annual Conference of the Canadian Nuclear Association and 9th Annual Conference of the Canadian Nuclear Society, Winnipeg, Manitoba, Canada *Sponsor:* Canadian Nuclear Association and Canadian Nuclear Society *Contact:* M.G. Wright, Chairman, CNA/CNS Planning Committee, Atomic Energy of Canada, Ltd., Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, Canada R0E 1L0 Telephone (204) 753-2311.

June 12-17, 1988
Annual Meeting of the American Nuclear Society, San Diego, Calif. *Sponsor:* American Nuclear Society *Contact:* Meetings Dept, American Nuclear Society, 555 N. Kensington Ave., LaGrange Park, Ill. 60525 U.S.A.

June 26-29, 1988
INMM 29th Annual Meeting, Bally's Hotel, Las Vegas, Nev. U.S.A. *Sponsor:* Institute of Nuclear Materials Management *Contact:* Beth Perry, INMM, 60 Revere Dr., Suite 500, Northbrook, Ill. 60062 U.S.A. Telephone (312) 480-9573.

August 1-19, 1988
Seventh Annual Battelle International Program in R&D Management, Battelle Memorial Institute, Columbus, Ohio *Sponsor:* Battelle Memorial Institute *Contact:* Dr. William D. Hitt, Director, Battelle International Program in R&D Management, Battelle Memorial Institute, 505 King Ave., Columbus, Ohio 43201-2693 U.S.A.

January 11-13, 1989
INMM Spent Fuel Management Seminar VI, Loew's L'Enfant Plaza, Washington, D.C. U.S.A. *Sponsor:* Institute of Nuclear Materials Management *Contact:* Beth Perry, INMM, 60 Revere Dr., Suite 500, Northbrook, Ill. 60062 U.S.A. Telephone (312) 480-9573.

The events listed in this calendar were provided by Institute members or taken from widely available public listings. We urge INMM members, especially those from countries outside the United States, to send notices of other meetings, workshops or courses to INMM headquarters.

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