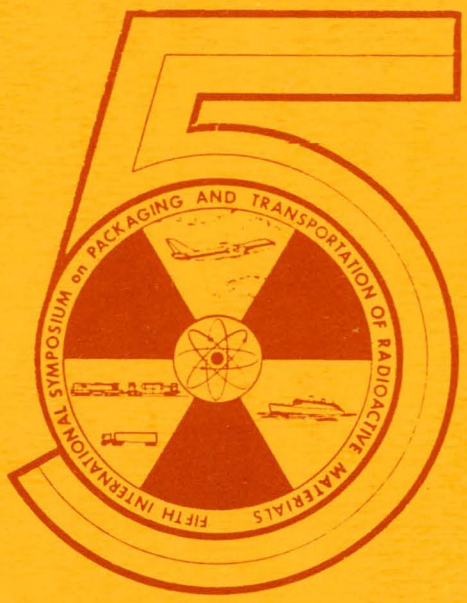
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PROCEEDINGS

OF THE

FIFTH INTERNATIONAL SYMPOSIUM

PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS

MAY 7-12, 1978

LAS VEGAS,

NEVADA, U.S.A.

VOLUME I

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PREFACE

The Fifth International Symposium on the Packaging and Transportation of Radioactive Materials was held May 7-12, 1978. This series, begun in 1965, has now developed into a significant opportunity for the interchange of information on the technology and politics of radioactive material transportation. The program for this Symposium was structured in an attempt to accomplish two objectives. First, to inform those in attendance of the current state of the art and thinking in our profession and, second, to stimulate your thought toward solutions for the problems we face. Since the previous meeting in Miami Beach in 1974 the interest in this field has grown considerably as witnessed to by the fact that there were over 150 papers submitted for consideration in this program.

In these bound proceedings you will find published 123 papers, the banquet address, and the detailed discussions by the panel on the Role of Risk Assessment.

The Symposium was considered a success. A total of 500 people registered at the Symposium, from 13 countries. The sessions were well attended and discussions were lively.

CONF-780506

5th International Symposium on PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS

May 7-12, 1978
Sahara Hotel, Las Vegas, Nevada, USA

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The Vital Line

Dr. George W. Cunningham
Acting Program Director for Nuclear Energy
U.S. Department of Energy

Good morning. I am indeed pleased to have the opportunity to address this group today. The challenge that is presented to those of us who are engaged in transportation of nuclear materials is immediate and urgent and is well served by the focusing of international attention such as this. I feel confident that what I will discuss this morning, while reflecting experience specific to the United States, will be relevant in many instances to all countries. The problems that face us are not unique, and collective efforts to solve them can be most beneficial. Looking back, the progress that we have made since the First International Symposium was held in Albuquerque in 1965 has been most gratifying. Looking ahead, cooperative endeavor will enable us to exchange experience, evaluate alternative approaches, and, thereby, continue to extend our accomplishments.

Before I discuss transportation of nuclear material, I'd like to share a few facts and figures that give a pretty good picture of how the nuclear industry as a whole is developing in the United States today. In 1977, the 67 operating U.S. commercial light water reactors generated 260 million megawatt-hours of electricity, an increase of 30 percent over the 1976 figure, and between 12 and 13 percent of the total U.S. electric production. This percentage figure is rather dramatic when it is compared to the figure of 1.4 percent for the nuclear contribution as recently as 1970. Electric utility generating capacity projections (including nuclear and nonnuclear) indicate a growth of from 260 gigawatts in 1976 to 1200 gigawatts in the year 2000. The projections of nuclear capacity indicate growth from the present 50 gigawats of installed nuclear capacity to 110 gigawats in 1985 and 330 gigawatts in the year 2000. This means that by the year 2000 nuclear power would account for about 25 percent of the total U. S. capacity. The utilities expansion program in the U. S. remains based on nuclear power, and the $55.2 billion spent on nuclear plant construction in 1977 is expected to rise to $55.8 billion in 1978. Thus, we see an expanding nuclear industry with increasing demands being placed on each of its elements. A critical link that makes possible the functioning of all current and proposed nuclear activities from mining through final disposal is the transportation of nuclear materials.

Let us turn our attention to some near-term transportation requirements. As you are aware, there exists today a growing backlog of nuclear materials, including spent fuel, DOE and commercial transuranic waste, and DOE high level waste. The transportation requirements for some or all of this material represents a potential burden which could fast become the limiting factor both in current operations and in maintaining the nuclear option. Concerns about transportation are reflected in a variety of forms: Congressional legislation; changes in Federal regulations; state and local jurisdictions developing restrictive transport regulations; heightened public concern; the paucity of container manufacturing capability; licensing reviews; and the need for Government decisions on system operations and parameters. The costs of nuclear shipments are increasing rapidly. These costs are both direct and indirect, the latter reflecting the cost of operational delays, public hearings and litigation, and overly restrictive procedures

The Presidentally-approved policy announcement on October 18, 1977, whereby DOE proposed, under certain conditions, to take title to and store spent fuel from private power reactors, underscores the urgent need for coordination and cooperation within the transportation industry. This policy would encourage the shipment, beginning in the mid-1980's, of substantial quantities of spent fuel to central repositories. The timely availability of the required transportation capability and equipment is essential to the success of our waste management program.

The evolution of a viable, balanced, and integrated transportation system depends upon the resolution of several critical and interrelated problems. Logistical problems -- such as types and numbers of casks and their handling systems, carriers, routes, and schedules -- that are encountered in serving hundreds of fuel cycle facilities in widely separated geographical locations are in themselves substantial, as are technological and safety requirements. Compounding these difficulties are the policy problems such as Government vs private ownership and responsiblities as well as institutional problems such as uncertainty and changes in Federal, state, and local regulations, packaging certification; and safeguard requirements. Providing public information and answering responsible intervenors is another area which will require an increased effort by both Government and industry. Of these, the policy and institutional problems could potentially present the most formidable obstacles. Although the other problems are important, in my opinion they are manageable if and only if the policy and institutional ones are resolved or significantly reduced. These two areas dictate the approach and direction of solutions to problems in the other areas.

In organizing our efforts to ensure the development of a transportation system that satisfies the regulatory requirements and logistics as well as the public concerns, there are several special issues that require attention. I would like to discuss a few of these critical issues with you, outline some of the steps already taken, and suggest some possible paths that might facilitate the accomplishment of our objectives.

NEED FOR INCREASED EMPHASIS AND COORDINATION

Historically, shipments of fuel cycle materials have moved with relative ease for a small fraction of the cost of nuclear power. Transportation has not presented major technical problems and, accordingly, the attitude that transportation problems should and will be solved by the private sector has prevailed. As a result, it has, at times, been handled in a fragmented manner and, in a sense, has been taken for granted. This attitude failed to recognize that the quantity of nuclear materials transported today constitues only a small fraction of the total goods moved by the overall transportation industry and, as such, efforts toward resolution of the related nuclear transportation problems have not been and are not likely to be forthcoming from this source.

Clearly, this attitude must be radically revised in the face of the growth that I have previously described. Transportation must be treated as a major element in any nuclear fuel cycle if such a pattern of growth is to be maintained. One requirement is a clear delineation of responsiblity. We cannot allow a situation to persist whereby a lack of structure within the system permits a haphazard approach to this challenging opportunity. Transportation is one of the most visible sections in the fuel cycle because of its frequency and proximity to the public on a nationwide scale. Therefore, the resolution of the problems that might deter the development of a viable transportation system deserve and demand the dedication and cooperation of all segments of Government and the nuclear industry. This may well entail numerous changes in existing practices and procedures that will require lengthy implementation. It most certainly will involve controversy, difficult negotiations, protracted administrative and judicial proceedings, and major public policy decisions.

GOVERNMENT LEADERSHIP

Where appropriate, the Government must provide the lead and support that is rightfully expected. As an initial step, a DOE nuclear transportation program has been established to provide the necessary coordination of transportation-related activities. The development of integrated plans and programs are now in an early stage of development. The responsibility for the overall planning and direction of transportation functions has been centralized and assigned to the recently created Division of Fuel Storage and Transport. This office will keep abreast of current requirements and will be able to contribute constructively to pending and proposed legislation/regulation that affect nuclear transportation. It will provide a focus and leadership for overall transportation issues. Coordination with the Department of Transportation and the Nuclear Regulatory Commission is underway. The development of a reliable nuclear transportation industry must be one of our highest priority efforts.

DEVELOPMENT OF A SYSTEMS APPROACH

One measure that could appreciably contribute to the alleviation of the institutional problems I have mentioned is the development of a systems approach. Historically, transportation has received little consideration in plant design and operations. Transuranic and high level waste facilities have been designed without consideration of transportation parameters and limitations. As a result, a safe, secure, reliable and affordable transportation system, designed to support the complete nuclear fuel cycle must now be developed to support the expansion of the nuclear industry.

Lead time for large casks of new designs run from six to eight years for design, licensing, fabrication, and testing. This means that beginning in 1985, radioactive material shipment schedules may not be attainable unless decisions are made now on parameters and specifications for canister size, shape, and heat output; spent fuel logistics; transuranic waste container optimization; and cask standardization. The time required for licensing review alone is presently three to five years and increasing. Studies must be completed based on the new non-reprocessing policy for truck vs rail vs water transportation systems; on the need for standardized designs; and on the need for single or dual purpose casks to ship both spent fuel and high level waste. Standardization offers many advantages: efficiency in handling, maintenance, decontamination, and testing; lower capital costs for casks and handling hardware; increased safety in handling; shorter turnaround times; lower labor costs; and better public acceptance. There is need to provide incentives to expeditiously restore a national capacity to fabricate large casks. The simultaneous solution of all of these related problems is a necessary aspect of the realization of our goal -- a smooth and safely operating nuclear transportation system.

TECHNOLOGY

Although our past record in technology is one to be proud of, there are some areas of concern that merit our attention.

Before I outline these concerns, let me briefly review a few past events that illustrate the productive benefits of our worldwide cooperation over the years. Design and handling of casks for shipping fuel was an important topic at the Atomic Energy Commission Symposium for Chemical Processing of Irradiated Fuels held in October, 1959. Shortly thereafter, the AEC organized an ad hoc committee to provide advice to the budding nuclear power industry and to develop recommendations on design and performance criteria for casks. After extensive tests, certain criteria were formulated. From the recommendations of this committee and the deliberations of IAEA panels came the present regulatory drop and puncture tests. The present fire test specification of complete exposure to an environment of 800°C was derived largely from British investigations of rail and highway accidents. A three day symposium was held in the United States in December, 1962, on the more general subject of packaging and regulatory standards for

shipping radioactive materials. Several countries already had active programs in this area, and the IAEA was busy developing regulations. The AEC then sponsored an International Symposium which was held in 1965 at Albuquerque, New Mexico. Developments since that time have been well documented in the proceedings of IAEA meetings and the four International symposia. The regulatory requirements have settled down to some extent. A large body of information which substantiates the validity of these requirements has been developed and is being applied to the design of containers which meet the requirements so that we are now in a period of refinement. More sophisticated methods are being used for design and for demonstration of compliance, and more attention is being focused on situations of very low probability.

With such refinement of methods, we may be approaching the point of diminishing returns, and should carefully consider the potential benefits before undertaking programs. For example, a container which survives the hypothetical accident conditions would also survive credible real accidents with no serious release of contents. The crash tests performed last year at Sandia Laboratories demonstrated good evidence of the degree of safety provided by packages designed to meet existing standards. However, additional tests should be performed to demonstrate that analytical techniques can accurately predict package failure. This will enhance the credibility of the nuclear transportation industry. The uncertainties and the risks need to be kept in proper perspective in the design and regulatory processes. Regulatory requirements continue to undergo scrutiny. There is no doubt that some weak points and some unnecessarily stringent points will be found. However, it should be recognized that the radiological hazard in transportation is so small that attempts at further reduction in hazard could increase the number and complexity of containers and could at the same time result in increased nonradiological hazards.

On the other hand, there are some design features for which additional information is needed. In particular, the limitations of different types of gaskets and seals, correlations between leakage of powders and standard leak tests, methods of small-scale modeling for testing closures, the distribution of dynamic loads and resulting stresses in packages subjected to impact conditions, the behavior of packages and contents in fire and other energy intensive environments, and the relationship between impact on essentially unyielding surfaces and impact under realistic conditions appear to be worthwile subjects for continued investigation. Maintaining the excellent safety record while providing viable transportation systems is of course a primary goal. To this end, program items of high priority include the proper operation and maintenance of shipping containers and a better understanding of the actual conditions to which packages may be subjected during transport.

The translation of baseline technology into pragmatic action is an activity which must be reassessed on a continuing basis. As I suggested earlier, the application of technology is complicated by policy and institutional issues. At present, availability of containers is seriously limited. The current commerical cask inventory in the U.S. can transport only about one-third of the present output from reactors, and there are no industry commitments to build more. Preliminary conceptual designs exist for high level waste casks, but detailed design has not been addressed. Transuranic waste is now shipped only in a limited number of DOE-owned railcars which are not available to industry and in a few overpacks which are privately owned.

Industry capability to build lead or uranium shielded casks is limited, and unless fabrication orders are forthcoming, it may soon disappear. The potential production capacity needed to meet 1985 requirements does not presently exist in the U. S. There is a need to decide whether to modify existing package designs or to design new ones in view of the changing transportation picture, i.e., the average heat load vs spent fuel currently expected to be shipped is lower than earlier projections due to longer decay times. I repeat, the problems we face are institutional as well as technological, relating to systems interfaces and definition of contents.

PUBLIC ACCEPTANCE

Many people feel that they cannot accept either the need for nuclear shipments or the safety of those shipments. This problem of acceptance is experienced by all facets of the nuclear industry. Fear of nuclear energy is easy to identify but difficult to combat. It is human nature to be suspicious and fearful of the unknown. Nuclear technology is highly complex, and only a small percent of the public has a good grasp of what it's all about. This lack of understanding, often abetted by biased journalism, has given the very word "nuclear" a negative connotation. It is associated in the public mind with two devastating bomb blasts that succeeded in bringing World War II to an end. We read even today of possible residual effects from contact with radiation during the testing of the bombs.

For some people, the unparalleled safety record and exhaustive proof testing established by the nuclear industry will never completely eliminate their negative feelings. This sentiment, if utilzied, can stagnate the growth of nuclear power to the point of paralysis. These attitudes affect the transportation segment of the industry by resulting in restrictive Federal, state, and local regulations and legislation, public overreaction to minor accidents, and restrictive tariffs by railroads. To alleviate this situation, testing programs should be expanded and more public visibility obtained. For example, recent good publicity on ERDA-DOE full-scale crash tests has helped. There is a critical need to put the very low nuclear material transportation risks in perspective for the general public. By aggressively developing, evaluating, and applying tough health and safety standards and by generating additional baseline technology to meet these standards, Governments can assure the people that nuclear power is a safe source of energy. Accomplishment of our goals depends on an accurate perception of the facts and a clear-sighted view of the contribution which nuclear power can make to this Nation's and the world's future.

In closing, we must wake up to the fact that the energy problem challenges our entire civilization. Our economy -- food, jobs, industry, income, homes, health, recreation, and transportation -- all of these and more depend on energy. An understanding of this - by Government, by industry, and, perhaps most importantly, by the people - is crucial to finding the necesary reponses to this problem. Nuclear power is an indispensible part of the answer. The transportation of nuclear material is a demanding and necessary element of this nuclear option. It is indeed the vital link for all phases of the entire nuclear fuel cycle; and, as such, the effectiveness of its operation must be ensured. This is a real challenge. We can, and must, meet it.

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NUCLEAR TRANSPORTATION: PROBLEMS REQUIRING ATTENTION AND ACTION

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The extent to which the nuclear power controversy has affected the daily operations associated with this energy source may be seen through its effects on nuclear transportation. In preparing this address I found myself thinking about the earlier part of this decade when I could have made a speech based on the simple theme that the vital role of the transportation sector in the nuclear fuel cycle necessitated the provision of efficient and reliable carriage. I probably would have exhorted the industry to establish standards and quality assurance programs, noted that historically shipments of fuel cycle materials have moved with relative ease for a small fraction of the cost of nuclear power, and sat down.

Today, however, we find a much different situation. In recent years concern has been expressed regarding the safety and security of nuclear materials carriage. Transportation rapidly is becoming a major limiting factor in both current operations and in the maintenance of the nuclear option. These concerns have manifested themselves in a number of ways and threaten the well-defined operational and legal framework governing shipments. Today we shall explore some of these impediments and what we can do to solve them.

Just recently my firm completed a study of the legal and regulatory factors affecting the establishment of international spent fuel storage facilities. We found that almost every country active with nuclear power regulates nuclear transportation activities in one way or another. Given the widespread adherence to the suggested regulatory requirements of the international Atomic Energy Agency, transportation does not appear to be a problem in the possible establishment of international storage facilities. Indeed, we found it reasonable to expect that whatever transportation licenses are required should be obtained. Nevertheless, while the legal procedures in most nations are well-defined and have posed no impediments in the past, the existing structures have not been thoroughly tested in the highly controversial climate which currently affects nuclear activities in general. An example of what may happen is currently occurring in the United States and therefore can be expected in time to be experienced in other nuclear power countries.

The state of nuclear transportation in the United States in not good. The reasons for this are several. They run the gamut from pure politics to lack of political courage to genuine concerns that transport of nuclear materials will degrade the environment or imperil human life. Concerns for the environment and human life are vital and necessary, but past history indicates that we can provide safe carriage. We all know that during the history of commercial nuclear power operations in this country, no accidents have occurred involving public injury. In the same period on American highways alone over 900,000 people have been killed by motor vehicles and more than 80 million have been

injured by this highly popular invention. Yet there is no popular movement to "ban the auto." Nevertheless, the debate over nuclear power and the transport of radioactive materials continues.

An overview of the restrictive measures which various entities have imposed or attempted to impose illustrate that the controversy has gone beyond the political level and has affected groups not ordinarily thought responsive to sensational-type charges. For instance, in 1975 the Congress passed legislation banning the air transportation of plutonium until the Nuclear Regulatory Commission could certify to the Congress that " a safe container has been developed and tested which will not rupture under crash and blast-testing equivalent to the crash and explosion of a high-flying aircraft." The Department of Energy has taken over its own shipments of strategic special nuclear materials, excluding the private sector entirely. The NRC has tightened up its requirements considerably -- both in quality as well as quantity of supporting data. In the last several years state and local governments have set increasingly stringent conditions for the transport of nuclear materials. Finally, carriers have attempted to restrict their carriage of these materials. The Air Line Pilots Association has sought to ban the transport of radioactive materials on passenger flights. And groups of railroads have contended that either they are not common carriers for the transport of nuclear materials or, if common carriers, that the carriage must be by the use of special trains and special operating procedures at great additional expense.

The expansion of restrictions is perhaps best illustrated by the increase in state and local jurisdictions asserting additional controls over shipments within their boundaries. In the past, states have been content to exercise their inherent police power over such shipments in a conservative manner. State highway departments, for instance, have almost uniformly regulated gross vehicle weights, axle weight and vehicle dimensions. Some have required pre-notification and special routings. All of these requirements have imposed minimum hardship on nuclear materials transport.

Recently, however, states and localities have become more aggressive in their attempts to regulate the carriage of nuclear materials. The Oregon Energy Facilities Council has recently promulgated regulations governing the issuance of site certificates which requires the Council to consider the transports of wastes from the facility in determining whether the facility reduces to the extent practicable the risk of injury to the public health and safety "... in Oregon, or in adjacent areas that are directly impacted." The agency intends to make this finding only after examining the proposed facilities compliance with the requirements of agencies, including the NRC. Connecticut passed a law in June 1976 which regulates the transportation of radioactive materials shipped from or through the state to a waste disposal site of facility. The New York City Board of Health has issued a regulation which prohibits the transportation of radioactive materials within the city limits, absent the issuance of a Certificate of Emergency Transport. Such certificates would be issued only "for the most compelling reasons involving urgent public health and safety concerns." Obviously, this standard all but bans shipments through the City. Suffolk County, New York is considering similar legislation as is Cincinnati, Ohio.

The effect of these constraints is significant. Not only do they raise the possibility that varying state and local regulation will greatly increase the cost of the carriage of nuclear materials, but they also could lead to a decrease in safety due to a moratorium on such shipments or the use of indirect routes. When New York City enacted its ordinace in 1976, Brookhaven National Laboratory on Long Island used trucks to transport spent fuel from its High Flux Beam Reactor through New York City on its way to Savannah River for reprocessing. As a result of the ordinance, Brookhaven began barging the fuel to New London, Connecticut for reshipment. Now New London has imposed its own ban on nuclear shipments through that city, and it is unclear where Brookhaven will turn next. At the very least, we are beginning to see a checkerboard pattern emerge which contains certain squares through which no carrier of radioactive materials may go.

A related concern, one also caused by the present uncertainty surrounding nuclear power, is the lack of places available to receive shipments. The Barnwell facility is in a state of limbo; the Morris facility is being used as a storage pond within its limited capacity; and the West Valley facility is closed. Low-level waste sites also are closing down with disturbing rapidity. In the early 1970s six sites were available throughout the United States; by 1974 this number decreased to four; and now we have only two sites -- Beatty, Nevada and Hanford, Washington -- still accepting low-level wastes. The shipment of these wastes from facilities in the East to the Far West is costly and could substantially affect nuclear power operations in the United States. This has created a growing backlog of nuclear materials to be shipped.

Assuming that a facility operator finds a carrier who is willing to transport his shipments, a route not affected by overly restrictive state and local regulation and a facility capable of accepting the shipment, he still must obtain a cask in which to ship the material. This may not be an easy task in the years ahead. The Department of Energy Task Force report on waste management found that at present there is a serious lack of availability of containers for spent fuel, transuranic waste, and high level waste. The current commercial cask inventory can transport only about one-third of the spent fuel output from reactors and there are no industry commitments to support the building of more. One conceptual design exists for a high level waste cask but none has been built, and no safety analysis has been prepared. Transuranic waste is now shipped only in DOE-owned railcars. These are neither licensed for nor available to industry and are limited in number. To transport the current inventory of transuranic waste stored retrievably from the Idaho National Engineering Laboratory to the Waste Isolation Pilot Plant (WIPP), would take over 12 years if started now, using the best equipment currently available. To transport the transuranic waste predicted to be stored by 1985 would take over 28 years using all of the railcars available. This situation, of course, can only be exacerbated by the growing number of nuclear materials and wastes which must remain on-site pending a decision where to ship them.

These then are the problems. What can we do about them? First, a basic need, often ignored, is to educate the public of the extremely impressive safety record associated with the transport of nuclear materials. These materials have been transported worldwide for the last thirty years and thus far there have been no accidents resulting in any injury or significant radiation exposure from such activities. Last year more than one million shipments of nuclear materials occurred in the United States alone and few incidents were associated with them. The transportation industry and those in government responsible for its regulation have succeeded admirably in fashioning a well-defined operational and legal framework in which to conduct their activities. The constraints on transportation operation which we see developing have not been imposed by extreme anti-nuclear forces. For the most part they are responsible citizens who are responding to public concerns, concerns which too often in the past have gone unrefuted. Unfortunately, people's knowledge of nuclear energy involves little fact and an inordinate amount of fear. The only way we can counter these fears is to trust the public to understand the difference between headlines and fact. Transportation cannot be assured without public confidence and acceptance that it is a safe operation. It is our job to see the facts to support this public confidence is disseminated.

Second, the Federal Government can take numerous steps to reduce the uncertainties associated with nuclear materials transport. An obvious initial step is the resolution of the broad issues now affecting the fuel cycle. More specifically related to the transportation sector, it should reexamine the existing allocation of responsibilities among the various agencies and, where appropriate, revise the interdepartmental memoranda of understandings in an attempt to reduce duplication and lack of coordination. Agencies should be encouraged to resist the temptation to expand their jurisdiction, following the example of the Interstate Commerce Commission which recently concluded that it should follow NRC and Department of Transportation (DOT) Safety Standards rather than developing their own. Any they should be discouraged from acting for their own parochial political

interests when broader national energy policy matters are at stake, as was possibly the case with the recent DOT decision to uphold the New York City ban.

The Federal Government also should be concerned about spent fuel cask capability and have a stand-by inventory to forestall the problems of cask shortages. DOE's own figures illustrate the problem; today cask supply is capable of moving 600 metric tons (MTU); by 1983, 1700 MTU must be transported; by 1990, 14,000 MTU must be shipped. While the demand exists in the years ahead, the manufacturing capacity is not expanding given the abnormal economic risk associated with the nuclear industry today. Both government and industry should proceed now to remedy this problem.

The Federal Government, more specifically the NRC, should move to stabilize the regulatory process and conduct a thorough confirmatory research program. Recent full-scale crash tests of shipping casks have been completed and are reassuring. The NRC should build upon this base and conduct a Rasmussen-type study applicable to all modes of radioactive materials transport. These studies can be factored into the regulatory process so as to avoid unnecessary design changes and backfits. This is especially necessary to stabilize the Commission's regulations governing physical security.

Finally, although it is my opinion that the Atomic Energy Act preempts state and local control of the transport of nuclear materials except on non-nuclear requirements applicable to all shipments of any kind, the Federal Government still should work with the states and localities to ensure that their concerns are taken into account. While the recent Department of Transportation ruling concerning the New York City ordinance was incorrect, in my opinion, I do belive the Department made a wise decision in announcing that will hold a rulemaking to address the extent to which routing requirements are necessary to insure the general safety of radioactive shipments. There is a useful and important role for state agencies in the transportation of radioactive materials in the areas of compliance and emergency response in event of accidents. Cooperation from state agencies, nationwide, in compliance and inspections would assist both NRC and DOT and be an important contribution to our efforts to assure the public that its safety is not being neglected. In addition, state programs to develop emergency response plans, procedures and capabilities are essential even though we do not expect significant radiological incidents. However the states, for their part, must recognize that the nature of transportation activities is such that it requires uniform national treatment.

While the recommendations I have made have been directed toward the Government sector, few, if any, of the proposals can be enacted without the support of the nuclear industry. In this regard an industry commitment to standardization offers many advantages: efficiency in handling, maintenance, decontamination, testing; lower capital costs for casks and handling hardware; increased safety in handling; shorter turnaround times and lower labor costs; and better public acceptance. Strict assurance to quality assurance programs offers many of the same advantages. The onus of providing a relatively inexpensive, workable transportation system is on neither the Government or the industry exclusively. Rather both must take action, remembering that the safety record of nuclear materials carriage is not just fortuitous, but the result of a mutually responsible industry and government.

TRANSPORTATION

SYSTEMS

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TO STANDARDIZE OR NOT TO STANDARDIZE -- THE PROS AND CONS OF STANDARDIZED DESIGNS FOR CASKS AND CASK TRANSPORT SYSTEMS

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ABSTRACT

There are many different cask designs that have been approved, built, and used over the years. Some special-use vehicles have been built. In the U.S., there are now over a dozen "current use" casks. With a sharp increase in the number of spent fuel shipments coming up in the next decade, and the beginning of high-level waste shipments in the late 1980's, there could be a proliferation of cask and transport vehicle designs. But is this a good idea? Should there be more casks of fewer designs, or are custom-designed casks and vehicles a better way to go? This paper examines the advantages and disadvantages of standardized designs for casks and cask transport systems. Factors of safety, economics, efficiency, and public acceptance are considered. Examples of other transport hardware standardization are given and evaluated. The effects of cask standardization on reactor and processing plant design and operations are examined, as well as the effects of plant standardization on the transport system. This is a "chicken and egg" situation, and will require close cooperation between designers of transport system hardware and plant hardware.

BACKGROUND

There is a growing trend towards more standardization in nuclear power stations[1/, 2/]. The standardized plant offers much in the way of improved safety, materials costs, procurement times, reliability, economy, and shortened licensing time. We see standardization all around us, in the form of electrical sockets, CB antennas, hand tools, and ball point pen refills. We see some standardization in shipping cask design, but not much. Should there be more? In the nuclear transport business, should we match this industry trend toward more standardization? There are many different cask designs that have been approved, built, tested, and used over the years. There are 46 different designs in the current DOE package directory[3/]. There are a half-dozen different commercial spent fuel cask designs in current domestic use[4/]. Some special-use vehicles have been used. With a sharp increase in the number of spent fuel shipments coming up within the next decade, and with the beginning of high-level defense waste shipments in the late 1980's, there could be a proliferation of cask and transport vehicle designs. But is this a good idea? Should there be fewer casks of fewer designs, or are custom-designed and special-purpose casks and vehicles a better way to go? With the current delays in the nuclear fuel cycle,[5/] it is the perfect time to take action to institute an aggressive cask standardization program, *if* it is to be done at all. But *should* it be done at all?

We found it interesting to read a Westinghouse brochure[2/] which discussed the standardization of nuclear power plants. We took some of the words from that brochure, paraphrased the text, and substituted the words "casks" and "vehicles" for "nuclear power plants." We did this as a first step to see if the overall benefits of standardization of casks and vehicles would parallel the benefits of standardization in nuclear power plants. We came up with the following:

> "Good management implies standardization of parts and components, and also of casks and vehicles. To the regulatory authorities, standardization means a reduction in the amount of original material to be reviewed for safety compliance. NRC and DOT would have fewer people to deal with and less work to do. To the cask supplier, standardization means more easily understood designs, more repetitive shop work, and less customizing, all of which means lower cask costs. Components may be manufactured on an even flow basis, and the attendant benefits of reduced cost, predictable schedule, and consistent high quality may be achieved.
>
> "To the architect-engineer, standardization of casks means the ability to use existing known designs and, hopefully, preapproved designs. It means firm design information and more efficient procedures and equipment in the plants. Costly and time-consuming repetitive analysis and design of handling areas and equipment are minimized. Earlier completion of preliminary plant design, early licensing applications, and early plant start-up all mean profits.
>
> "But the greatest benefits are to the shipper and the receiver. Standardization means confidence in the safety, reliability, maintainability, licensability, and predictability of the construction, start-up, and operating schedules. Standard components permit equipment interchangeability between shipping and receiving plants; which provides flexibility to meet operating schedules."

On the face of it, there appears to be a good parallel here. But perhaps it would be more relevant to examine the specific advantages and disadvantages. In doing so, it would be useful to set a few of the basic ground rules in my discussion of standardization.

First, a basic design principle that would need to be incorporated in all stages of development of standardized transport systems is that only proven materials, equipment, and systems would be used[1/]. All experimentation, design, and testing must be done prior to, and separate from, the operation of the shipping and receiving facilities. For example, one should not start up a high level waste storage facility using a cask and cask handling system that has not been thoroughly debugged and refined.

Second, there are several degrees and methods of implementing standardization of casks or vehicles:

- Duplicate casks/vehicles.
- Cask/vehicle/facility interface requirements.
- Compatible/standard hardware (yokes, seals, trunnions, valve attachments).
- Centralized design/procurement/fabrication.
- Centralized management control of hardware.

The final ground rule is that standardization need not be absolute in order to be useful. Two alternative designs may not be as standardized as just one design, but it's still not as diverse as six or seven or twenty. Intentional deviations from a standard are far better than having no standards at all.

There are real dollars to be saved by standardizing. Custom designs are expensive, and each requires a separate set of analyses and tests. By limiting the number of separate designs to the minimum practicable, savings can be realized in several areas:

Design and Analysis:

A large cask may cost nearly a million dollars to design, analyze, and perform safety tests. The more casks there are to spread these costs over, the lower the design cost per cask.

Licensing:

Once the design is complete and the safety analysis report written, the long and expensive NRC licensing review begins. NRC licensing fees are significant, and a NRC cask review routinely takes 3-6 years. The applicant spends much time meeting with NRC, preparing answers to questions, and in redesign and reanalysis. Time is money. This will be discussed in more detail later.

Materials and Equipment Costs:

Castings and machinings are cheaper if standardized and made in quantity; 8 or 10 individual sets, all different, are expensive and time-consuming to make. Spare parts management is easier with a smaller inventory and greater interchangeability. Plant equipment can be tailored to a particular cask and vehicle design, thereby saving money on the capital cost of the plant. With several casks of a single design, both the shipper and receiver each needs only a single set of handling yokes and other hardware, plus a spare somewhere. This saves on storage space at the facility and saves the cost and damage risk of having to ship the handling hardware back and forth.

Vehicle Costs:

A single design means a single tiedown system, and a standard approach to each vehicle. The interchangeability of vehicles means better availability of rolling stock, and more efficiency in its use.

Handling Costs:

If the shipper and receiver can plan on seeing the same or duplicate casks and vehicles each time, handling time and costs can be reduced significantly[6]. Operations become routine rather than special. Plant operational procedures can be tailored to the casks and vehicles, thus realizing even more savings; proven methods and procedures mean more efficient use of time and less loss due to damage.

Fewer Casks:

Standard designs will result in greater efficiency of use of the casks, and thereby reduce the total number of casks needed.

Three major economic disadvantages of standardization are frequently mentioned. The first is a concern that adherence to a fixed rigid design may tend to discourage experimentation with new cask concepts which might otherwise lead to savings.

The second disadvantage is that the lack of custom designs means that the cost per shipment might go up because of design inefficiencies or operational limitations. For example, a long cask is not efficient for shipment of short fuel; a large cask designed to hold seven assemblies is not efficient for shipping only one or two assemblies. This latter problem can be largely overcome by designing in some flexibility. Optimization for one reactor may result in inefficiencies for another reactor. A 75-ton cask can't be shipped by truck, and a 20-ton cask

might move more economically by truck rather than by rail. So -- two separate designs, one for truck and one for rail, might be appropriate. The next step could be two truck casks -- one short and one long; and two rail casks along the same lines. Both truck casks could use the same handling hardware, and perhaps the same tiedown skid, as could each of the two rail casks share their hardware.

The third disadvantage is that existing casks could be made obsolete by adoption of a standard that they did not meet. As much as casks costs, we can't disregard the reality of this economic penalty. However, this penalty can be avoided by the use of a "grandfather clause." There are still few enough casks[7] that we can live with a few mavericks, and still be much better off than having dozens of different designs. Another way to avoid this problem would be to standardize on those existing casks, at least for spent fuel.

There are a couple of other economic factors worth mentioning. The costs of the lack of system engineering and the resultant lack of standardization can be exemplified by the experience of Philadelphia Electric's Peach Bottom nuclear plant during a single shipment of BWR spent fuel by truck to Idaho in April 1977[8]. It required a detailed exchange of information between the shipper, the receiver, and the cask manufacturer. This cost money. In the future, if it has to be done frequently because of the use of different cask designs, those costs would be multiplied. The general cask handling, loading, and unloading procedures supplied by the cask manufacturer had to be rewritten by both Peach Bottom and Idaho to include the specific details unique to their own sites. This cost money. The loading area at Peach Bottom wouldn't accept the entire 55-foot tractor/trailer rig, so the operation had to be carried out with the rig extending part way into the car bay access corridor. There was insufficient overhead room to dolly the crane to keep the crane cable vertical as the cask was lifted from the trailer; the rig was slowly driven forward beneath the crane as the cask was lifted. With standardized equipment and procedures information available to the reactor designer, this design limitation could have been avoided. Or, if the cask designer could have worked with standardized reactor handling facilities, the reactor system design could have been incorporated into the design of the cask transport system. Lack of standardization cost money. There were many delays in handling due to unfamiliarity by the cask loading crew with the procedures and equipment. This cost money. Philadelphia Electric concluded:[8] "The utility should employ one cask design and one loading crew so that speed and confidence increase with experience. All cask compatible fittings should be received well in advance and all auxiliary equipment should be given a functional test at the utility and at the cask vendor (and the receiving site) prior to cask receipt." (parenthetical phrase added). The message is clear, but the designers of casks, reactors, and processing plants don't seem to have read it -- they're still not really looking at joint design of a total system -- standardized and compatible. Perhaps we can do better in DOE in the design of spent fuel storage facilities and high-level waste repositories.

SAFETY

The economics of cask transport are usually hidden in total operating costs, although the transport fraction is increasing[7]. But safety, or lack of it, is usually more visible to the public. Nuclear accidents make headlines, even if minor. One of the biggest and more tangible advantages of standardization is handling safety.

When one is able to set up a routine procedure, using consistent procedures, a uniform philosophical approach, and standardized hardware, one generally makes fewer mistakes.

Constant handling techniques and practices lend themselves to a good training program. Simple rules, consistently followed, mean good compliance and fewer mistakes. It's easier to learn one check list well than it is ten, and one quickly develops a "feel" for routine operations that make it easier to spot potential problem areas. There would be less "jury rigging" and "make-do" practices because of the number of new and unexpected handling situations would be minimized.

Standardized hardware would lead to better quality assurance. QA efforts can be concentrated on a few operations, with fewer opportunities for error.

Better work can be done in safety analysis if the analysts can concentrate on a single system or even two or three, rather than a dozen. This will improve safety, and hopefully also increase public confidence by common, well-known, tested, and proven methods and equipment.

Contamination control is better with standardized casks. It's easier to miss something if you don't know by experience where the potential problem areas are.

From a design viewpoint, standardized hardware makes it easier and cheaper to provide for safety-related items such as cooldown equipment, radiation level (and exposure) control, fuel storage and decay time, cask handling area and equipment, door and air-lock size, etc. Spare parts management can be more efficient if the number of parts is kept to a minimum. With only a few parts to deal with, one is more likely to replace a worn part with a new one rather than to try to make do with a poor substitute.

On the other hand, the cliche, "familiarity breeds contempt," is often applied to the principles of safety engineering. In jobs which are totally repetitious, safety procedures may end up being complied with as a matter of rote rather than conscious analytical action. Cask handling could fall into this category. Perhaps daily operations with the same old hardware might lead to some lackadaisicalness. But right now, "variety is the spice of lift." Workers aren't likely to become complacent about repetitious and routine procedures when there's a different checklist for each cask. The importance of this disadvantage seems small in comparisonto the advantages.

RELIABILITY

We might call this section "Predictability," for what we want is to have a hardware system whose reliability is predictable and positive. A standardardized system allows one to "fine-tune" the operation much better than he can with a plethora of designs. Down-time costs money -- a cask might rent for several thousand dollars a day. One needs short turn-around times and low maintenance. This comes with experience. Look at the history of the Volkswagen. By sticking to the same basic design for years, Volkswagen was able to consistently produce a car of high reliability and low defect. Major components were standard, and interchangeable from year to year. With proven designs, Volkswagen engineers could concentrate on refined solutions to small problems. Detroit gives <u>its</u> engineers a whole new set of opportunities each year to make errors! And then follows up with massive vehicle recalls to correct the errors.

LICENSING

People who have been in on a major cask licensing effort recently understand that it can take 3-6 years to get a new design licensed. They may not understand <u>why</u> it takes so long, but they accept it as a present fact of life. It costs the shipper money -- up to $75,000 for a cask license. But the cost to the taxpayer is many times that $75,000. It takes many hundreds or thousands of man-hours on the part of the applicant and his subcontractors. Multiply all of

that by the number of cask designs. Fewer designs mean more emphasis on those few, and a better job. A smaller workload at NRC also means shorter review times (although a believer in Parkinson's Laws might dispute that).

OTHER FACTORS

Standardized casks will provide a better long-term statistical data base. Problems can be more easily isolated and solved; retrofits will be easier and cheaper. The total number of spent fuel shipments in the U.S. is still small -- perhaps 4,000 or so in the last 25 years. The operation isn't really "old hat" yet. Failure and deficiency reports will have more meaning with a broader data base. The experience of a single cask design, evaluated over millions of miles, may well be more meaningful than the experience of more designs over fewer miles. The experience of the Navy in cask standardization can serve as a shining light to guide a standardization program. The experience in other standardized container programs can be helpful, too: uranium hexafluoride cylinders, steel drums for birdcages and low-level waste, the 6M and 6L DOT specs.

In spite of the growing trend toward standardized nuclear power plants, there is still some resistance from reactor designers and theoreticians who don't want to be told to follow certain design limits or physical characteristics. "Don't fence me in!" Many operators still want their shipment security blanket -- their own shining stainless steel status symbol -- their cask.

Choosing a single standardized design could raise questions of patent and proprietary rights. If the Government should settle on just one commercial design, this could cause problems involving royalties or manufacturing rights for other commercial firms. Of course, if the Government makes the design, the problems would be eased because of public accessibility to the patent rights.

VEHICLES

Many of the advantages and disadvantages of standardized casks would be equally applicable to the design, fabrication, and use of the transporting vehicle. In the transportation industry, there is already a significant degree of standardization of vehicles, vehicle sizes and configurations, and vehicle components. A 40-foot box car which is owned by one railroad can move freely over the rails of any other railroad. One needs only to look at the makeup of a freight train to note the cars of different ownership in the train. Standardized trackage and rail car coupler systems make this possible.

Trucking operations are not totally different. Trailers can be interchanged between different trucking companies because of a standardized "fifth-wheel" system. Tire sizes and widths are standardized, as are many engines and some other vehicle components. Repair and replacement parts are largely interchangeable. So there already is a great deal of standardization in trucks and rail cars.

In the case of vehicles for radioactive material shipments, the term "dedication" might be more appropriate than the term "standardization." An often-voiced complaint by traffic managers is that they frequently can't get a car to meet their shipping schedule. Unwarranted delays can cause program problems -- storage and extra handling for example -- with attendant extra direct and indirect transportation-related costs. What is significant to the shipper is the overall cost benefit of having a readily available (dedicated) vehicle to meet his own specific shipping requirements. The cost effectiveness of a "dedicated" transport vehicle is measured by increased program efficiency and transportation economy. Minimized handling and standardized loading/ handling procedures provide increased safety; a spin-off, but no less valuable benefit.

Where nuclear materials are concerned, a transport vehicle is more likely to be "dedicated" for transporting heavy shipping casks[9/], such as those in the 75-ton and heavier weight range. Shipments in this weight range generally are moved by rail.

It might be useful to look briefly at a specific historical case of spent fuel shipments, in casks weighing over 100 tons, on railroad-owned general duty cars, as compared with the same shipments on shipper-owned "dedicated" rail cars.

CASE HISTORY

A depressed center flat car with a rated carrying capacity of more than 125 tons was needed to accommodate the concentrated weight and transport characteristics of a heavy cask in one of our nuclear programs. Private industrial shippers of nonnuclear articles such as generators, transformers, and heavy machinery also used rail cars of this type. At that time, there were relatively few railroad cars of this type in service which could accommodate these loads, and we competed with those nonnuclear industrial shippers for the cars.

Because of this heavy demand, long lead times were needed to get cars for loading, and expedited off-loading was necessary to return the cars to the railroads for other service. Transportation under those circumstances was time consuming, costly, and programmatically burdensome.

For example, the total elapsed time for a one-way 1,000 mile trip, from the time that a car was ordered from the railroad until it was delivered at destination and released back to the carrier, ranged from 60-95 days.

In addition to the 30 or more days to get a car placed for loading, there were lengthy delays at origin and destination inspecting the car, preparing it for loading, loading/unloading, and monitoring (and perhaps decontaminating) the car prior to returning the car for other service. The total holding period for loading/unloading and servicing the car (about 25 days) exceeded the normal 5-6 day time allowed by the railroads for loading and unloading the car. The adverse economic impact was substantial.

Because of the ongoing and growing need for this type of a rail car, we decided to buy some rail cars for our own exclusive use. We made this decision because, despite detailed shipment planning and control, unexpected program changes at origin or destination resulted in detention of the railroad cars beyond normal requirements. We were incurring significantly high car detention, demurrage, and use charges. In addition, since the cost to offload, reload, secure, and inspect spent fuel casks prior to shipment was (at that time) about $30,000 (1970 dollars), it was decided to permanently affix the shipping cask support structure to our own rail cars.

ADVANTAGES AND DISADVANTAGES

Some of the relative advantages/disadvantages of a shipper-owned (dedicated) car as compared with a railroad-owned (general duty) car are shown below:

		Railroad-owned Cars (General Duty)	Shipper-owned Cars (Dedicated)
1.	Availability	Short supply. Compete with other nuclear and nonnuclear shippers. Long lead time needed to get car. Additional cask usage charges due to delays.	Car always on hand. Car can be used for training personnel in cask loading/unloading procedures.
2.	Services at Origin	Detention of car for inspection, preparation, and car loading - about 15 days. Increased potential for cask damage during loading/handling operation. Accrual of high detention, demurrage, and use charges.	Detention of car about 3-4 days. Potential for cask damage eliminated. Almost no additional accessorial service cost.
3.	Services at Destination	Detention of car for off-loading and decontamination about 10 days. Accrual of high detention, demurrage, and use charges. Increased potential for cask damage during offloading operation.	Detention of car about 2-3 days. Potential for cask damage eliminated. Almost no additional accessorial service cost.
4.	Unforeseen program delays	Accrual of detention, demurrage, and use charges. Holding of cars erodes good shipper/carrier relations.	No adverse effect.
5.	Total Shipment Time (1,000 mile trip)	60-95 days (total elapsed time).	10-12 days.
6.	Costs	Cask Amortization ($3,000/day) $180,000 - $285,000 Rail Freight $5,000 (does not include special train) Detention, Demurrage, Use Charges - $15,000	Cask Amortization ($3,000/day) $30,000 - $36,000 Rail Freight (does not include special train) $5,000 Detention, Demurrage, Use Charges - None

CONCLUSION

What's the bottom line on the trade-offs of cask and vehicle standardization?

We believe the two should be treated differently. First, vehicles are already adequately standardized, in our opinion. But dedicated vehicles, whether trucks or rail cars, or even ships, can provide sufficient economic and operational benefits to justify their use in any cask transport system.

Second, as to casks themselves, there is no question in our minds as to overwhelming superiority of standardized casks. Whether you are operating a reactor, a spent fuel storage facility, a reprocessing plant, a high-level waste repository, a truck line, or a railroad, the advantages of being able to handle a known and common cask far outweigh the disadvantages. Those advantages involve safety, economics, and operations, as well as the legal and public acceptance aspects.

Next, the time to embark on a program of standardized casks is <u>today</u>. Washington's former football coach, George Allen, often said, "The future is now." It's just as true in casks as it was for the Redskins.

Operationally, there are too many shipments coming up in the future for everybody to do his own thing[9]. Ten years from now, there are likely to be several thousand shipments of spent fuel per year. Now is the time to decide on a standardization policy -- <u>before</u> a large scale cask design and fabrication program is undertaken. Cask standardization <u>now</u> could be a major contributing factor to a standard nuclear fuel system throughout the nuclear fuel cycle -- fabrication of fuel assemblies, reactors, fuel storage, reprocessing, and transport.

At the present time, most spent fuel shipments come out of the Navy program and move into a Government reprocessing plant in Idaho. The Government has standardized the system. Little commercial spent power reactor fuel is moving. Some research reactor fuel is going to the DOE reprocessing plants in Idaho and Savannah River. One future option is for the Government to become involved in the spent fuel storage business in a big way; or, spent fuel storage facilities might be owned and operated by the nuclear industry. With that in mind, let's look briefly at who might shepherd this standardization effort for high-level waste and power reactor spent fuel casks.

THE SHIPPER

Shipments of spent fuel will be coming out of 50 or so reactors in 1985, or earlier. Those reactors are already built or under construction, so it's a little late to expect them to be much interested in cask standardization. Further, there is little they can do to change what they already have. In any case, they're probably not in a position of leadership in this area. Nor is anyone else in the nuclear industry, unless the Atomic Industrial Forum would care to take on the task. I've seen no indication that will happen. No individual commercial shipper is in a position to speak for the others, or to make the commitments on a matter of such importance. I question whether AIF is really in a position to do so, although perhaps they could do if there was a change in their goals and objectives.

The other big shipper is the Government. Shipments of high-level waste will be coming out of DOE reprocessing plants in Idaho, Savannah River, and perhaps Hanford. Since the Government is both the shipper <u>and</u> receiver for these wastes, DOE might be in a good (and perhaps the best) position to take the lead in standardization of high-level waste casks and hardware for its own operations, working with the transportation industry to ensure a total systems approach.

THE CARRIER

Carriers are flexible people who and will try to accommodate any load a shipper has. Their equipment is fairly standard now. Although there would be some benefit to the railroads in dedicated cars and standard tie-down and handling gear, there's little incentive for them to take the lead in cask standardization. He needs to be involved in the standardization effort, but not to control it.

THE RECEIVER

The receiver's facility is where all casks focus in. All roads lead to Rome, and Rome becomes the center of control. In this case, Barnwell notwithstanding, Rome is not yet built, and so is a prime candidate to be the target of a major standardization effort. If a comprehensive and pervasive standard design effort can be initiated now, in the early stages of facility design, the benefits can be reaped both before and after startup. It would be satisfying to know that the fuel storage facilities (whether the Governments' or industry's and the Federal waste repository would start off operations with casks and handling equipment that all look alike, and where both the shipping and receiving plants are designed around a single cask system! Conversely, the cask system would be designed with the initial plant parameters in mind. Nirvana to a cask man!

But who would take the lead in bringing this about? Well, how about the Government? It looks more and more like Uncle Sam might be a major receiver as well as being a major shipper. Because of that status, and perhaps also by default, I think that the U. S. Government, and specifically DOE, is perhaps in the best position to take on this responsibility, if it is going to be done at all. There just doesn't seem to be any other reasonable option. In my personal opinion, nobody else is in the same unique position to pull the whole thing together.

SUMMARY

Standardization of casks offers many advantages in safety, economics, operations, reliability, interchangeability, licensing, and public acceptance. There are a few disadvantages, but in my opinion these are far outweighed by the advantages. Standardized casks should be used with dedicated truck/trailers and railcars in order to gain significant operational, economic, and safety benefits. Because of the growing U.S. Government involvement in the back end of the nuclear fuel cycle, I believe that the Government (and specifically the Department of Energy) is probably in the best position to direct the standardization program, if there is is to be one at all. But for the program to be useful and effective, the effort should begin now.

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Dup

CONCEPTUAL DESIGNS OF SPENT-FUEL SHIPPING CASKS FOR THE U.S. BREEDER REACTOR TECHNOLOGY PROGRAM*

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INTRODUCTION

Conceptual designs of spent fuel shipping casks for use by facilities participating in the Breeder Reactor Program have been developed by Sandia Laboratories, Albuquerque, New Mexico. Some aspects of conceptual designs that are reviewed in this paper are interface requirements and cask design features. The major emphasis of the work to date has been to develop cask designs which would be suitable for use at the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant (CRBRP). However, the advanced technology utilized in developing spent-fuel shipping cask designs for these facilities results in many cask features which would be required by all breeder reactor plants.

INTERFACE REQUIREMENTS

Spent-fuel shipping casks must fill the dual role of being both plant equipment and transportation packages.[1] Shipping casks are an integral part of the plant facilities during loading and unloading of spent fuel. The methods of loading/unloading, sealing, controlling contamination, cooling during loading/unloading, adding supplemental shielding to reduce personnel exposure, and handling and testing of casks affect both the cask design and plant features at the system interface. Details of the interfacing systems should be established by jointly optimizing plant and cask systems. Ideally, this will result in the development of a cask system which has the flexibility to be utilized efficiently at diverse locations. A standardized shipping system for all new plants in the Breeder Reactor Program could result in interface simplification, e.g., reduced turnaround time for cask loading/unloading.

Interface considerations between spent-fuel shipping casks and breeder reactor plants are shown in Table 1. While there are some differences in handling methods between FFTF and CRBRP which complicate interfacing, advanced facilities could use standardized interface features which would tend to simplify handling at the plant. The transportation mode selected will affect spent-fuel shipping cask (SFSC) designs. Maximum weight and size limitations are examples of constraints placed on the shipping cask by the mode of transport. Truck transport has a maximum-payload weight restriction of about 22 tonnes, unless overweight permits are obtained from state agencies. Rail shipments with payloads of about 100 tonnes can be handled routinely. Because of the larger payload-to-weight ratio available with rail shipping casks, rail casks are expected to be used for a large fraction of breeder reactor fuel.

*This work is supported by the United States Department of Energy.

Table 1

Breeder Reactor Spent-Fuel Shipping Cask Interfacing Features

	FFTF	CRBRP	Advanced
Loading/ Unloading	Rotating basket or cask to index one position at a time. Dry unloading in vertical position.	Rotating basket to index one point or hot cell concept to have simultaneous access to all fuel positions. Dry unloading in vertical position.	Hot cell concept to have simultaneous access to all fuel positions. Dry unloading in vertical position.
Sealing	Small plug seals in lid.	Small plug seals in lid or large seal for entire lid.	Large seal for entire lid.
Contamination Control	Valves	Bagging	Bagging
Cooling While Loading/ Unloading	Not needed for low-heat producing assemblies	Canister fuel, or forced gas system	Canister fuel
Supplemental Shielding	Included with interface adapter	Included with interface adapter	Included with interface adapter

CASK FEATURES

The conceptual designs which have been developed at Sandia Laboratories[2] divide SFSC features into the following groups:

1. Canisters,
2. Baskets,
3. Shields,
4. Heat transfer methods,
5. Structure, containment, seals, and
6. Operational methods and interface compatibility.

Canisters

Fuel which generates decay heat greater than three kilowatts is expected to be placed in canisters. Canistering fuel for shipping serves several purposes. First, heat transfer will be enhanced by including a liquid or pressurized gas in the canister. Second, all spent fuel with questionable cladding integrity is contained to prevent contamination of the cask, to provide an additional level of containment, and to provide an assured minimum degree of structural integrity for the fuel during shipment. Third, cleaning of the fuel prior to shipment may not be required because residual amounts of sodium would be diluted and contained in the canister coolant. Canister

coolants currently being considered are: pressurized helium, sodium, heat transfer salts (HTS), and lead-bismuth mixtures. Canistering, of course, has some negative impact on plant operations since potentially tedious sealing and leak checking will increase plant operating costs.

A re-usable canister which has been designed by Sandia Laboratories is shown in Figure 1. This canister utilizes a mechanical seal for closure. A welded closure would decrease the canister cost and require a smaller diameter canister lid. Since neither FFTF or CRBRP currently have provisions for welding all fuel canisters, and because of high welding-equipment costs, limited canister re-usability, and weld inspection requirements, a mechanically sealed canister was designed.

A canister testing program is expected to begin during the latter part of FY 78 which is intended to demonstrate that the canister can survive more severe environments than would be experienced in a cask during a shipping or handling accident.

Baskets

The basket of a SFSC is defined as the internal structure which holds the fuel assemblies or canisters in the cask cavity. All baskets which have been designed at Sandia remain fixed in the cask cavity. This is in contrast to the rotating basket designed by the Aerojet Manufacturing Company for an LMFBR spent-fuel cask.[3] A cask with a rotating basket would be less desirable because of increased potential for operational problems such as jamming, reduced payload, and greater fabrication cost.

Several basket concepts have been developed by Sandia for use in spent fuel-shipping casks to be utilized in the Breeder Reactor Program. All concepts have the following in common:

1) Capacity for twelve assemblies (nine assemblies in the outer ring and three assemblies in the inner ring)

2) Capability for accommodating canistered assemblies,

3) Adequate length for CRBRP-type fuel (a spacer would be located in the bottom of the basket for shorter fuel).

The currently recommended basket concept is simple. Basically, it is a large aluminum block. The aluminum block (which may extend over a distance greater than the heated length of the fuel or fill the entire cavity) enhances heat transfer by conduction to the cavity walls. The aluminum would be clad with stainless steel for corrosion protection, and stainless steel end plates and basket inserts would provide the required structural support.

Other basket designs which have been developed include features to enhance heat removal during loading and unloading. An example is a basket which incorporates a system that permits gas to flow through the individual fuel assemblies during loading and unloading. Details of these alternate basket systems are found in References 2 and 4.

Shields

Most existing light-water spent-fuel shipping casks utilize uranium or lead as a gamma shield and water or a hydrogeneous resin as a neutron shield. The primary gamma-ray shielding material utilized in conceptual designs for the breeder reactor program is a uranium - 2% molybdenum alloy. The addition of a small amount of molybdenum improves the structural properties of the uranium. Because of the intense neutron emission from breeder reactor spent-fuel, the ability of the neutron shield to remain intact following an accident is important. In order to ship shorter cooled fuel and to avoid using additional thicknesses of uranium or steel as inefficient neutron shields, some some neutron shielding must remain following any postulated accident scenario. As a result, the development of alternate solid neutron shields has been pursued at Sandia. The solid neutron-

shield materials currently being investigated are boron carbide in a copper matrix, borated beechwood (marketed under the tradename of Permali JN by the Permali International Group), and boro-silicone rubber (marketed by Reactor Experiments, Inc.). Other materials have been identified and are being evaluated.

The boron carbide neutron shield is composed of commercial grade B_4C powder in a copper matrix. This grade of B_4C typically contains excess free carbon and iron impurities in the material. The effective non-stoichiometric compound is $B_{2.77}C\ Fe_{0.015}$. The B_4C powder is hot pressed with copper powder with the expected loading to be 60 to 70 percent B_4C by volume (v/o) in the copper. This shield material will maintain integrity under accident fire conditions and is expected to have a higher thermal conductivity than other solid neutron shield options. This will aid in heat removal during normal transport.

The borated beechwood and boro-silicone rubber would degrade somewhat during accident fire conditions. In addition, the poor heat transfer properties of these materials make it necessary to augment heat removal by placing copper fins through the material. An experimental program is currently underway to evaluate the neutron transmission and accident resistance properties of these heterogeneous neutron shields.

A cross section of a proposed shipping cask concept is shown in Figure 2. This SFSC concept utilizes a composite uranium, B_4C in a copper matrix, and borated beechwood shield. Examples of shield thicknesses needed for a 12-assembly cask are shown in Table 2.

Heat Transfer Methods

Various methods of removing decay heat from the core region of spent-fuel assemblies have been considered in the conceptual designs of spent-fuel shipping casks. These methods are reviewed in detail in References 4 and 5. The preferred method of removing decay heat is to utilize totally passive cooling. The problem of removing decay heat during loading and unloading fuel is often a controlling design parameter. Table 3 gives a summary of the thermal performance of breeder reactor spent-fuel shipping casks with different loading and shielding options.

Structure, Containment, and Seals

Several containment features are common to all of the conceptual designs being developed at Sandia Laboratories. These features are:

1. Double Containment - Two steel shells and two lids enclose the cask cavity. A third level of containment occurs when fuel canisters are used.
2. The inner lid is a shielding and containment lid, and the outer lid is a structural and containment lid.
3. The seal on the inner lid is currently a large-diameter Graylock seal.*
4. All valved penetrations for cooling, sampling, and testing will penetrate at most one containment barrier.
5. Impact limiters are an integral part of the structural lid on each end.
6. Structural materials which maximize corrosion resistance and maintain sufficient ductility at all applicable temperatures to satisfy accident-condition requirements will be used.
7. There are four lifting trunions on the lid end (top) and two lifting trunions on the bottom end.
8. There are two tie-down lugs on each end with one end incorporated into a tilting fixture on a rail car.

*Manufactured by Gray Tool Co., Houston, Texas.

Table 2
Maximum Radial Thickness of Radiation Shield Layers Required for Shipping Twelve 180-Day Cooled CRBRP Spent Fuel Assemblies

Option	Cask Type (Designated by Neutron Shield Material)	Thickness of[a] Gamma Shield (Depleted U) cm(in)	Thickness of Neutron Shield[b] (Material Designated By Layer in Multi-layer Designs) cm(in)
1	B_4C[c] - only	6 (2.4)	28 (11.0) 75 v/o B_4C, 25 v/o Cu
		4 (1.6)	37 (14.6) 50 v/o B_4C, 37.5 v/o Cu 12.5 v/o Void
2	Beechwood[d]	8 (3.2)	21 (8.3)
3	B_4C/Beechwood[e]	7 (2.8)	6 (2.4) 75 v/o B_4C, 24 v/o Cu + 15 (5.9) Wood
4	B_4C/Water[f]	7 (2.8)	5 (2.0) 75 v/o B_4C, 25 v/o Cu + 20 (7.9) Water (non-borated)

Notes:
[a] Estimated uncertainty in the U thickness is ± 0.5 cm.
[b] Estimated uncertainty in the neutron shield thickness is ± 1 cm.
[c] Commercial grade B_4C assumed in all cases. Excess free C and Fe impurity result in the nonstoichiometric equivalent compound $B_{2.77}C\ Fe_{0.015}$.
[d] To meet post-accident dose rate regulations at least 10% of the beechwood must remain in place.
[e] Post-accident dose rate requlations are met assuming complete loss of beechwood.
[f] Post-accident dose rate regulations are met assuming complete loss of water.

Operational Methods and Interface Compatibility

The current cask conceptual designs were based on dry loading/unloading in a vertical position. For FFTF and CRBRP, the cask would arrive at the site, road dirt would be removed, minor inspection procedures would be performed, the impact limiters would be removed, and the cask would be removed from the rail car for loading (future facilities may develop a method of loading the cask while it is on the rail car). The outer lid would be removed, and the top of the cask would be sealed against the interface adapter. The inner lid would be removed or rotated, and the cask filled with spent fuel. Following loading, the process would be reversed with several additional steps for controlling contamination and verifying complete containment by leak checking and sampling. The cask would be filled with pressurized helium prior to shipment.

In addition to handling and loading operations, each facility must be able to remove any radioactive contamination of the cask received on its site and perform minor maintenance such as seal replacement. If canisters are used in the cask, operational methods for handling and servicing canisters must be developed.

Table 3
Summary of Thermal Performance of Breeder Reactor Spent Fuel Shipping Cask with Different Loading and Shipping Options

	Loading and Shipping Configuration	Assemblies In Cask	Decay Heat Capability (kW/Assembly)
Low Performance	I. Uncanistered assemblies. Load -- 1 atm argon Ship -- 1+ atm helium	9 to 12	0.8 to 1.0
	II. Uncanistered assemblies. Load -- 1 atm helium Ship -- 1+ atm helium	12	1.4
	III. Assemblies canistered in approximately 5 atm helium Load -- 1 atm argon Ship -- 0.5+ atm helium	12	1.4
Intermediate Performance	I. Assemblies canistered in 5 to 8 atm helium. Load -- 1 atm argon Ship -- 0.5+ atm helium	9	2.0 to 2.5
	II. Assemblies canistered in sodium or other liquid. Load -- 1 atm argon Ship -- 0.5+ atm helium	12	4.0 to 8.0

CONCLUSIONS

A cask which would accommodate FFTF, CRBRP and future breeder reactors would be similar to the concept shown in Figure 2. The recommended options for this cask are listed in Table 4. The expected cost of this cask is currently from three to three and one-half million dollars per cask. One goal of the conceptual design development effort at Sandia was to make the cask as simple and maintenance free as possible.

There is some incentive for developing separate cask systems for each breeder reactor installation. While a common cask could not be utilized for all currently planned facilities, the differences in fuel assembly lengths and decay heat output would probably make a separate cask for each facility desirable. For example, an FFTF cask would carry a large number (30) of the shorter, low heat-producing assemblies while a cask for CRBRP would be very similar to the cask in Figure 2 with the options in Table 4, since CRBRP spent-fuel characteristics control the design of a dual purpose cask.

Finally, the development of conceptual cask designs for an FFTF and CRBRP cask has shown the importance of maintaining communication between the cask and facility designers at an early stage of system development. A more operationally efficient and cost-effective system can be planned

and implemented by establishing and maintaining an interchange of information from a program's inception. The standardization of an optimized system should would be the goal for advanced facilities.

Table 4

Recommended Features for an FFTF/CRBRP Spent-Fuel Shipping Cask

The cask should:

1) Be transported by rail with a 90 tonne gross weight,

2) Have at least a 12-assembly capacity,

3) Be capable of transporting assemblies with a significant amount of decay heat in canisters,

4) Have a fixed solid aluminum basket with removable stainless steel inserts,

5) Utilize a U-2% Mo gamma shield,

6) Utilize B_4C in a copper matrix and beechwood or boro-silicon rubber as a neutron shield (the amount of B_4C is partially dependent on its cost and fabricability into large blocks), and

7) Utilize stainless steel for the cask structure.

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FIGURE 1

RE-USABLE SPENT FUEL CANISTER

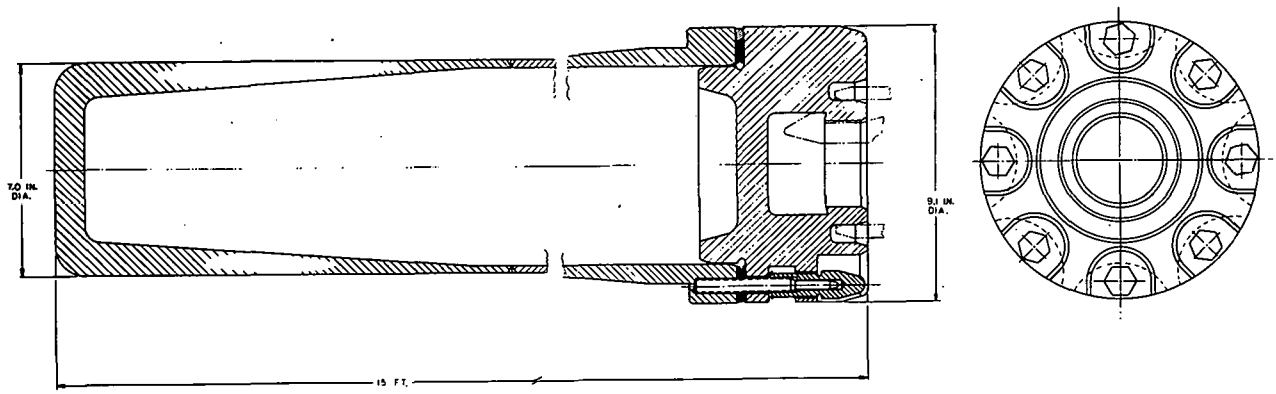
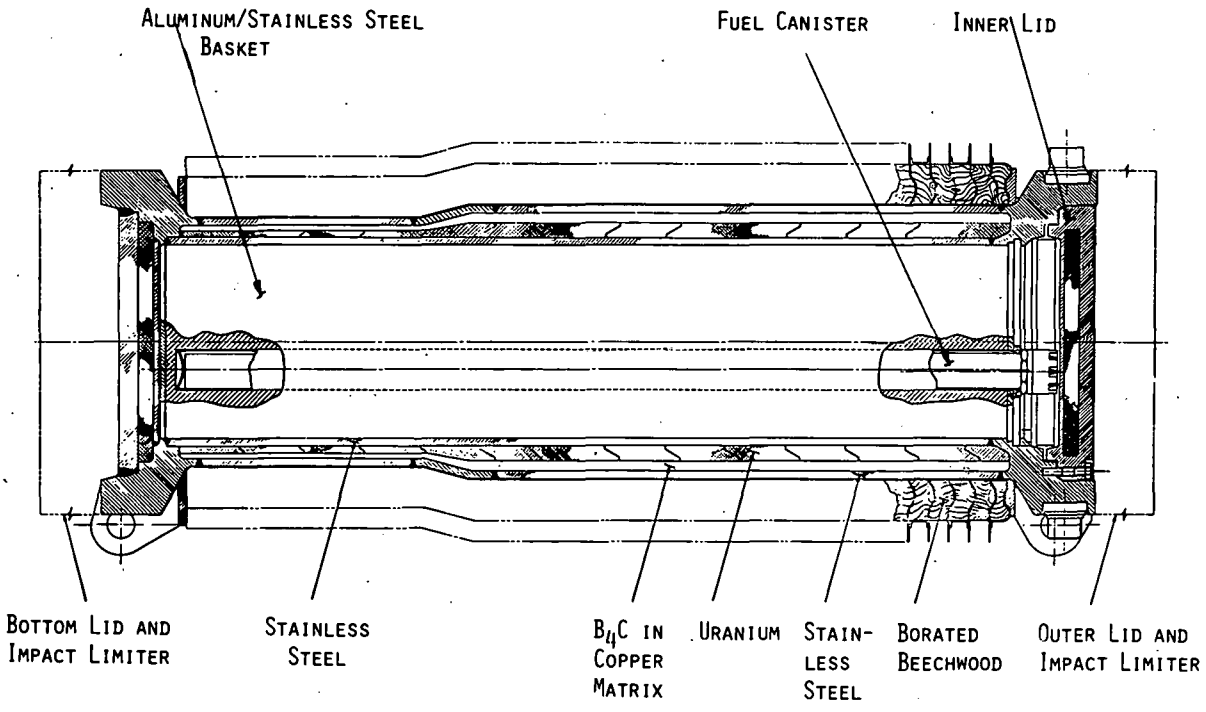


FIGURE 2

BREEDER REACTOR SPENT FUEL SHIPPING CASK



THE HZ-75T SPENT FUEL SHIPPING CASK

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Hitachi Shipbuilding and Engineering Co., Ltd.

Hitachi Shipbuilding and Engineering Co., Ltd. (HZ) is building HZ-75T spent fuel shipping cask, which is the largest ever licensed and built in Japan and associate following design features:

1) low shipping cost
2) ease of operation and maintenance
3) shipping of 2nd generation reactor fuels

HZ-75T was originally designed by NL Industries, Inc. and Battelle Columbus Laboratories to the IAEA, USA and Japanese Regulations for the transport of radioactive materials. The design was partially modified in the course of design and licensing follow to meet the revisions of the regulations and reflecting the state of art. The following description presents the design phylosophy, the basic cask design and operating parameters and additional information.

I. Design Phylosophy

HZ carried out an extensive study on the transportion of spent fuels from 2nd generation power reactors of light water type, paying special attention to the transport by sea going vessels as well as by trailors. HZ established with the assistance of both NL Industries, Inc. and Battelle Columbus Laboratories design phylosophy which encompass the following cask features:

1. Large Capacity

It was found desirable from the transportation study that the cask accommodates as many fuel elements as practicable while satisfying all of safety and other technical conditions. The capacity of the cask depends on the inplant handling capacity, transport conditions, size of fuel elements and level of radioactivity. Considering these factors, HZ assumed 80 Tons as maximum weight of loaded cask and selected 7 PWR or 17 BWR fuel elements as the capacity of the cask.

2. Stainless Steel Structure

Not only the interior surface but also the exterior of the HZ-75T is all made of stainless steel for the ease of decontamination of the cask. It will be of great advantage for maintenance as no rust will generate during the long ocean voyage.

3. Neutron Shielding

Water and ethylene glycol mixture was selected as neutron shielding material and contained in the external annulus of the cask body. This neutron shielding is necessary for a cask to transport high burn-up fuel elements of 2nd generation power reactors.

4. Large Heat Removal

Stainless steel clad copper fins are welded on the exterior of the cask cylinder to dissipate estimated 84 KW of decay heat for high burn up, short cooled 7 PWR fuel assemblies.

The selection of abovementioned fins makes it possible for HZ-75T withstand the pressure rise even without auxiliary cooling system.

5. Auxiliary Cooling System

Air and water cooling system are provided for the HZ-75T, taking account of both land and sea transport. Air cooling is applied for land transport and air and/or water cooling for sea transport. Water can be much more effectively used for the casks stowed in the enclosed hold of sea going vessel.

II. Design Description

The HZ-75T cask is designed to ship up to 7 PWR fuel elements or up to 17 BWR fuel elements irradiated to design exposure. The HZ-75T cask consists of stainless steel packaging with lead gamma shielding, water neutron shielding and heat rejection fins to accommodate high burn-up/short cooled fuels.

1. Cask Body

The HZ-75T cask inner cavity which forms the containment vessel, is a stainless steel cylinder, 952 mm in inside diameter and 4556 mm in length with 19.1 mm thick walls, having its bottom end sealed with a 107 mm thick stainless steel plate. The upper end is welded to the closure flange. Surrounding the inner cavity, a gamma ray shielding is formed by the lead filled 184 mm annulus between the inner cavity and intermediate cylinder with 12.7 mm thick wall. A neutron shielding is formed by the 150 mm wide annulus between the intermediate and the outer cylinder filled with a 45 - 55 volume percent water - ethylene glycol mixture.

The auxiliary water cooling tubes are also provided in this neutron shielding annulus.

Circumferential stainless steel clad copper fins are welded to the exterior of the outer cylinder for the heat dissipation and the impact protection. The fin is 152 mm high and 6.4 mm thick. All structural components are made of stainless steel and all external and internal surfaces of cask are stainless steel.

The cask body with closure head in place has an overall nominal length of 5051 mm.

2. Closure head

The cask top is closed with the bolted flange-type closure head having double O-ring-seals.

The closure head consists of stainless steel plates and lead gamma ray shielding, and is held in place by 24 - 40 mm diameter closure bolts. Because of variations in fuel length, the inner cavity spacer is provided. This spacer is used for PWR fuel elements only and is mounted on the closure head.

3. Fuel Baskets

Two kinds of removable fuel basket are designed to be provided for the HZ-75T cask. One is a 7-channel PWR unit and other is 17-channel BWR unit. The 7-channel PWR unit has three 235 mm and four 225 mm square cross section channels for PWR fuel elements. The three larger channels will accommodate fuel cans for containing defective or contaminated elements. The 17-channel BWR unit has seventeen 156 mm square cross section channels for BWR fuel elements. Seven fuel cans may be loaded in this basket. In both fuel baskets the structural components consist of stainless steel and the wall between adjacent channels contain neutron absorbing material (boral plates) clad with stainless steel for criticality control.

4. Trunnions/Lifting Devices

A pair of 200 mm diameter stainless steel trunnions welded to the outer shell near the top of the cask are used to lift the cask. A second pair of similar trunnions welded to the bottom impact limiter are used to pivot the cask from the vertical to the horizontal position. These trunnions are positioned 50 mm off of the cask center plane to cause the cask pivot into preferred orientation. Pads welded to the outer shell near the midsection of the cask are provided for attaching tie-down devices or for attaching lifting trunnions for low overhead facilities.

The cask lifting yoke is a steel structure which engages a building crane hook at the top and the HZ-75T cask lifting trunnions at the bottom. This yoke is used for all cask handling operations including removal and replacement on the skid, transfer of cask to the fuel

storage pool and closure head removal and installation.

5. Impact Limiter (Top/Bottom)

During transport the ends of the cask are protected by multilayer circular aluminum tubes impact limiters which are encased by stainless steel shells. These are attached by screw fittings and may be removed during fuel loading. These impact limiters are designed to provide protection of the main cask body and closure from impact forces.

The overall dimension of top impact limiter is 2270 mm in diameter and 774 mm in length and the bottom is 2270 mm in diameter and 671 mm in length.

6. Heat Transfer Fins

For land transport the heat transfer from the HZ-75T cask to the environment is done by forced convective heat transfer from circumferential fins which are welded to the cask body.

87 stainless steel clad copper fins are used for this purpose. These fins have a thickness of 6.4 mm and a radial height of 152 mm. The center to center spacing is 50 mm. All exterior surface of fin is stainless steel for easy decontamination and maintenance. The fins are welded to the cask body, providing the fin-to-cask surface necessary for low resistance heat flow across the interface. Sufficient heat transfer surface and forced cooling system are provided to dissipate up to 84 kilowatts of decay heat.

Even at loss of forced cooling conditions, heat transfer fins will provide sufficient heat transfer surface to dissipate up to 84 kilowatts. Therefore, the HZ75T cask will sustain safety margins at the accident conditions. The heat transfer fins are also effective for the impact protection.

7. Neutron Shielding

External annulus of the cask body, consisting of intermediate and outer cylinders, is filled with water and ethylene glycol mixture. A nominal 150 millimeter thick layer provides sufficient ability to shield neutron radiation from high burn-up fuels. Thermal expansion chamber is provided at the upper end of annulus.

8. Valves and Ports

There is a stainless steel drain line at the bottom of the containment vessel which terminates in a valve box on the side of the cask near the bottom. A stainless steel sampling line extends from the cover cavity to a valve box on the side of the cover cavity near the top.

The drain line and sampling line are each closed with a Worcester Type stainless steel ball valve housed in a valve box. The outlet of

each valve is closed with a stainless steel pipe plug. The valves are protected by a valve box cover plate and impact limiter.

There is leak test hole in the closure head. This hole provides access for pressurization or evacuation of the available void space which exist between the inner and outer O-ring seal in the closure head.

9. Skid

The cask will be supported in a skid mount which is designed for truck, rail car, or barge. The lower end of the cask is supported in trunnion sockets; the upper end rests in a cradle. Pads at each end of the skid mount absorb longitudinal acceleration or deceleration forces. A mechanical air cooling system is mounted in the separate skid which utilizes two blowers, each capable of supplying the required air flow, exhausting through bellows between the skid beams beneath the cask which permits air flow vertically across the cask fins.

III. Manufacturing

The HZ-75T cask is being manufactured under the quality control system based on ASME Code, Sec III with fabrication, inspection and testing procedures approved by the competent authority (Science and Technology Agency) to assure that the unit is manufactured in accordance with the approved design.

The fabrication, inspection and testing procedures are based on the requirements of ASME Code, Sec III. Class 2 vessel except where these requirements are not compatible or conflict with the approved design and STA's specific requirements for fabrication, inspection and testing of shipping container.

In addition to normal inspection and testing for vessels, the following in-process tests carried out under the witness of STA inspector to assure the integrity of products as shipping container.

(1) Shielding Integrity Test

(2) Over Load Test for Lifting Yoke and Trunnions

(3) Thermal Test

(4) Handling Test (Functional Test)

(5) Leak Test

To ensure shielding integrity of the cask HZ has developed special equipment for gamma probing and scanning test, which gives a sensitivity of 0.5 percent for gamma probing and 3 percent for gamma scanning, using ^{60}Co gamma source and NaI (Tl) scintillation counter.

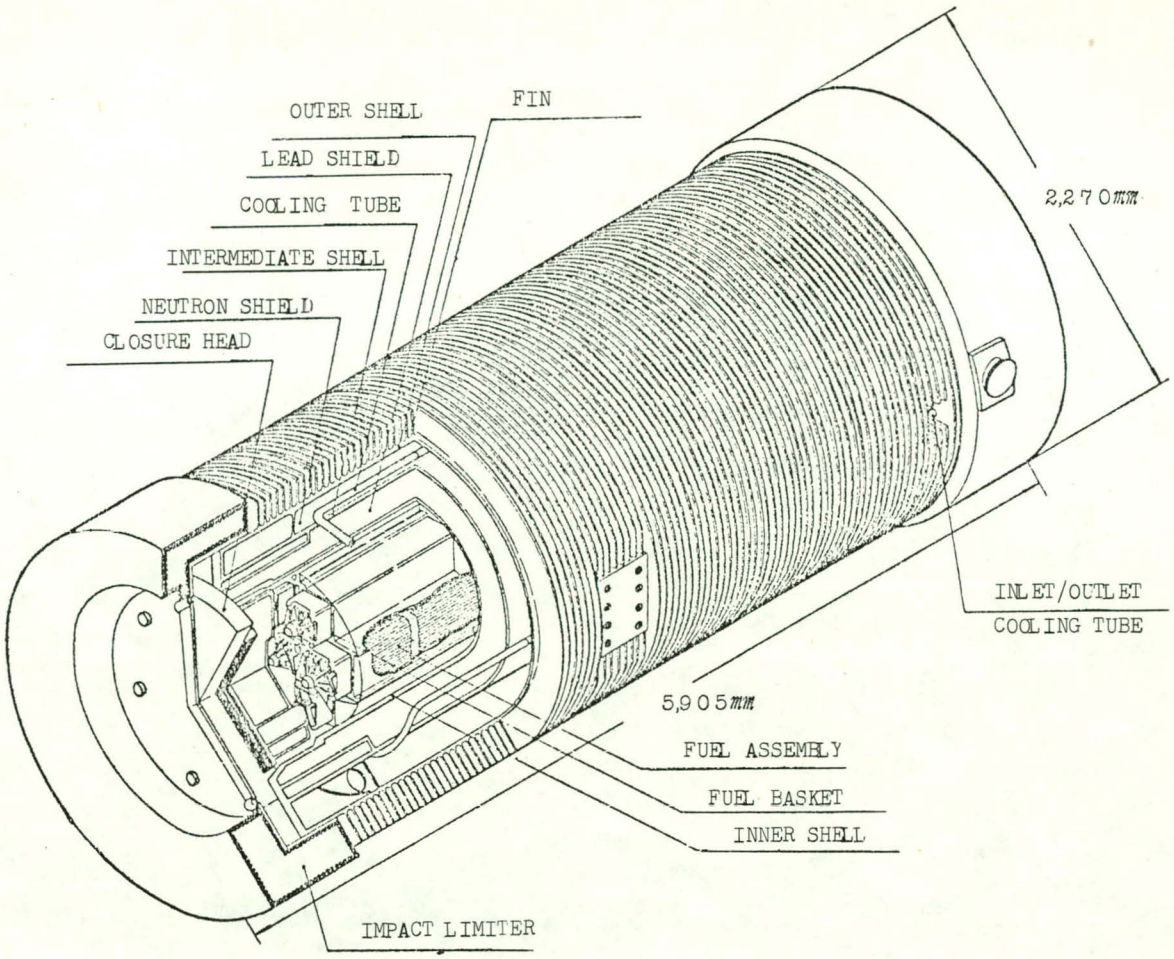


Figure 1. Bird's-eye view of HZ-75T cask.

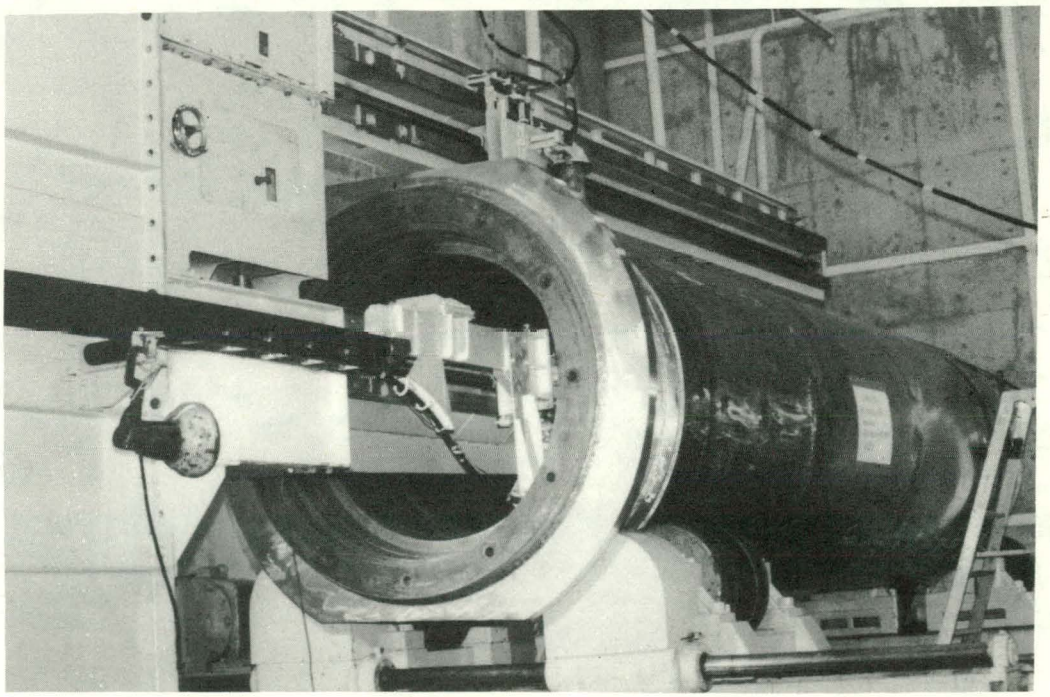


Figure 2. Gamma probe equipment.

Summary of Major Design Parameters

1. Classification — Type B (M)

2. Dimension

Length (including Impact Limiter)	5.91 m
Outside Diameter	2.27 m
Cavity Diameter	0.95 m
Cavity Length	4.54 m

3. Weight

Lifting Weight in Reactor (Aux.) Building	approx. 75 ton
Cask on Transporter	approx. 80 ton

4. Capacity

PWR Fuel	7 assemblies
BWR Fuel	17 assemblies

5. Design Parameter of Fuel

	PWR	BWR
Fuel	UO_2	UO_2
Cladding	Zircaloy	Zircaloy
Enrichment, U235	3.4 %	2.7 %
Average Burn-up (max.)	33,000 MWD/MTU	27,500 MWD/MTU
U Weight per Assembly	450 kg	195 kg
Cooling Time (min.)	120 days	120 days
Assembly Length	4,203 mm	4,470 mm

Dup

A WASTE TRANSPORTATION SYSTEM PLAN FOR GEOLOGIC NUCLEAR-FUEL-CYCLE WASTE REPOSITORIES*

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INTRODUCTION

A program plan for the procurement of shipping casks to service federal commercial nuclear waste repositories was initiated at Sandia Laboratories for the Office of Waste Isolation, Union Carbide Corporation. Since geologic waste repositories are expected to begin cold testing in the mid 1980's, prototype shipping casks will be required at that time. Implementation of the plan developed as a part of this program will provide licensed prototype truck and/or rail shipping casks with subsequent production units on a schedule that is consistent with repository objectives.

Additional activities of the program were:

1. To provide information on the details of how casks affect and are affected by interfacing systems,
2. To prepare conceptual designs for casks to be used in the shipment of high level waste and cladding waste, and
3. To describe operational characteristics, maintenance requirements, and interfaces associated with cask use.

CASK PROCUREMENT PLAN

The cask procurement plan addresses the need to develop shipping systems for three categories of nuclear waste:

1. Light Water Reactor (LWR) spent fuel assemblies with a minimum cooling time of one year and nominal fuel composition and burnup. The assemblies may be canistered or uncanistered.

2. High-level waste (HLW) cooled for a minimum of five years following removal of spent fuel from a reactor with the waste concentration and composition as defined by Kee, et al,[1] (waste solidified in borosilicate glass with a concentration of 0.085 m^3 (3 ft^3) waste per 1 MT of heavy metal charged to a light water reactor. The waste is assumed to contain ~100% of the fission products from the spent fuel excluding ^3H, I, and Br, ~100% of the transplutonic actinides from the spent fuel, and approximately 0.5% of the U and Pu originally in the spent fuel).

*This work is supported by the U. S. Department of Energy through the Office of Waste Isolation, Union Carbide Corporation.

3. Cladding waste (CW) from the recycle chop and leach process, consisting of Zircaloy compressed to 70% of theoretical density and contaminated with 0.1% of the Pu, 0.05% of the transplutonic actinides, and 0.05% of the fission products originally in the spent fuel.

Casks capable of shipping spent fuel exist and are in service at the present time. Transportation systems for shipping HLW or CW have not been designed. The system plan[2] includes ten activities associated with procuring casks to transport the three types of waste previously defined. These activities are:

1. Procurement Management,
2. Inventory Definition,
3. Design,
4. Testing,
5. Licensing,
6. Fabrication,
7. Legal Considerations,
8. Maintenance and Repair,
9. Decommissioning and Disposal, and
10. Accident Recovery Capability.

The procurement activities are assumed to extend over a period of approximately 8 years, 1978-1986. During this period, changes and updates to the plan are expected and contingencies in the plan allow for changes in policy direction, time schedules, and logistics. Cask procurement options, along with the range of lead times needed to complete prototype fabrication, were examined in detail and are shown in Table 1.

Table 1
Cask Procurement Lead Times

Option #	Option	Range of Lead Times to Obtain Prototype Unit (Weeks)
1	Ship spent fuel using existing casks:	
	truck	210-260
	rail	240-340
2	Ship spent fuel by modifying existing casks:	
	truck	350-400
	rail	390-490
3	Ship spent fuel in newly designed casks:	
	or	
	Ship high-level waste in newly designed casks:	
	rail	440-460
4	Ship clad waste in newly designed casks:	
	rail	410-440

The lead time associated with procuring a given cask depends on a number of factors including the complexity of the cask design, mode of transportation (truck, rail, barge) selected, payload of the cask, and fabrication lead-times. An economic optimization within the range of lead-times has not been performed. The stated lead-times assume that fabrication of a prototype is not initiated until a license is obtained. Estimates of times required for licensing efforts are included in the procurement schedule.

Options 1, 2, and 3 are alternative paths for procuring casks to ship LWR spent fuel to a geologic repository. A decision associated with this goal is whether or not canistering of spent fuel is permitted or required. Options 1, 2, and 3 are not mutually exclusive. However, the long lead time associated with Option 3 and the diversity of LWR spent-fuel shipping cask designs available commercially (see Tables 2 and 3), could result in selection of Options 1 or 2 over Option 3. Despite the apparent advantages of 1 or 2, there are two variables which may result in the economic justification of option 3. First, the presence of a large backlog of long-cooled LWR spent fuel may justify the expense of designing and licensing new casks because of the higher payload-to-gross-weight ratio which would be achieved with reduced shielding and heat removal requirements of long-cooled fuel. Second, the initiation of Pu recycle in LWR fuel would increase the thermal and neutron source strengths which, coupled with a possible requirement for one year cooling of spent fuel may justify designing and licensing new spent fuel shipping casks.

Depending on the logistics and economics of the overall shipping problem, it appears that the use of several different casks will be desirable. In particular, rail casks should offer the advantages of larger payloads and lower freight charges per ton-mile compared with truck casks. However, many existing and planned nuclear power plants do not have rail access. Therefore, in the nearterm, truck casks, or at least a truck cask with the capability for connecting to rail heads, will be required. In the long term, the relative economic merits of rail casks vs the cost of installing railheads must be examined.

To procure casks to ship HLW, Option 3 should be employed. The long lead time required to design, license, and fabricate a HLW cask implies that, in order to have prototypes available by the mid-1980's, initiation of the design effort should begin immediately. Similarly, Option 4 must be followed to procure casks to ship CW. The lead time to procure CW shipping casks should be slightly shorter than the time needed to procure HLW casks. However, the need of both HLW and CW casks is dependent on future national policy regarding reprocessing.

CASK CONCEPTUAL DESIGNS

An important objective of the program is the preparation of conceptual designs for casks to be used in the shipment of HLW, CW, and recycle spent fuel. Preliminary work has been done on a number of designs. Important criteria utilized in preparing conceptual designs of waste and spent fuel casks are the NRC requirements specified in 10 CFR 71[3]. These requirements include radiation dose rate limitations for shipping containers under normal operation (total dose rate less than 10 mrem/hr at 2 m (6 feet) from the cask surface, 200 mrem/hr at the surface) and post-accident (total dose rate less than 1000 mrem/hr at 1 m (3 feet)) conditions.

Table 4 presents estimates of neutron and gamma source strengths for HLW, CW, and recycle spent fuel calculated with the ORIGEN computer code[4]. The source values labeled present generation pertain to fuel once-burned in a reactor and the waste obtained if such spent fuel is reprocessed. The U-recycle sources refer to the equilibrium waste and spent fuel resulting from a number of U-only recycling processes. The Pu-recycle values correspond to the equilibrium sources for the waste and spent fuel with both Pu- and U-recycle. These three distinct source categories were examined to estimate the effect of a recycle economy on waste and spent-fuel shipping casks.

Table 2. Existing LWR Spent Fuel Shipping Cask Designs

Cask Identification	Owner/ Supplier	Payload (# Assemblies) PWR	BWR	Mode of Transportation	Gross Weight (tonnes)
NLI-1/2	NL Industries Barnwell, SC	1	2	Truck	22
NAC-1[a]	Nuclear Assurance Corporation Atlanta, GA	1	2	Truck	24
NFS-5	Nuclear Assurance Corporation Atlanta, GA	2	4	Truck	25
TN-8[b]	Transnuclear White Plains, NY	3	0	Overweight Truck[e]	36
TN-9[b]		0	7		
IF-300	General Electric Co. San Jose, CA	7	18	Rail	74
NLI-10/24	NL Industries Barnwell, SC	10	24	Rail	83
TN-12[c]	Transnuclear White Plains, NY	12	32	Rail	97
NAC-2[d]	Nuclear Assurance Corporation Atlanta, GA	7	12	Rail	80
NAC-3[d]		12	32	Rail	100

[a]Formerly Nuclear Fuel Services NFS-4.

[b]The TN-8 and TN-9 are versions of the same cask designed exclusively for PWR and BRW fuel, respectively.

[c]The TN-12 is not yet licensed for use in the U.S.

[d]The NAC-2 and -3 are currently under design.

[e]Overweight permits are required resulting in increased freight cost which must be amortized by the increased payload compared with legal weight trucks. The permit requirement introduces uncertainties in obtaining permission to ship along certain routes.

Table 3
Estimated Spent Fuel Shipping Cask Cost and Fabrication Lead Times

Several leasing or purchase prices depend on cask capacity and whether or not technical services are provided by the supplier. Some suppliers require a maintenance or service contract with lease or purchase. Some suppliers permit casks to be reserved for use at a specified future time. Some suppliers offer a lease with an option to buy. Prices range approximately as follows (the cost ranges quoted are believed to be reasonably accurate. Firm quotes are available only from the suppliers):

Cask Type	Purchase Prices($M)	Lease or Rental Prices	Fabrication Lead Times (mo)
Truck	0.5 - 1.25	$500-1600 per day or $200k-500k per year	12 - 24
Rail	2.0 - 4.0	$2K-4K per day or $0.7M-1M per year	24 - 42

A two-component shield concept was employed in developing cask designs. This concept was chosen as a compromise between the increased shielding efficiency and decreased thickness of multi-layer shields, and decreased heat transfer capability and increased fabrication costs of such shields. Candidate shielding materials included U or Pb for the primary gamma shield and borated beechwood (Permali JN)[5], water, boro-silicone rubber[6] or NS-2 polymer[11] as the neutron-shield materials. An important problem associated with these neutron-shield materials is their loss of hydration as a consequence of elevated temperatures such as those encountered in a fire accident. Simulation of this problem in evaluating potential cask designs was achieved by assuming complete removal of the neutron shield material in the post-accident shield analysis of each design.

One- and two-dimensional radiation transport calculations were performed on a series of candidate cask designs to determine the shielding requirements for HLW and CW shipping casks. Initial analysis was done with the ANISN computer code.[7] Two-dimensional analyses were performed with the DOT computer code[8] for r-z geometry. An 11 group, P_1 cross section set generated by GAMLEG[9] was used to analyze the transmission of primary gamma-rays. Transmission of neutron and secondary gamma rays was analyzed with a 19-neutron, 13 gamma-ray group, P_1 cross section set collapsed from the DNA Few-Group Library[10].

The basic HLW cask design was cylindrical with a 3-m (10-ft) long, 1.07 m (42.25-inch) diameter cavity capable of containing seven, 30-cm (12-inch) diameter, HLW canisters in an aluminum support basket. The cask structural material was stainless steel which completely encased the primary gamma-ray shield. The neutron shield was assumed attached to the exterior surface of the cask and was penetrated with Cu heat conduction fins. Only the thicknesses of the shield layers were varied in the analysis. The model utilized for the CW cask design was essentially the same as the one for the HLW cask except the cask cavity diameter was increased to 1.25 m (49 inch) to accommodate nine, 30 cm (12-inch) diameter, CW canisters and no neutron shield material was employed.

Table 5 summarizes the ANISN results for candidate shield combinations. The corresponding estimated dose rates determined for present generation HLW and CW for the shield materials considered are also shown. These combinations satisfy the 10 CFR 71 dose-rate requirements. It was also determined that these potential shield combinations were acceptable for the safe shielding of U-recycle HLW and CW. This is an expected result since the radiation source strengths for U-recycle wastes are not significantly different than those for present generation wastes.

Table 4. Gamma Ray and Neutron Sources Per One Metric Ton Heavy Metal (MTHM) Charged to a Light Water Reactor

	Present Generation			Uranium Recycle			Plutonium Recycle		
	High Level Waste	Clad Waste	PWR Spent Fuel	High Level Waste	Clad Waste	PWR Spent Fuel	High Level Waste	Clad Waste	PWR Spent Fuel
Average Group Energy (Mev)	Photon Source (gammas/sec) by Group								
3.25	1.25+9	1.25+6	4.21+10	2.42+9	2.41+6	8.14+10	3.93+9	1.96+6	6.76+10
2.75	3.97+10	3.94+7	1.33+12	7.65+10	7.59+7	2.57+12	1.24+11	6.20+7	2.14+12
2.38	5.09+11	5.07+8	1.75+13	9.85+11	9.84+8	3.39+13	1.58+12	7.92+8	2.76+13
1.99	2.14+12	2.43+9	1.59+14	4.84+12	5.50+9	3.61+14	5.18+12	2.59+9	1.54+14
1.55	3.95+13	2.86+10	3.78+14	5.55+13	4.01+10	5.31+14	5.56+13	2.83+10	4.34+14
1.10	2.63+14	1.02+14	1.25+15	2.88+14	1.16+14	1.34+15	3.87+14	9.68+13	1.62+15
0.63	6.46+15	5.01+12	2.97+16	7.66+15	5.94+12	3.53+16	7.84+15	4.94+12	3.04+16
0.30	2.54+14	2.03+11	4.68+15	3.57+14	2.74+11	6.34+15	3.65+14	1.96+11	5.19+15
Group Energy Range (Mev)	Neutron Source (neutrons/sec) by Group								
3.01-10.0	4.47+7	2.34+4	6.79+7	3.79+7	1.99+4	5.75+7	1.23+9	7.33+5	1.70+9
1.83-3.01	5.12+7	2.71+4	7.36+7	4.39+7	2.30+4	6.66+7	1.43+9	8.48+5	1.97+9
1.11-1.83	4.82+7	2.53+4	7.33+7	4.09+7	2.14+4	6.20+7	1.33+9	7.91+5	1.84+9
0.55-1.11	4.29+7	2.25+4	6.53+7	3.65+7	1.91+4	5.53+7	1.18+9	7.04+5	1.64+9
0.111-0.55	2.69+7	1.41+4	4.10+7	2.29+7	1.20+4	3.47+7	7.43+8	4.42+5	1.03+9
0.053-0.111	4.60+6	2.41+3	6.99+6	3.91+6	2.04+3	5.92+6	1.27+8	7.55+4	1.75+8

a. Sources are for one year old spent fuel and five year old high level waste and clad waste.

Table 5. Shielding Requirements for Non-recycle High Level Waste and Clad Waste as Estimated by One-Dimensional Calculations Corrected for Geometry

High Level Waste Gamma Shield	Neutron Shield	Radiation	Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
7 cm U	none	neutron (post-accident)	95	326
		gamma (post-accident)	231	
12 cm Pb	none	neutron (post-accident)	560	964
		gamma (post-accident)	404	
7 cm U	25 cm beechwood	neutron	1.2	4.6
		gamma	3.4	
7 cm U	26 cm H_2O	neutron	1.1	3.1
		gamma	2.0	
7 cm U	25 cm NS-2	neutron	2.2	4.5
		gamma	2.3	
7 cm U	26 cm boro-silicone rubber	neutron	1.9	4.4
		gamma	2.5	
12 cm Pb	25 cm beechwood	neutron	1.9	8.0
		gamma	6.1	
12 cm Pb	26 cm H_2O	neutron	1.6	6.4
		gamma	4.8	
12 cm Pb	26 cm NS-2	neutron	2.8	5.9
		gamma	3.1	
12 cm Pb	26 cm boro-silicone rubber	neutron	2.7	6.0
		gamma	3.3	
Clad Waste				
5.5 cm U	none	gamma	7.7	
13.0 cm Pb	none	gamma	3.7	

Pu-recycle HLW post-accident shielding requirements were also determined using ANISN. The neutron source for Pu-recycle HLW is approximately 30 times larger than that of present generation waste. This seriously affects the amount of shielding required. It was found that a cask capable of shielding Pu recycle waste would greatly over-shield either U-recycle or present generation waste.

Two-dimensional analysis was then used in evaluating the HLW conceptual cask design employing depleted U and beechwood shields. This particular shield material combination was considered since it results in the smallest diameter cask. Using the computer code DOT, it was found that 7.5 cm of U in the cask wall, 6.5 cm of U in the cask ends, and U rings 6.0 cm thick and 12.5 cm high placed at each end of the cask cavity were necessary to meet the 10 CFR 71 post-accident dose rate condition at all locations about the cask. Uranium rings at the cask ends were utilized to alleviate the problem of radiation streaming through the steel end forgings. To satisfy dose-rate limitation for the normal operating condition, it was determined that 35 cm of beechwood were required on the cask sides and 25 cm on the ends. These values reflect a consideration of the calculated k_{eff} of 0.09 for the cask, a 40% increase in the dose rate due to ground scatter effects, a 20% uncertainty in the radiation source definition, uncertainty in cross section values, and uncertainty introduced by the discretizing procedure used in the DOT computer code. The results are considered to be accurate to within $\pm$ 1 cm for any shield dimension. A cask of these dimensions and materials would have an overall loaded weight of approximately 50 tonnes.

The conceptual design study for a LWR spent fuel shipping cask involved a re-evaluation of an existing cask design using the present generation, U-recycle and Pu-recycle radiation sources. The emphasis here was to determine if this cask was adequate for each of the recycle spent fuel types. Both one year cooled PWR and BWR spent fuel were considered. One-dimensional transport analysis indicated that the cask was acceptable for shipping U-recycle and present generation spent fuel but was undershielded for Pu-recycle spent fuel. The difficulty with the latter was again attributed to the significantly greater neutron source for Pu-recycle spent fuel than that for present generation or U-recycle spent fuel. Thus the design of new spent fuel shipping casks will be desirable from a shielding viewpoint if plutonium recycle occurs. LWR shipping casks are designed to transport relatively short-cooled (150 day) spent-fuel; and if long-cooled (5 years or longer) spent-fuel is to be transported, then a new cask may be desirable. A significant improvement in the ratio of spent fuel payload to cask weight can be obtained for a cask which will transport long-cooled spent-fuel in comparison to one designed for transporting short-cooled fuel. A substantial reduction in shipping charges during the cask's operating lifetime (approximately 20 years) will more than offset the additional design and licensing charges.

SHIPPING CASK INTERFACES

It is essential that facility and cask designers exchange information on interface features. Facility designers tend to design loading and unloading equipment based on previous generation casks without considering current cask design information.

During the development of the system plan study an effort was made to exchange information between cask and repository designers. An important part of the cask procurement plan is the effective implementation of transportation system/ repository interfaces. Interfaces which must be implemented are:

1. Mechanical couplings,
2. Supplemental shielding,
3. Contamination control,
4. Cooling during loading/unloading,
5. Cask maintenance methods,
6. Cask testing and inspection methods,
7. Loading/unloading orientation, and
8. Payload details (e.g., canisters, etc.).

Appropriate design reviews can be used to minimize cask turn-around times once a system is in operation.

CONCLUSIONS

The program described in this paper has resulted in the development of a plan which can be utilized to procure prototype shipping casks needed to service a geologic waste repository in the mid-1980's. Implementation of one or more of the plan options should begin now due to the long lead times and limited manufacturing capacity in the United States. Design, licensing, and fabrication of prototype shipping systems should be completed in a timely manner so that high-level policy makers will not have to eliminate options which are consistent with national policy strictly on the basis of hardware unavailability.

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CONCEPTUAL DESIGN OF A SHIPPING CASK FOR RAIL TRANSPORT OF SOLIDIFIED HIGH LEVEL WASTE

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Reprocessing of spent fuel from commercial light water reactors (LWR's) has been deferred indefinitely while the international safeguards issues associated with plutonium extraction and recycle are being debated. If it is eventually decided to proceed with the reprocessing option, much work remains to be done to develop commercial facilities and equipment to solidify the high level liquid waste (HLLW) produced during chemical reprocessing of LWR fuel; to transport the waste from the reprocessing plant to a federal radioactive waste repository; and to isolate the waste from the environment until the radioactivity has decayed to nonhazardous levels. The Department of Energy has conducted a variety of research activities in these areas to insure that reprocessing and waste disposal technology will be available if the decision to use this technology is made.

A shipping cask uniquely designed to transport solidified high level waste (SHLW) from reprocessing spent LWR fuel has not yet been developed. Battelle, Pacific Northwest Laboratory (PNL) has conducted a program for the Department of Energy to develop a conceptual design for a rail cask for transporting SHLW, and to produce models and graphical displays based on the conceptual design for use at technical meetings and in public information displays. The results of this work are summarized in this paper.

The design of a shipping container for radioactive materials is dependent on a variety of factors or design constraints, including federal regulations, the characteristics of the material to be transported, limitations of the transport mode, and operating considerations. Economics and logistics considerations dictate that a shipping cask should be able to transport the maximum volume of SHLW possible within these design contraints. The general approach used to develop the conceptual cask design was to:

- Review current technology in high-level waste solidification.
- Select a reference or baseline canister design and associated characteristics based on the current technology.
- Establish design constraints due to SHLW properties, the railroad transportation environment, and requirements for handling or storing the canisters.
- Identify design constraints imposed by current federal hazardous material transportation regulations.
- Review current technology in spent fuel casks.

- Conceptually design a system to maximize the volume of SHLW conveyed in light of the constraints identified.

The important design constraints for a SHLW cask are summarized in Table 1. These constraints are of three types - federal regulations, waste properties and transportation system haracteristics. Waste properties are based on fuel cycle characteristics and HLW solidification technology that currently appear most likely to be used in the future. Spent fuel is assumed to be reprocessed 1.5 years after reactor discharge, and the SHLW is assumed to be shipped to the federal repository 5 years after it is separated from the fuel. Limitations of the railroad transportation system were determined from contacts with railroad officials. These limitations are intended to avoid special handling requirements because of the weight or size of the shipment. The most important federal regulations that apply to a SHLW cask design are the radiation dose limits and the Type B packaging test criteria. The federal regulations essentially provide an envelope within which the other design criteria can be met.

CASK/RAILCAR DESCRIPTION

An artists's representation of the conceptual cask/railcar system developed in this study is presented in Figure 1. The cask has a capacity of nine canisters of the reference solidified high-level waste. The maximum heat removal capability of the cask is 50 kw heat is removed by natural convection from the cooling fins on the cask body. The loaded cask weighs about 110 metric tons and is shipped on a conventional 6-axle depressed-center flat car that has been modified by the permanent installation of the cask tiedown system. The gross shipping weight of the cask and railcar is about 160 metric tons.

Details of the cask structure and internal design features are presented in Figure 2. The cask body is 4 m long with an overall length, including cooling fins, of 4.4 m. The cask cavity is 1.5 m in diameter and the outside diameter of the cooling fins is 2.5 m. The inner cask cavity contains a solid aluminum magazine to position the waste canisters and transfer the heat from the canisters to the main cask body. Each canister is held in place by a three-part positioner that is pressed against the canister by leaf springs. An aluminum honeycomb impact absorber is placed at each end of the canister when it is loaded into the cask. The cask is designed to be shipped dry (i.e., water is not required in the cask cavity for heat transfer).

TABLE 1. Important Constraints on the Design of a SHLW Cask

Federal Regulations

Radiation Dose Rate - 1000 mr/hr at 1 m
Type B Package Thermal Test - 800°C fire for 30 min
Type B Package Impact Test - 9 m drop onto unyielding surface
Type B Package Puncture Test - 1 m drop onto steel probe

Waste Properties

Canister Size - 30 cm dia x 35 m length
Waste Age - 6.5 years since reactor discharge
Waste Decay Heat - 30 watts/ℓ
Waste Radioactivity - gamma 6 x 10^{13}/sec
- neutron 1 x 10^{7}/sec

Railroad Transportation System Characteristics

Maximum Cask/Railcar Gross Weight - 180 metric tons
Maximum Cask/Railcar Dimensions - Standard railroad clearance outline
Normal Transport Forces - Stresses on tiedowns - 10 g axial
- 5 g lateral
- 2 g vertical

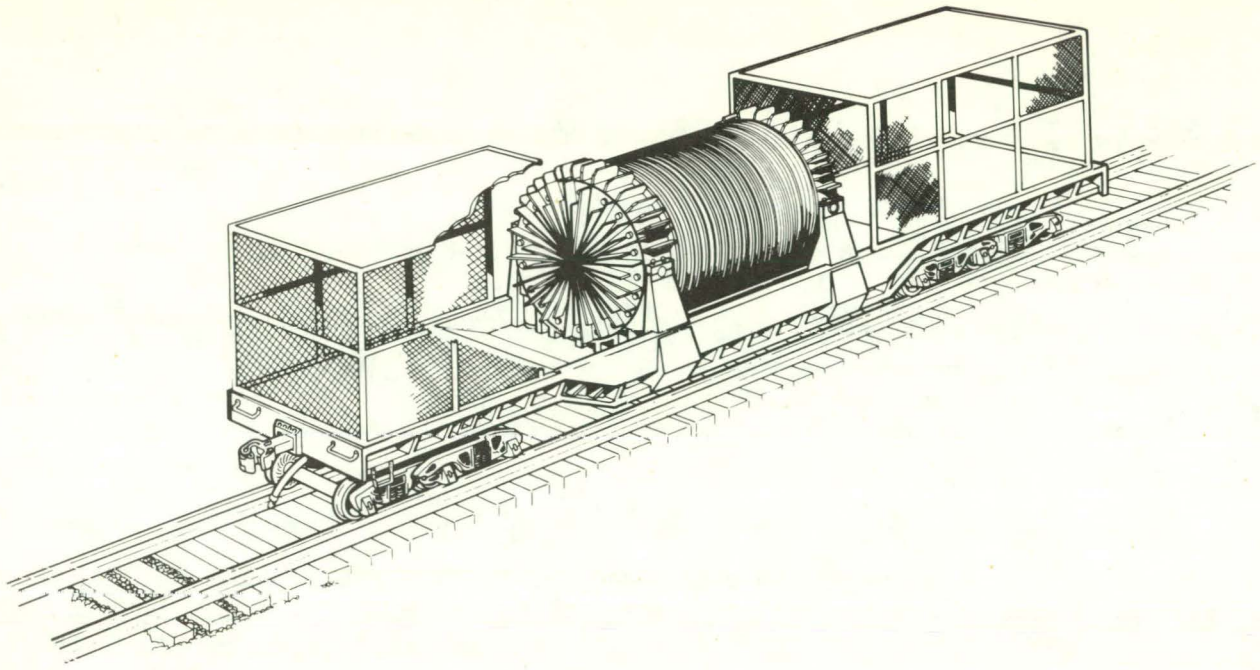


FIGURE 1. Conceptual Solidified High-Level Waste Shipping Cask and Railcar

The cask body is a series of concentric shells. The inner wall, constructed of 1.9-cm thick stainless steel, is surrounded by a lead gamma radiation shield 20-cm thick. The gamma shield is contained by a 5-cm thick stainless steel structural wall. A 10-cm thick borated water neutron shield surrounds the other structural wall. The neutron shield water is divided among four compartments. Each compartment is connected to a 100 ℓ expansion tank that permits the water to remain at a pressure of about 1 atmosphere over the maximum range of temperatures expected during normal operation. Each compartment is also supplied with a pressure relief valve that prevents the neutron shield water pressure from exceeding 5 atmospheres. Ethylene glycol is added to the shield water to prevent freezing when the cask is shipped empty. The cask body is completed by a 1.2-cm thick stainless steel outer wall and a series of radial cooling fins 1.2-cm thick and 10-cm high.

The stainless steel cask lid is constructed in two pieces. Double lid construction has been used on some large spent fuel casks. This concept provides a stronger closure since more area is available for attaching the head bolts. The inner lid of the conceptual cask contains a 12.7-cm thick layer of depleted uranium for gamma radiation shielding. Depleted uranium is used to provide additional strength in the lid if the cask is involved in an accident. The outer lid contains a 7.6-cm thick layer of solid hydrogenous material for neutron shielding. The stainless steel fins on the outer lid function as impact absorbers in an accident. Each lid is attached to the cask body by twenty 5-cm diameter stainless steel bolts. A metallic o-ring gasket is used to seal the outer lid to the cask body. The cask bottom is not removable, but is otherwise similar to the lid. Four 18-cm diameter trunnions are mounted on the cask. The trunnions extend 14 cm from the cask body and support the cask during handling and shipping.

The cask tiedown and handling systems are permanently mounted on the railcar. The tiedowns consist of four massive columns that support the cask during shipment. The trunnions on the cask body rest on the support columns. The cask is clamped in place at each end by a steel semicircular retaining ring 2.5-cm thick that is bolted to the top of the support columns. Energy- absorbing impact fins are welded to the steel rings. A support pad is located under the front and rear of the cask to provide additional restraint if the cask and tiedown assembly is subjected to large axial accelerations during shipment.

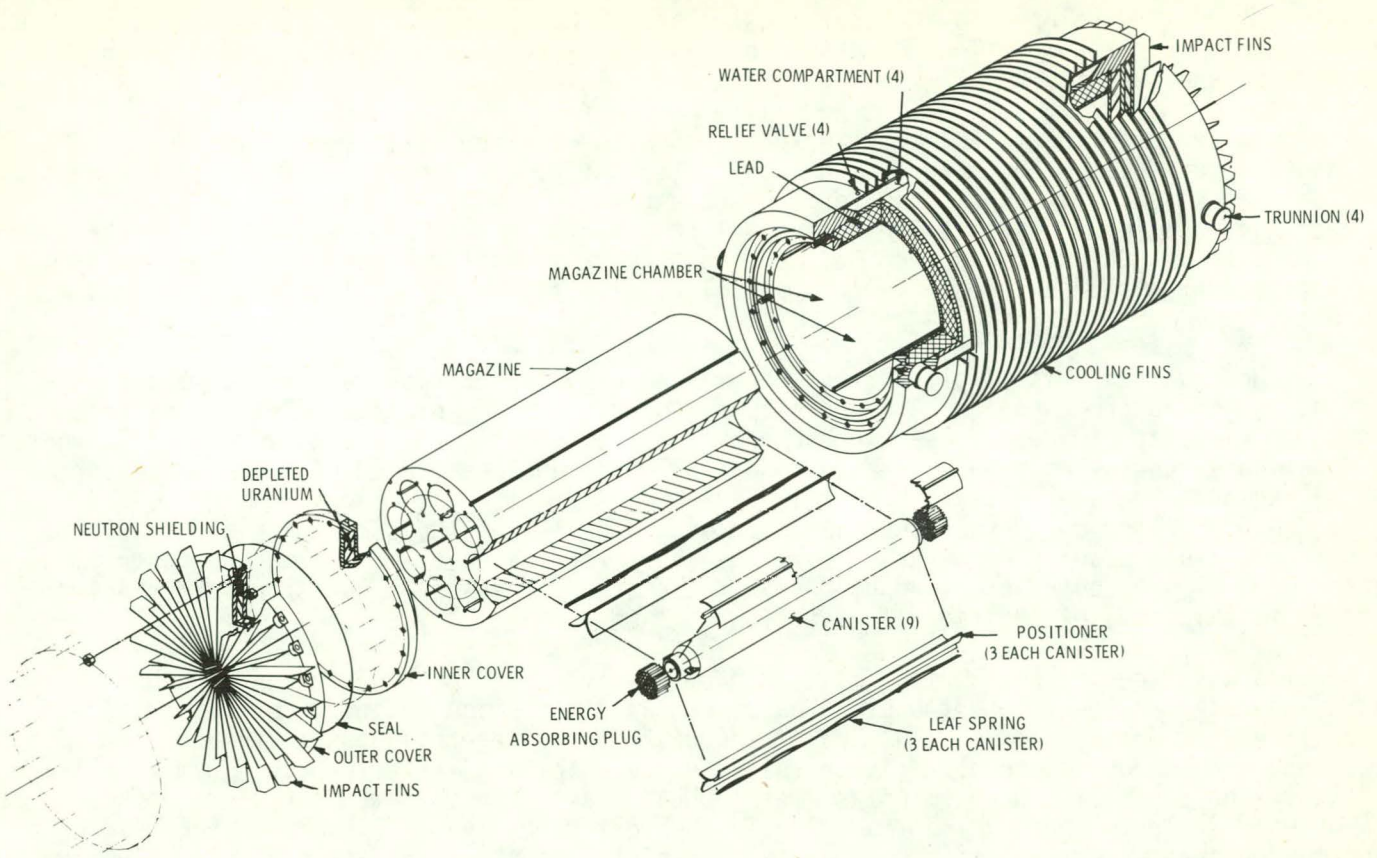


FIGURE 2. Conceptual Solidified High Level Waste Shipping Cask Structural Details

The cask is unloaded from the railcar by removing the retaining rings, lifting the cask with the installed hydraulic jacks so that the forward set of trunnions clears the support column, and attaching a specially designed lifting cradle suspended from a large crane to the trunnions. The cask rotates about the rear trunnions while it is lifted to the vertical position with the crane.

The depressed center railcar shown in Figure 1 is a standard design with some modifications to accommodate the conceptual cask. The deck of the car is steel to facilitate attachment of tiedown equipment and permit easy cleanup of any radioactive contamination that might occur. The support columns and associated bracing and the hydraulic jacks and auxiliary equipment are permanently mounted on the car. An expanded metal cage covers the cask during shipment. This cage keeps debris off the cask and prevents people from contacting the heated outer surface of the cask. The cage is constructed in two pieces that slide back onto the car deck during cask loading and unloading operations.

BASES FOR REFERENCE DESIGN

Much of the cask design is defined by the constraints discussed previously. Design alternatives are available for satisfying many of these constraints. The choice among design alternatives is usually dictated by economics. The task of the designer is to satisfy the design constraints in the most cost-effective manner. A detailed economic analysis was beyond the scope of this study, so the choice of design approaches and construction materials is based primarily on historical experience with spent fuel casks. In some instances, the choice was ased on simplicity or clarity for the models that were subsequently produced. The major areas

of consideration in developing a SHLW cask design are cask structure, heat dissipation, radiation shielding, and cask mounting and handling. The bases for the designs selected in these areas are outlined briefly below.

The primary determinant of cask structural strength requirements are the Type B package accident test criteria. The conceptual design relied on the basic shell configuration used for spent fuel casks that have been shown to meet these test criteria. Preliminary calculations have been made that indicate that the conceptual cask design would survive the puncture test and the 9-m drop test. Calculations were not made for the crush and immersion tests because it was felt that these tests produce much lower stress levels in the cask than the puncture and 9-m drop tests.

The conceptual cask is supplied with energy-absorbing impact fins on the ends and on the retaining rings that hold the cask to the support columns during shipment. These fins are designed to minimize damage in a severe transportation accident. The radial cooling fins would also function as energy absorbers in an accident. Impact absorbers used on large casks for carrying radioactive materials are generally balsa wood, steel fins, or rolled steel shapes. Fins were used for the conceptual SHLW cask design primarily for the simplicity and resulting clarity in a visual display. Internal impact absorbers fabricated from an aluminum honeycomb material were also added at each end of the SHLW canisters because the canisters are readily adaptable to this type of additional protection.

A compatible thermal environment must be provided for the SHLW canisters during both normal transit and under accident conditions. A passive heat dissipation system is preferred because it simplifies the design and construction of the cask and is generally easier to license. The design approach to develop an effective passive heat dissipation system was to minimize all air gaps within the cask. The problem of asymmetric heat transfer due to the canisters being inserted in an oversize hole in the aluminum magazine in the center cavity was minimized by providing a spring-loaded three-piece positioner around each canister as shown in Figure 2. The aluminum magazine was designed such that differential thermal expansions would minimize the gap between the magazine and the inner wall of the cask shell as the temperature rose. Cooling fins were provided on all outside surfaces to facilitate heat transfer.

A thermal analysis of the conceptual cask under normal operating conditions and postulated accident conditions was performed. Two cases were analyzed: 1) a steady state analysis to determine temperatures at key positions in the cask during normal operation, and 2) a transient analysis to determine temperatures in the cask as a function of time if the cask were exposed to a 1000°C fire for one hour. Results of these calculations are summarized in Table 2.

The steady-state analysis shows that the waste centerline temperature, canister surface temperature, and lead shielding temperature could be maintained at acceptable levels in the conceptual cask. The analysis also shows that the cask surface temperature during normal operations will generally be 15-25°C higher than the regulatory guideline of 80°C. This condition is also present in many spent fuel casks currently in use. This necessitates installation of the screen shown in Figure 1 to keep personnel away from the hot surface of the cask.

The transient thermal analysis showed that the SHLW would maintain its containment integrity during exposure to a 1000°C fire for one hour. The analysis assumed that the neutron shield water would be released through the pressure release device soon after the fire began. Because the waste has a relatively low heat generation rate and the cask body has a large heat capacity, the temperature of the waste does not change significantly during exposure to a 1000°C fire for one hour. Some melting of the outer layers of the lead shield may occur at the end of the one-hour fire, but this would not significantly affect the radiation dose rate at the surface of the cask. The analysis indicates that the conceptual cask would remain undamaged during the thermal test required by NRC and DOT regulations (800°C fire for 30 minutes).

Typical shielding materials used in large shipping casks are lead, iron and depleted uranium. Because of the many factors that must be considered, the choice of a shielding material usually involves trade-offs with other cask design parameters. Detailed economic, engineering and safety analysis studies are required to select the optimum shielding material for a

TABLE 2. Results of Thermal Analysis of Conceptual Cask Design

Location Within Cask	Steady State Temperature[a] (°C)	Temperature (°C) During Exposure to 1000° Fire		
		20 min[b]	40 min[b]	60 min[b]
Center of SHLW[c]	450	450	450	450
Canister Surface[c]	415	415	415	415
Lead Shielding	190	200	240	305
Neutron Shield[d]	125	--	--	--
Cask Surface	105	1000	1000	1000

[a] Ambient air temperature is 50°C.

[b] Time after start of fire

[c] Hottest canister

[d] The neutron shield is assumed to be released through the pressure relief device in an extended fire.

particular cask. Such studies are beyond the scope of efforts performed for a conceptual design. The shielding material selections were based on past experience with spent fuel cask designs and engineering judgment of a reasonable balance between the many competing factors.

Lead was selected for the gamma shield material in the shell of the cask because 1) it is generally the least expensive gamma shield material, 2) the temperature during operation was well below the melting point (330°C), 3) the thermal analysis showed that significant melting with the resultant loss of shielding did not occur during exposure to a 1/2-hour fire at 800°C, and 4) the strength of the shielding material was not critical to cask integrity. Depleted uranium was selected for the closed end and lid of the cask body because of its inherently greater strength and resultant resistance to axial impact or puncture forces. The use of uranium in the ends of the cask body has the additional advantage of serving as a neutron reflector.

EFFECT OF CHANGES IN WASTE CHARACTERISTICS

The conceptual cask system is based on a reference SHLW canister with fixed physical characteristics. The design developed in this study provides some flexibility to accommodate SHLW canisters with different configurations and properties, but significant changes in some waste properties would require a different design. The SHLW characteristics identified as most influential on cask design are the heat generation rate and the gamma radiation level. These characteristics establish the heat dissipation and shielding requirements.

The heat generation rate in the SHLW declines significantly during the first few years after the fuel from which the waste is created is removed from the reactor. The conceptual SHLW cask system described in this paper can dissipate up to 50 kw of decay heat with a passive heat dissipation system. If the age of the waste were less than approximately three years, an active system such as those based on spent fuel casks would be required to maintain the same shipping capacity. Higher thermal flux due to either younger waste or a stronger composition might also require the use of materials other than lead as a gamma shielding material because operating temperatures in the shield for a passive heat dissipation system would approach the melting temperature of lead.

Other waste characteristics can also significantly impact cask design. The volume of the anister assembly per unit volume of SHLW can change significantly from the reference value used

if a protective overpack is added to the canister before shipment or if the HLLW is diluted with larger quantities of inert material during solidification. Steel protective overpacks could add up to 15 cm to the diameter of the 30-cm diameter reference canister. The outer protective barrier would decrease the allowable heat flux from a basic 30-cm diameter canister. Overpacks adding more than 2 to 5 cm to the canister diameter would significantly reduce the capacity of a multiple-canister cask due to both greater shielding weight (the shielding layer would be at a great diameter) and constraints on the overall diameter to comply with railroad clearances.

The conceptual cask design is based on a reference SHLW canister diameter of 30 cm. Canister diamters ranging from 15 cm to 60 cm are being considered in HLLW solidification studies. The conceptual cask could be used to transport waste canisters of different diameters from the reference canister by fabricating a new aluminum magazine and inserting it in the cask cavity. The effect of alternative canister diameters on the cask design parameters is shown in Table 3. Larger canister diameters produce the most efficient shipping system because more waste can be transported per shipment and loading and unloading operations would be simplified with fewer canisters in each shipment. Further analysis would be required to determine if the conceptual cask had sufficient shielding and heat removal capacity to accommodate three 60-cm diameter canisters of 5-year-old waste, but this canister size would be favored on the basis of transportation system efficiency.

TABLE 3. Comparison of Alternative Canister Sizes

Canister Diameter (cm)	Maximum Number Of Canisters in Reference Cask	Waste Volume[a] Per Shipment (m^3)	Heat Generation Rate[b] (kw)
30	9	1.58	46
45	4	1.58	46
60	3	2.10	63
23	15	1.48	44
15	27	1.18	36

[a] 80% of theoretical canister volume

[b] Based on 5-year-old reference waste

SPENT FUEL TRANSPORTATION - A REQUISITE FOR ANY ALTERNATIVE IN THE BACK-END OF THE NUCLEAR FUEL CYCLE

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INTRODUCTION

In a paper given at the 1977 ANS Winter meeting in San Francisco, William A. Brobst of DOE warned of the upcoming bottleneck in spent fuel transportation capacity. The nuclear industry will find itself hard pressed to relieve this bottleneck until the present state of uncertainty in the back-end of the fuel cycle is resolved. Several companies, Exxon Nuclear Company, Inc. included, have been developing optimized spent fuel transporter designs with a primary objective of shipping short-cooled spent fuel from LWR reactors to reprocessing plants. These same transporters that would be efficient in a fuel cycle where reprocessing is a key step, may not be optimum in an alternate fuel cycle.

ALTERNATE FUEL CYCLES

With the present deferral of reprocessing, several alternate fuel cycles are being investigated. These include variations of the uranium-oxide fuel cycle to the use of new fuels such as thorium. All have one thing in common - the spent fuel will ultimately have to be transported from the LWR of origin to another facility for processing or storage. The historic baseline uranium-oxide fuel management scheme has included reprocessing as a key element. In this scheme, the incentive is to remove discharged fuel from reactor basin storage to a reprocessing plant as soon as possible in order to utilize the recoverable plutonium and uranium. To support this goal, efforts were aimed at designing spent fuel packages that could transport fuel cooled as little as 0.25 years. Most of the alternate uranium-oxide fuel management schemes being investigated lack this incentive to transport short-cooled spent fuel - in fact, the incentive is to allow the fuel to cool in the reactor basin for an extended period, perhaps in excess of 10 years, in order to minimize the design impact on the receiving facility.

SPENT FUEL PARAMETERS AS A FUNCTION OF FUEL CYCLE

Three of the basic differences in these alternate fuel management scenarios that will affect transportation systems are the amount of cooling time in the reactor basin prior to being shipped off-site (from 0.25 to greater than 10 years), the initial fuel enrichment (from 2.5% fissile up to 4.0%), and the quantity of spent fuel that will ultimately be shipped. This wide range in decay times produces significant differences in two of the spent fuel characteristics that directly impact package design-the gamma and thermal source terms.

Figure 1 shows the decrease in gamma source strength as a function of decay time for typical R fuel. After 5 years of cooling, the gamma source has dropped to less than 10 percent of what it was at 0.25 years.

Figure 2 shows the decrease in thermal source strength as a function of time for typical PWR fuel. As with the gamma source, the thermal source after 5 years of cooling is less than 10 percent of what it was at 0.25 years.

IMPACT ON CONTAINER DESIGN

The variation in gamma and thermal source strengths as a function of time can have a significant impact on package design. This can be illustrated by Exxon Nuclear Company's experience in designing an efficient highway transporter under contract with the Boeing Engineering and Construction Company. Early in the design effort, three basic criteria were imposed on the transporter:

1) A minimum payload of 2 PWR assemblies in order to increase efficiency over existing packages;

2) legal weight (80,000 pound GCVW) to minimize operational constraints; and

3) accommodate short cooled fuel in order to support a planned reprocessing plant.

These criteria impact several aspects of package design and cost.

The high gamma source resulting from items 1) and 3) and the legal weight requirement dictated that depleted uranium be used as the gamma shield because of its high density. Also, the high thermal source resulting from items 1) and 3) necessitate the use of fins on the cask exterior in order to provide adequate heat dissipation capacity. Exxon Nuclear Company believes that the added cost of these features is justified if the transporters are used in conjunction with a fuel cycle containing reprocessing as an integral step. However, this may not be the case if one of the alternate fuel cycles which permits the transport of long decayed spent fuel were to be adopted.

Figure 3 shows potential options which are available for optimizing package payload depending on the cooling time of the spent fuel to be transported. If depleted uranium is retained as the gamma shield material, preliminary trade-off studies indicate that the payload may be increased to 3 or 4 PWR assemblies at 5 and 10 years cooling time respectively. Alternatively, the 2 PWR assembly payload could be maintained and the depleted uranium gamma shield replaced by less expensive lead or steel at 3 and 5 year cooling times.

Potential design changes are also available due to the decreased thermal source if longer decayed spent fuel is to be transported. If the fins could be eliminated from the highway transporter, significant benefits would result from reduced system weight, lower fabrication costs, and improved operation due to ease of decontamination. The impact would be even greater for rail system design where the present high heat load requires both fins and an active cooling system to maximize operational efficiency. If one or both of these could be eliminated by transporting long decayed fuel, the savings in design, fabrication and operating costs would be substantial.

If a fuel management scheme is adopted which utilizes fuel enriched significantly higher than the traditional uranium-oxide fuel cycle, Exxon Nuclear would have to significantly alter its present spent fuel transporter designs. Providing the necessary spacing to assure subcriticality may preclude the ability to design a legal weight system with twice the payload of existing systems.

The task of designing an efficient packaging system for the transportation of spent fuel is drastically dependent on the cooling time prior to shipment and the initial enrichment. Besides the payload, choice of gamma shield material and type of heat dissipation system, other areas affected are: cavity coolant, neutron shield material and system weight. Exxon Nuclear will have to perform an entirely new set of system optimization tradeoffs if spent fuel with long decay time or higher enrichment is to be shipped routinely.

QUANTITY OF SPENT FUEL REQUIRING SHIPMENT

As noted above, the fuel management scheme also affects the demand for spent fuel transportation services. In its 1977 forecast (reference 1), ERDA stated that the annual spent fuel discharge rate from the LWR's operating at the end of 1977 will exceed 1,000 MTU/year. By the mid-1980's, ERDA predicted that the annual LWR discharge rate would increase to 4,000 MTU/year. For a fuel management scheme which includes reprocessing, these annual discharge rates will be equivalent to the demand for transportation services at equilibrium. However, if a scheme is adopted which includes interim storage, this demand would double since ultimately the spent fuel would have to be moved a second time. This would result in an equilibrium demand for spent fuel transportation services in excess of 2000 MTU/year for LWR's operating at year end 1977 and 8000 MTU/year by the mid-1980's.

CONCLUSIONS

The capacity of existing spent fuel transportation equipment is 500-1000 MTU/year. The annual discharge from operating reactors already exceeds the capacity and will continue to outgain it. The bottleneck alluded to by Mr. Brobst is not far off. In order for the existing and potential suppliers of spent fuel transportation equipment to help alleviate this shortfall in capacity, some stability must be brought to the back-end of the fuel cycle. Until the criteria for designing an efficient transportation system can be developed, companies will be reluctant to invest substantial capital for a system that may soon be outdated.

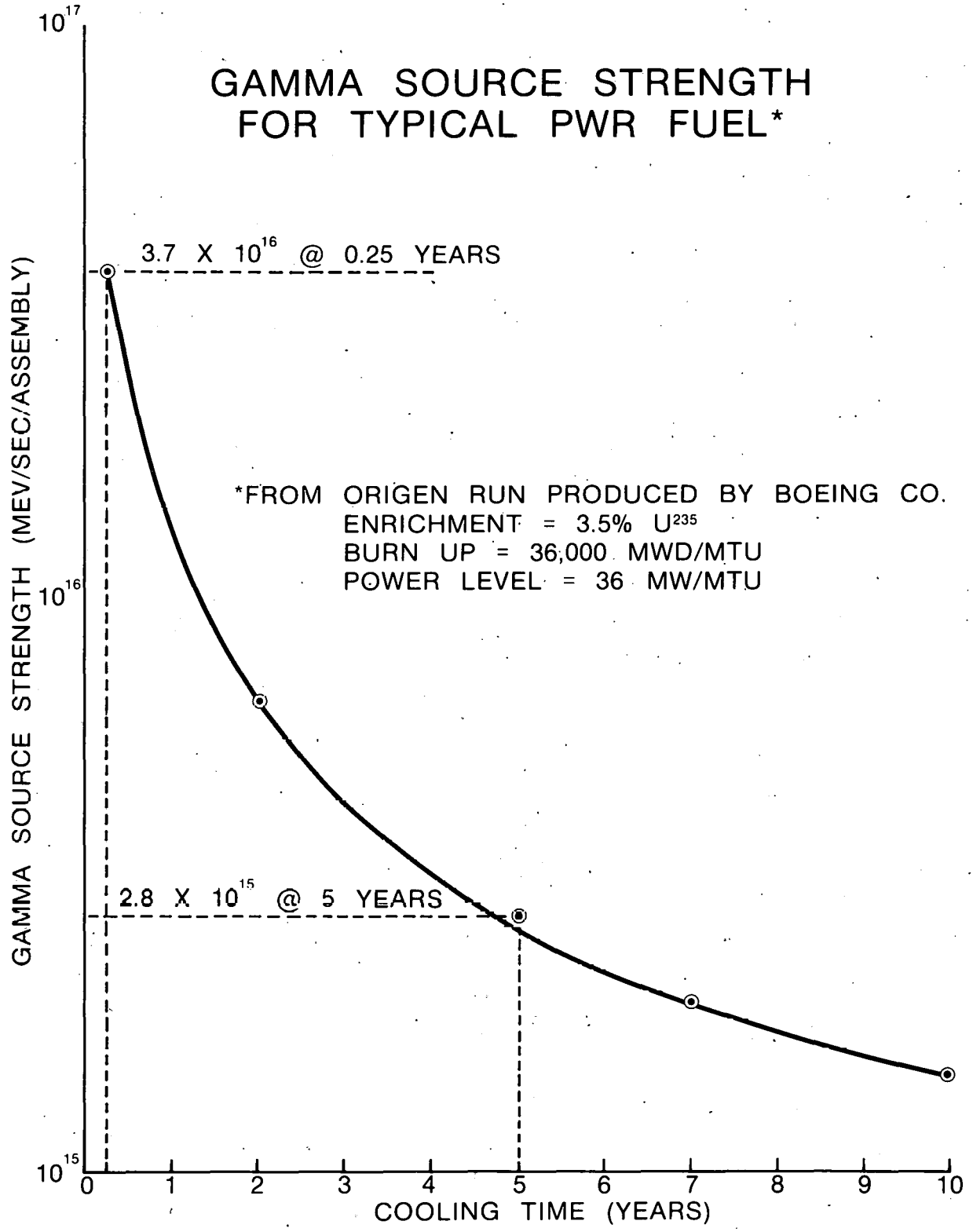

GAMMA SOURCE STRENGTH
FOR TYPICAL PWR FUEL*
3.7 X 10^16 @ 0.25 YEARS
*FROM ORIGEN RUN PRODUCED BY BOEING CO.
ENRICHMENT = 3.5% U235
BURN UP = 36,000 MWD/MTU
POWER LEVEL = 36 MW/MTU
2.8 X 10^15 @ 5 YEARS
GAMMA SOURCE STRENGTH (MEV/SEC/ASSEMBLY)
10^17
10^16
10^15
0
1
2
3
4
5
6
7
8
9
10
COOLING TIME (YEARS)

FIGURE 1

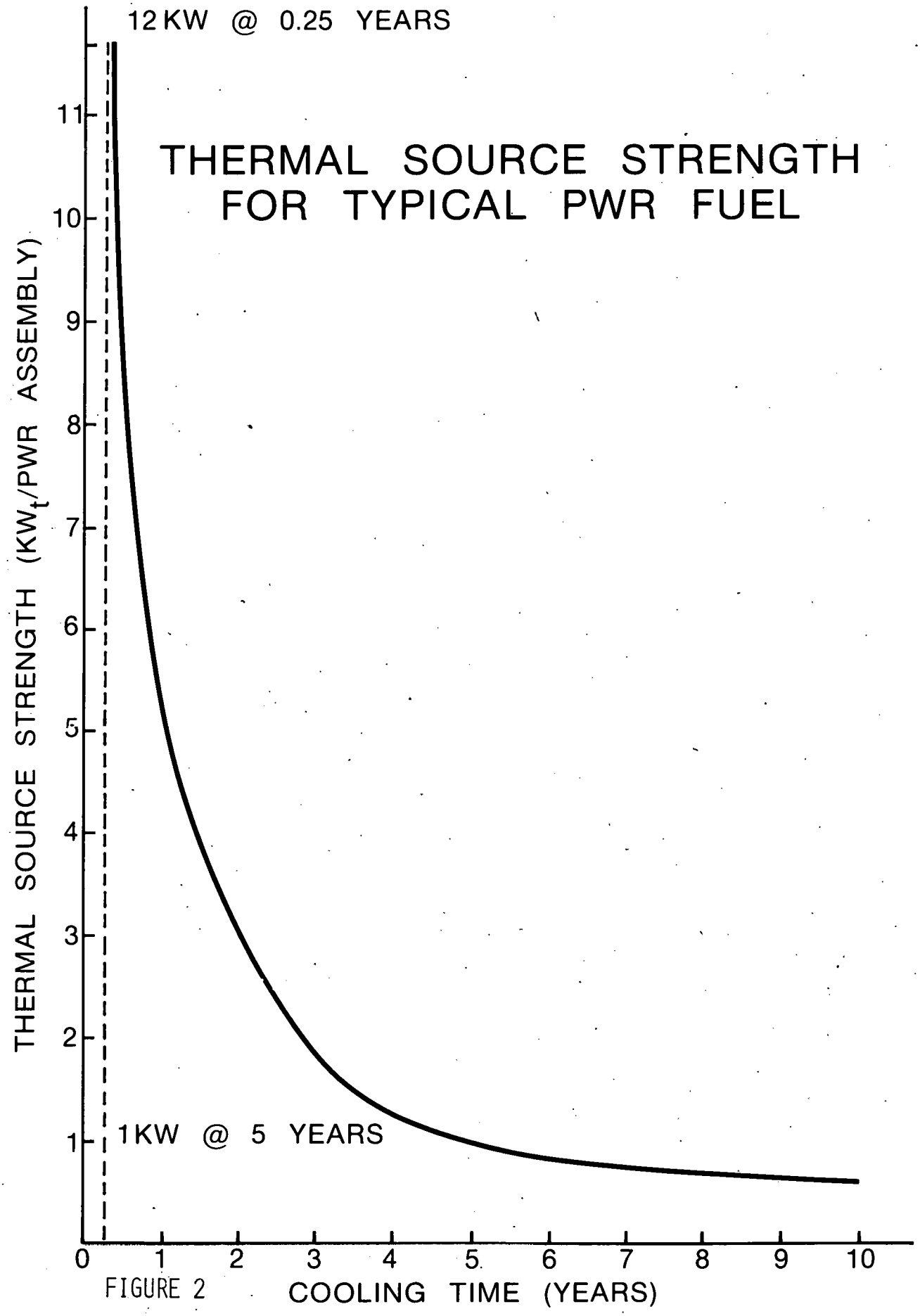

12 KW @ 0.25 YEARS
THERMAL SOURCE STRENGTH FOR TYPICAL PWR FUEL
THERMAL SOURCE STRENGTH (KW_t/PWR ASSEMBLY)
11
10
9
8
7
6
5
4
3
2
1
1KW @ 5 YEARS
0
1
2
3
4
5
6
7
8
9
10
COOLING TIME (YEARS)

FIGURE 2

PAYLOAD OF LEGAL WEIGHT TRUCK CASK VS SPENT FUEL COOLING TIME

GAMMA SHIELD MATERIAL	NUMBER OF PWR FUEL ASSEMBLIES* FOR COOLING TIME OF (YRS)			
	0.25	3	5	10
DEPLETED URANIUM	2		3	4
LEAD		2		
STEEL			2	

*ESTIMATED BY BOEING ENGINEERING AND CONSTRUCTION COMPANY DURING PRELIMINARY TRADEOFF STUDIES.

FIGURE 3

Dup

OAK RIDGE, Y-12 PLANT DT-14 FISSILE MATERIAL SHIPPING CONTAINER (A SAFE AND ECONOMICAL METHOD OF PACKAGING ENRICHED URANIUM)

H. E. Crowder
Union Carbide

In October, 1966 the Atomic Energy Commission issued regulations for the packaging and shipping of radioactive materials. These regulations, identified as AEC Manual Chapter 0529, created a new era for containerization of fissile materials. The most stringent section of the regulation is the hypothetical accident conditions that a container must withstand before it can be used for the shipment of enriched uranium. The container must withstand a puncture test which consists of dropping the loaded container onto a six-inch diameter steel rod from a distance of 40 inches. The container is then dropped from a distance of 30 feet onto an unyielding surface, and the contents of the package must not be exposed, and the lid of the container must stay intact. After the drop tests have been accomplished the container must withstand 1475° F for a period of 30 minutes. During the fire test, the acceptable temperature inside the inner container will vary, depending on the type of enriched material being shipped. For instance, uranium solutions are shipped in a polyethylene bottle, therefore, the temperature must be kept below 200° F.

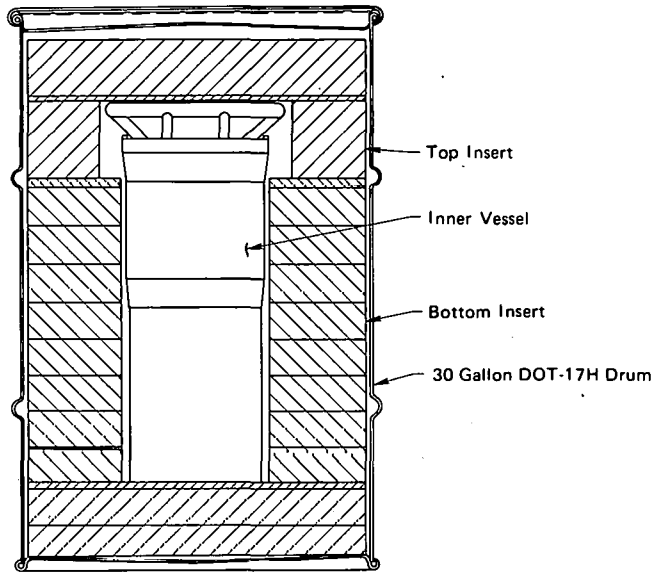
Through years of container testing, Union Carbide's Y-12 Plant has developed a family of fissile-material shipping containers using cane fiberboard as an insulator. During fire testing of these containers, a minimum of three inches (wall thickness) of cane fiberboard has been shown to limit the temperature of the inner container to a safe value. The highest temperature received by the inner container has been less than 250° F.

The family of fissile material shipping containers developed by the Y-12 Plant are referred to as DT (Drum Type) Containers. At this time, there are fifteen "Drum Type" containers that have been tested and accepted by the Department of Energy.

The "Drum Type" containers are fabricated with easily acquired materials such as: a DOT Specification 17H metal 30-gallon or 55-gallon drum; cane fiberboard insulation, better known by its brand name "Celotex;" plywood; and an identification plate. The inner-containment vessel is standard steel tubing with an "O"-Ring grooved flange. The lid and bottom consist of steel sheet stock.

The DT-14 is a unique shipping container — it provides a safe containment for shipments of enriched uranium in solid form, in solutions, and as oxides. This container will enable Union Carbide's Y-12 and K-25 Plants in Oak Ridge to eliminate the use of several types of fissile material shipping containers such as: the DOT-SP-5560 solution container, the DOT-SP-5765 shipping container, and the DOT-SP-5463 (Foamglas) container.

The DT-14 is a superior container. In the past, shipments of enriched uranium were made using loose vermiculite as an insulator between the outer and inner containers. The loose vermiculite settled during shipment and did not provide ample



CROSS SECTION OF THE DT-14 SHIPPING CONTAINER.

protection for the enriched uranium under hypothetical accident conditions. Therefore, the use of loose vermiculite has been prohibited after June 15, 1978. The use of loose vermiculite has other unfavorable aspects – the need to load and unload material around the inner container, the need to clean inner container for reuse, and the administrative control necessary to ensure that containers are properly filled with vermiculite before shipment. Also, if the outer container was accidentally punctured, the loose vermiculite would be lost, and the inner container would have inadequate insulation.

The DT-14 is a practical container. This container was designed and tested to meet the Code of Federal Regulations Number 49, the Department of Energy Manual Chapter 0529 regulations, and International Atomic Energy Agencies regulations for Type "B" packages. This container has also been designed and tested with three inner containment vessels, (1) the M-101/M-102 vessels which will be used by Union Carbide Plants at Oak Ridge, Tennessee (2) an "Alternate" inner-containment vessel, which can be used by facilities not having access to the M-101 or M-102 vessels.

The DT-14 can be recycled and reused without excessive maintenance. Physical damage to the outer container is very unlikely, because the inner liner is a 14 lbs/ft.^{3} density cane-fiberboard insulation. Under normal handling conditions the DT-14 will last indefinitely as a safe and economical container. Decontamination and cleaning is made easy because the inner containment vessel is removable.

A major benefit of the DT-14 shipping container is its ability to handle a larger payload in a smaller outer container. The outer container shell is a 30-gallon Specification 17H metal drum, and can safely transport a payload of 30 kg of enriched uranium in solid form. This container has also been approved for the shipments of enriched uranium solutions and oxides.

With the introduction of specialized transportation systems, such as the Safe-Secure Trailer (SST), the reduction of the overall package size and the increased payload of the DT-14 offers a sharp decrease in transportation cost, as well as reduced storage requirements and cost.

The DT-14 could conceivably be used for the shipment of U-233, with the proper criticality safety analysis and approval. Considering the versatility and expected long life of the DT-14, a fabrication cost of approximately $200 per container is very reasonable.

A ten minute color film report covers the testing of the DT-14 per the Hypothetical Accident conditions of Department of Energy Manual Chapter 0529. The first section of the film shows the assembly of a new container before testing. The second section of the film shows the puncture test, accomplished by dropping the container onto an upright 6-inch diameter steel rod from a distance of 40 inches. The third section shows the 30-foot drop test; dropping the container from a distance of 30 feet onto an unyielding surface. In this test, the container must strike the bolt and locking-ring area, because this area is considered to be the most vulnerable. The fourth section of the film shows the containers entering a furnace at 1475° F. Here, the containers are monitored for 30 minutes, then removed. Upon exit from the furnace, the film dramatically shows the gases being expelled through the holes in the metal container. The fifth section of the film illustrates the physical damage of the drop and fire tests. The last section of the film shows the leak test of the inner vessels. Narration accompanying the film explains in detail the acceptable conditions of each testing procedure.



DT-14 SHIPPING CONTAINER BEFORE TESTING.



DT-14 SHIPPING CONTAINER AFTER TESTING.

URANIUM-LEAD SHIELDING FOR NUCLEAR MATERIAL TRANSPORTATION SYSTEMS

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SUMMARY

The basis for the selection of shielding materials for spent fuel shipping containers is described with comments concerning the favorable and unfavorable aspects of steel, lead, and depleted uranium. A concept for a new type of material made of depleted uranium and lead is described which capitalizes on the best cask shielding characteristics of both materials. This cask shielding is made by filling the shielding cavity with pieces of depleted uranium and then backfilling the interstitial voids with lead. The lead would be bonded to the uranium and also to the cask shells if desired. Shielding density approaching 80 percent of that of solid uranium could be achieved, while a density of 65 percent is readily obtainable.

This material should overcome the problems of the effect of lead melting in the fire accident, high thermal gradients at uranium-stainless steel interfaces and at a major reduction in cost over that of a solid uranium shielded cask. A development program is described to obtain information on the properties of the composite material to aid in design analysis and licensing and to define the fabrication techniques.

INTRODUCTION

Historically, the materials used for shielding emitters of gamma and neutron radiation have included water, concrete, iron, lead, uranium and tungsten. For portable containers to ship high-burnup, power-reactor fuel, the selection of the shielding material depends on many factors. Shielding efficiency, material cost, fabricability, and shield weight are important factors. Also thermal and mechanical properties must be compatible with a design which can tolerate normal and postulated accident shipping conditions. To a degree these factors are interrelated. The gamma shielding efficiency of a material is roughly proportional to its density or its atomic number. Neutron shielding requires moderating capability and neutron capture capacity. Thus for both gamma and neutron shielding a composite shield is an efficient method of incorporating all of the necessary properties.

Selection of the higher density material (increased shielding efficiency) results in lighter weight shipping containers for a given internal volume. In addition to the saving in shipping cost of a lighter weight cask, road weight limits for truck casks and crane and rail car limitations for rail casks must be considered.

A rule of thumb used by designers when uranium is used in place of lead as a shielding material is that a truck cask is 20 percent lighter and a rail cask is 15 percent lighter. This advantage may be used to increase capacity rather than reduce weight. For a legal weight truck cask with uranium shielding, two PWR or three BWR elements may be shipped as compared to 1 PWR or 2 BWR elements in a legal-weight lead shielded cask.

COMPARISON OF COMMONLY USED SHIELDING MATERIALS

For gamma shielding materials the three commonly used materials are steel, lead and depleted uranium (DU). Tungsten might be considered but it is difficult to fabricate, high in cost and is not readily available in large quantities.

Steel is used as a shielding material because of its low cost, adequate strength at high temperature, availability, good thermal conductivity, well established fabrication techniques and good data base for licensing analyses. Negative aspects of steel are its lower density (0.28 lb/in.^{3}), high buildup of secondary gamma radiation, the possibility of thermally induced stress corrosion cracking (stainless steel), brittle behavior at low temperatures (mild steel), and the large physical size. The heavier steel casks (weight about 1-1/2 times that of an equivalent capacity lead shielded truck cask) are not attractive economically if shipping costs, both loaded and empty, are computed on vehicle gross weight.

The favorable points in the use of lead as a gamma shield material include its higher density (0.41 lb/in.^{3}), good availability, low fabrication cost, and impact energy absorption capability. Problems associated with the use of lead include its low melting point (622 F), low structural strength, cold flow characteristics and poor thermal conduction at any gaps between the lead and containment walls caused by shrinkage.

Depleted uranium (DU) is selected as a shield material for its high density (0.68 lb/in.^{3}), high melting point (2070 F), structural strength and availability. Negative aspects of use of DU are the high fabrication cost relative to lead and steel, limited casting size, lack of acceptable mechanical and physical property data and high thermal gradients at interfaces with cladding materials. Of lesser importance is the need for protection from oxidation by air or water and problems of handling and inventory accountability of DU by nonlicensed fabrication facilities.

In the above discussion little mention has been addressed to the costs of these materials. The material cost of shielding materials is only a part of the total cost of cask shielding since the fabrication cost of a particular material may be much greater than the cost of the raw material. In addition, use of a less expensive shielding material could result in a greater total shielding cost if structural additions or more analysis for licensing is required for a given material. Also, the cost analysis for one or two casks of a single type is different in the relative weighting of cost elements than that for a production run of 20 casks. For these reasons the authors do not consider simple material cost comparisons to be a valid selection criteria for shielding materials.

Thus, in selection of a shielding material, it is necessary to weigh the advantages and disadvantages of different materials relative to the overall design requirements for the cask. No single shielding material is best for all applications.

URANIUM-LEAD MIXTURE

A promising method of achieving much of the weight reduction of a uranium shielded cask at a fabrication cost approaching that of a lead cask is by use of a uranium-lead mixture. This method would use small pellets or chunks of depleted uranium metal compacted into the shielding space and then backfilled with molten lead. The fabrication technique would have the simplicity of a poured lead shield.

The lead could be made to bond to the uranium pellets and the shield walls (or to fins on the outer wall) to provide a good thermal path completely across the shield. The filling technique must assure a uniform dispersion of uranium pellets which can be completely infiltrated with the lead matrix. By optimizing the range of pellet sizes and the filling technique, one may expect to obtain about 80 v/o uranium in the shield space. Even lesser volume fractions of uranium will result in significant cask weight savings. For equivalent shielding around a given size cask cavity a lead shield may weigh 20 percent more than a uranium shield and a 60 v/o uranium-lead shield may weigh 5 percent more than the all uranium shield. In addition to the cask weight saving over that of lead, the uranium-lead matrix system would provide the efficient thermal path of some all-lead shields and should overcome the thermal expansion stress problem of uranium and stainless steel. This type of shielding could adequately survive the fire accident condition where molten lead would be lost in the case of a breach in the shielding containment. The uranium pieces would not melt or leak out and would provide adequate shielding for the fire accident condition even if all of the lead in the matrix were lost. Also with the more brittle uranium in the ductile lead matrix one may expect to overcome problems of the cold flow of lead and the brittle fracture characteristic of uranium.

Potentially this material has many other uses. Examples would be for shielded areas in nuclear facilities where lead is inadequate, where uranium metal cannot dissipate the heat, and for readily castable high density weights such as are used on certain aircraft parts and other machinery.

DEVELOPMENT REQUIREMENTS

The problems to be solved to develop the uranium-lead mixture into a practical shielding material lie in five main areas.

- Determine acceptable or optimum characteristics of the uranium and a general forming process.
- Develop an economical means of treating the uranium surface to assure wetting by molten lead.
- Define techniques for handling the uranium and producing packed beds in containers.
- Define acceptable methods for infiltrating the bed of uranium with molten lead.
- Determine physical and mechanical properties of the uranium-lead mixture to the extent necessary to permit adequate design and analysis for licensing and use.

Industry experience is available to be applied to many of these problems and some preliminary investigations have been made on the others. This information makes the development program largely the application of industry "state-of-the-art" techniques and promises a superior shielding material for the industry. A discussion of required development tasks follows.

Characteristics of the Uranium

The required physical character of the uranium for the matrix material is primarily a function of the size, range of sizes, and shape of pieces which permit formation of high density packed beds. However, several secondary factors must also be considered and may be controlling in determining the optimum characteristics. The size of the uranium pieces is influenced by several factors. The pieces must be significantly smaller than the smallest dimension of the shield in which they are used and must be small enough that the mixture may be considered homogenous. These influences toward small sizes must be in balance with the need to provide sufficient interparticle spaces to facilitate infiltration with molten lead. Also, the surface area of the uranium should be limited to minimize problems of surface treatment and to limit the area for interaction of the uranium with the lead. The shape of the uranium pieces should afford efficient packing in beds with a high degree of confidence of reproducing uniform density beds.

The shape of the pieces will also be influenced by the process used for treating the surfaces for wetability; if electroplating is used, one should avoid pieces with sharp edges and points. However, the greatest influence of size and shape will probably be the method for most economically producing the uranium. At this point in the concept it is believed that the uranium will be in the form of chunks, pellets, or cylinders rather than particles.

Surface Treatment of the Uranium

An intimate bond between the lead and uranium such as that produced by wetting is considered essential to achieve acceptable physical and mechanical properties of the mixture. The as-produced uranium, being a reactive metal, can be expected to have surface layers of oxide or other products which will inhibit the wetting of the uranium surface by the molten lead. Thus, some treatment of the uranium surface is needed to promote wetting. The coating of the uranium with some material which will accept wetting by the molten lead appears to be one acceptable surface treatment. A number of coating techniques should be investigated; among these are electroplating, vapor deposition, and spray coatings. Preliminary considerations indicate that plating using barrel plating techniques is a promising economical approach for coating large quantities of uranium pellets with metal coating amenable to wetting by molten lead. Successful coatings of nickel, tin, and copper have been made.

Handling Techniques

Methods are well established for handling (pouring, scooping, packing, mixing, etc.) materials such as gravel, iron ore, lead shot, steel punchings. However, pellets of materials with over double the density of steel can be expected to have different handling characteristics and their potential handling problems need investigation. These problems can best be evaluated and solved by experimental approaches. Sample lots of the proposed uranium pieces should be used to evaluate the suitability of current methods, or modification of current methods, for the various required handling operations to be encountered. These include moving and storing multi-ton lots of uranium pellets, mixing or blending of different sized pellets, continuous feeding into and out of coating equipment, and filling and compacting into typical shielding geometries for shipping containers (3 to 6-inch wide annular regions up to 15 feet long).

Lead Filling Techniques

The filling with lead of the voids in a packed bed of uranium pellets in a shield chamber should be conducted with precision and under controlled conditions. Lead melts at 622 F and is generally poured in the temperature range of 700 to 800 F. In order to infiltrate small passages the walls must be heated to keep the lead fluid. In this case, massive components (possibly 16 feet long and 5 feet in diameter) must be preheated to nominally 750 F. The heating must be done uniformly to avoid sensitizing the steel walls at localized hot spots and to minimize gradients which could cause distortion of the walls or the packed bed. Conversely, the system should be preferentially cooled from the bottom up while additional lead is added to the top to allow liquid to contact the solidified phase and prevent the formation of shrinkage voids. The system should be evacuated and outgassed during heating to limit the oxidation of surfaces which would inhibit wetting and to avoid dross formation of the lead which would limit its flowing and penetration into small areas. The actual pouring could be done under vacuum or inert cover gas which is a technique presently used in some lead pouring operations.

The development of the filling method would of necessity be scaled up from studies conducted with small prototype sections. A necessary part of the lead filling method will be inspection methods to verify that adequate filling has been achieved to give acceptable shielding, thermal, and mechanical properties.

Physical and Mechanical Properties

Since this composite is a new material, it will be necessary to determine those physical and mechanical properties which are necessary to characterize the material to assist in the sign, the analytical defense of the integrity, and the fabrication of composite-filled spent el shipping containers.

Some concern has been raised in relation to the possible pyrophoric nature of uranium-lead mixtures. The literature shows that the intermetallic compound UPb is highly pyrophoric and that UPb_3 is pyrophoric. The uranium-lead phase diagram shows that temperatures much higher than a casting temperature of 800 F are required for solubility of uranium in molten lead or for the formation of intermetallic compounds. Also, shipping containers in which lead has been cast around large inserts of uranium have been successfully fabricated. Another factor opposing reactions in the present concept is the anticipated need for a metallic coating on the uranium which would limit the contact of the uranium by the lead. Because of the above concerns, the reactions between molten lead and uranium, as well as lead and potential coating materials should be investigated to temperatures which may be anticipated during the casting process and maximum thermal accident conditions.

Some of the thermophysical properties of the uranium-lead mixture can be predicted from the properties of the individual materials and their relative amount in the mixture. This is true for specific heat and, to a lesser degree, the thermal conductivity. However, the thermal expansion of lead is sufficiently higher than that of uranium to make this bulk property less certain. These properties need to be verified over the maximum anticipated temperature range. Where radiant heat transfer may exist between the shield material and an adjacent wall, one could justifiably assume the surface emissivity to be that of lead.

It is believed that the shielding capacity of this mixture can be analyzed by assuming it is a homogenous mixture of uranium and lead. The validity of this assumption should be verified by comparing the results of experimentally measured attenuation factors of different energy gamma sources with the values predicted by analytical techniques.

A thorough characterization of the mechanical properties of the uranium-lead mixture to allow analytical prediction of the integrity of the shield and to analytically evaluate the interaction of the shield with the containment vessel as a result of the combination of forces which may be imposed on the system during normal and postulated accident condition. The shield material is a heterogeneous composite in which the normal tensile, compression, and shear strengths do not have a true meaning. It will be necessary to obtain these mechanical property data as a function of temperature and of strain rate to permit analyses of static and dynamic stresses over a range of temperature. The data should also be determined for a range of U-Pb mixtures.

The uranium-lead composite material cannot be considered a structural material and must depend on adjacent structural materials for support. The strength of samples of the material will be sensitive to the size of the sample until its size is large in respect to the size of the pellets in the mixture. It may be necessary to consider the interaction with adjacent structural walls in defining meaningful strength data. The tensile strength of the material can be expected to be controlled by the strength of the lead matrix unless the strength of the lead-to-uranium bond is considerably lower. When subjected to compressive loads, forces can be transmitted through the relatively higher strength pellets at point to point contacts which may tend to force the pellets apart and apply tensile stresses to the matrix. Thus, the compressive strength of the composite may be a function of the tensile strength of the matrix or the matrix-to-pellet bond strength. These types of data can be obtained with properly defined specimens in standard static testing machines.

When an impact occurs on a localized area on a packed bed of pellets, forces may be expected to be transmitted through the bed like those in an array of racked pool balls when struck by the cue ball. These forces will produce high localized stresses at the points of contact of pellet to pellet and pellet to wall. The lead matrix will tend to dampen the forces. A careful evaluation of this effect is necessary to allow analysis of the interaction of the shield material h the adjacent walls of the cask, particularly the containment vessel walls. These effects

can best be determined with impact or drop tests in which deceleration rates and energy dissipation may be related to full size shipping casks. The important information to be gained from the drop tests is in regard to attenuation of "g" forces through the composite material, localized and generalized deformation in the composite and adjacent structural walls, and void formation above the composite (analogous to slump in a lead shield).

In summary, the potential advantages of the composite appear to more than offset the efforts required for material characterization to gain the simplicity of fabricability of lead with most of the advantages of a uranium shield.

Dup

A SHIPPING CONTAINER FOR TRITIATED WATER

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INTRODUCTION

A reusable shipping container for Type B and large quantities of tritiated water has been designed and tested at Mound Facility. An inner and an outer container are used for shipping up to 100,000 Ci of tritium in water absorbed on molecular sieve, silica gel or commercial clay absorbent. The package is commonly called configuration-5 of the model AL-M1 shipping containers. It is the newest member of the AL-M1 family which currently also includes configurations 1 and 3. These other configurations are both for plutonium-239 and uranium-235. The common feature uniting the AL-M1 family is the 55-gal outer drum assembly which is the same for all configurations. The inner containers are quite dissimilar.

A "Safety Analysis Report for Packaging" or SARP was prepared to qualify the new shipping container for DOT/DOE approval.[1] That document is a formal safety analysis of all three AL-M1 insulated drum shipping container configurations. Together these are officially identified as USA/9507/BLF (ERDA-AL). The Certificate of Compliance has been issued and the SARP has been submitted to NRC for review. This report deals mainly with a description of the configuration-5 inner container.

In addition to the prototype used in testing, thirteen of these containers have been fabricated. One is retained at Mound Facility and twelve will be used at Sandia Laboratories, Livermore, to collect tritiated wastewater. This wastewater whill be shipped to Mound for recovery or disposal of the tritium.

PACKAGE DESCRIPTION

The AL-M1 configuration-5 shipping container is shown in Figure 1. The complete package consists of three main assemblies - the insulating outer drum, the insulating spacer and the inner container. In addition to the descriptions contained in the SARP [1], detailed engineering drawings of the AL-M1 containers are available from Mound Facility. For the configuration-5 these are:

Mound Facility is operated by Monsanto Research Corporation for the U. S. Department of Energy under Contract No. EY-76-C-04-0053.

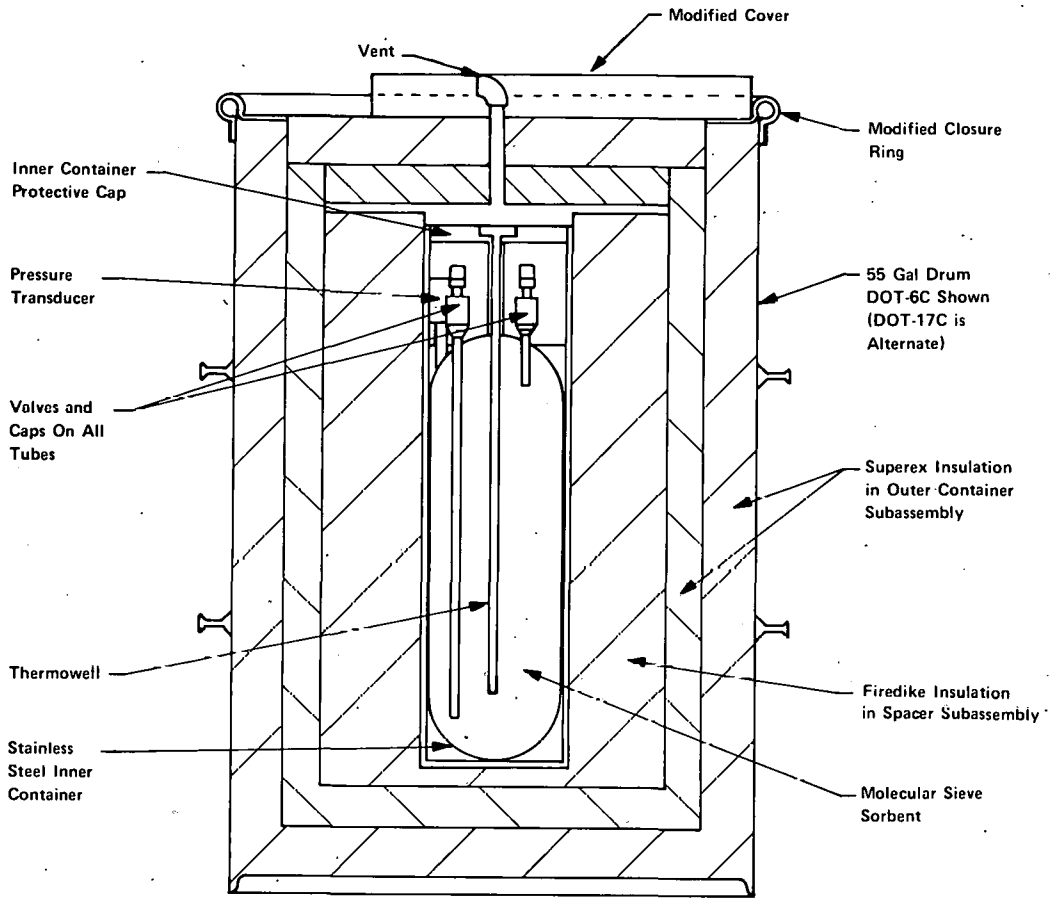


FIGURE 1 - Sketch of AL-M1 configuration-5.

Item	Drawing No.
Insulating drum assembly	AYD 740425
Insulating spacer	AYD 740626
Inner container	AYD 740198

The drum is a DOT 17C, 55-gal 16 gauge steel drum with modified cover and closure ring. The drum assembly is insulated with Superex and lined with stainless steel forming a cylindrical cavity 15-in. in diameter and 25-in. high. The insulating spacer fits into this cavity. This spacer consists of a glued stack of Fire-dike insulation lined with stainless steel which forms the cavity for the inner container.

A photograph of the configuration-5 inner container is shown in Figure 2. It is a cylindrical vessel nominally 6 5/8-in. outside diameter by 23 7/8-in. overall height when assembled. It is made of 316L stainless steel for compatibility with tritium and corrosion resistance.

The container has a removable cover as is shown in Figure 3. This cover is held in place by the bail (also shown removed in Figure 3), which is screwed onto a threaded center post which projects through the cover. The bail also serves as a handle for manual lifting of the assembled container. The cover has two self-sealing quick-disconnect fittings on top for connection to tritium monitors; this permits checking for possible contamination under the cover before it is removed. The cover has two purposes: physical protection of the valves, fittings, and pressure transducer at the top of the primary containment vessel; and provision of secondary tritium containment for these same elements. For this latter purpose, the cover is sealed with o-rings. There are four such o-rings: one face seal at the bottom circumference of the cover, one near the top of the post, and two sealing the quick-disconnect fittings in their wells at the top of the cover.



FIGURE 2 - The assembled AL-M1 configuration-5 inner container.

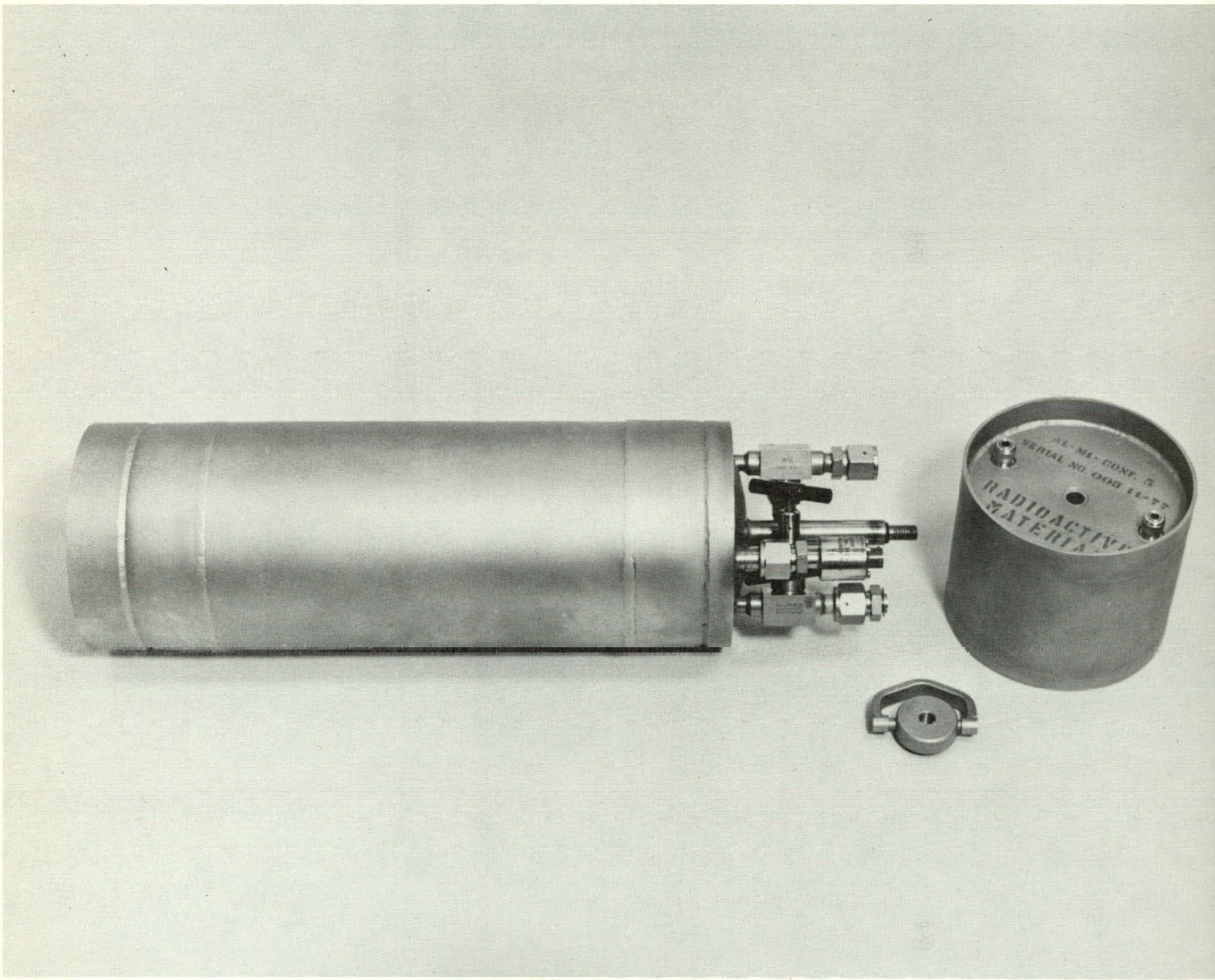


FIGURE 3 - The disassembled AL-M1 configuration-5 inner container showing the cover and bail removed.



FIGURE 4 - The top of the primary containment vessel of the AL-M1 configuration-5 inner container. The two valves with capped disconnect fittings, the pressure transducer, the center post, and the circumferential o-ring face are shown.

Figure 4 shows the top of the primary containment vessel. The two valves are of the bellows type with metal seats and stems, and are attached to metal gasketed disconnect fittings. These fittings are used for connection to external systems for loading and unloading tritiated water; they are capped for shipment. Also during shipment, the valves are closed, with the seats "facing" the vessel for protection against possible bellows failure. The pressure transducer is attached with a disconnect fitting. Removal of the transducer gives direct access to the interior of the vessel.

Figure 5 is a cross-sectional sketch of the inner container showing the inlet and outlet tubes. The thermocouple well is also shown. The interior of the primary vessel is filled with a granular or pelletized solid adsorbent for water. To date, the containers have all been filled with about 5.6 kg of type 13X molecular sieve pellets.

TRITIUM CONTENTS

The quantity of tritium to be shipped in each configuration-5 container is arbitrarily limited to a maximum of 100,000 Ci or about 10 g of tritium. This quantity does not generate enough heat from radioactive decay (3.3 W) to cause any significant temperature rise in the inner container.

In addition to this limitation on total tritium, the quantity of tritiated water is limited. The quantity of water to be shipped is determined by weight difference between the filled and unfilled package. For three common materials evaluated as sorbents, the allowed maximum quantities of water are shown in Table 1. No free liquid is allowed.

The column labeled "sorbent quantity" indicates the minimum weight of sorbent which should be loaded into the inner container. The column labeled "maximum allowed water loading" represents quantities which ensure that no free liquid will exist within the container. In actual tests, equivalent quantities resulted in loaded sorbents which did not cling to container surfaces and retained the normal free-flowing properties of dry granular solids.

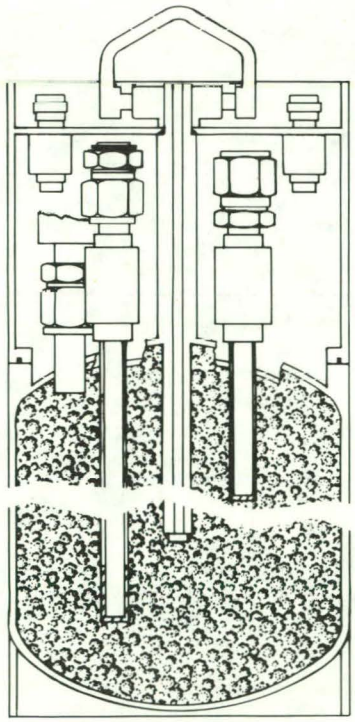


FIGURE 5 - Sketch of the AL-M1 configuration-5 inner container.

Table 1

WATER LOADING LIMITS FOR THREE SORBENTS

Sorbent Material	Sorbent Quantity (kg)	Maximum Allowed Water Loading (g)
Molecular Sieves 4A, 5A, or 13X	5.6	2000
Silica Gel	6.4	2000
Commercial Clay Absorbent[a]	4.2	2500

[a] A typical montmorillonite clay, Florco, from the Floridin Company, Pittsburgh, Pa., was tested.

The formation of helium-3 from natural tritium decay amounts to a generation rate of less than four liters/year for 100,000 Ci of tritium. With a void volume of 20% in the sorbent, the pressure in the 10-liter inner container would thus rise only about 30 psi in one year.

The tritium must be in the form of tritiated water (HTO); organic compounds, e.g., decontamination solvents such as alcohol, must not be present in amounts greater than 1% of the water content of the package. This limitation ensures that excessive pressure buildup due to radiolysis will not occur over long storage periods.

The total pressure in the configuration-5 inner container several months after loading and sealing would (at normal conditions) be roughly the sum of three partial pressures. These partial pressures would be due to: 1) air or inert gases trapped in the container at the time of loading; 2) helium-3 decay product; and 3) hydrogen from radiolysis. At that time the pressure could be expected to be 15 to 20 psig.

During accident fire conditions this pressure would rise for two reasons. First, the heating of the gaseous contents would increase the absolute partial pressures. Second, the vapor pressure of the sorbed water would become significant at the elevated temperature. A configuration-5 container which is loaded and sealed and then subjected to Type B hypothetical accident fire conditions several months later is expected to have a maximum internal pressure of substantially less than 100 psig. The exact magnitude of this pressure depends upon several variables.

The configuration-5 inner containers are made of 316L stainless steel to provide maximum resistance to corrosion. However, they are not intended to contain corrosive agents, and the user must ensure that none are introduced. In particular, halogen compounds, e.g., decontamination and degreasing solvents, are to be avoided. These include, but are not necessarily limited to, halogenated hydrocarbons such as Freon propellants and solvents like trichloroethylene. Hence, halogenated compounds which can be introduced into water or decomposed should not be used in systems which will generate tritiated water to be loaded into these containers.

For loading (filling), the inner container is attached to the filling system with Cajon fittings. The inner container is loaded either by admitting tritiated water directly or by passing air, containing the HTO, through the dry sorbent. Direct loading with water is done slowly, because sorbency is exothermic. Figure 6 shows the pressure drop across the sorbent bed when loading is done with moist air at various flow rates.

After loading, the valves to the container are closed with a torque wrench. Before being disconnected from the filling system, the tubing and Cajon fittings above the valves are kept under vacuum until dry. Blind Cajon nuts (caps) are attached to each Cajon fitting to provide secondary containment. The surface of the container is then decontaminated, allowed to dry, and probed with a tritium monitor for leakage or surface offgassing. If tritium is observed, the

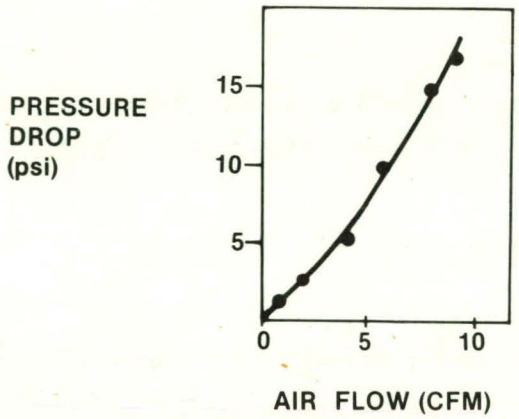


FIGURE 6 - Pressure drop vs flow for loading AL-M1 configuration-5 inner container by admitting moist air.

decontamination procedure is repeated. If no tritium is detected with the monitor, then surface wipes are taken. Further decontamination is performed as needed until these wipes are below 1000 dis/min beta. No leakage is permitted.

The protective cover is attached to the configuration-5 inner container to protect the valves and transducer from physical damage. Through the quick-disconnect fittings, the leakage from the cover and also from the container is determined by pressurizing the cover to 20 psig. No change in pressure should be observed after 30 min. The pressure is then released through a tritium monitor. If a tritium concentration reading greater than 180 $\mu Ci/m^3$ is obtained, the cap must be removed in order to look for a leak or to repeat the decontamination. The test is then repeated.

If the external surface of the capped configuration-5 inner container yields a surface wipe count of less than 1000 dis/min beta, the container is inserted into the insulated sleeve which is already in place in the outer 55-gal drum assembly.

For unloading after shipment, a configuration-5 inner container is removed from the insulating sleeve and checked for contamination. It is then assayed by calorimetry and stored in a hood until removal of the tritiated water is begun. The pressure is monitored if the container is not emptied within 30 days after filling.

REFERENCE

1. R. A. Watkins et al., "Safety Analysis Report for Packaging (SARP): USA/9507/BLF (ERDA-AL), Model AL-M1," MLM-2447, Mound Facility, Miamisburg, Ohio, September 30, 1977.

Dup

CARGO-RESTRAINT-TRANSPORTER (CRT)—A NOVEL APPROACH TO THE HANDLING OF CARGO ON DOE'S SAFE-SECURE-TRAILERS (SST)

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INTRODUCTION

Early in 1976 the Energy Research and Development Administration, now the Department of Energy, introduced to the Y-12 Plant and other installations a new transportation system for the shipment of special nuclear materials. This system consisted of a uniquely designed tractor-trailer combination that was dedicated to the safeguard transportation of special nuclear materials. The cargo space of the Safe-Secure Trailers (SST) is limited in size due to the design of the trailers. Experience has shown that the SST's have functioned well in their intended use as a safeguard transportation system.

OLD SHIPPING METHODS

Prior to the SST introduction, Y-12 shipped special nuclear materials primarily in rail cars couriered by AEC and ERDA couriers. Some shipments were made by standard tractor-trailer rigs. The securing of the cargo in the rail cars generally required the services of at least two carpenters and several material handlers. The cargo was physically secured by shoring the containers in place with lumber, generally consisting of 2 x 4's, 2 x 6's, and sheets of plywood. This operation was time consuming and costly. We often wondered just how many houses we helped to build at other installations since in most cases the shoring material was not returned to Y-12. This method of shipping materials is now considered to be a crude method, but crude or not the operation was successful and without incident.

SST LOADING AND TIE-DOWN TECHNIQUES

In regard to problems in securing the cargo in the rail cars, the SST's have also presented many problems in the loading and securing of containers in the limited cargo space of the trailers. Every time an SST backed up to our loading dock, a surprise awaited us when the doors were opened. Since Y-12 is a central shipping and receiving area within the DOE operation, we have seen a variety of tie-down techniques and cargo-loading configurations.

Figure 1 shows an SST loaded with shipping containers, and you can see by the amount of lumber that was used that many manhours were used in securing the cargo. Keep in mind also that since each container was handled individually, many trips in and out of the SST were made—and probably by mechanized equipment. The unloading operation by our people was time consuming and also required the services of a carpenter to remove the lumber.

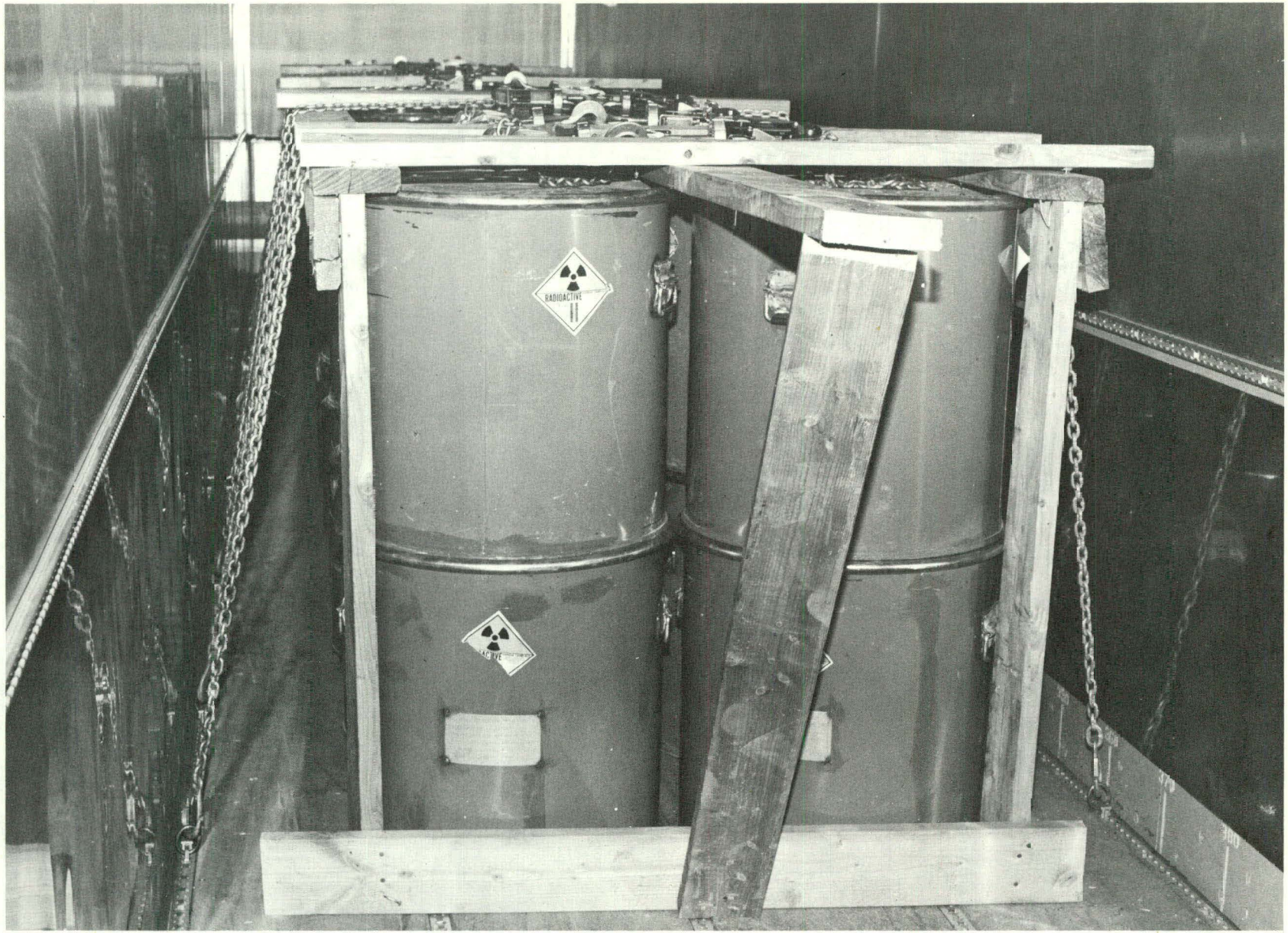


Figure 1. TYPICAL TIE-DOWN METHOD.

The type of tie-down equipment that is furnished on the SST's has also created many problems. Some early model trailers were furnished with typical log chain ratchet assemblies, locking bars to span across drums and a few nylon strap assemblies. Later models have only the chain assemblies. Stacking of containers was accomplished through the use of stacking adapters for small, drum-type containers. Stacking adapters for 55-gallon drum-type containers were not provided. Some shipments of 55-gallon drums were received and these containers were stacked in two layers; but, since the containers are not designed to nest into each other, this loading technique was deemed to be an unsafe method.

INITIAL DESIGN CONCEPTION

After receiving several shipments that caused our warehousing people considerable trouble in unloading, we started a research program into the design of a cargo-handling system that would eliminate some of the problems generally encountered, but would also standardize the handling equipment and reduce the time involved in the actual loading and unloading operations. To obtain additional information and ideas in handling of cargo in limited spaces, I visited Delta Airlines in Atlanta, Georgia and observed their handling of cargo on the 747's. Delta uses cargo packs that are preloaded in a warehouse and then loaded into the cargo compartment of the 747's. This operation reduces the time that the aircraft is sitting idle at the loading dock. Ted Burger, my co-author, and I incorporated some of these techniques in our cargo packs that we have designed at Y-12 for use on the SST's. Our cargo pack is named the Cargo-Restraint-Transporter (CRT). Since we primarily ship products out of Y-12 in drum-type containers, our CRT is designed basically to handle 30-gallon and 55-gallon drum-type containers.

BASIC DESIGN CRITERIA

DOE tie-down requirements for shipments of Special Nuclear Material in the SST are as follows:

1. Containers (including contents) weighing over one ton must be tied down in accordance with the criteria set forth in RDT Standard F8-11T as modified by consideration of the SST structure and equipment. This standard requires that the static stresses in the tie-down equipment will not exceed 75% of the yield strength when the carrier is subjected to sustained accelerations of 2g's forward or backward, 1g sidewise, 3g's vertically up, or 2g's vertically down.

2. Containers (including contents) weighing less than one ton must be secured against movement under conditions normally incident to transportation and in a manner which would be accepted as good industrial practice.

Since our primary shipping item is contained in a 30-gallon drum weighing 120 pounds, making the total weight of a loaded CRT 1650 pounds, we only have to meet the good industrial practice requirement. The CRT tie-down system will also meet the g force loadings as stated in RDT Standard F8-11T for gross weight up to one ton.

CRT DESCRIPTION

Briefly let me describe the basic components of the CRT's. The CRT as shown in Figure 2 is a multi-container double-tiered cargo transport system that consolidates cargo items of various configurations into a single unit. Each CRT can handle eight 55-gallon containers or ten 30-gallon containers or a combination of either size. The system consists of three major components—base, center, and top sections. Casters on the base section provide mobility in the warehouse as well as eliminating the need for using fork trucks in the SST loading operation. Unique outrigger wheel assemblies on the sides of the base section guide the CRT inside the SST cargo compartment preventing damage from occurring to the fragile sidewalls of the trailer. A collapsible wire mesh cage, as shown in Figure 3, that can be positioned in the upper tier in place of the drums provides the capability of shipping an assortment of packages as cargo of opportunity. An additional feature of the CRT is the fact that it can be shipped in its empty state and all of the necessary tie-down gear can be stowed inside during transport by either SST's or commercial transport trucks.

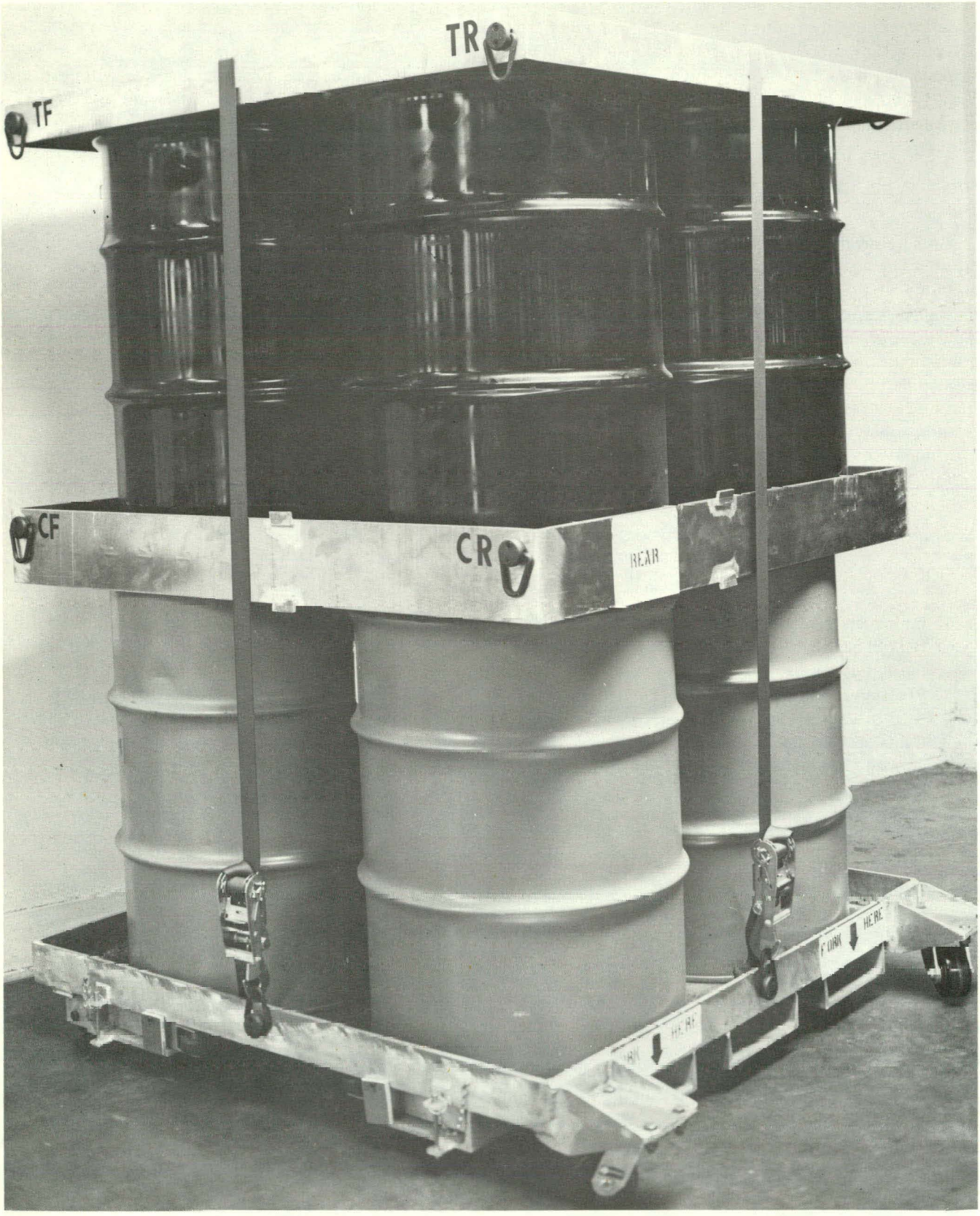


Figure 2. CRT LOADED WITH SHIPPING CONTAINERS ON BOTH TIERS.

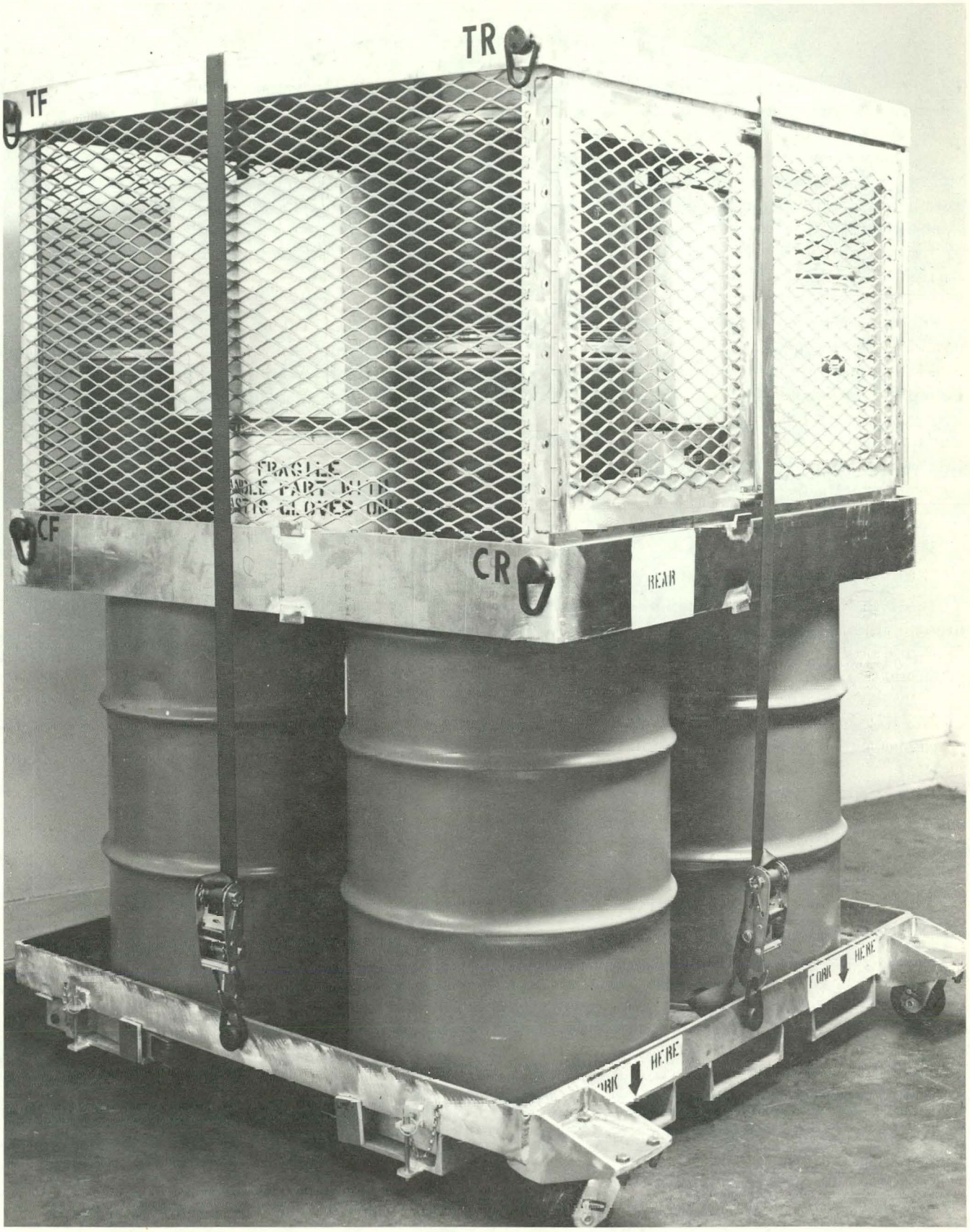


Figure 3. CARGO OF OPPORTUNITY CAGE IN PLACE ON THE UPPER TIER OF THE CRT.

CRT LOADING OPERATION

A jib crane is used to load the drums onto the CRT. Many of the drums can be loaded manually, but this equipment allows us more flexibility and safety. The base section is positioned under the jib crane and the drums are placed on it. The center section is then placed on top of the drums. This section is also the base section for the top tier of containers. After placing the shipping containers on the center section, the top of the CRT is then put into position and two nylon straps are attached to the base section and firmly secured. In preparation for the actual loading of the CRT in the SST, two strap assemblies are attached to predesignated points. The prepositioning of these two straps will eliminate the need to get to the front of the CRT when it is loaded in the trailer. The clearance between the CRT and the sidewalls makes it difficult to get to the front of the CRT. Thus, all of our preplanning is done so that all ratchet takeup assemblies will be accessible from the rear of the CRT.

Prior to moving the CRT into the trailer, some preparations must be made inside the trailer. Four strap assemblies are positioned as indicated, and a floor channel lock is secured in the predetermined position in the center track. The outrigger wheel assemblies on the sides of the CRT are fully extended, and the CRT is moved into the SST cargo area so that the front of the CRT engages the prepositioned floor channel lock. The strap assemblies are then attached to the floor lugs that were positioned in both of the outer tracks, and the ratchet tightening mechanisms are used to secure each of the straps to a uniform tension. A floor channel lock is positioned at the rear of the base section and secured to the center track. The CRT is now secured in place as shown in Figure 4. Additional CRT's can be loaded as shown in Figure 5 by a repeat of the same loading technique as the first cargo pack.

SUMMARY

This system we believe is quite unique and will require training of loading personnel. We have already developed an instructional manual on the loading techniques required to load the CRT's in the SST's. Copies of this manual have already been distributed on a limited basis and are available through proper channels to interested parties. The title of this publication is ***Cargo-Restraint-Transporters (CRT) Handling Instruction***, Document Y/OA-1999.

The cost of these cargo packs was approximately $1500 per unit. High? No, we don't really believe that this cost is unreasonable. Let's review just a few of the major assets that this system will provide:

1. Standardization of loading and tie-down techniques.
2. Cost savings in more efficient use of warehousing employees.
3. Cost savings to the transportation by reducing the time of loading operation.
4. Improved quality assurance of containers and contents.
5. Reduction of cost for returning empty containers to shipper.

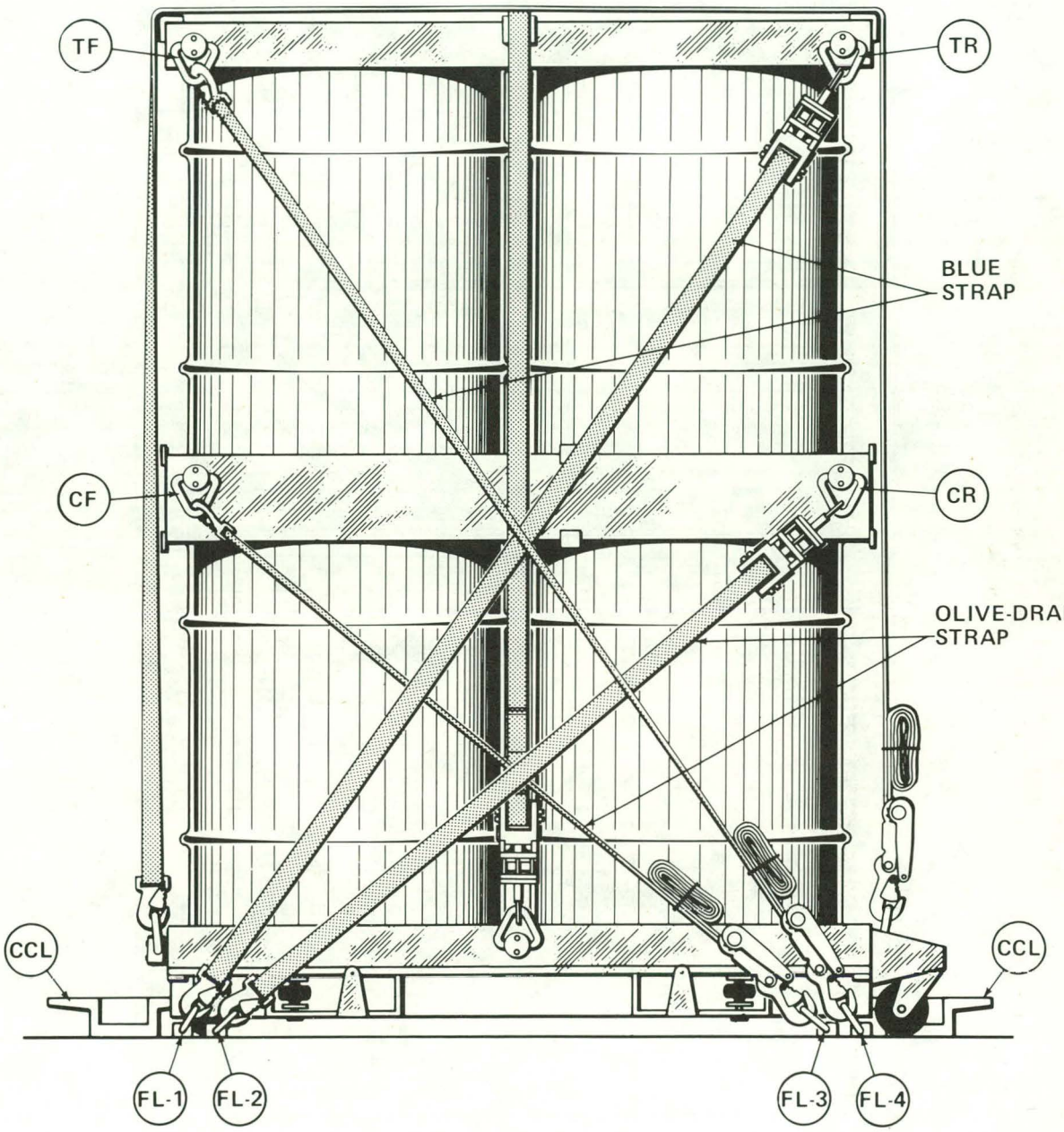


Figure 4. CRT SECURED IN TRAILER.

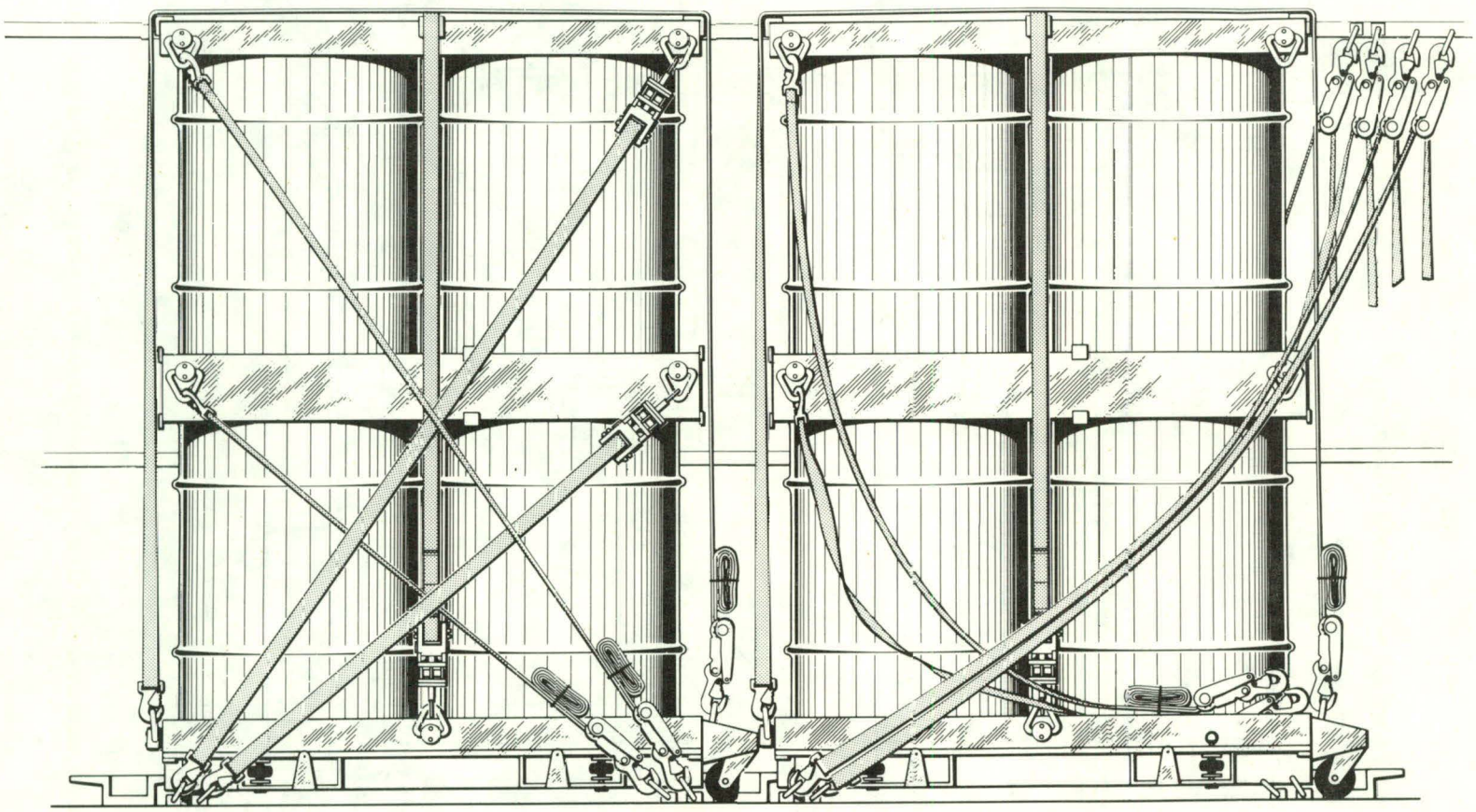


Figure 5. TWO CRT's POSITIONED IN TRAILER. CRT ON THE LEFT SECURED; ONE ON RIGHT PRIOR TO SECURING.

INVESTIGATION OF BLANKET-TYPE INSULATION AS A REPLACEMENT FOR VERMICULITE IN THE DOT-6L SPECIFICATION CONTAINER[a]

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SUMMARY

As required by ERDA directive IAD 0529-2, the currently-used DOT-6L specification container, which is filled with loose vermiculite as an insulating medium, must be modified, replaced, or provided with an insulating material that would not be lost in an accident that punctured the drum wall. Because modification or replacement of the 6L container would be costly, a substitute insulating material that is not free-flowing was sought. Based on availability, cost factors, and ease of installation, two blanket-type insulating materials, Kaowool and Fiberglas, were selected for thermal tests. The insulating qualities of these materials were compared with those of vermiculite using furnace tests that simulated the temperature environment of a 1475°F (802°C) fire (as prescribed by 10 CRF Part 71, Appendix B).

Neither Kaowool nor Fiberglas insulated the contents of the 6L container as well as vermiculite. The centerline temperature was 23° higher with Kaowool and 173° higher with Fiberglas than with vermiculite. The tests also indicated that none of the insulating materials, including vermiculite, could maintain the centerline temperature inside the container below 300°F (149°C). (Current regulations allow solid materials that decompose at or above 300°F to be packaged in the container.) The heat input into the inner 2R pipe was attributable to the low thermal resistance of the steel rod supports and braces that are welded to the 2R pipe and in contact with the drum wall. In order to maintain a centerline temperature below 300°F, the 6L container design would have to be changed to isolate the inner 2R pipe from the rod spacers and braces by an insulating material.

Considering the cost of converting the large inventory of 6L containers in use and the inability of any of the tested insulating materials to keep the centerline temperature below 300°F, consideration should be given to changing the regulations so that only solid materials that do not decompose below 750°F (399°C) could be packaged in the 6L container. No container design change would be necessary and blanket-type insulation could be used to replace the loosely-packed vermiculite.

(a) Work supported by the U.S. Department of Energy, formerly the Energy Research and Development Administration, under contract EY-76-C-06-1830.
(b) Operated for the U.S. Department of Energy by Battelle Memorial Institute.

INTRODUCTION

Risk analysis studies[1] performed on radioactive material containers that use loose vermiculite packaging as a thermal shield have indicated that losses of the vermiculite, due to puncture or crushing accidents followed by a fire, could increase the probability of a release of radioactive material. Such a release could result from overpressurization of the internal container vessel due to excessive heat. Based on these studies, a directive (ERDA IAD 0529-2) was issued instructing shippers using containers with loose vermiculite packing to discontinue use of these containers as soon as practicable where a suitable alternative is available. Alternatives might include conversion from a Specification 6L to a Specification 6M container, or replacement of the loose vermiculite with a nonflowing insulation.

The 6L container (Figure 1) is a steel drum with an inner 2R pipe centered by eight steel rod spacers or three steel spiders (not shown) and with loose vermiculite poured around the inner pipe for thermal insulation. The DOT-6M container is a similar type of container (Figure 1), but its inner pipe is held in place by solid celotex rings that serve as the insulating barrier. The two containers provide almost identical packaging capacities.

The advantages of using one container rather than the other are currently not clear.

- It has been documented[2] that the 6M container can withstand high impact forces without loss of the material packaged inside it. However, because the 6L container has not been tested under similar conditions, no comparisons can be drawn.

- The 6L container does offer some protection and flexibility that the 6M container currently lacks: the rod spacers that support the inner pipe in the 6L container act somewhat like shock absorbers and help protect the inner packaging from vibration damage, while the pipe inside the 6M container is free to bounce inside the celotex cavity and in some cases has caused damage to the inner packaging.[3] However, the use of appropriate packaging methods (rubber dampers, etc.) could provide protection from damage to the contents of the 6M container.

- The 6L container can be used to ship fissile material (Pu, ^{235}U) containing hydrogenous material with hydrogen/fissile ratios up to 20. At present, the 6M container cannot be loaded with fissile material if the hydrogen/fissile ratio exceeds 3. Here again, the difference between the 6L and 6M containers could be eliminated by performing the appropriate criticality analysis and petitioning the Department of Transportation and the Nuclear Regulatory Commission for approval to use the 6M container at a higher H/X ratio.

These points indicate that the 6L container could be replaced by the 6M container with no sacrifice of performance or flexibility. However, the problem of what to do with the existing



FIGURE 1. 55-Gallon DOT-6M Container (left) and 65-Gallon DOT-6L Container

6L containers remains. Like most companies or laboratories that ship any quantity of fissile material, the Pacific Northwest Laboratory has a fairly large inventory of 6L containers. Because conversion to 6M containers would be costly, we decided to look into insulating materials that are not free-flowing and might be an adequate substitute for vermiculite.

SELECTION OF INSULATING MATERIAL FOR TESTING

For convenience of handling and application, vermiculite is hard to surpass. This material is also lightweight, inexpensive, and thermally stable in the temperature range of concern. The use of vermiculite allows the inner pipe to be removed from the barrel so that if the drum is punctured or torn, the inside container can be salvaged and inserted into a new drum, and if the inner pipe needs repair or modifications, it can easily be removed.

A substitute insulating material should possess the same qualities; therefore, the criteria used to select potential substitutes were: availability; ease of application; cost of using the material; thermal properties; and density. We also looked for an insulating material and installation method that would allow the inner pipe to be removed from the drum.

The option of removing the inner pipe from the drum precludes the use of insulations that are poured or formed around the pipe. Consequently, foam glass or vermiculite that is mixed with an elastomer and poured around the pipe inside the drum were ruled out because they would cement the pipe in place. Table 1 shows the various insulating materials that were considered and some of their properties. A rigid type of insulation such as celotex requires considerable cutting and fitting to properly insulate the pipe. The cutting, glueing, and assembly of such an insulation would normally have to be done by either an in-plant shop or a commercial fabricator, and the fabrication costs associated with insulating the 2R pipe using rigid insulation are quite high. On the other hand, a blanket type of insulation does not require any special fabrication or installation techniques. Insulating the inner pipe with these kinds of material is a do-it-yourself project and the installation costs are small.

Because of their availability, ease of application, thermal properties, and low densities, and the low cost of using the materials, blanket-type insulating materials were chosen as likely candidates to replace vermiculite. The blanket insulating materials shown in Table 1 were then tested to determine whether they would stand up under a 1475°F temperature for 30 min and still properly insulate the inner 2R pipe.

THERMAL TEST WITH INSULATING MATERIALS

To establish some basis for comparison between vermiculite and the blanket type of insulation, a thermal test with vermiculite as the insulating medium was in order. The thermal test with vermiculite would also determine whether the centerline temperature inside the 6L container was being maintained at 300°F or below. This was of interest for the following reason. Prior to December 1974, the 6L container had only 4 rod support and 4 braces if the volume exceeded 55 gal (208 ℓ). In December 1974, the regulations were changed (FR 45246, 12/31/74) to require 8 rod supports and 8 braces for a volume over 55 gal. The authorized loading requirements were also changed. Prior to June 1975, only solid materials that do not decompose below 750°F could be loaded in the 6L container. After June 1975, solid materials that do not decompose below 300°F could be loaded into the container. Thus the temperature requirements for the material in the container were being lowered while the heat transfer into the modified container was increasing because of the additional rod supports and braces.

The container selected for the thermal test was a 65-gal (246-ℓ) 6L container. Thermocouples were attached at the bottom, center, and top of the inner pipe, on the center of one of the steel braces, and on the outside of the drum (Figure 2). The centerline temperature was determined by inserting a thermocouple into a metal can filled with MgO. The can was positioned approximately halfway between the top and bottom inside the pipe by using other metal cans for spacers. A typical arrangement of the metal cans alongside the pipe is shown in Figure 3.

TABLE 1. Properties, Commercially Available Forms, and Cost Factor of Using Insulating Materials

Material	Thermal Conductivity, BTU/hr-ft-°F	Thermal Stability In Air	Density, lb/ft^3	Composition	Available Forms	Cost (1977), \$/ft^3	Approximate Fabrication Costs/Drum
Celotex	0.031 (100°F)	Mechanically degrades above 280°F; bursts into flames at 425°F in air	15	Sugar cane fibers bonded with organic glue	1/2-in. to 3-in.-thick solid board	3.50	\$800 to \$1000
Fire dike	0.031 (100°F)	Burns at 1400°F in air; self-extinguishing	20-22	Mineral wool and slag closely bonded	1/2-in., 5/8-in., 3/4-in.-thick solid board	5.58	~\$1000
Thermo 12	0.028 (100°F) 0.063 (1000°F)	Service temperature 1200°F; incombustible	11	Lime and diatomaceous silica	1/2-in., 1-in., 2-in.-thick solid board	10.89 (2-in.)	\$1000 to \$1200
Ceraform	0.028 (100°F) 0.067 (1000°F)	Stable up to 2300°F; incombustible	11	Alumina silica refractory bonded by organic binders		36.00 (1-in.)	\$1000
Kaowool	0.010 (100°F) 0.071 (1000°F)	Stable up to 2300°F; incombustible	6	Alumina silica	1/4-in., 1/2-in., 1-in., 1-1/2-in., 2-in.-thick blanket	23.00 (1-in.)	None
Fiberglass	0.024 (100°F)	Service up to 435°F depending on binders	0.6	Glass fibers and organic binders	1-in. to 6-in.-thick 1-1/2-in. to 3-ft-wide blanket	1.20 (1-in.)	None
Vermiculite	0.04 (100°F)	Incombustible	7	Hydrous silicate [illegible]	Bagged, granular	1.25	None

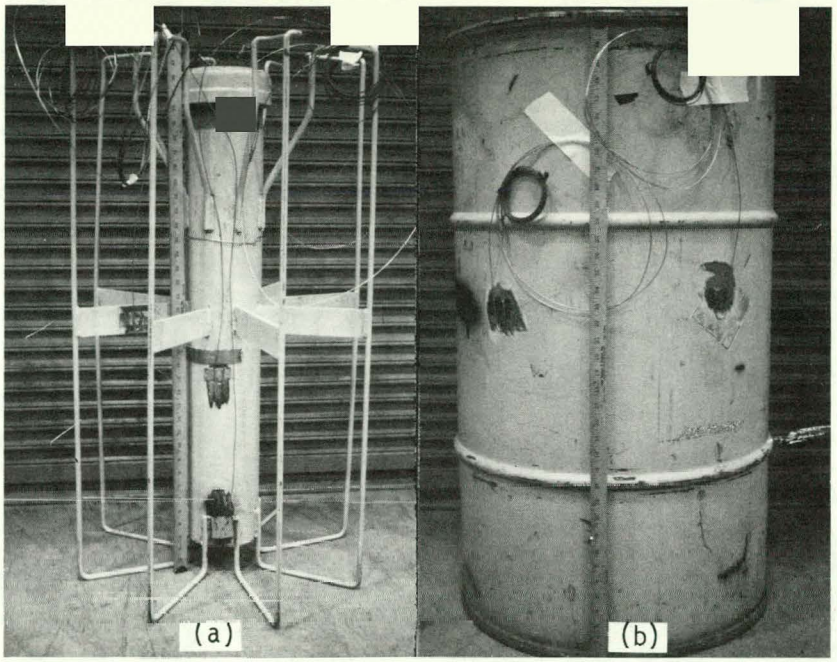


FIGURE 2. Thermocouple Placement on the DOT-6L Container. (a) On inner pipe assembly. (b) On drum.

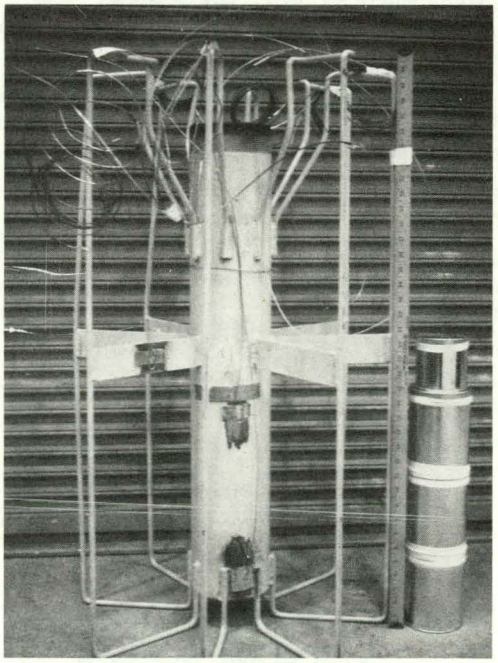


FIGURE 3. Arrangement of Metal Cans That Were Placed Inside Inner Pipe (2R). Topmost can filled with MgO.

For the thermal test with vermiculite, the inner pipe assembly was placed inside the drum so that the rod supports were in contact with the bottom of the drum. The rod spacers also touched at certain points along the sidewall of the drum. Vermiculite was then poured into the drum to fill all the empty spaces, and tamped and vibrated to ensure maximum packing density. The drum was then placed in a preheated oven (Figure 4) and subjected to a temperature environment as prescribed in 10 CFR Part 71, Appendix B. It was left in the furnace at temperature for 30 min after the furnace recovered. The temperature was recorded using a multipoint recorder. After 30 min, the drum was removed from the furnace and the temperature was monitored as the drum cooled in air for at least 3 hr or until the centerline temperature began to decrease.

The results of the thermal test with vermiculite are shown in Figure 5. While the drum was in the furnace, the centerline temperature reached about 200°F (93°C). However, after the drum was removed from the furnace the temperature continued to increase for about 90 min, achieving a maximum temperature of 307°F (153°C). Although this temperature is not much above 300°F, it could cause problems if the material packaged in the 6L container decomposed at 300°F during a fire-type accident. Plutonium (IV) oxalate and plutonium (IV) nitrate in solid form decompose at 302°F (150°C), and the regulations allow these compounds to be packaged in the 6L container. These tests, however, show that they would probably decompose in a fire-type accident even though the 6L container's insulation was intact.

The two blanket-type materials tested for insulating quality were Kaowool and Fiberglas. The Kaowool was wrapped around the inner pipe, using widths varying from 2 in. to 17 in. (5 cm to 43 cm) and a thickness of 1 in. (2.5 cm) (Figure 6). Wrapping the Kaowool proved to be quite tedious. The material is somewhat like Fiberglas: it has a tendency to irritate the skin after prolonged contact. The material also tears quite easily so that the wrapping has to be done with care. Typical installation time was about 8 hr.

The results of the furnace test are shown in Figure 7. Overall, the Kaowool did not insulate as well as the vermiculite, resulting in a maximum centerline temperature 23° higher than in the vermiculite test. The initial resistance to the heat input was higher with Kaowool then with vermiculite, but in time the resistance became less. The curve shows that higher heat input was through the stiffeners and through the rod support on the bottom of the pipe. This higher input can be attributed to the Kaowool having a higher conductivity than the loose-fill vermiculite, and to the vermiculite's ability to fill in the small gaps between the rod supports and the drum sidewall. The Kaowool could not be stuffed in the small gaps around the rod supports; because of the tight fit, the Kaowool shown wrapped on the rod supports in Figure 6 had to be removed so that the pipe assembly could be inserted into the drum. There was no evidence of deterioration or degradation of the Kaowool during the furnace test; the material appeared the same after the test as it had before.

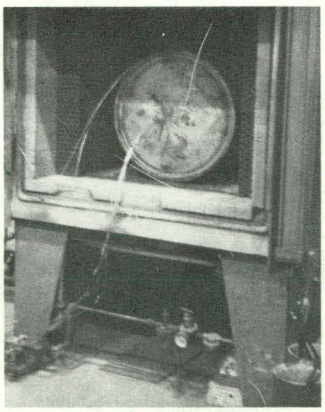


FIGURE 4. DOT-6L Container Inside Electric Furnace Prior to Furnace Heatup

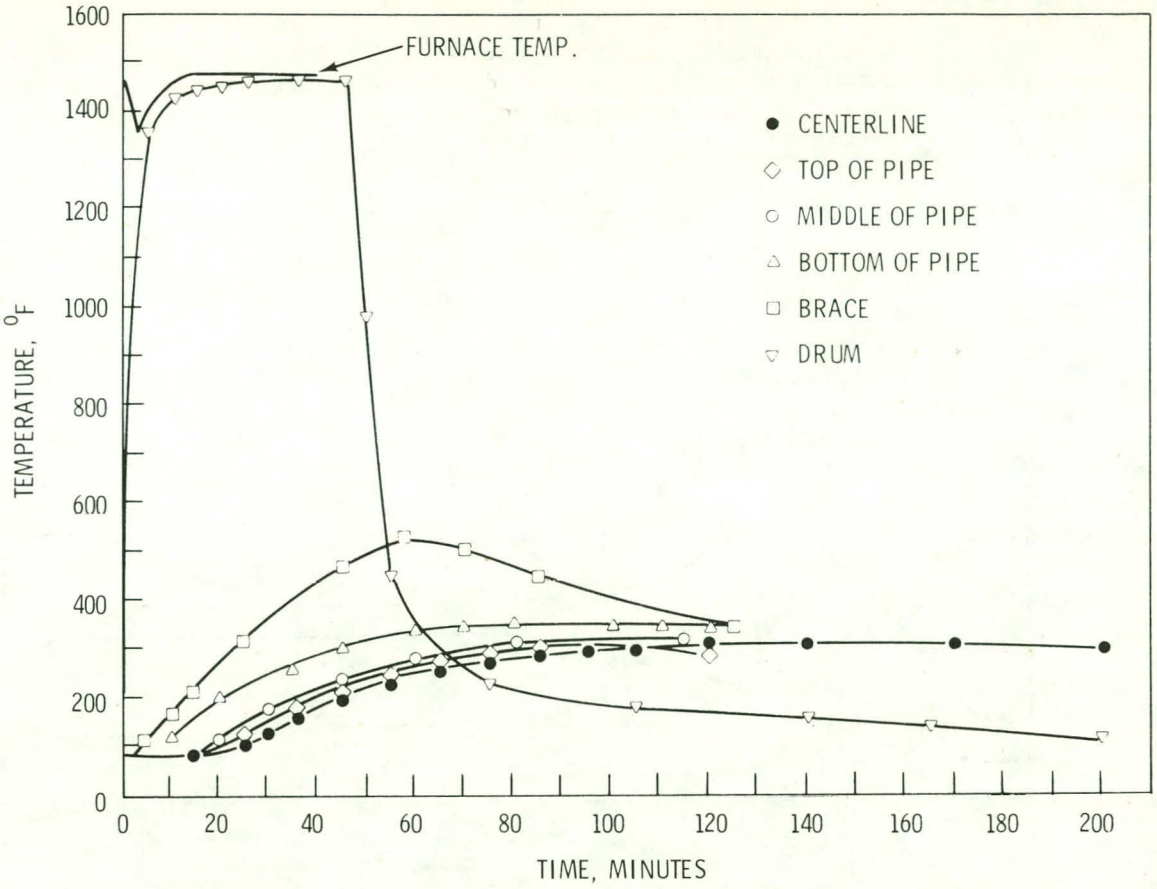


FIGURE 5. Temperature-Time Data for Vermiculite Furnace Test

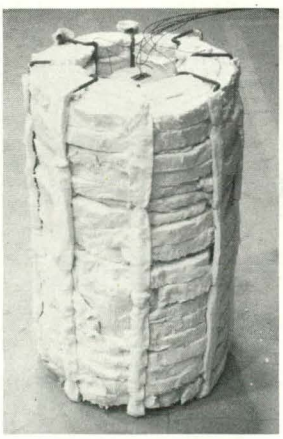


FIGURE 6. Kaowool Insulation Wrapped Around Inner Pipe Assembly

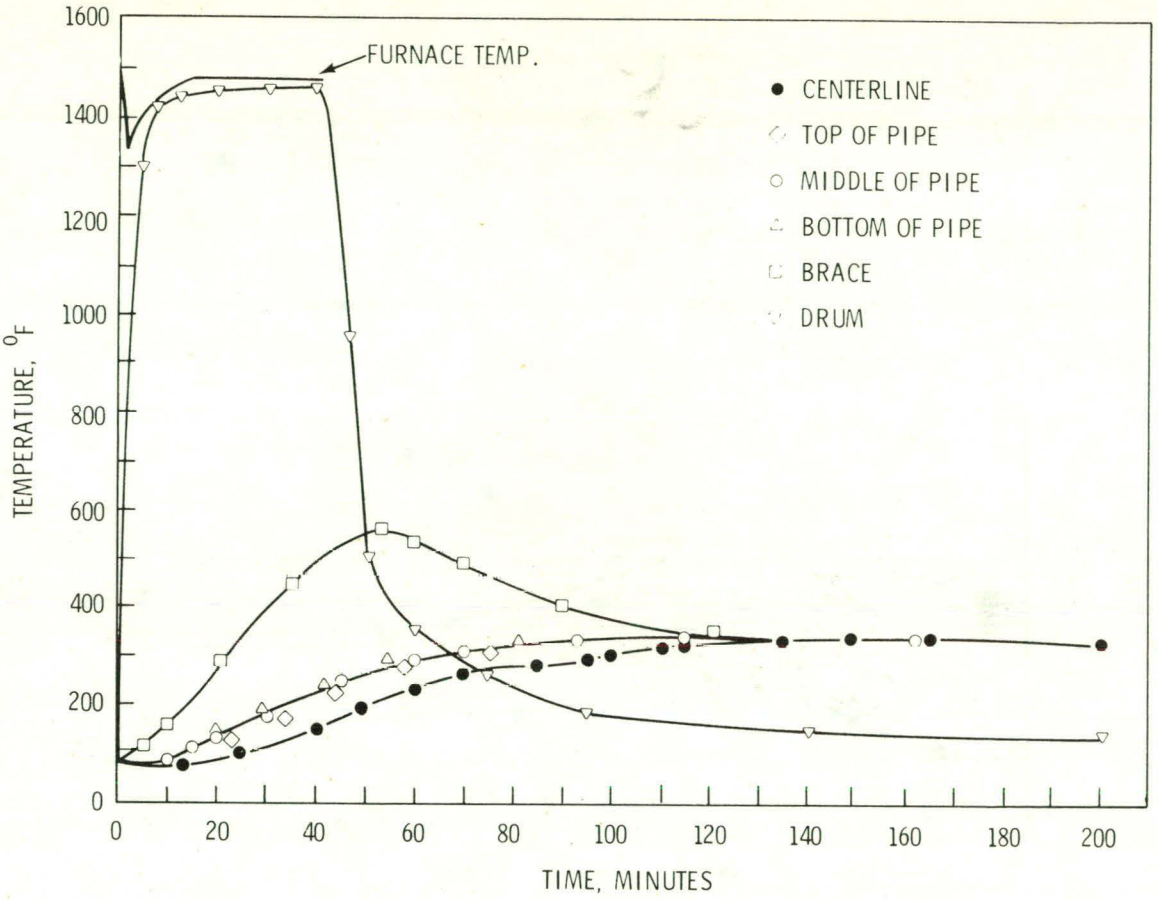


FIGURE 7. Temperature-Time Data for Kaowool Furnace Test

The use of Fiberglas as an insulating medium might be questioned, since Fiberglas is known to burn if heated to high enough temperatures (because of combustion of the binders that are used in manufacturing the Fiberglas blanket). However, guessing that only the Fiberglas next to the drum wall would smolder inside a closed drum, we decided to perform a test with this insulation. The Fiberglas was stuffed into the container as shown in Figure 8, using square blocks 12 in. by 12 in. by 1 in. (30.5 x 30.5 x 2.5 cm) cut from conventional 3-ft-wide by 1-in.-thick (0.9-m-wide by 2.5-cm-thick) Fiberglas blanket insulation. The fluffiness or low density of the material meant that a considerable amount (~21 ft^3 or 0.6 m^3) was packed into the container. Because the material compressed to a small volume, packing the container took longer than expected, or approximately 3 hr. Using a denser material, such as Kaowool, to stuff the container in this fashion would take less time.

The insulating ability of the Fiberglas was inferior to that of both the vermiculite and the Kaowool, as shown by Figure 9. The heat input to the center of the container was higher at all stages of the furnace test than during the vermiculite or Kaowool tests. We had expected the heat input at the initial stages of the test to be similar to that for vermiculite, since the thermal conductivities of these materials are nearly identical. A possible explanation for the variation seen could be that the Fiberglas starts to offgas and change conductivity as soon as the container is exposed to a hot environment. Certainly this is the case at the later stages of the furnace test. When the furnace was opened to extract the container, flames were coming from around the lid and thermocouple port. The burning continued for about 10 min after the container was removed from the furnace. Inspection of the Fiberglas after the drum had cooled showed that the Fiberglas near the top was considerably charred (Figure 10) and had lost its fluffy character. The thickness of the material had decreased from about 1 in. to about 3/8 in. (2.5 cm to 0.95 cm). Thus the insulation had lost most of its air-holding capacity and the thermal conductivity increased. The maximum centerline temperature was some 173° higher with the Fiberglas than with the vermiculite fill.

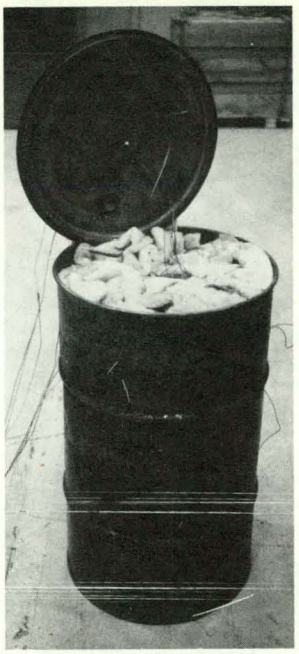


FIGURE 8. Fiberglas Insulation Stuffed Into a DOT-6L Container Prior to Furnace Test

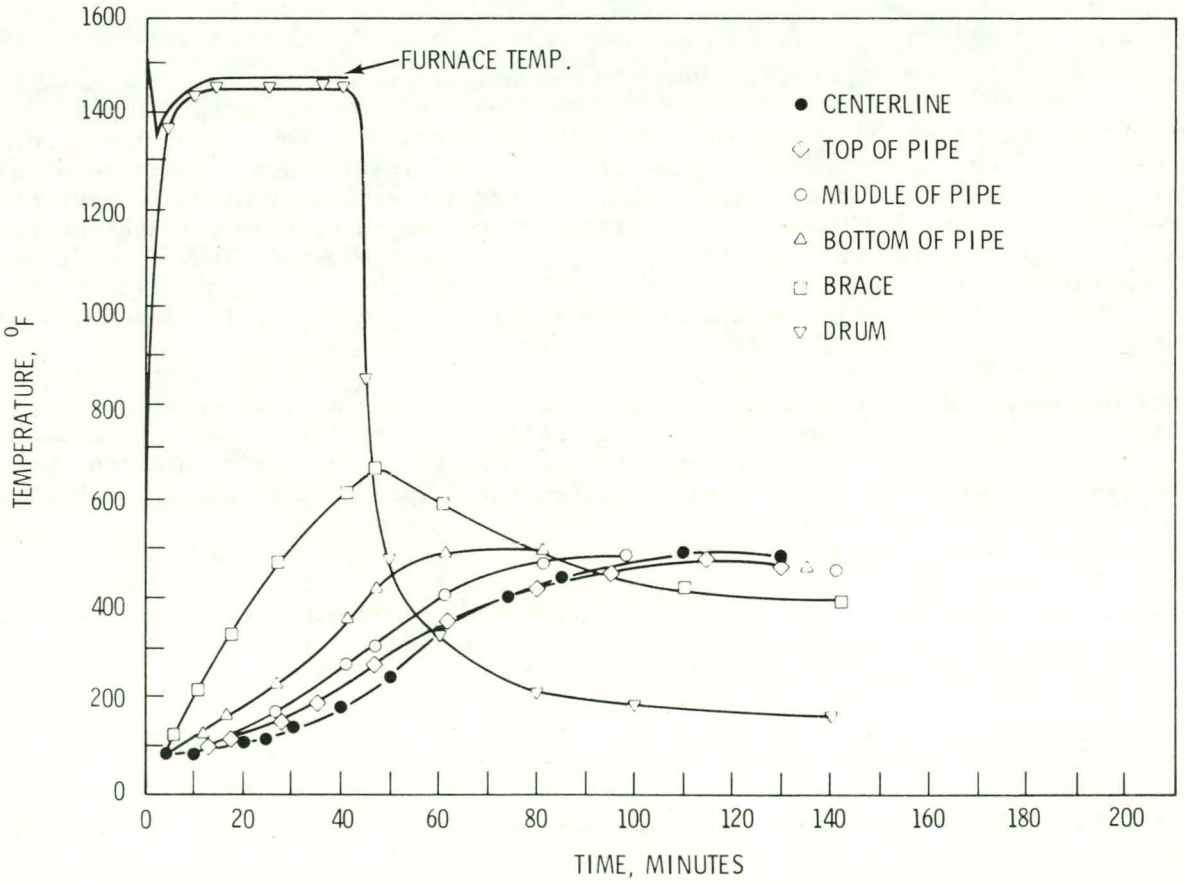


FIGURE 9. Temperature-Time Data for Fiberglas Furnace Test



FIGURE 10. Charred Fiberglas Insulation After Furnace Test

In summary, neither of the blanket-type materials insulated the inner packaging as well as vermiculite.

OPTION AND RECOMMENDATIONS

In light of the thermal test performed with the 6L container, several options can be considered. Use of the container under its present authorized loading constraints requires that the container be redesigned. None of the insulating media tested prevented the centerline temperature from exceeding 300°F because of heat input to the inner pipe through the rod supports, which contact the drum wall and the inner pipe. A possible modification of the container to insulate the inner pipe would be to attach the rod supports to a collar that could be bolted to the inner pipe, allowing insertion of an insulating material between the collars and the inner pipe to prevent high heat flow.

Another option would be to change the regulations on decomposition temperature back to the limitations prescribed prior to 1975, that is, allow only materials that would decompose at temperatures above 750°F to be loaded into the container. With this regulation change, the blanket-type insulating materials could be substituted for the vermiculite without a change in the container design.

A regulation change that would allow only the dioxides of uranium and plutonium to be loaded into the container might also be considered. This change would eliminate the need for any insulation, since the dioxides of uranium and plutonium are stable above 3632°F (2000°C). However, this option would restrict the flexibility of the container.

Another option might be to replace the 6L container with the 6M container. Through the appropriate criticality analysis, the 6M container could be approved for the same kinds of moderated fissile material loadings for which the 6L container is now authorized. However, this option would eliminate the use of all existing 6L containers.

Given the sizable inventories of 6L containers at different sites around the country, the most desirable option seems to be a change in the regulations to allow loading of only solid materials that do not decompose below 750°F. If this change is effected, the 6L container can still be used with a blanket-type insulation that is inexpensive and easy to apply.

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2. L. L. Bonzon, Final Report on Special Impact Tests of Plutonium Shipping Containers: Description of Test Results. SAND 76-0437, Sandia Laboratory, Albuquerque, NM, February 1977.

3. J. M. Tayler, "Impact Analysis of Metal Cans Damaged During a Radioactive Materials Shipment." Proceedings of the Fourth International Symposium on Packaging and Transportation of Radioactive Materials, September 22-27, 1974, Miami Beach, FL, Conf-740901-P1, September 1974.

THE DESIGN, LICENSING AND FABRICATION OF A 64,000 POUND, TOP AND BOTTOM LOADING, "TYPE B" SHIELDED OVERPACK

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ABSTRACT

The NUPAC OH-142 shielded overpack has been designed to provide a flexible package capable of transporting 142 cubic feet ($4m^3$) of medium-to-high level wastes. It was primarily designed for the transport of spent ion exchange resins from Canadian reactors to the Quadricell retrievable storage facility. During the design phase it was determined that with minor modifications its function could be expanded from a pure top loading transportation shield to one of a top and bottom loading transfer cask. As such it is used to transport liners of ion exchange resins to the Quadricell where, through the use of its bottom unloading feature, the shield can be placed atop the Quadricell, thus allowing a liner to be lowered directly into the storage cell. The resultant transfer provides minimum operator exposure.

The OH-142 consists of a steel and lead shield providing approximately four inches (10 cm) of lead shielding. Both the top and bottom are removable. Two crushable foam impact limiters are provided to reduce the effects of the hypothetical accident condition. In addition, their insulating ability provides thermal protection to the critical sealing areas. Temperatures along the body areas are maintained within limits thru the use of a thin layer of intumescent mastic. At elevated temperatures the mastic begins to expand and off-gas, forming a sacrificial char. The resultant formation provides ideal fire protection.

Because of the size and cost of the package, it was impractical to consider testing to demonstrate compliance with the IAEA or NRC regulations. Therefore, a detailed analytical approach was taken. Foam impact limiters were analyzed using a specially developed computer program. The program takes into consideration the stiffening or strain hardening characteristics that the foam exhibits during large deformations. The effects of the puncture condition were evaluated using a finite element routine. This program takes into consideration not only local shearing, but also large plate deformations into the plastic range. Temperature profiles were established through the use of a three dimensional thermal program. Phase changes of the foam and lead are accounted for, as well as the ablative characteristics of the intumescent materials.

Licensing time for the package was approximately one year. During that time a number of changes were made to the design to make the package simpler to analyze. Many of these changes did not improve the design, but did reduce the amount of time required for regulatory review.

Once the license had been received, fabrication of the steel shells began. The central shell was rolled, welded, and lead filled. Impact limiters were foamed-in-place and fitted to the package. The intumescent insulation was applied to the body. All of the above steps were carefully controlled by a detailed Q.A. Program. Upon completion of the package, it was fitted to its special trailer and transported to Ontario for its first usage.

To date, two packages are in service. One at Ontario Hydro's facility and the other at an ERDA lab in Richland, Washington.

INTRODUCTION

The NUPAC OH-142 package was created because of the need to ship large disposable containers of depleted ion exchange resins. Specifically, it was designed to transport 100 cubic foot liners of resins from Canada's Bruce Generating Station to the above ground Quadricell storage complex. Activity levels of the resins dictated that the package be able to accommodate Type B and Large Source Quantities.

The original design called for a trailer-mounted, top loading cask providing 142 cubic feet of capacity and four inches of lead shielding. A 100 cubic foot disposable liner placed within the cask is used to collect and contain the resins. Once the liner was filled and transported to the Quadricell area, it was anticipated that the lid would be removed, and a special helmet or transfer cask would be placed over the liner. By means of a remote grapple, the liners would be raised into the transfer cask. This assembly would in turn be positioned over the Quadricell. The liner would then be lowered into the cell and disconnected.

In reviewing the overall system it was determined that the function of the transfer cask could be combined into the basic transportation shield, if it could be both top and bottom loading. The cask was then redesigned and licensed as a Type "B" cask to provide a closure system on each end. In this manner the liner would be top loaded, filled through a secondary lid, and then bottom unloaded.

OH142 "TYPE B" SHIELDED OVERPACK
MK 2

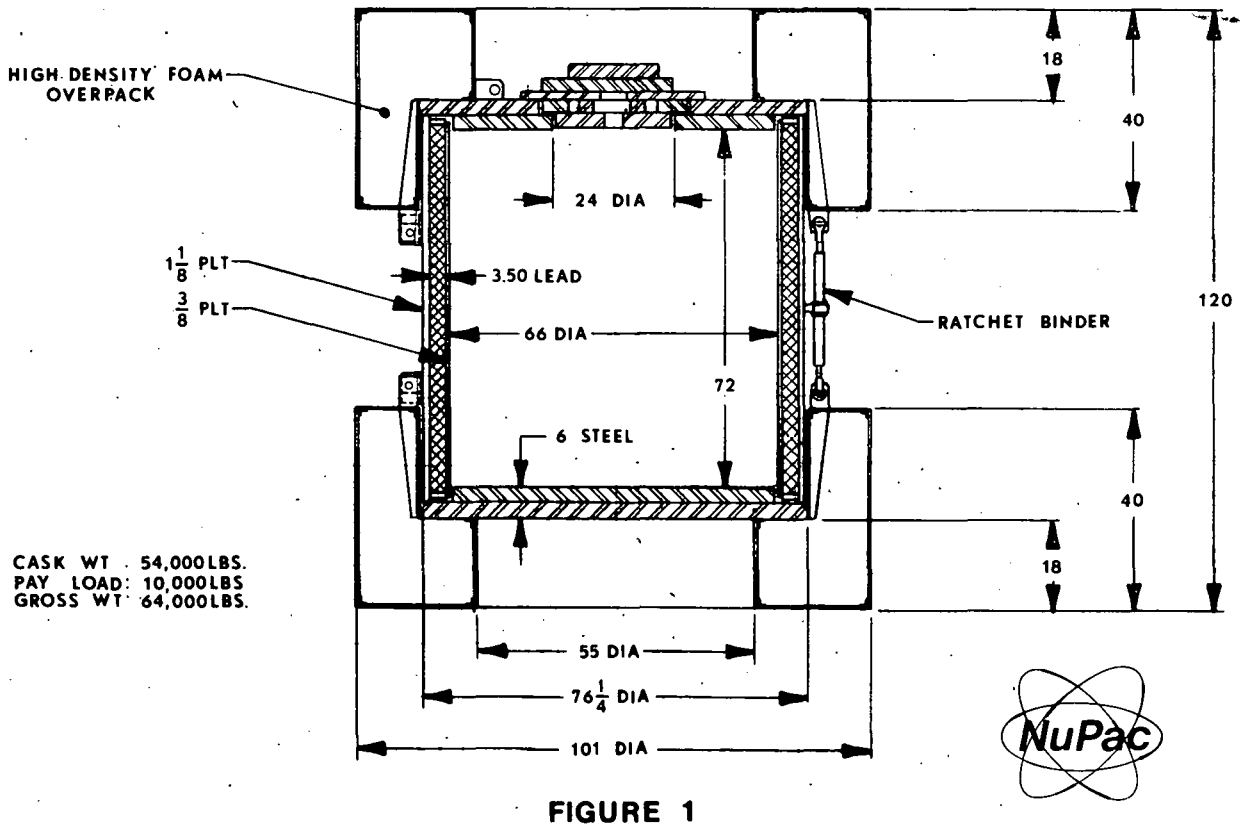


FIGURE 1

From Figure 1 it can be seen that the shield has an inside cavity of 66 inches in diameter by 72 inches high. Shielding is provided by 3 1/2 inches of lead sandwiched between 1 1/8 and 3/8 inch steel plates. The top and bottom are constructed from two 3 inch thick solid steel plates. The total weight of the shield and overpack is 54,000 pounds, which, when combined with the 10,000 pound payload, produces a 64,000 pound gross package weight. Each end of the package is protected by means of a crushable foam impact limiter. The impact limiter and lid are secured by means of eight high strength ratchet binders. These binders provide an easy method of securing the lid without the problems associated with lost bolts and nuts or stripped threads. More importantly the time required to open the OH-142 is a fraction of that needed for a bolt-on design. This time savings is directly related to Reduced Operator Exposure, since the operator need not remain in direct contact with the cask for extended periods of time.

Two basic configurations have been designed and licensed. The first designated the Mark I is a top loading only cask. It is provided with a single primary lid which contains six 19 inch diameter secondary lids. The secondary lids provide a means of individually loading small ion exchange columns. The second package has a single secondary lid and provides both top and bottom loading capabilities. These packages carry an NRC Certificate of Compliance Number 71-9073 and are licensed for international use. A description of the analysis and fabrication sequence follows.

HYPOTHETICAL ACCIDENT CONDITIONS

The Model OH-142 package has been designed and its contents are so limited that the performance requirements specified in 10 CFR 71.36 will be met, if the package is subjected to the hypothetical accident conditions specified in Appendix B of 10 CRF 71.

To demonstrate the structural integrity of the package and its ability to withstand the hypothetical accident conditions, a detailed computerized analysis was conducted.

Free Drop

It was suggested that, for conservatism, the one foot drop condition associated with normal handling be combined with the 30 foot accident condition to provide total drop height of 31 feet. Therefore, the analysis was conducted on the package using the full gross weight and a 31 foot drop.

The high density foam contained within the impact limiters is designed to crush on impact, thus absorbing and distributing the load.

Three drop conditions for the package were evaluated, i.e., end, corner, and side. For each condition the failure mechanism that would produce maximum load as well as maximum deformation was reviewed. In these evaluations maximum and minimum mechanical properties were used to produce the most conservative results.

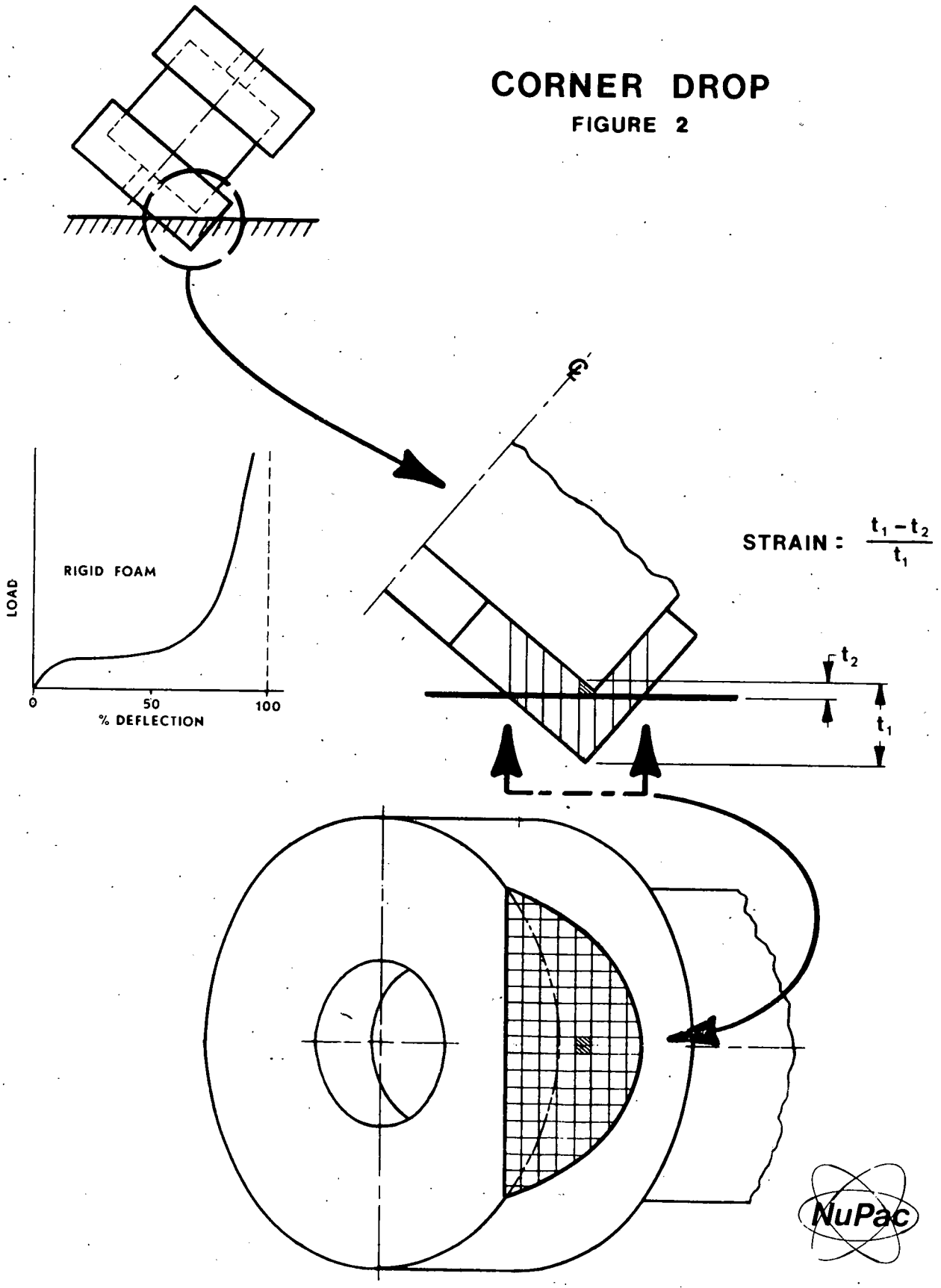
In the case of corner drop, the deformations experienced by the overpack are difficult to approximate with a simple analysis. In order to account for the strain hardening which takes place in the foam, a detailed computer program (CYDROP) was developed.

CYDROP treats the corner impact of a cylindrical package upon an unyielding surface. The package itself consists of a cylindrical payload portion surrounded by a larger cylindrical volume composed of a crushable media. As long as the deformations of the crushable media are modest, the problem may be approximately solved by assuming that uniform crush stress exists over the surface of the crush plane (contact surface). CYDROP was developed by NUPAC specifically to address problems of large deformations of this crushable media and to treat geometries where the cylindrical overpack envelope possesses axisymmetric cylindrical voids.

The large deformation behavior of the crushable media is accommodated by determining the actual strain of the crushable media at a point. This strain is used to determine the

CORNER DROP

FIGURE 2



CYDROP NUCLEAR PACKAGING 14.45.38 77/09/25 PAGE 2

CRUSH DEPTH (IN)	++ CRUSH PLANE ++ AREA (IN2)	VOLUME (IN3)	++++ IMPACT ++++ FORCE (LBS)	ACCEL. (G)	++++++ ENERGY ++++++ KINETIC (IN-LB)	STRAIN (IN-LB)	RATIO (SE/KE)	DISTRIBUTION OF STRAIN RATIOS BY PERCENT OF CONTACT AREA LE.70	GT.70 LE.80	GT.80 LE.90	GT.90 LE.95	GT.95
1.00	33.6	8.	7511.	.1	23872000.	1878.	.000	100.00	0.00	0.00	0.00	0.00
1.50	61.5	32.	20788.	.3	23904000.	8953.	.000	100.00	0.00	0.00	0.00	0.00
2.00	94.4	71.	41761.	.7	23936000.	24590.	.001	100.00	0.00	0.00	0.00	0.00
2.50	131.4	128.	69438.	1.1	23968000.	52389.	.002	100.00	0.00	0.00	0.00	0.00
3.00	172.0	203.	102638.	1.6	24000000.	95408.	.004	100.00	0.00	0.00	0.00	0.00
3.50	215.9	300.	140552.	2.2	24032000.	156206.	.006	100.00	0.00	0.00	0.00	0.00
4.00	262.7	420.	182312.	2.8	24064000.	236922.	.010	100.00	0.00	0.00	0.00	0.00
4.50	312.1	564.	227408.	3.6	24096000.	339352.	.014	100.00	0.00	0.00	0.00	0.00
5.00	364.1	733.	275454.	4.3	24128000.	465068.	.019	100.00	0.00	0.00	0.00	0.00
5.50	418.3	928.	325933.	5.1	24160000.	615414.	.025	100.00	0.00	0.00	0.00	0.00
6.00	474.6	1152.	377426.	5.9	24192000.	791254.	.033	100.00	0.00	0.00	0.00	0.00
6.50	532.9	1403.	429861.	6.7	24224000.	993076.	.041	100.00	0.00	0.00	0.00	0.00
7.00	593.0	1685.	484059.	7.6	24256000.	1221555.	.050	100.00	0.00	0.00	0.00	0.00
7.50	654.8	1997.	535687.	8.4	24288000.	1476490.	.061	100.00	0.00	0.00	0.00	0.00
8.00	718.2	2340.	593708.	9.3	24320000.	1758839.	.072	100.00	0.00	0.00	0.00	0.00
8.50	783.1	2715.	646936.	10.1	24352000.	2069000.	.085	100.00	0.00	0.00	0.00	0.00
9.00	849.4	3124.	705899.	11.0	24384000.	2407209.	.099	100.00	0.00	0.00	0.00	0.00
9.50	917.1	3565.	767761.	12.0	24416000.	2775624.	.114	100.00	0.00	0.00	0.00	0.00
10.00	985.9	4041.	824105.	12.9	24448000.	3173590.	.130	100.00	0.00	0.00	0.00	0.00
10.50	1056.0	4551.	890594.	13.9	24480000.	3602265.	.147	100.00	0.00	0.00	0.00	0.00
11.00	1127.1	5097.	961483.	15.0	24512000.	4065284.	.166	100.00	0.00	0.00	0.00	0.00
11.50	1199.2	5679.	1030083.	16.1	24544000.	4553175.	.186	100.00	0.00	0.00	0.00	0.00
12.00	1272.3	6297.	1104672.	17.3	24576000.	5096864.	.207	100.00	0.00	0.00	0.00	0.00
12.50	1346.2	6951.	1182835.	18.5	24608000.	5668741.	.230	100.00	0.00	0.00	0.00	0.00
13.00	1421.0	7643.	1271717.	19.9	24640000.	6282379.	.255	100.00	0.00	0.00	0.00	0.00
13.50	1496.5	8372.	1364653.	21.3	24672000.	6941471.	.281	100.00	0.00	0.00	0.00	0.00
14.00	1572.8	9140.	1468632.	22.9	24704000.	7649792.	.310	100.00	0.00	0.00	0.00	0.00
14.50	1649.6	9945.	1568501.	24.5	24736000.	8409076.	.340	100.00	0.00	0.00	0.00	0.00
15.00	1727.1	10790.	1681230.	26.3	24768000.	9221508.	.372	100.00	0.00	0.00	0.00	0.00
15.50	1805.0	11673.	1810680.	28.3	24800000.	10094486.	.407	100.00	0.00	0.00	0.00	0.00
16.00	1883.5	12595.	1967184.	30.7	24832000.	11038952.	.445	100.00	0.00	0.00	0.00	0.00
16.50	1962.4	13556.	2134168.	33.3	24864000.	12064290.	.485	98.45	1.55	0.00	0.00	0.00
17.00	2041.7	14557.	2341443.	36.6	24896000.	13183192.	.530	96.13	3.87	0.00	0.00	0.00
17.50	2121.3	15598.	2563596.	40.1	24928000.	14409452.	.578	93.75	6.25	0.00	0.00	0.00
18.00	2201.2	16679.	2844687.	44.4	24960000.	15761523.	.631	91.85	8.15	0.00	0.00	0.00
18.50	2281.3	17799.	3154357.	49.3	24992000.	17261284.	.691	89.70	8.72	1.59	0.00	0.00
19.00	2361.6	18960.	3472857.	54.3	25024000.	18918088.	.756	87.10	10.48	2.42	0.00	0.00
19.50	2442.1	20161.	3883307.	60.7	25056000.	20757129.	.828	85.69	9.65	4.66	0.00	0.00
20.00	2522.7	21402.	4363359.	68.2	25088000.	22818795.	.910	84.34	8.97	6.68	0.00	0.00
20.50	2603.3	22684.	4891323.	76.4	25120000.	25132466.	1.000	81.42	10.13	6.77	1.68	0.00
21.00	2683.9	24005.	5525357.	86.3	25152000.	27736635.	1.103	78.13	11.57	7.80	2.50	0.00
21.50	2764.5	25367.	6274154.	98.0	25184000.	30636513.	1.218	75.83	11.43	9.84	1.23	1.68
22.00	2845.0	26770.	7119670.	111.2	25216000.	34034969.	1.350	74.41	11.24	9.14	2.69	2.51
22.50	2925.4	28212.	7960613.	124.4	25248000.	37805040.	1.497	73.08	11.38	8.47	4.16	2.92
23.00	3005.6	29695.	8831894.	138.0	25280000.	42003167.	1.662	71.37	11.42	8.53	4.21	4.47
23.50	3085.7	31218.	9829162.	153.6	25312000.	46668431.	1.844	70.19	9.63	10.25	4.18	5.75

FIGURE 3

corresponding stress from an implicit tabular definition of media stress-strain characteristics. The total crush force is found by a double integration over the contact area of the crush plane. (Reference Figure 2)

Strain energy absorbed by the crushable media is determined by integrating the crush orce and its associated deformation. The package is assumed to be at "rest" when the computed strain energy value equals the applied kinetic (drop) energy. In addition to the variation in mechanical properties, the effects of the central hole in the overpack were evaluated.

From the sample printout (Figure 3) it can be seen that columns 6 and 7 provide the total kinetic energy and absorbed strain energy for incremental crush depths. When these values become identical the package is in equilibrium. Column 8 provides this ratio for rapid evaluation. From Case No. 1 it can be seen that the package comes to rest after reaching a crush depth of 20.50 inches. From column 8 it can be seen that it would require an additional 84% more energy for the package to bottom out at 23.50 inches.

Loads or accelerations experienced by the package are also plotted as a function of crush depth. Accelerations were found to vary from a low of 72.2 g's to a high of 76.4 for the full range of conditions.

These impact accelerations produce internal loads on the lid of the package. Binders and their support brackets were designed to react these loads.

Puncture

A 40 inch drop onto a six inch diameter pin can occur in three separate regions of the package, i.e., overpack area, ends, and side walls between overpacks.

Since the overpack is backed by side wall or end type construction, any impact in this region would be less severe.

ORNL-NSIC-68 was used for the side wall evaluation.

The top or lid of the package is constructed of two thicknesses of three inch steel plates for a total thickness of six inches. For the Mark I configuration, the top end plates contain six access holes spaced equally about the center.

Puncture resistance of this configuration was examined by a non-linear large deflection "ANSYS" finite element analysis. The "ANSYS" solution employed a three dimensional model to represent the geometry and properties of the end plates. The model used represented a 30° "pie slice" of the total lid assembly.

Maximum pin penetration (deformation of plates) was determined by plotting the load-deformation characteristics of the central nodes. Integrating this load-deformation relation produces a description of plate strain energy versus deformation depth.

The kinetic energy relation of the dropped cask was also plotted. The intersection of these two energy relations defines the deformation depth at which the strain energy (or work done on the plates) equals the available kinetic energy. At this deformation depth, 5.05 inches, the deformation is arrested and can proceed no further.

Thermal Analysis

In order to evaluate the package for both a normal transport as well as the hypothetical accident condition, a conventional thermal analyzer program was used. A model consisting of a sixty-eight node lumped parameter was created. A single node was used to represent the source. For conservatism, the OH-142 shield was assumed empty; that is, the fraction of heat absorbed by the payload during the hypothetical fire exposure was neglected.

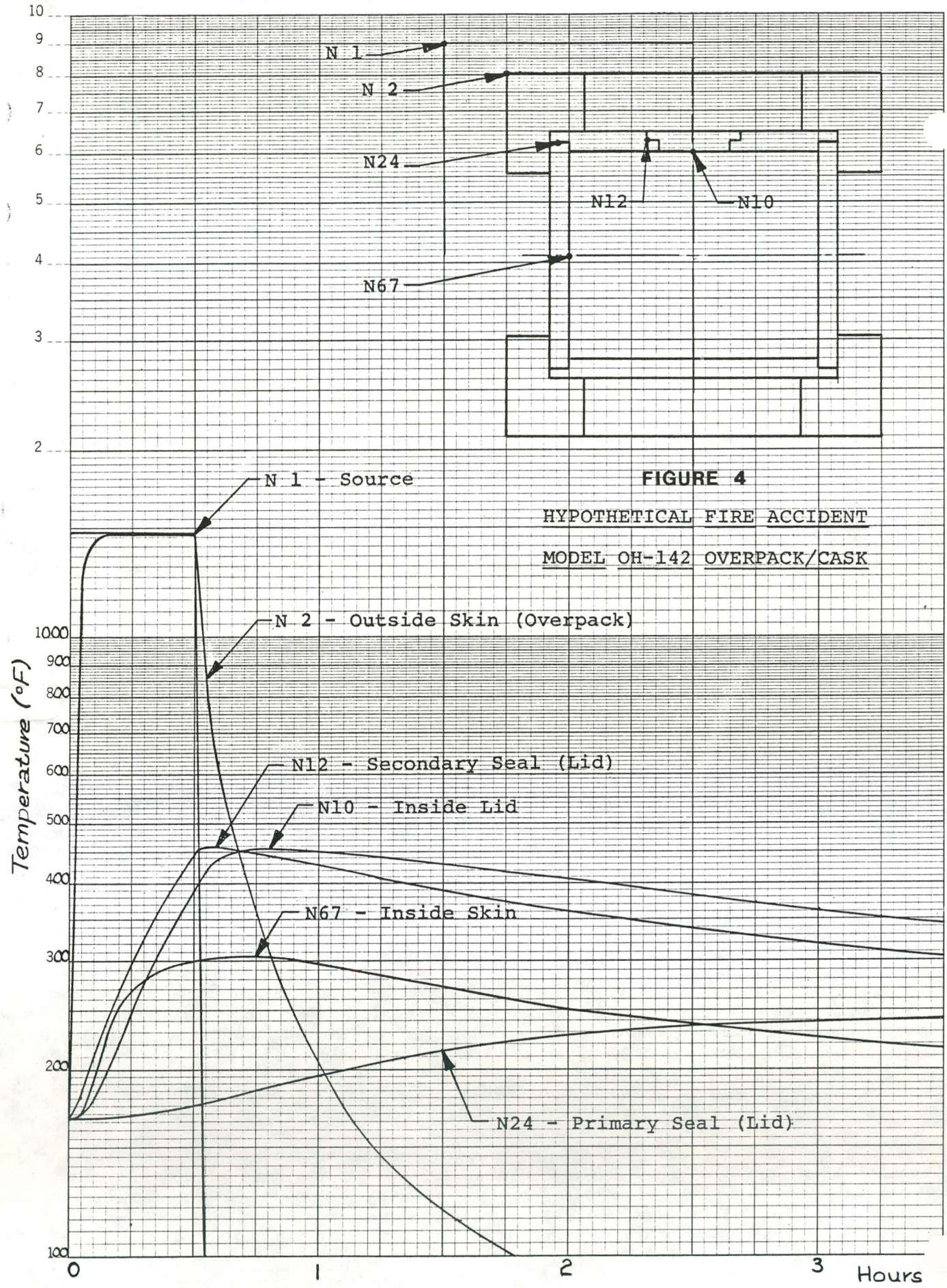


FIGURE 4

HYPOTHETICAL FIRE ACCIDENT

MODEL OH-142 OVERPACK/CASK

The cylindrical shell of the shield is thermally protected from fire by a 3/16 inch nominal thickness of Albi-Clad 89 intumescent mastic which exhibits a four to one expansion upon exposure to fire (@ 300°F). A comprehensive discussion of the experimental demonstration of the thermal performance of this coating is summarized in the Union Carbide Report No. K-1661, "Fissile Material Container and Packaging Development and Test Program", April 1, 1966. The experimental thermal response of the container discussed in this reference compares closely with that predicted by the OH-142 shield thermal analysis.

From the analysis, under steady state conditions and assuming a twenty-four hour day at maximum solar heat load, the maximum package termperature was found to be 168°F.

When subjected to the hypothetical accident conditions, lead temperatures remained well below critical melting point. The primary and secondary seal temperatures rose to 243°F and 458°F respectively. These are well below the 500°F allowable operating temperature for the silicone rubber seals. (Reference Figure 4).

FABRICATION

Fabrication of the package is broken down into three main items: the steel shell, lead pouring, and installation of the rigid foam.

Since the shells are fabricated from carbon steel, normal forming and welding procedures are applicable. Dimensional tolerances for roundness, concentricity and parallelism of the two shells are closely controlled in order to assure uniform thickness of the lead shielding. Quality Assurance throughout the fabrication program plays an important role. The casks are manufactured to RDT F2.3T standards which closely follow the 10 CFR 50 Appendix B program.

Once the cask body has been assembled, the lead cavity is pressure tested to assure tightness. Molten lead is introduced only after the cask has been heated to a predetermined temperature. Lead is pumped into the cavity. By means of controlled cooling, it is allowed to solidify from the bottom up. Additional molten lead is added to the top as required to compensate for shrinkage.

To substantiate the integrity of the lead pour, the cask is subjected to a 100% gamma scan.

The last remaining item is the assembly of the overpack. For this a rigid, high density polyurethane foam is used. The foam is formulated into two liquid components. These are proportioned together, blended and poured into the overpack cavity. There, reaction is exothermic which causes a phase change or gasification of an internal blowing agent. The resultant expansion produces the actual foaming action. By pouring the ingredients into the overpack cavity, the foam expands, filling the internal voids, and bonds to all contacting surfaces. Within minutes it solidifies into a hard rigid structure. The bonding action serves to tie or unitize the complete structure. As noted, the foam used for this package has a compressive strength of 1000 psi and a density of approximately 20 pounds per cubic foot. At this point the overpack shells are capped off and the cask is ready for final assembly.

Two such casks have been manufactured and are in service. A MK I configuration cask is currently being used at the Hanford site for the multiple transport of small ion exchange containers. The second, a MK II, is in service at Ontario Hydro where it is being used to transport large disposable liners of ion exchange resins.

Dup.

TRU WASTE TRANSPORT ECONOMICS: AN OVERVIEW

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INTRODUCTION

There are currently three predominant methods used to transport Transuranium (TRU)** contaminated waste. These are: 1) ATMX Railcars - 500 and 600 series, 2) Super Tigers® and 3) Poly Panthers®. These are briefly described below.

Both the ATMX-500 and 600 series railcars are massive doubly walled steel railcars which provide the equivalent protection of a Type B package. (For definition of Type B, see 49 CFR 173.398.)

ATMX-600 - Rapid loading and unloading of the 9x9x50 feet cargo space is achieved by prepackaging the TRU waste into standard 20-foot steel cargo containers. The ATMX-600 railcar will hold two cargo containers, each carrying 70 55-gallon drums and up to 44,800 lb gross weight.

ATMX-500 - These railcars do not utilize inner cargo containers and are divided into three inside bays, having dimensions of 16 (l) x 9.25 (w) x 6.25 (h) feet. Drums are typically palletized to facilitate loading and unloading. A typical load consists of 128 55-gallon drums (however, space can accommodate 192 drums), 12 fiberglass boxes (4x4x7), or a combination of palletized drums and boxes.

A Super Tiger is an overpack authorized for Type A, Type B, and "Large Quantities" (for definitions, see 49 CFR 173.398) of radioactive materials having outside dimensions of 8x8x20 feet. Maximum payload is approximately 28,700 lb with a gross weight of 45,000 lb.

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*Mound Facility is operated by Monsanto Research Corporation for the Department of Energy under Contract No. EY-76-C-04-0053.

**Definition per DOEM-0511: Transuranium-Contaminated Solid Wastes - Those contaminated with certain alpha-emitting radionuclides of long half-life and high specific radiotoxicity to greater than 10 nanocuries per gram (10 microcuries per kilogram), subject to the following condition and understandings: a) The radionuclides included are U-233 (with daughter products), plutonium and transplutonium nuclides except Pu-238 and Pu-241. (NOTE: That Pu-238 and Pu-241 waste should be handled as transuranium-contaminated waste when so indicated by Pu-239 impurities or when required by local burial criteria.)

A Poly Panther is a rectangular overpack authorized for Type A and Type B quantities of radioactive materials having outside dimensions of approximately 6.2 (l) x 5.6 (w) x 7.6 (h) feet. Maximum payload is approximately 3300 lb with a gross weight of 6100 lb.

The relative economics of these transportation modes are demonstrated in Table I. The data in Table I is for relative comparison purposes only. It must be recognized that these figures represent average costs and there are many variables which can affect an individual shipment. It should be noted that LLL owns their own Super Tiger; hence, no rental costs are included in their cost figure of $0.57 per cubic foot per 100 miles. Mound Facility does ship a small amount of TRU waste, which falls within the "Low Specific Activity" criteria, by commercial van. The reason for the lesser cost by highway for this material is that, because it is reasonably light, approximately 150 55-gallon drums can be placed on a van as opposed to the 120-140 drums loaded per ATMX railcar shipment. In addition, the freight rates are less than for ATMX railcar.

As may be seen from Table I, ATMX railcars offer a significant cost advantage for transporting Type B quantities of TRU waste; however, it must be remembered that for Argonne, Battelle and LLL the use of ATMX railcars was not very practical for reasons such as small volume generation and lack of available rail facilities. The cost figure, $0.71 per ft^3 per 100 miles for the Super Tiger could be reduced somewhat by longer term (monthly, semi-annual, etc.) rental contracts for the Super Tiger.

The rental question is one which must be resolved. Should the nuclear industry rent the TRU transportation units or will it be more cost effective in the future to purchase these units?

The primary factors influencing transport costs are:

1) freight rates of transport mode
2) effective cargo (weight and volume) management
3) effective utilization of available space (package design)
4) transport mileage, and
5) rental fees or initial capital outlay.

Factors 1, 2, and 3 are discussed below. Factor 4 is a constant and the objective is to choose the most economical and appropriate transport mode. Factor 5 has already been mentioned as a question requiring further evaluation and a decision.

FREIGHT RATES

The freight rates, charged per 100 pounds, currently charged to Mound Facility for transporting radioactive materials are:

Rail - $7.47 per hundred pounds (40,000 lb minimum) Miamisburg, Ohio to Scoville, Idaho.
Highway - $5.84 per hundred pounds (32,000 lb minimum).

Table I

COMPARISON OF RELATIVE TRU WASTE TRANSPORT ECONOMICS

	Transport Mode	Cost per ft^3 per 100 Miles
Argonne National Lab.	Poly Panther	$0.33
Battelle Columbus Lab.	Super Tiger	$0.71
Lawrence Livermore Lab. (LLL)	Super Tiger	$0.57
Mound Facility	ATMX-600 and 500	$0.17
	Highway Van	$0.15
Rocky Flats	ATMX-600	$0.16

When considering possible future TRU transport systems (see paper No. 52: "Development of a Safe TRU Transport System [STRUTS] for DOE's TRU Waste") one of the first questions is, "What will the future freight rates be?" The following information and discussion is taken essentially from materials presented at a one-day seminar, "Transportation Strategic Planning," presented in July 1977 by Drake Sheahan/Steward Dougall, Inc., Marketing and Physical Distribution Consultant The basic approach is to predict Carrier/Operating Costs for rail and highway, predict the required Return On Investment (ROI) for rail/Operating Ratio (OR) for highway, and predict Taxes. The sum of these three figures represents the total revenue required by a carrier per year. The tonnage carried is then predicted and the Freight Rate estimates are achieved by means of the following equation:

$$\frac{\$\text{Total Revenue Required}}{\text{Tonnage}} = \text{Predicted Freight Rates (\$ per unit of weight)}$$

Freight rates can be predicted for future years by the same method. This was done by Drake Sheahan/Steward Dougal, Inc., for the following years: 1977, 1978, 1979, 1980, 1981 and 1982. Only the determination of Carrier Operating Costs will be followed in-depth. Carrier Operating Costs were grouped into five major categories:

1) labor,
2) fuel,
3) materials & supplies,
4) depreciation, and
5) miscellaneous expenses.

Table II provides data on the relative percentage of each of the five categories of the total operating carrier costs.

Drake Sheahan predicts the behavior of each of these cost categories through 1982 are as outlined below:

Cost category: 1) Labor

Rail	Highway
Labor costs, as a percentage of total operating expenditures, are expected to remain approximately the same.	Labor costs, as a percentage of total operating expenditures, are expected to increase.

Table II

COMPOSITION OF CARRIER OPERATING COSTS 1976 AND 1982 (PREDICTED)

	Relative Percentage of Total Carrier Costs			
	Rail		Highway	
Cost Categories	1976	1982	1976	1982
1) Labor	55.1	54.1	61.4	66.3
2) Fuel and Oil	8.6	9.1	9.9	9.7
3) Materials and Supplies	19.1	19.3	13.4	11.4
4) Depreciation	5.0	3.7	2.8	2.2
5) Miscellaneous	12.2	13.8	12.5	10.4
	100.0	100.0	100.0	100.0

Cost category: 2) Fuel

- Since the oil embargo, fuel costs have escalated faster than any other operating cost.
- All modes have seen fuel costs double since 1973.

Rail	Highway
Fuel costs increased from 4% of total operating expenditures in 1973 to 9% today.	Fuel costs increased from 5% of total operating expenditures in 1973 to 10% today.

- Relatively moderate rate increases (6-7% per year) are predicted for the next few years.

Cost category: 3) Material and Supplies

- This category includes all items except fuel and oil.

Rail	Highway
This category represents the second largest operating cost and is due mainly to maintenance of rights-of-way.	Smaller than for rail and seen to be decreasing slightly.

- $3 billion in 1975.
- $9 billion needed for nation-wide upgrading.

Cost category: 4) Depreciation

Rail	Highway

- Expected to decline for both, with rail more significant than highway.
- Net investments have actually declined for the rail industry for the last 20 years.

Cost category: 5) Miscellaneous

- This category includes such items as insurance, loss and damage claims, operating licenses and miscellaneous taxes.

Rail	Highway
This is expected to increase in relative importance.	This is expected to decrease in relative importance.

A discussion predicting required Return On Investment (ROI) for rail, Operating Revenue (OR) for highway, or Tax expenditures is not really pertinent to the subject of this paper and will not be discussed in detail. However, after these three cost categories are summed and divided by the predicted tonnage, the freight rates required to keep the industry reasonably profitable can be predicted. Figure 1 graphically depicts the overall behavior predicted for freight rates based on the analysis of these data.

From these figures it can be seen that rail and motor freight rates are predicted to increase at approximately the same rate (rail slightly greater than motor). The major conclusion is that freight rates through 1982 are not expected to offer any major advantages to one transport mode over the other.

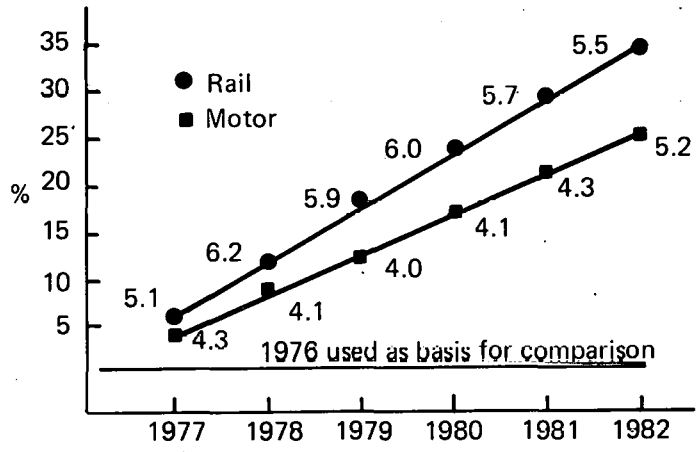


FIGURE 1 - Neither rail nor highway freight rates are expected to increase drastically.

EFFECTIVE CARGO MANAGEMENT

The primary point to be made here is that the minimum weight upon which transport costs are based must be exceeded. Typical minimum net weights for highway are approximately 22,000, and for rail, approximately 40,000 lb. It should also be noted that transport ($/ft^3) decreases only to a certain point and then levels off. Thus, from a purely economic viewpoint, there is no benefit at all in going from a transport unit capable of transporting 150 packages (uniform weight) to a larger transport unit capable of transporting 300 packages (of the same weight).

An example of cost savings through cargo management is in ATMX-500 shipments from Mound Facility. Typically, the 500 series cars are used to transport the 4x4x7 feet fiberglass reinforced plywood boxes of TRU waste. A full load (12 such boxes) rarely exceeds the 40,000 lb minimum, so the ATMX-500 loads are planned to be mixtures of the lighter boxes and heavier drums.

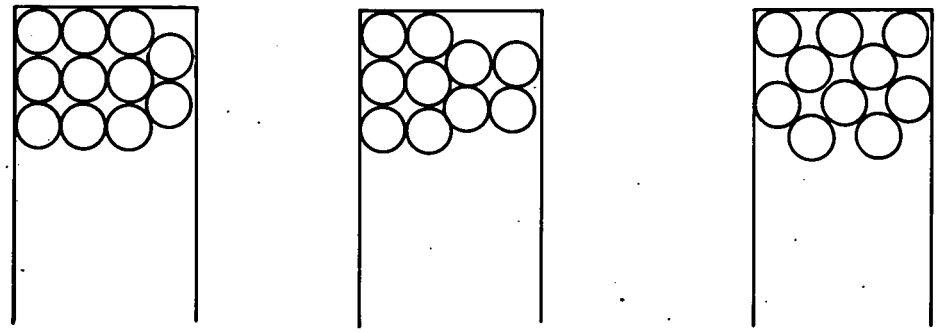
When the minimum weight cannot be achieved, space utilization becomes the important factor. If a shipper is paying for a minimum weight, it is necessary to design packages and shipping modes which will allow the maximum number of cubic feet.

An example of this is the use of fiberglass boxes vs 55-gallon drums in an ATMX-600 railcar. Only 140 55-gallon drums can be shipped for a total of 1029 ft^3. However, placing this waste in fiberglass boxes allows an ATMX-600 to accommodate 24 fiberglass boxes, at 112 ft^3 each, for a total of 2688 ft^3.

EFFECTIVE PACKAGE DESIGN

Custom package design based on the transport mode will be important. But, a better idea is to integrate the package design with the transport unit design to achieve maximum space utilization.

An example of inefficiency is the open-head 55-gallon drum when used in highway vans and cargo containers. The 55-gallon open-head drum is 24 inches in diameter, but the inside of a highway van or cargo container is 90 inches. Thus, only three rows of 55-gallon drums can be

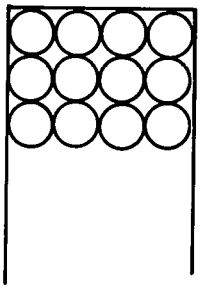


Maximum of 70
55-gallon drums
(35 per layer)

Maximum of 68
55-gallon drums
(34 per layer)

Maximum of 60
55-gallon drums
(30 per layer)

FIGURE 2 - Typical arrangements of 55-gallon drums in 90-in. van are inefficient.



Maximum of 80
55-gallon drums
(40 per layer)

FIGURE 3 - If 55-gallon drums were 22.5 in. in diameter, a more efficient arrangement would be possible.

placed in any one row across the transporter. Examples of spacing in a typical cargo container (8x8x20 feet) are shown in Figure 2. If the drums were approximately 22.5 inches in diameter, then the arrangement in Figure 3 could be utilized.

The reduction in diameter could easily be compensated for by an increase in height, since excess height is available in both highway vans and cargo containers. This would provide not only more effective volume utilization, but also a better load configuration. The load configuration shown in Figure 3 is expected to result in a minimal interaction of drums, and therefore minimal damage in transport, under normal transport conditions.

As mentioned previously, when the bulk density is low, the volume utilization becomes the main factor. To be most cost effective, then, it is necessary to ship as many net cubic feet of waste as possible.

MISCELLANEOUS FACTORS

There are many other factors affecting TRU waste transport cost. These are divided into two categories: 1) those factors which shippers can control through management, engineering, systems planning and applied research and development; and 2) those factors which are not directly under control, but which shippers can influence to varying degrees.

1) Factors which shippers directly control
   a) Effective design of transport vehicles.
   b) Cost of returning empty overpacks -- a non-productive cost which is incurred after the overpacks have served their purpose.
   c) Small quantity generators.
   d) Productivity per unit of manpower time.
   e) Technical Innovations.
   f) Integration of the package and the TRU Transporters.
   g) On-site handling, staging and storage requirements.
   h) The ease of use, durability and lifetime of the transport units.

2) Factors which shippers influence only to varying degrees
   a) More stringent environmental standards.
   b) More stringent packaging and transport regulations.
   c) Special restrictions such as requiring special shipment escorts and other unnecessary and very costly items such as the "sole-use train".
   d) Safeguards and security requirements.

These factors are discussed briefly either below or in the following paper: "Development of a Safe TRU Transport System (STRUTS) for DOE's TRU Waste."

Factors Which Shippers Directly Control

a) More effective design of transport vehicles
   i) Reducing the weight of the transport vehicle through use of ultra-light material of construction would allow a greater net payload.
   ii) Designing transport units (highway and rail) to allow a greater usable payload volume such as single-drop trailers where three layers of 55-gallon drums could be placed on a van instead of the current two layers commonly achieved.

b) Having to pay for return of empty overpacks
   The ATMX railcar transport costs are based on a round trip. One problem with overpacking to achieve regulatory compliance is having to pay for the return of the empty overpacks. This will probably always be true for highway transport. One possibility is utilizing overpacks by rail which are an integral part of the railcar and, thus, the car and the overpack are returned for one round-trip price. Another possibility is a national (DOE, NRC, etc.) pool of overpacks. Thus, overpacks would go to a staging area for use in shipments back to their originating area.

c) Small quantity generators on sites/areas having limited facilities need consideration. This is discussed in the second paper.

d) Productivity per unit time of manpower
   This factor is not expected to increase significantly in the future and really is reflected in the previously discussed labor costs as part of carrier operating costs.

e) Technical Innovations (from Drake Sheahan/Steward Dougall, Inc.)
   These are expected to center around enacting new environmental standards, safety and enhanced operating efficiency. The diesel engine will continue to be the predominant power source for both highway (larger, over-the-road vehicles) and rail for the next five years. Braking systems for highway, electronic braking, automatic detection of hot boxes and journals, and detection devices for derailed wheels are the typical expectations. No major breakthroughs are expected in freight car technology.

Table III

COST IMPACT IF "SOLE-USE TRAIN" WERE IMPOSED

Number of Rail-Cars per "Train"	Cost per Cubic Foot	Cost per Cubic Foot per 100 Miles
1 Railcar	$43	$2.19
4 Railcars	$14	$0.71
8 Railcars	$ 9	$0.46

f) Integration of the package and the TRU Transporters

g) On-site handling, staging and storage requirements

These are discussed in the second paper.

h) Ease of use and durability

These units must be easy to use and must effectively withstand the rigors of transportation. They must also "look strong, sturdy and have a 'low profile'."

Factors Which Shippers Influence Only to Varying Degrees

a) More stringent environmental standards (from Drake Sheahan/Steward Dougall, Inc.)

Large trucks have generally been exempt from emission and noise standards developed for passenger vehicles. It is not clear whether the large commercial tractor-trailers will be made subject to increasingly stringent standards. If so, the cost of compliance would be high and Technical Innovations would become very important.

b) More stringent packaging and transportation regulation

What if some proposed changes in performance requirements were imposed:

1) 1-hr fire test vs current 1/2-hr fire test?
2) 2000°F in fire test vs current 1475°F?

It is expected that the regulations will become more and more complex and compliance will become more and more costly.

c) Special restrictions imposed

Special escorts are now required in some eastern states for certain shipments of radioactive materials. The railroads have proposed such extreme restrictions as "sole-use trains" for radioactive materials. These "trains" would carry only radioactive materials, have severe speed limitations, follow special routing, etc. These restrictions are not needed, from a safety or any other standpoint. The major impact of these restrictions would be a cost of approximately $20 per mile for the engine alone. In addition, the shipper would have to pay normal freight charges.

An example of the cost impact of such restrictions is given in Table III. The first column demonstrates that this mode would be very costly and that a number of cars per train would be essential. It now appears that these restrictions will not be imposed. Recent court decisions have ruled against the railroads being able to impose such restrictions. This decision is expected to be upheld.

d) Safeguards and Security

The impact of regulations in this area on TRU transportation is not clear at this time; however, the potential for major impact is very evident.

SUMMARY

There is a need to promote uniformity, systems planning, and integration into the packaging and transportation of TRU waste. Economics is not promoted as the major factor; the assurance of transport safety is, of course, the major emphasis area. But, the high degree of safety in the area of packaging and transportation of radioactive materials has been proven for 30 years and is consistently demonstrated by the industry's outstanding safety record. Cost effectiveness can be and must be achieved by consideration and application of the ideas presented in this paper.

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DEVELOPMENT OF A SAFE TRU TRANSPORTATION SYSTEM (STRUTS) FOR DOE'S TRU WASTE

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INTRODUCTION

The transportation of DOE's Transuranium** (TRU) waste is highlighted in the Defense Waste Program Management Document as one of the key elements in the effective management of TRU waste. In addition, there are several major needs common to the radioactive materials transportation area as a whole which are just as apparent with respect to TRU waste transportation:

- Lack of shipping container standardization,
- Lack of uniform analytical evaluation techniques in determining package performance with respect to test (performance) criteria,
- Lack of public and carrier acceptance and understanding of safety aspects,
- Lack of a functional transportation data base, and
- Lack of a thorough analysis of "TRU Waste Transport Economics".

These can be summarized by stating that a need exists for an integrated, long-term DOE Transportation Plan.

Mound Facility, recognizing the long lead times required to develop such transport systems and recognizing the need for an operational STRUT System to support the start-up of the Waste Isolation Pilot Project (WIPP), began such a program last year. In addition to the long lead time and WIPP operational support, there are two more important reasons for starting the transportation study now.

First, the volume of TRU waste is increasing and will continue to increase because of major decontamination and decommissioning operations, under way or planned, throughout the DOE complex. NOTE: There are greater than 1000 ATMX railcar loads of waste at Idaho National Engineering Laboratory (INEL) right now... and 10 ATMX railcars available today for transport of such waste.

*Mound Facility is operated by Monsanto Research Corporation for the Department of Energy under Contract No. EY-76-C-04-0053.

**Definition per DOEM-0511: Transuranium-Contaminated Solid Wastes - Those contaminated with certain alpha-emitting radionuclides of long half-life and high specific radiotoxicity to greater than 10 nanocuries per gram (10 microcuries per kilogram), subject to the following condition and understandings: a) The radionuclides included are U-233 (with daughter products), plutonium and transplutonium nuclides except Pu-238 and Pu-241. (NOTE: That Pu-238 and Pu-241 waste should be handled as transuranium-contaminated waste when so indicated by Pu-239 impurities or when required by local burial criteria.)

To transport this volume of TRU waste would take approximately eight years of constant transportation. A volume of approximately 2600 ATMX railcar loads is predicted by 1985. To transport this waste using all ten ATMX railcars currently available would take 22 years.

Dedicating all the available Super Tiger resources to meeting this TRU transportation need does not provide much help. There are only six Super Tigers in the entire country. Three are owned and rented by Nuclear Engineering Co. of Louisville, Ky., and the other three are owned by DOE contractors. If all six were put into dedicated service, the TRU transport times would be reduced from eight years to six years and from 22 years to 15 years, respectively.

Secondly, we may not have several years to develop the STRUT System. ATMX railcars are currently under scrutiny and may not be authorized for use even through FY-1978. A long-term answer must be developed.

Mound Facility has been analyzing this problem for several years and has been actively pursuing answers to key questions impacting a future STRUT System. A detailed approach has been prepared and presented to the Department of Energy. The key elements of this plan are presented in this paper.

These elements include:

1) Acceptance - Public, Carrier and Regulatory.
2) Transport data base formation and analysis that takes into account the following questions:

    - What transport modes will be most effective in future years?
    - What volume of TRU waste will have to be transported?
    - What will be the condition of the retrieved TRU packagings - DOT Specification 7A?
    - What changes are expected in the transport regulations and what impact will these changes have upon future packaging and transportation of TRU contaminated waste materials and radioactive materials as a whole?

3) Systems Analysis to integrate TRU packaging, DOE contractor (large and small generators), WIPP, retrieval sites, safeguards and security, and carrier needs and requirements into the developing STRUTS.
4) Development of the actual STRUTS hardware to:

    - meet regulatory requirements,
    - meet the start-up schedule of WIPP,
    - meet the full-scale operational needs of WIPP,
    - ensure maximum integration with the "Standard TRU Package," and
    - provide transport units to meet all portions of the TRU waste transport needs.

5) Custom implementation of the STRUT System to ensure a smooth transition at the contractor sites, WIPP, the retrieval sites and with the carriers and the public.

These five elements will be discussed in greater detail in the following sections of the paper. Mound Facility began data assembly and analysis in several of these areas as early as FY-1976, and in FY-1977 contracted with an industrial consultant, Protective Packaging, Inc., Louisville, Kentucky, to assist in the data assembly and analysis.

ACCEPTANCE IS THE FIRST AND MOST ESSENTIAL OBJECTIVE

Public, carrier and regulatory acceptance must be achieved. Regulatory acceptance, the licensing and authorization for use, would emphasize actual performance testing of the STRUT System units. Modeling, testing and analysis, in the same fashion as is currently being done by Sandia Laboratories, would be very important. Those responsible for the "final approval" of the license (along with representatives of the news media, the public, carriers, legislators, etc.) would be invited to view and participate in the hypothetical accident testing and would be able to see first-hand the compliance with the regulatory requirements. Pictorial

documentation, such as movies, slide presentations, still photos, and brochures, could be prepared and utilized for the next phase of acceptance: Public and Carrier Acceptance. It is expected that the services of a professional public relations firm would be utilized in developing a formal plan to achieve the maximum in effective communications and overall acceptance.

Resources such as professional societies, DOE contractor and nuclear industry personnel, and college and university personnel could be organized first. One possibility is to identify several levels of communication targets forming a pyramid in shape, with each lower level encompassing a greater portion of the public. Presentation would be made starting at the top of this pyramid; resources such as films, slide presentations, automated slide presentations, pictorial brochures, and handouts would be provided and personnel at a particular level would be requested to approach the next target level of communication. This communication pyramid would continue until a very extensive level of public and carrier understanding and acceptance had been achieved. Every appropriate technical and governmental meeting would have exhibits depicting the STRUTS development program, its objectives and planned approach. Industry and trade associations would be heavily utilized in this proposed communication plan.

A FUNCTIONAL TRANSPORTATION DATA BASE IS ANOTHER KEY ELEMENT

One question which must be resolved is whether to buy the transport units, with the associated large capital outlay, maintenance, storage, repair and replacement costs or, alternately, to rent these units from industry. It must be understood that while economics can determine how much it will cost to transport TRU waste and how one method compares in cost to another method, economics by itself may not be the deciding factor. In the case of small generators or sites with limited facilities, the "best" method from an inventory and operational standpoint may cost somewhat more than the most "economical" system.

Another question to be resolved concerns waste characterization: Can the types of TRU waste be grouped into several specific categories based on physical and nuclear characteristics, e.g., high-, intermediate-, and low-density materials, Transport Index (TI), number of curies, Fissile Class? This information would be necessary for effective cargo management and for determining the different types of transport units necessary for an effective STRUT System. For example, large highway vans might be used for Low Density LSA (10-100 nCi/g) TRU waste; small highway vans, for High Density LSA (10-100 nCi/g) TRU waste; large highway STRUTS units, for Low Density Type B and L.Q. TRU waste; and rail units, for High Density Type B and L.Q. TRU waste.

Assay techniques and technological advances in this area could have a significant effect on transport economics. LSA and Type A quantities of radioactive materials can generally be transported via highway (with no elaborate containment or overpacks) at a lesser cost than can Type B packages. Some TRU waste that really meets the LSA criteria or Type A Quantity criteria is being transported as Type B quantities because analytical techniques are not available to distinguish between LSA, Type A, and Type B quantities once the material is packaged. Since it would be cost effective (and decrease the required usage of the more sophisticated Type B and Large Quantity transport units) to ship materials as LSA or Type A, an emphasis on this less expensive transport mode could be expected if the technology advances to allow this determination. These questions--How much is generated? How much has been generated? Is that already generated marked adequately to allow future segregation? What is the likelihood of such advances?-- must be assessed and appropriate plans made.

The number of transport units required to meet the TRU transportation needs is a critical number. This number depends on a number of factors, the first of which is the design capacity within WIPP for accepting and handling incoming TRU waste. This is expected to be the limiting factor. The sum of the incoming TRU waste will be the sum of the routine TRU waste generation plus that retrieved from sites such as Idaho National Engineering Laboratory (INEL). The transport of retrieved waste is expected to be more "straightforward" than for that portion being generated at the contractor sites and requiring transport to WIPP, considering that:

a) transport (turn-around) times for the TRU transport units would be well-known and a definite schedule could be established;

b) cargo management will be easier since there is just one center for the coordination of handling and loading of the waste and the scheduling of transport units; and

c) the other incoming TRU waste stream is made up of several sources and types of waste; therefore, the size of packagings, weight of packagings, composition with respect to curie content, and isotope and radiation levels will vary substantially.

Predicting turn-around times for shipments from a multitude of points will be much more difficult. It would seem reasonable that a logistics study would point out advantages in a dual or multi-modal system encompassing possible transport by highway units to one or a series of points where the waste could be consolidated into more economical transport modes. This would most likely benefit generators of small volumes of waste. Thus, accurate data on future annual generation of TRU waste and the types and physical characteristics will be very important. Ease of use, minimum handling, inspection, maintenance, and ease of loading and unloading will all affect this turn-around time.

The development of a STRUT System would initially assume a specified quality of package to be retrieved. Mound Facility's plan calls for planting "test drums" in the retrieval arrays. This would start immediately, and when retrieval was begun (several years hence) these test drums would be removed and tested to determine their ability to meet specified transportation performance criteria. This would provide test data upon which packaging and transportation decisions could be made. These test packages would contain nonradioactive materials to simulate the actual contents of the TRU waste drums. These drums would be placed in retrievable storage arrays. In addition, the data currently being gathered at these sites would be analyzed to allow predictions of package performance characteristics upon retrieval.

Early in 1977 when Mound Facility began looking at the question; "What will be the optimum transport mode(s) in FY-1984 and thereafter?" the rail question was definitely not clear. Some railroads were trying to say, "We aren't going to carry radioactive materials at all." Others were saying that such materials would be carried only by "single use trains," a situation which has several severe disadvantages, described briefly in the preceding paper (no. 51).

The possibility of these two restrictions being imposed has been reduced by recent court decisions; however, the possibility of further restrictions (regulatory or carrier imposed) has to be continually assessed.

It would seem that a combination of rail and highway transport units would most likely be an optimum system, but air and water would also have to be examined extensively. Some western states allow use of double and triple bottom trailers. This concept, if generally accepted by all states, could swing the balance in favor of the highway transport mode. Possibly "standard" routes could be established with increased authorized gross weights for highways and/or rail.

In the fluid environment of regulations, the impact of significant changes expected in the regulations governing radioactive materials is of utmost importance. This may well be the area requiring the most attention, since several early decisions on STRUTS will be made based on the direction of regulations.

SYSTEMS ANALYSIS WILL ENSURE TIMELY COMPLETION AND EFFECTIVE IMPLEMENTATION

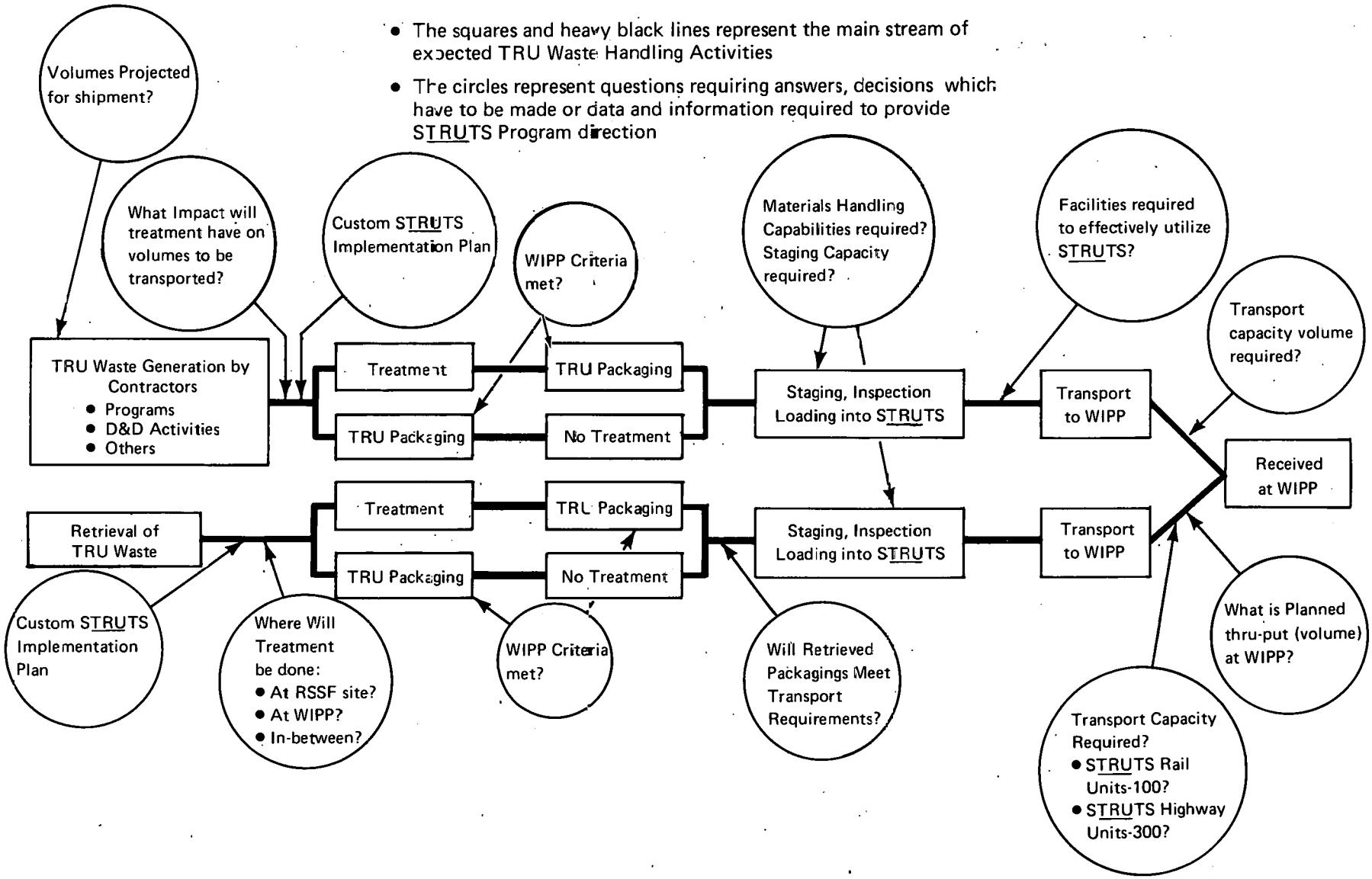
Figure 1 depicts the intent of the system analysis in general terms. The heavy lines represent the main flow of TRU contaminated waste to WIPP:

1) from on-site generation, and
2) from the inventory stored at 20-year retrieval sites.

The circled areas represent questions which must be answered; data which must be generated, assembled, and analyzed; or decisions which must be made in order to allow the timely development of a STRUT System.

The overall theme of the development program should be an integrated system. So often ackage development within the nuclear industry has been very parochial in nature. Too many imes, redundant technologies have been studied and developed for specific purposes when more

FIGURE 1 – SYSTEM ANALYSIS OF DOE TRU WASTE TRANSPORTATION



effective communications or willingness to utilize technologies developed by others would have saved considerable time and effort and money. Uniformity has been needed in the past, is needed today, and must be incorporated in STRUTS.

Another problem has been the recent decay in technology development in the area of packaging transportation of radioactive materials. Industry has found that it is generally uneconomical to invest in research and development of radioactive materials packaging. The reasons are readily apparent, e.g., lengthy time required to receive license and approval for use, high development costs, lack of ensurance that regulatory changes will not negate yet-to-be completed research and development, uncertainty about future regulatory impacts. These are all summarized by high development costs and less-than-adequate return on investment.

STRUTS would integrate the standard TRU package design (planned for development by DOE) into the overall transport system. No longer can the industry afford to have such intimately related programs proceeding independently of one another. Representatives of WIPP, DOE contractors, the TRU retrieval sites, and the carriers must have roles on the project management team developing STRUTS. This integration would be accomplished by a Systems Analysis approach.

The proposed systems analysis would:

a) identify the major functions or components in the TRU waste transport chain;
b) analyze these components and determine the critical path; and
c) develop a detailed plan to achieve the overall objective - the safe and economic transport of TRU waste.

The initial emphasis would be on preparing the way for effective implementation, at both the contractor and WIPP operations level, of the STRUTS hardware system. Hardware needs and requirements would be identified based on contractor operations (types of waste, volumes, radiation) and incorporated into the STRUTS design criteria.

Custom STRUTS implementation plans would be developed for DOE contractors (as appropriate) beginning in FY-1979. Individual sites would be analyzed and a TRU waste materials handling model established. Parameters of interest in this detailed analysis include an evaluation of waste staging facilities, railroad interface, materials handling equipment and packaging materials, and cost effectiveness.

The objective should be to ensure that the implementation of the final STRUTS System is as efficient and as cost effective as possible. The site specific material handling models will point out advance preparations, and equipment and facility procurement which must be made. The development of these site models will continue until all sites are completed.

In addition to these site models, other specific components such as WIPP operational characteristics, retrieval plans at INEL and possible contractor sites, waste treatment plans (and their expected efficiency and effect on waste volumes), commercial spent-fuel storage, and, later, possible reprocessing plans will be analyzed. Such analyses will yield data important to the successful implementation of a STRUT System. These data include announcements of required transport capacity and implementation cost. The data will also enable development of an implementation plan and provide direction to STRUTS hardware development.

DEVELOPMENT OF ACTUAL STRUTS HARDWARE

The results of the data collection and analysis would provide the direction for the Phase II portion of this program, the actual development of the STRUTS hardware. As mentioned before, the final transport "system" will likely consist of several different units--e.g., small and large highway transport units, small and large rail units, possibly a combination of both--to meet most effectively the wide variety of needs, current and expected. TRU waste transport needs include those for:

- Low Specific Activity (LSA)
- Type A quantities
- Type B quantities
- High and Low density materials
- Odd size packaging from special activities such as decontamination and decommissioning
- Small volume generators
- Sites with limited facilities (e.g., no rail sidings, no capability to handle heavy packaging, etc.)
- Sites with special local and state restrictions on transportation (e.g., Brookhaven and New York City's special restrictions).

CUSTOM IMPLEMENTATION OF THE STRUT SYSTEM

The major emphasis of the entire program should be the development of a system which most effectively meets the DOE and industry TRU waste transport needs. To ensure maximum effectiveness in this area, individual TRU waste handling plans would be developed by the DOE contractors to cover the areas of packaging, on-site movement, staging, inspection, and loading into the transport units. These materials handling analyses and plans, initiated even before the STRUTS hardware effort was begun, would answer many questions and provide a wealth of data on how best to meet the operational requirements of WIPP, DOE contractors and the retrieval sites. Thus, implementation of STRUTS would be easy, because it would be custom designed for the TRU waste handling operation by individual sites and by WIPP.

SUMMARY

Transportation, the link between TRU waste generation and WIPP and a vital link in the overall TRU waste management program, must be addressed. The program must have many facets: ensuring public and carrier acceptance, formation of a functional and current transportation data base, systems integration, maximum utilization of existing technology, and effective implementation and integration of the transport system into current and planned operational systems.

SEA TRANSPORT

OF

RADIOACTIVE MATERIALS

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LICENSING EXPERIENCES, RISK ASSESSMENT, DEMONSTRATION TEST ON NUCLEAR FUEL PACKAGES AND DESIGN CRITERIA FOR SEA GOING VESSEL CARRYING SPENT FUEL IN JAPAN

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INTRODUCTION

The transportation of nuclear fresh fuel has increased rapidly following the completion of nuclear power stations at several sites in Japan. In addition, as the reprocessing plant at Tokai has begun to operate, the transportations of spent fuels between some sites and Tokai have been started by sea-going vessel as the normal routine. In order to secure the safety of transportation by sea, the assessment of environmental safety and the establishment of a design criteria of sea-going vessel carrying spent fuel casks. Based on the criteria, HINOURA-MARU (G/T. 1290) was reconstructed and now is put on the route. To make sure the integrity of spent fuel cask, Nuclear Safety Bureau made also a plan to carry out a long term demonstration test on spent fuel package, in addition to scale model tests which were carried out since 1967.

In this presentation the current status in Japan with regard to licensing experiences, risk assessment study on sea transportation, a long term demonstration test project and also the design criteria for sea-going vessel will be reviewed.

LICENSING EXPERIENCES

Safety Criteria

It was ten years ago that Japan AEC had set up " Safety Examination Criteria for Nuclear Fuel Packagings " as domestic manual based upon the IAEA Regulations for the Safe Transport of Radioactive Materials 1967 edition. This manual had been long used by regulatory staff of Science and Technology Agency (STA) for examination of packagings both for domestic and international use. JAEC quickly responded to the issue of the IAEA Regulations 1973 revised edition and, through one year long deliberation by Technical Advisory Group, newly set up " Safety Criteria for the Transport of Radioactive Materials " and at the same time made a decision that domestic regulations related to the transport of radioactive materials should be amended to adopt the new criteria in January, 1975

Licensing

Due to the needs to clarify the allocation of responsibility required as administrative procedure in the 1973 IAEA Regulations, STA and Ministry of Transportation (MOT) made

memorandum of mutual understanding in April 1976 which was the basis for all the work carried out afterwards. Setting up of the memorandum called for STA to fully exercise its responsibility as the competent authority for package licensing. Examination of application is done by its Nuclear Safety Bureau, Nuclear Material Regulation Division which was created in January 1976 and an advisory committee of experts. The number of package designs which have been so fa licensed under the new criteria are about 30. Table 1 shows the list of examination. Some experience of audit on testing following the requirements has been obtained. As to the certificates issued for the international transport, an effort has been paid to pass them by the same format as those in U.K. , U.S.A. and wherever it possible.

Regulations

Following the decision of JAEC, STA and MOT which were competent authorities in the IAEA regulations carry out an extensive review of their regulations and regulatory practice. Regulations for sea transport, road transport, rail transport, air transport and for transport by nuclear undertaker and all the concerned have been amended by November 17, 1977. By this change and a set of orders issued simultaneously by the relevant branches, there remains no inconvenience any more for full implementation of the regulation in accordance with the 1973 IAEA regulations. New regulatory system has been effective on and after January 1, 1978.

Features of Japan's System

1. Approval of Package Design

It is required for user of type B(M), B(U), Fissile packages and special form material to obtain from STA the certificate that their design comply to the technical requirement in the regulations

2. Registration of Packaging

Packagings for B(M), B(U) and Fissile packages are required to be registered by STA prior to thier use. As a prerequisite of this registration, an inspection to check the integrity of the package should be done by the user. Therefore it's provided by STA that inspection for type B packages are audited by the nuclear facility safety inspector from Nuclear Safety Bureau. This system is expected to be useful in identifying and examining the safety of each package from time to time.

3. Approval of Package

Type B(M), B(U) and Fissile packages are required to obtain approval from STA as they are formed using the registered packagings for shipment in accordance with the design certificate. This application is simplified by using design certificate and the registration card in addition to the specification of the content.

4. Submission of Shipment Plan

Carriers of type B(M), B(U) and Fissile Class packages are required to submit their shipment plan to STA and also to MOT except Fissile Class I and II for each operation. This shipment plan are practically requested also from National Police Agency, local governments and other agencies.

5. Approval of Shipment

Type B(M), B(U) and Fissile Class III packages must obtain approval from MOT in advance their sea and air transport.

ASSESSMENT OF ENVIRONMENTAL SAFETY OF SPENT FUEL DURING SEA TRANSPORTATION

The investigations are intended to guarantee safety of sea transport. They have been carried out by Central Research Institute of Electric Power Industry (C.R.I.E.P.I.) under sponsorship of Nuclear Safety Bureau, Science and Technology Agency. Surveys have been done as follows;

(1) investigation of accidents on the sea transportation,

(2) behavior of radioactive material at the accident,
(3) evaluation of radionuclides effects on the environment.
The detailed presentation is made by the authors in this symposium.

CURRENT STATUS OF ROAD TRANSPORTATION OF NUCLEAR FISSILE MATERIAL AND ACCIDENT PROBABILITY INVESTIGATION

The purpose of this investigation is first to investigate present status of road transportation of nuclear fissile materials and accident probability and secondly to contribute to the estimation of safety of road transport, depending upon such background that road transport of fresh nuclear fuels increases in accordance with the expansion of nuclear power. The investigation plan is as follows:

1. Investigation on actual situation of overland transportation of nuclear fuel material
(1) Considering transportation of nuclear fuel materials, to survey the transport mode, accident mode, accident frequency and accident probability,
(2) Loading and unloading accident; Depending on carrying method on road, to investigate the accident mode during loading and unloading.

2. Survey of transport accident
(1) Surveying current status of road transportation mode, volume, route, radiation exposure control for carrier, to understand the actual situation of road transportation
(2) Forecasting transport volume; from the above mentioned survey to forecast transport volume in coming five years and to analyse the problem for safe transport including road control for transportation of nuclear fuels.

DEMONSTRATION TEST PROJECT ON THE RELIABILITY OF SPENT FUEL PACKAGE

History of R & D in Japan

In order to obtain the basic data for shipping cask design, some experimental studies have been performed since 1967. The results which were obtained by those tests were presented at the past international seminar and symposium in detail. In those studies some different scale models of horizontal cylindrical prototype cask weighing 80 tons were used to make clear the scale effect through the drop, puncture and fire tests, which was one of the features of Japanese research and development. Particularly the collaps test in high pressure water up to 500 bars was carried out to investigate the structural integrity of cask in the deep sea and to prevent the release of radioactive contents into sea water. The result was presented at IAEA International Seminar on Transport Packaging for Radioactive Materials, Vienna, on August 23-27 1976.

Subject and Sequence

In order to confirm the validity of analogy between the integrity of prototype cask and the extrapolated results from model tests, Science and Technology Agency proposed a long-term Demonstration Test Project on the Reliability of Spent Fuel Package utilizing full scale and one half scale models. To demonstrate the reliability of shipping casks the tests will be performed in accordance with the schedule shown in Fig. 1. Testing criteria are based on " Safety Criteria for the Transport of Radioactive Materials " which were proposed by Japan Atomic Energy Commission in 1975 and was essentially identical with the IAEA Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition.

The tests will be performed mainly under the accident conditions of transport of Type B package as stipulated in the IAEA Regulations and partly under the normal conditions of transport. The test items are as follows:
(1) Drop test from 9 m and higher height,
(2) Heat removal test,

Table 1. LIST OF PACKAGES (Science and Technology Agency, Japan. Dec. 26, 1977)

No.	Indentification Mark	Name of Packages	Description
1	J/1/AF	RCC-1	2 PWR fuel elements
2	J/2/B(U)F	TN912/A	FBR fuel element
3	J/3/B(M)F	SB/23	Research Reactor (KUR) spent fuel elements
4	J/4/B(U)F	GEmodelBP-2	Pu fuel rod for irradiation experiment
5	J/5/B(M)F	HZ-75T	7 PWR or 17 BWR spent fuel elements
6	J/6/AF	RCC-1	2 PWR fuel elements
7	J/7/AF	RCC-1	"
8	J/8/B(M)	SB/23	spent fuel piece for post irradiation examination
9	J/9/B(U)F	DOT-6M	Pu fuel rod for irradiation experiment
10	J/10/AF	NFI-II	2 PWR fuel element
11	J/11/B(M)F	UNIFETCH-L	Research Reactor (TAIC Reactor) spent fuel elements
12	J/12/B(M)F	JFC-75Y20T	Research Reactor (JPDR) spent fuel elements
13	J/13/B(M)F	JMTR'W5200	Research Reactor (JMTR) spent fuel elements
14	J/14/B(M)F	EXCELLOX-3	12 BWR spent fuel elements
15	J/15/B(M)	SB/23	spent fuel piece for post irradiation examination
16	J/16/B(U)F	TN 9180	2 ATR fuel elements (PuO_2-UO_2 fuel elements)
17	J/17/AF	OGL-77Y1.5T	coated particle fuel for irradiation experiment
18	J/18/B(M)F	EXCELLOX-3	14 BWR spent fuel elements
19	J/19/B(M)	TK	Calder Hall Type Reactor spent fuel
20	J/20/AF	RAJ	2 BWR fuel elements
21	J/21/AF	EXCELLOX-3A	5 PWR spent fuel elements
22	J/22/AF	NT-II	2 ATR fuel elements (UO_2 fuel element)
23	J/23/AF	BU-J	UO_2 powder, bulk, scrap
24	J/24/AF	BU-J	"
25	J/25/AF	21 PF-1	solid UF_6 (20A, B cylinder)
26	J/26/AF	21 PF-1	"
27	J/27/AF	21 PF-1	"
28	J/28/AF	21 PF-1	"
29	J/29/B(U)F	DOT-6M	high enriched U metal or U-Al alloy

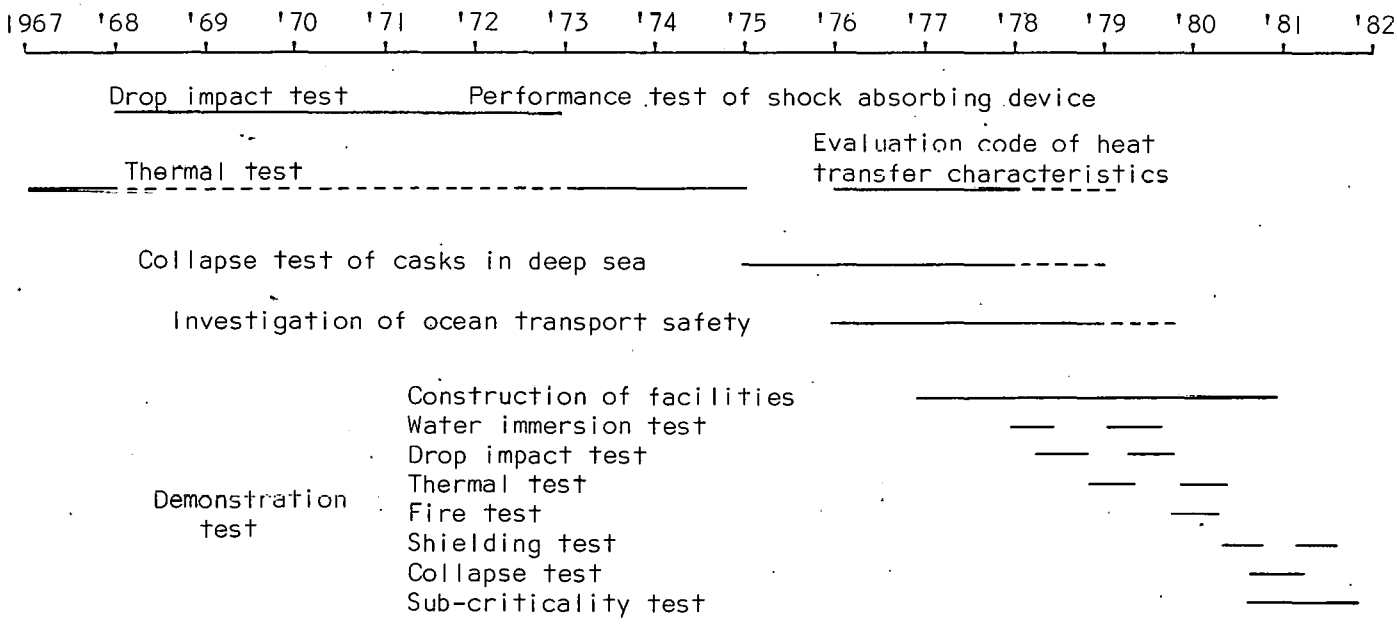


Fig. 1. Outline of packaging test and analysis and schedule of demonstration test project in Japan.

(3) Fire test (open fire test and furnace fire test),
(4) External pressure test for 300 - 500 bars corresponding to 3000 - 5000 m deep sea,
(5) Immersion test,
(6) Shielding test,
(7) Subcritical test,

Test Specimen and Budget

Test specimen will be designed to represent the larger shipping cask for fuels of light water reactor weighing approximately 100 tons and equipped with instrumentations which are adequate to the test condition. Also one half scale models will be fabricated and tested.

The budget for fiscal year 1977 is approximately 2.5 million dollars. The total expenditure for the project is estimated approximately more than 20 million dollars.

DESIGN CRITERIA FOR SEA-GOING VESSEL CARRYING SPENT FUEL SHIPPING CASK AND " HINOURA-MARU"

In the case of sea transport, a serious problem is the sinking of vessel and the leakage of radioactive nuclides from shipping casks in the deep sea. The counterplans to prevent the leakage of contents are of course now being investigated. For example, one of them is to equalize the pressure between inside and outside of cask in the deep sea adopting a pressure balancing valve, on which some experiment are now being carried out. Considering such situation the Bureau of Ship, MOT, established a design criteria for transport ship carrying packages of spent fuels, in order to make low the probability of sinking at sea where salvage operations are impossible and also of the radiation hazards under normal and accident conditions. Picking up some main requirements, those are as follows:

Structure of Ship

3,000 DWT vessel has been postulated for establishing the criteria, taking into account limitations of assumed harbors and quays at nuclear power station. In order to prevent sinking of casks, the criteria aim mainly to make minimum the probability of sinking. Therefore, the criteria shall meet fundamentally the provisions for Type 1 ship of IMCO for transportation of dangerous goods and also for nuclear ship.

(1) Collision protective structure

Although the damages of casks may not be caused in collisions accidents, the collision protective structures shall be adopted in accordance with the nuclear ship criteria. On the analysis of collision protective structure, T-2 tanker (Displacement 23,400 tons, speed 15 knots) shall be assumed as a striking vessel. The designed collision protective structure based on the above assumption can be effective in collision with a car ferry (6,000 G/T, 20 knots). It may be of course enough to retain the integrity when the designed ship collides against almost all sea-going vessels which navigate on the sea near the Japan Islands.

The values of energy to be absorbed by damage of two collided ships shall be calculated by formula given in the criteria, and the minimum absorbed energy should be larger than the kinetic energy lost during collision if the stem of striking ship reaches the longitudinal bulkhead on water line of the struck vessel.

(2) Arrangement of holds

Considering the safeguards in collision and the convenience to decontaminate the holds, double hulls shall be adopted. The location of longitudinal bulkhead has to be apart more than 1/5 X (bredth of ship) from the ship side as shown in Fig. 2. Assuming that the transport ship comes into collision with the other ship, it must be designed so that the location of fore bulkhead may have the distance of more than 0.15 X (length of ship) backwards from the fore perpendicular as shown in Fig. 3. In addition, it is not permitted the fore bulkhead serve as a collision bulkhead.

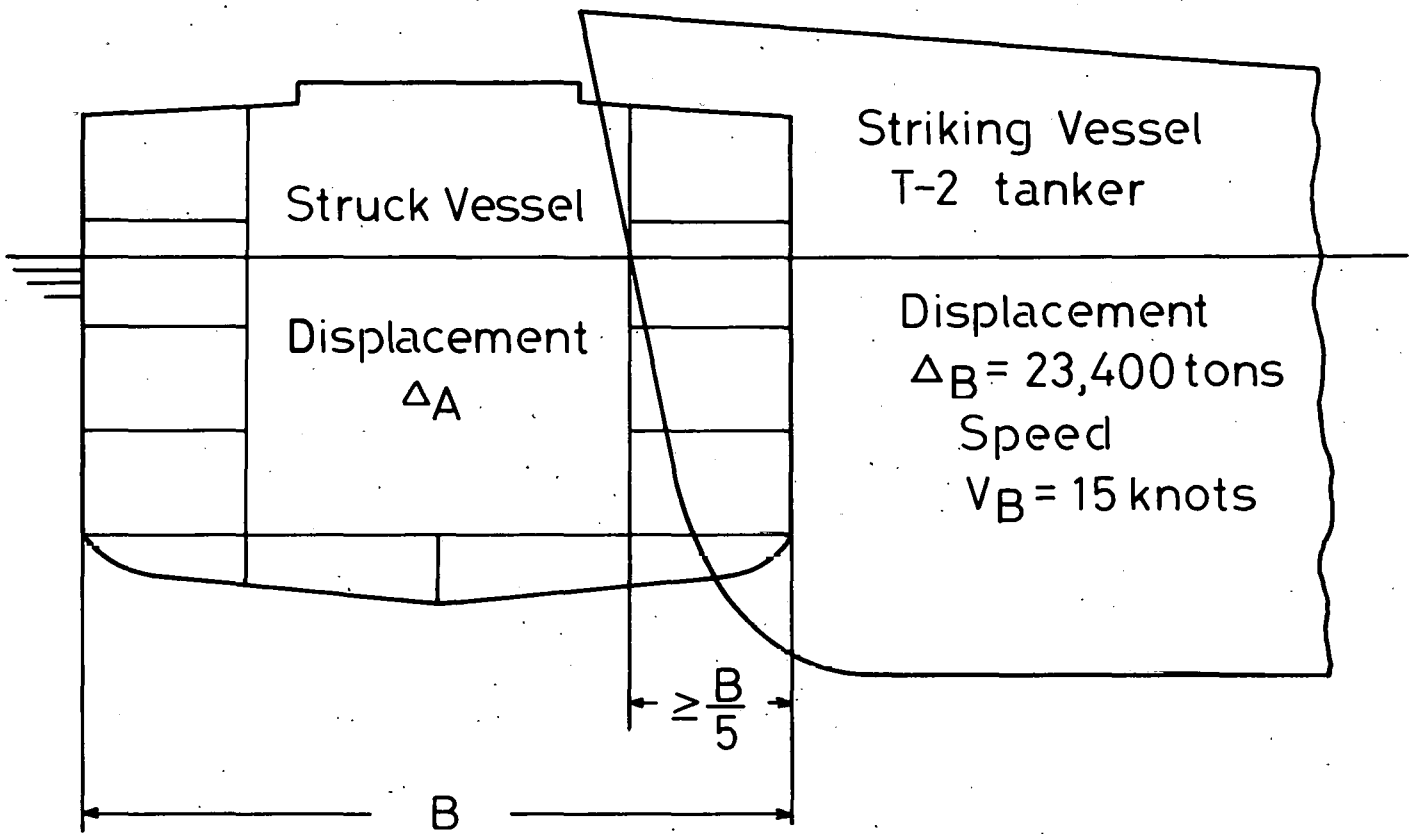


Fig. 2 Design Basis Collision

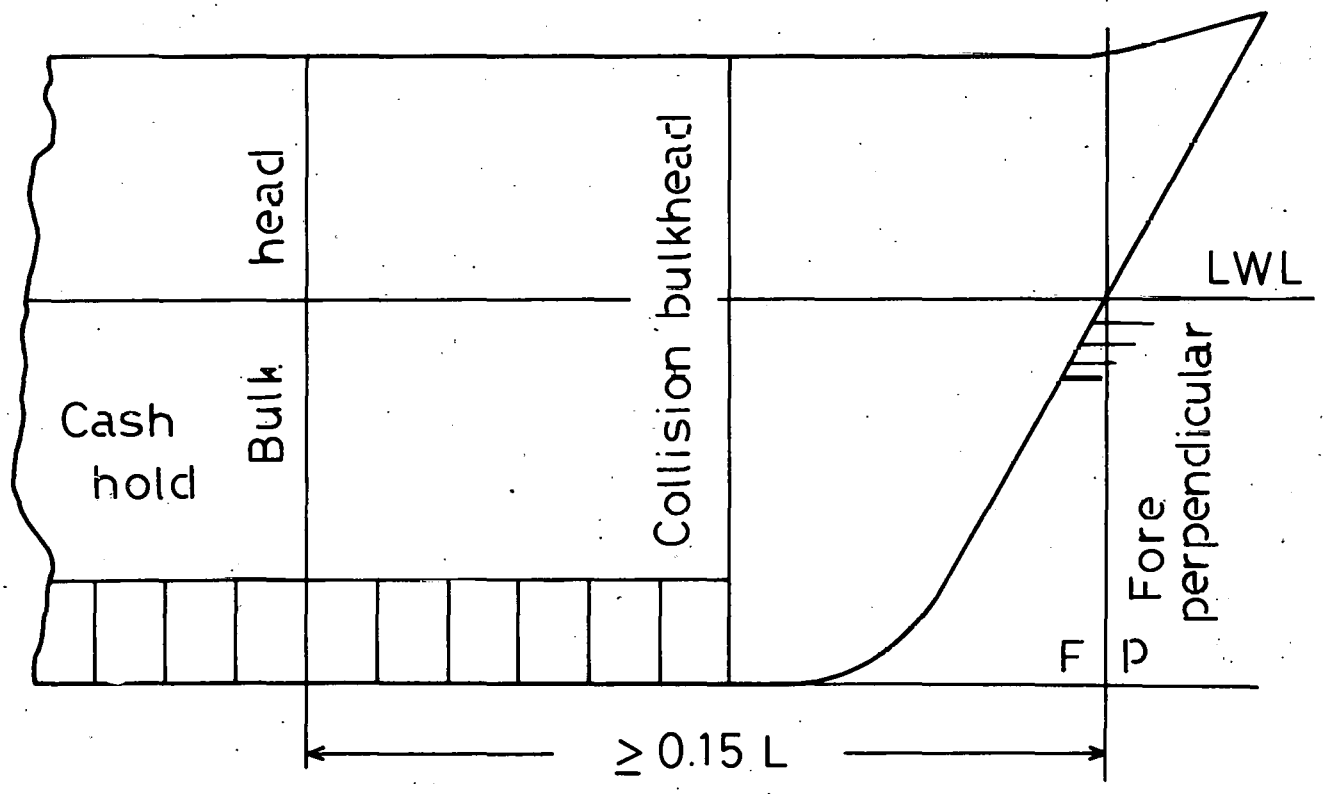


Fig. 3 Collision Protective Structure

(3) Stability after damage

It seems to be enough to have the stability range more than 20 degrees and the residual (GZ) over 100 mm under flooded condition. The heel generally cannot be over 15 degrees, though the exemption is provided. In order to prevent the list by nonsymmetrical flooding, the cross flooding facility is available.

(4) Grounding protective structure

To keep the integrity of casks at grounding, the depth of double bottom is required to be same order as that of nuclear ship.

Equipment

The cooling system must have enough capacity to maintain the ambient temperature around casks below 38°C. The tie-down equipment must be effective for the assumption of exteral force 1g (any direction), heel angles 45 degrees and trim angle 10 degrees.

Neccessary some kinds of radiation inspection apparatus, decontamiation facilities and also ventilation, drainage, fire extingushing facilities, emergency flooding equipment for ship fire, emergency electric sources, etc. shall be equipped.

" HINOURA-MARU "

Based on this design criteria, " HINOURA-MARU " was reconstructed for the purpose and was approved by the Japanese Government on July 1, 1977. The reconstruction work took three months from April to June, 1977. Fig. 4 shows the horizontal and vertical cross section of " HINOURA-MARU ". The main specifications of " HINOURA-MARU " are as follows;

Qualification	Japan Seas (Great Coast)
Gross Tonnage	1,290 tons
Overall Length	77.55 m
Length (between perpendicular)	72.00 m
Depth (until upper deck)	5.85 m
Depth (until erection deck)	7.75 m
Planning Full Load Draft	4.20 m
Full Load Navigation Speed	11 knots
Main Engine	2,500 P.S.
Loading Cask	Casks (Excellox-3A or HZ-75T)
Maximum Crew	Crew 17
	Owners 2
	Radioactive Supervisor 1
	Total 20

" HINOURA-MARU " repeated training for coming into port and leaving at all nuclear power station sities in October and November, 1977. On December 1, " HINOURA-MARU " loaded vacant casks at FUKUSHIMA Nuclear Power Station on board and sailed to Tokai and unloaded the casks at TOKAI Reprocessing Plant as the rehearsal for transportation of spent fuels. The first voyage as the normal operation was done successfully at the end of January 1978.

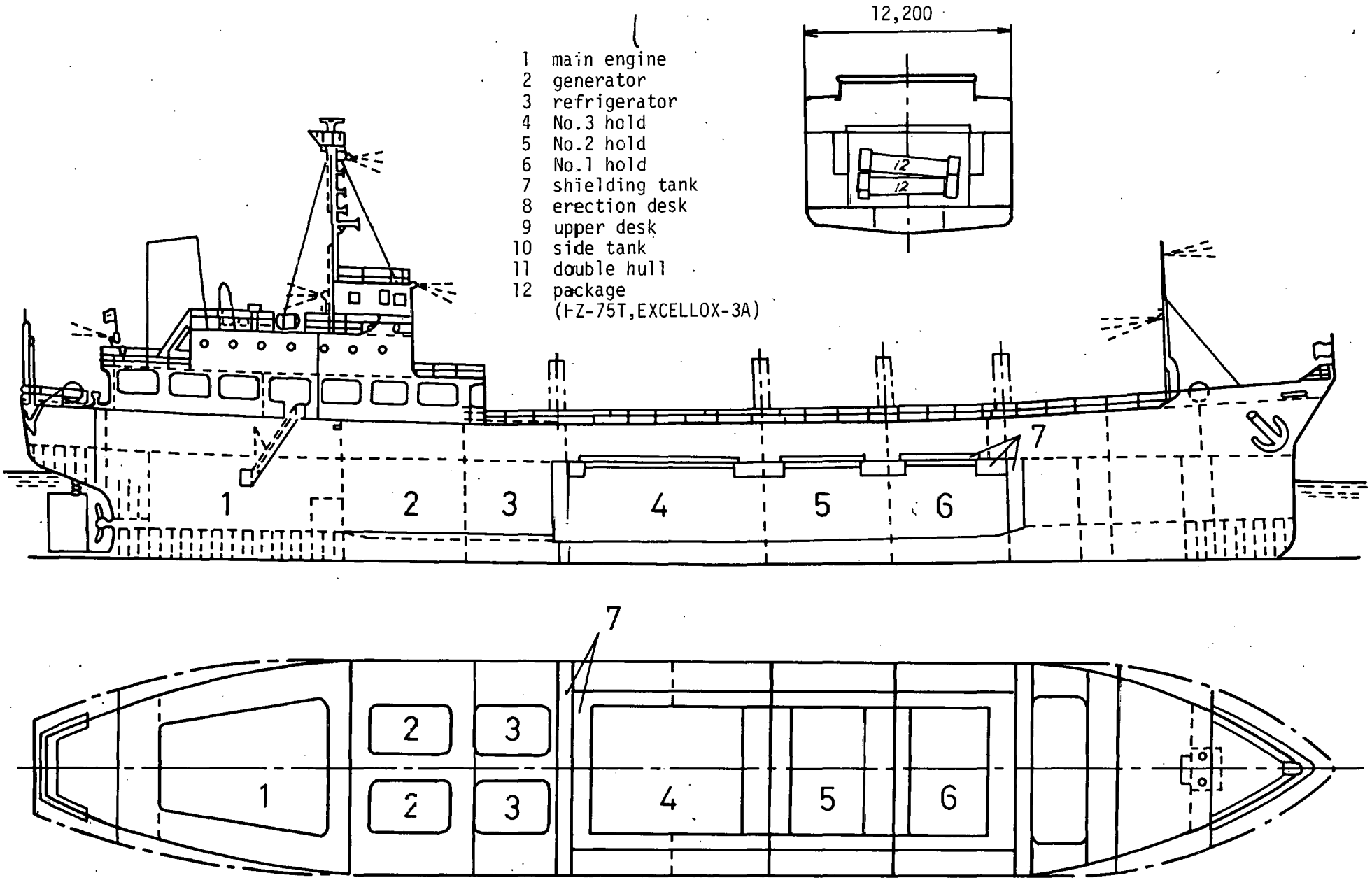


Fig. 4 General Arrangement of " HINOURA-MARU "

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CONSIDERATIONS IN THE TRANSPORTATION OF SPENT FUEL AND OTHER RADIOACTIVE MATERIALS BY SEA

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W. A. Brobst
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If you were to ask the average man on the street what hazardous material was the most dangerous in transport, what would he answer? If he lived near a railyard where a propane tank car blew up, he'd probably say, "propane." If he'd been evacuated from his home to avoid being gassed by chlorine leaking after a derailment, "chlorine" would be his answer. But if he'd been personally free from such disasters, the chances are high that he'd say "radioactive materials."

The three nuclear materials that seem to get the most public scrutiny these days are plutonium, spent fuel, and waste. This paper discusses one aspect of the first two of those materials -- the consequences of the loss at sea of plutonium packages and spent fuel casks.

To most of us in the nuclear transport business, the mention of nuclear spent fuel shipments by sea conjures up thoughts of strong steel and lead casks, double-hulled ships, careful loading and tie-down procedures, low radiation doses to handlers, and ultimate delivery of the product in good order. But for many people out in the rest of the world, the imagination plays different music. They think of ship collisions, crunching steel, long hot fires, damaged and sinking casks, leaks and ruptures, and contaminated fish and ocean waters. Is one group "right," and the other "wrong?" Or is the real situation somewhere between?

In the United States, the sea transport of spent fuel, plutonium, waste, and other nuclear products from one domestic port to another was mostly an academic consideration. Rail and highway transportation were easier and faster. There was some ocean shipment of spent fuel and uranium hexafluoride into the U.S. from overseas, but it was only a small fraction of the thousands of rail and highway consignments of those materials.

A couple of years ago, during the consideration by ERDA of the environmental impacts of the U.S. nuclear export program, questions were raised as to what might <u>really</u> happen to a spent fuel cask or plutonium package in case of a serious sea accident. Some 15 years earlier, the IAEA's panel of experts devised the Agency's first set of international nuclear shipment regulations. In their considerations, the panel looked at sea accidents in a rather general way, pooling their expert opinions and judgments, and concluding[1] that shallow water accidents would not breach the cask's integrity and deep water accidents would provide sufficient dilution of escaped material, or an ocean injection of contaminants at too deep a level to be a serious problem to the biosphere. Fires would quickly sink the ship, and smouldering fires were no threat to the package. Collisions produced more crush than impact; any collision serious enough to breach a large cask would sink the ship, putting the nuclear contents far out of harms way at the bottom of the sea.

The consensus of the experts at that time was that the IAEA regulations,[2] later translated and modified for U.S. use by the DOT[3] and the NRC,[4] provided adequate safety even in the event of serious sea accidents. Fairbairn and George stated[5] that packagings made to meet the IAEA test requirements would provide a reasonable degree of safety, with the probability of failure being reduced to an acceptable level. Sea accidents were not looked on as very significant. IMCO, the UN's maritime agency, seemed happy with the IAEA regulatory product, and stated[5] its acceptance of those regulations without much change.

During the ensuing years, things have been calm and the IAEA regulations have been well-accepted internationally. However, with the growing public uncertainty (and occasionally furor) over nuclear power, and coupled with the new national concern for environmental issues, the old opinions and judgments of the earlier gray-beards were cast by some into the shadows of doubt. Now, only hard data would be accepted, backed up by a comprehensive environmental impact analysis. Emphasis for a new effort was added by the impending increase in ocean transport of nuclear materials. Even more recently, the 1977 presidential policy of accepting foreign spent fuel for long-term storage in the U.S. has added further impetus for the new look into the question.

The DOE (then ERDA) decided to fund a modest study into what would happen to casks and plutonium packages in the unlikely event of a serious sea accident involving shipments of spent fuel or plutonium, and what the radiological consequences might be. Battelle Memorial Institute's Pacific Northwest Laboratories was asked to do this study. At the same time, the Japanese Government was considering a parallel study, so DOE's Transportation Branch agreed with the Japanese to work closely together, exchanging test and analytical data and opinions. The informal cooperative arrangements with the Japanese added much to the value of the study, and the Japanese results are also being reported at this symposium. The DOE is presently doing a specific environmental assessment of sea transport of spent fuel, as a part of the President's anti-proliferation program. The results of this Battelle study, which are summarized in the remainder of this paper, will play an important part in that assessment.

The Battelle study[6] estimated the radiological consequences to man through the marine food chain for some specific loss scenarios involving a plutonium shipping package and a large spent fuel cask. The study was not intended to be a comprehensive treatment of all the safety and environmental issues involved in shipping radioactive materials at sea, but a preliminary study such as this can lend perspective to the importance of the safety issues involved and provide valuable insights to direct further research in this area.

The oceans of the world can be conveniently divided into two regions -- the relatively shallow shelf surrounding the continents and major islands, and the deep ocean. The continental shelf generally does not extend more than 160 km from the shore, so this area comprises a relatively small fraction of the total ocean floor. It ranges in depth from tidal zones to about 1600 m, with an average depth of about 200 m. The edges of the continental shelf typically fall off rapidly to depths of 1000 m or more. Most of the ocean floor is then very deep, with an average depth of 4500 m.

Representative shipping containers were selected for plutonium and spent fuel shipments by sea. A 57-ℓ DOT specification 6M package was assumed for plutonium shipments. Consequences were estimated for a lost package containing 2.5 kg of recycle plutonium and one containing 4.5 kg of 95% ^{239}Pu. A spent fuel cask with a capacity of 3.1 metric ton (MT) of spent fuel was assumed. The reference fuel was PWR fuel with an average burnup of 33,000 MWd/MTU and an initial enrichment of 3.3% ^{235}U. Two possible pressure relief systems were considered for the cask -- a pressure relief valve only and a pressure relief valve plus rupture disc.

Four postulated loss scenarios were considered for the packages -- loss on the continental shelf and in the deep ocean of an undamaged package, and a package exposed to a severe fire. For the plutonium package a severe fire is defined as one that breaches the outer drum and thermal insulation in the package, and ruptures the cans inside the 2R vessel containing the plutonium. For the spent fuel cask, a "severe" fire was defined as one that causes a loss of coolant from the cask, resulting in failure of all the fuel cladding. The severe fire damage case was studied to provide an upper bound on the potential consequences of an accident involving a spent fuel cask for the loss locations considered in this study; even though it is not

clear at this point if a fire that could produce such damage can be realistically postulated in the ocean transport environment.

Package failure is defined in this study as a loss of containment integrity. The primary failure mechanisms for a shipping package lost into the sea are hydrostatic pressure and corrosion. Detailed calculations of the response of the shipping packages to these stresses were beyond the scope of this study, but conservative estimates were made of the package damage produced from these forces on the continental shelf and in the deep ocean.

For each of the four postulated plutonium package loss scenarios, a most likely failure mechanism or "material release scenario" has been developed. Since the outer drum of the 6M package and the produce cans containing the plutonium would likely fail at small differential pressures, no credit was taken for either as a containment barrier for a package lost into the sea. This also means that the material release scenarios for undamaged and fire damaged packages are essentially the same.

It was calculated that submersion to a depth of 3000 m or more would be required to collapse the inner 2R vessel. However, the 2R vessel considered in this analysis has a threaded pipe plug closure that could permit limited entry of sea water into the vessel at pressures encountered on the continental shelf. In descents to greater depths, inleakage is expected to equalize pressures and prevent collapse of the 2R vessel. The material release scenario for losses in the deep ocean and on the continental shelf are then essentially the same.

Release of the plutonium from the 2R vessel is limited by the rate at which sea water can be exchanged past the pipe threads on the closure plug. This rate can be affected by deformation of the threads from pressure forces during descent and eventual corrosion of the threads. Plutonium oxide particles of the size expected to be present in the container will pass through marine organisms with minimal retention, and therefore do not significantly concentrate in the marine food chain. The oxide must therefore dissolve to be a potential biological hazard. The solubility of plutonium dioxide in sea water is in the sub-mg/ℓ range.[7] The transport of plutonium from the 2R vessel into the sea is then limited by the solubility of PuO_2 and the rate of exchange of water between the inside of the 2R and the sea.

To determine the effectiveness of these barriers to release of the plutonium, specific knowledge is required about the leakage pathway from the 2R vessel, the surrounding environmental conditions and other parameters such as the damage to the package before sinking, the amount of thread deformation during descent, the loss location and the orientation of the package on the sea floor. Evaluation of all these parameters was beyond the scope of this study, so the rate of release of plutonium from a package lost into the sea could not be determined quantitatively. It was therefore conservatively assumed that all the plutonium in the package was released in soluble form within one year after the accident occurred for all loss scenarios.

Most likely material release scenarios were also developed for each of the spent fuel cask loss scenarios. Both the spent fuel cask and the fuel cladding can function as effective barriers to the release of radioactive material from an undamaged cask lost into the ocean. The response of the cask to hydrostatic forces depends on details of the cask construction. The cask lid seals are typically designed for internal pressures up to about 30 atmospheres. The resistance of these seals to external pressure depends on the type of seal used. Some seals could permit limited entry of sea water at depths typical of the continental shelf, while others may maintain their seal to depths where the pressure could be expected to collapse the cask walls. A similar situation exists for the pressure relief devices. Some spent fuel casks have used rupture discs as a backup for a pressure relief valve, while others use only a relief valve. The rupture disc could fail on the continental shelf, while most pressure relief valves are very resistant to external pressure and would maintain a seal at much greater depths. Because of these possible variations in cask design, two cases were considered for a cask lost to the continental shelf -- one assumes that the pressure relief device or cask seals leak on the continental shelf, and the other assumes that they remain in tact.

It has been estimated that if the pressure relief device and cask lid seal remain intact, the cask wall would collapse at a depth of 1670 m. It was therefore assumed that any cask lost to the deep ocean would fail by hydrostatic forces. Spent fuel cladding was calculated to fail

from hydrostatic forces at a depth of 2700 m for PWR fuel and 3100 m for BWR fuel. The cladding would therefore be expected to survive the pressure on the continental shelf, but fail in the deep ocean.

Corrosion becomes the important failure mechanism for the cask and the cladding when they do not fail from pressure forces. The cask wall is constructed of stainless steel with a minimum thickness of 5 cm. The corrosion rate for stainless steel in sea water is conservatively estimated to be about 0.1 mm per year. The cask wall would then provide a barrier to the sea water for about 500 years. Initially, the interchange of sea water between the interior of the cask and the ocean will be limited to small openings in the wall or past leaking seals or relief devices. This will limit the exposure of the fuel cladding to the sea water. Corrosion attack on the cask wall would tend to expose the entire cask contents to the marine environment after long periods of time. The zircaloy fuel cladding is very resistant to corrosion. The corrosion rate for the weld areas of the cladding has been estimated to be 2.5×10^4 mm/yr based on tests that were conducted for titanium alloys, that have similar corrosion behavior to zircaloy in sea water.[8] Thus it is possible that corrosion could open limited areas of the cladding in times on the order of 1000 years. General wasting away of the cladding would not be expected for tens of thousands of years.

The spent UO_2 pellets can be exposed to sea water by failure of the cask and cladding in a severe fire or by corrosion or hydrostatic failure of the cladding and cask in the ocean. There are two mechanisms for release of the radioactive material from the fuel into the ocean -- 1) washout of volatile and semivolatile materials in the fuel plenum, on the surfaces of the fuel and on the inside surfaces of the clad, and 2) leaching of fuel material into the sea water. The amount and type of radioactivity available for migration to man through the marine food chain is strongly dependent on the type of failure that has occurred.

For the case of a cask that is damaged by a severe fire before being lost into the sea, the majority of the radioactivity of concern is released by washout of semivolatile materials into the sea water. Semivolatile materials (principally cesium and iodine) are released to the fuel plenum during reactor operation and are also plated out on the inside surfaces of the cladding and the external surfaces of the fuel pellets. A significant fraction of the semivolatiles are also estimated to be released from the fuel during the fire. This material is assumed to plate out on structural surfaces of the ship and cask and along with the material in the plenum, to be dissolved into the sea water shortly after the ship sinks. The total release fractions for cesium and iodine are estimated to be 0.9 and 0.6 respectively. The remainder of the fuel would gradually be leached into the surrounding sea water. This process is estimated to take about 450 years.

For an intact cask and fuel that fail from hydrostatic forces after sinking into the deep ocean, the semivolatiles that were released to the plenum, inner cladding surfaces and fuel rod surfaces during reactor operation are again washed out soon after the cask sinks. This releases about 10% of the tellurium, selenium, antimony and iodine originally present in the fuel and about 15% of the cesium. The remainder of the fuel materials are again gradually leached out into the sea water over a 450 year period.

Two release scenarios were considered for a cask lost to the continental shelf. If the cask seal or pressure relief device fail on sinking, there will be an immediate release of semivolatile material that will be washed out of the approximately 1% of fuel that is assumed to have failed in the reactor, and a gradual leaching of the remainder of the radioactivity in this fuel into the surrounding sea water.

The radioactive material in the non-failed fuel would be released only after corrosion opens the cladding. This release would occur after several thousand years, when much of the radioactivity levels in the fuel had been reduced significantly by decay. For a cask that retained its containment integrity after sinking to the continental shelf, corrosion failure of the cask and the cladding would be required before a significant release could occur. Washout of semivolatiles from failed fuel is no longer a consideration, since most of these materials will have decayed away by the time the cask walls fail and expose the fuel to the sea water. The release scenarios for the spent fuel cask are summarized in Table 1.

TABLE 1. Summary of Spent Fuel Cask Release Scenarios

Loss Location	Pre-Loss Condition	Failure Mechanism	Time Between Loss and Release of Radioactivity	Principle Release Type
Continental shelf	Severe fire fails cask and fuel	Pre-loss damage	Immediate	Semivolatiles released by fire
Continental shelf	Undamaged; pressure resistant cask	Corrosion failure of cask and cladding	1,000 to 10,000 years	Leaching of fuel
Continental shelf	Undamaged; cask not pressure resistant	Pressure failure of cask; corrosion failure of fuel cladding	Immediate from failed fuel; 1,000 to 10,000 years from intact fuel	Semivolaties in plenum of failed fuel; leaching of fuel materials
Deep ocean	Severe fire fails cask and fuel	Pre-loss damage	Immediate	Semivolatiles released by fire
Deep ocean	Undamaged	Pressure failure of cask and fuel cladding	Immediate	Semivolatiles in fuel plenums

The potential radiation dose to man following release of plutonium or spent fuel materials into the sea has been estimated. Radionuclides released into the ocean tend to be dispersed rapidly by the natural mixing of ocean waters. However, biological mechanisms exist in the marine food chain for concentrating some radioactive substances. The primary potential hazard to man following the release of radioactive material into the ocean is through consumption of seafood contaminated with radioactivity. The population dose from consumption of contaminated seafood was estimated using the following relationship:

$$D = \frac{P}{2} \sum_i Q_i B_i (DF)_i$$

where the summation is over all radioactive species released, P is the productivity of the contaminated region, Q_i is the amount of a particular radionuclide released, B_i is the biological concentration factor of the radionuclide in marine organisms and DF_i is the dose conversion factor that relates dose to the amount of material ingested.

Seafood production of the oceans of the world varies considerably from region to region depending on the primary production of organic matter. Areas over continental shelves and those where the upwelling of nutrients takes place are regions where the primary production is great and fish and other organisms harvested by man for food are relatively abundant. Seafood production from a typical area of the continental shelf has been estimated to be about 1×10^{-4} kg/m^3 yr.[9] The harvest is assumed to be composed of either 100% fish or a mixture of seafood consisting of 50% fish, 25% crustacea and 25% molluscs. The edible quantity of seafood available was assumed to be 50% of the live weight of the harvest.

Areas such as the open ocean are generally much lower in productivity and hence in harvestable fish than the continental shelf. Although some fisheries do extend to the open ocean from the coastal water, it has been estimated that the productivity of these regions is about one thousand times less than the productive areas of the continental shelf.[9] The seafood and fish harvest from these areas is assumed to be 10^{-7} kg/m^3 yr.

The average dose to a member of the population consuming the seafood harvested from the contaminated region may be estimated if the extent of the contaminated area and the average individual consumption rate of the population are known. For this estimate, a typical coastal fishery is assumed to be 48 km wide and 96 km long. A 30-meter average depth is assumed for the ne in which seafood is harvested. This results in a fishery with a volume of 1.4×10^{11} m^3

yielding 1.4 x 10^7 kg of seafood per year. Using the average U.S. seafood consumption rate of 8 kg/yr and assuming that only 50% (by weight) of the harvested seafood is edible, approximately 870,000 people consume seafood from this region.

The estimated radiation doses to the population and to the average individual for the plutonium and spent fuel release scenarios are presented in Tables 2 and 3. All dose estimates assume that seafood from the contaminated area is consumed for one year and that noncontaminated food is consumed thereafter. Two doses have been calculated. The dose that occurs during the year in which the contaminated seafood is consumed and the 50-year dose commitment resulting from retention of ingested radionuclides in the body.

There is a significant degree of conservatism (pessimism) inherent in the dose numbers presented in Tables 2 and 3. The amount of radioactive material released for each case has been conservatively estimated. It has also been assumed that no attempt is made to prohibit or otherwise regulate the fishing activities in the contaminated area. For many of the loss scenarios, recovery of the shipping container before a significant portion of its contents are released is also a very real possibility. Despite this conservatism, the doses that have been calculated seem relatively modest. The maximum population dose calculated is comparable to the dose that population could be expected to receive from natural background sources. Although the distribution of the dose within the effected population was not determined, since the pathway to man is only through seafood consumption, the dose to most individuals would probably not vary drastically from the average individual dose presented in the tables.

This study has identified several areas where further research activities are needed to better understand the safety of transporting radioactive materials by sea. These areas include:

- Investigation of the probability that a ship carrying radioactive materials will be involved in an accident at sea, the geographic location of these accident forces that a shipping container would experience in an accident at sea.
- Analysis of the response of shipping containers to the accident forces and the stresses present in the ocean environment.

TABLE 2. Estimated Dose from Contaminated Seafood Consumption after Loss of a Single Plutonium Package at Sea

	Low Burnup Pu[a]				Recycle Pu[b]			
	Dose		Dose Commitment		Dose		Dose Commitment	
	Fish	Seafood[c]	Fish	Seafood	Fish	Seafood	Fish	Seafood
Total Population: (man-rem)								
Continental Shelf	9×10^{-3}	0.28	0.79	26	0.14	3.7	9.5	240
Deep Ocean	9×10^{-6}	2.8×10^{-4}	7.9×10^{-4}	2.6×10^{-2}	1.4×10^{-4}	3.7×10^{-3}	9.5×10^{-3}	0.24
Average Individual: (rem)								
Continental Shelf	1.0×10^{-8}	3.2×10^{-7}	9.1×10^{-7}	3.0×10^{-5}	1.6×10^{-7}	4.3×10^{-6}	1.1×10^{-5}	2.7×10^{-4}
Deep Ocean	1.0×10^{-11}	3.2×10^{-10}	9.1×10^{-10}	3.0×10^{-8}	1.6×10^{-10}	4.3×10^{-9}	1.1×10^{-8}	2.7×10^{-7}

(a) 4.5 kg PuO_2 per package with 95 wt% ^{239}Pu, 5 wt% ^{240}Pu.
(b) 2.55 kg Pu per package with 1.5 wt% ^{238}Pu, 58 wt% ^{239}Pu, 24 wt% ^{240}Pu, 11 wt% ^{241}Pu, 4.9 wt% ^{242}Pu, 1.0 wt% ^{241}Am.
(c) Mix of 50% fish, 25% crustaceans and 25% molluscs.

TABLE 3. Estimated Dose from Contaminated Seafood Consumption after the Loss at Sea of a Spent Fuel Cask Containing 3.1 MT Uranium

Total Population

Location of Loss	Pre-Loss Damage	Dose (man-rem) Fish	Dose (man-rem) Seafood[a]	Dose Commitment (man-rem) Fish	Dose Commitment (man-rem) Seafood[a]
Continental shelf	Undamaged; cask not pressure resistant	150[b]	160[b]	270[b]	1000[b]
Continental shelf	Undamaged; cask pressure resistant	~ 2	~ 15	~ 35	~ 765
Continental shelf	Severe fire	8.9×10^{4}	8.7×10^{4}	1.5×10^{5}	1.5×10^{5}
Deep ocean	Undamaged	< 150	< 150	< 150	< 150
Deep ocean	Severe fire	< 150	< 150	< 150	< 150

Average Individual

Location of Loss	Pre-Loss Damage	Dose (man-rem) Fish	Dose (man-rem) Seafood[a]	Dose Commitment (man-rem) Fish	Dose Commitment (man-rem) Seafood[a]
Continental shelf	Undamaged; cask not pressure resistant	1.7×10^{-4}	1.8×10^{-4}	3.1×10^{-4}	1.1×10^{-3}
Continental shelf	Undamaged; cask pressure resistant	$\sim 2.0 \times 10^{-6}$	$\sim 2.0 \times 10^{-5}$	$\sim 4.0 \times 10^{-5}$	$\sim 9.0 \times 10^{-4}$
Continental shelf	Severe fire	1.0×10^{-1}	1.0×10^{-1}	1.7×10^{-1}	1.7×10^{-1}
Deep ocean	Undamaged	$< 1.7 \times 10^{-4}$	$< 1.7 \times 10^{-4}$	$< 1.7 \times 10^{-4}$	$< 1.7 \times 10^{-4}$
Deep ocean	Severe fire	$< 1.7 \times 10^{-4}$	$< 1.7 \times 10^{-4}$	$< 1.7 \times 10^{-4}$	$< 1.7 \times 10^{-4}$

[a] Mix of 50% fish, 25% crustaces, and 25% molluscs.

[b] Includes initial 1% of failed fuel ($\sim$150 man-rem dose and $\sim$235 man-rem dose commitment for either fish or seafood) and eventual release from corrosion failure of fuel rods.

- Behavior of radioactive material in the ocean and its transport to man through the marine food chain, by resuspension from the water into the air and by direct exposure on beaches.
- Design of shipping packages to withstand the accident forces in the marine transport environment.
- Investigation of possible operational procedures to reduce the exposure of shipping containers to the forces that are produced in marine accidents.
- Investigation of remedial measures such as package recovery that could mitigate the consequences of a severe accident.

It should be noted that investigations in several of those areas are underway in Japanese programs being described in other papers presented at this meeting.

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Safety Evaluation on Transport of Fuel at Sea and Test Program on Full Scale Cask in Japan

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1. Introduction

With the operation of the reprocessing factory at Tokai Works, Power Reactor and Nuclear Fuel Development Corporation in Japan, the sea transport from all nuclear power station is planned for spent fuel. Spent fuel shipped contains a lot of radioactivity, and if an accident occurs during shipping and radioactivity is scattered into man environment, population may be affected. Therefore, it is necessary to immediately establish a transport safety policy to accompany the opening of the reprocessing plant. For this reason, the safety evaluation on transport of spent nuclear fuel at sea was determined and it will be carried out test program on fullscale cask for the safe transport of cask in support of this evaluation. This report will be described about these. The safety evaluation on transport of fuel at sea was carried out by the committee on study of safety evaluation of spent nuclear fuel sea transport (chairman Tadashi Nagakura) under contract with the Science and Technology Agency (STA), Japan Government.

2. Safety Evaluation on Transport of Spent Fuel at Sea

2.1 Investigation of Probability and Expectation of Sinking of Cask and Ship

In case of sinking of cask resultant from a shipwreck, the size and structure of ship affect what is the cause of the sinking during various cause such as collisions with other ship or underwater objection, strand, fire, disaster in the stormy sea and engine trouble. If the size of ship is greater than about gross weight 1,000 tons, it is shown that the sinking is occured by collision, specially, when smaller ship is collided with bigger ship, by statistics of disasters at sea. So, the probability and the expectation of sinking of transport ship are determined by the method as follow and flow sheet of evaluation method is shown in Fig. 1.

(1) Route

For this investigation, two sea lanes between the area of Wakasa Bay and that of Tokai are employed. One is the southern circuit by way of Ohsumi Channel and the other is the northern circuit by way of Tsugaru Channel. Turning points at route are assumed to be 5 and 10 nautical miles offshore from a cape jutting out into the sea, and the sea lanes are assumed to connect these points by almost straight lines.

(2) Transport ship

For the estimation of the posibility of sinking of cask, the sinking of cask with or without the ship, condition of fire of the ship and size of dangerous ship—dangerous ship is called vessels which run the transport ship down with collision—, the investigation is carried out about the size, structure and structural strength, performance and equipment of ship.

The transport ship, the Hinouramaru (displacement 2,600 tons), has an anti-collosion protective structure similar to that of a nuclear-powered ship, so that it will be safe against collision with a T-2 tanker. (See note.)

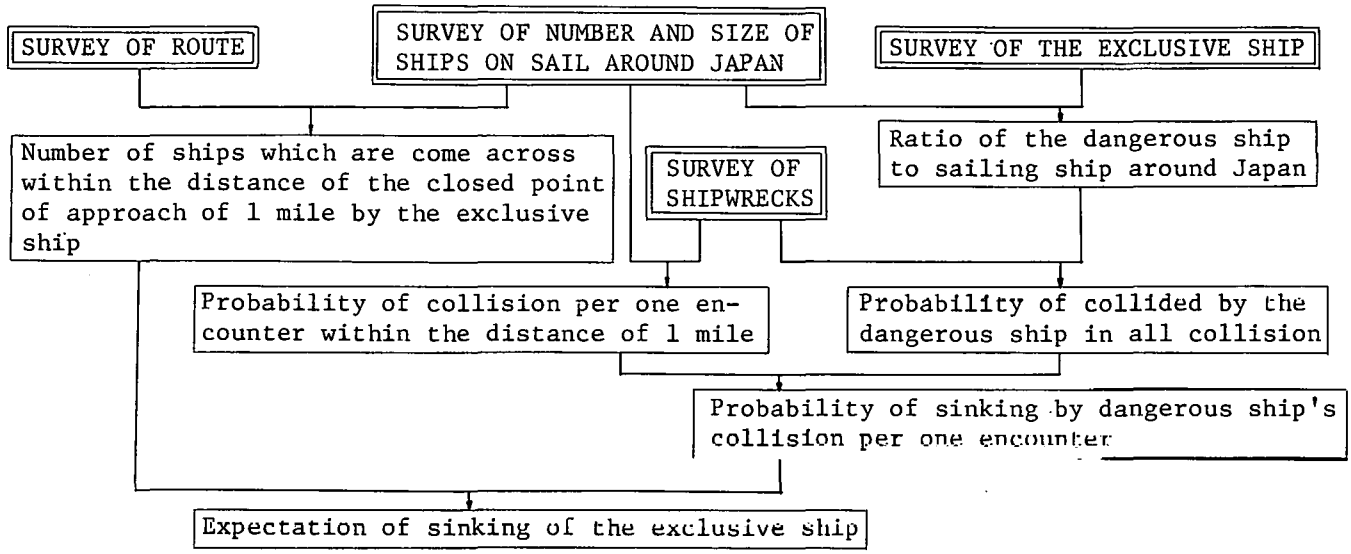


Fig. 1 Evaluation Model of Expectation and Probability of Sinking of Ship

(3) Statistics of ship and shipwreck

For the purpose of the estimation of number and size of ships which pass by the transport ship on the lanes and in order to determine the probability of the loss of the ship, it is surveyed that the shipping statistics such as number of Japanese ships, the number and distributions (along land and offshore direction) of Japanese coastal shipping, and sea tranffic accident statistics, such as kind and number of accidents (1973~1975), tendencies of accidents, and total loss sea accidents including sinking, are determined by classification of the accident type and tonnage of ship.

It is considered that about 10% of all collisions against ships having more collision energy than that of the T-2 tanker are damaged to this ship such as loss of function of ship (total loss accidents) and also, number of sinking is no more than part of total loss accident number. However, because of the lock of available data concerning sinking, it is assumed that all of the ships with total loss accident are sunken in this evaluation, as it can turn result to safety side.

In the types of accidents except collision, there are stranding, fire, accident in stormy seas, collision with an underwater obstruction, or engine failure. However, these accidents don't lead this ship to sink, so that these are excluded in this evaluation.

(4) Probability and expectation of sinking

Based on aforementioned, the incidence of sinking of this ship is determined as follow, when this ship passed by other ships, sinking probability P of this ship per one time of passing by other ship is determined by

$$P = P_1 \times P_2 \times P_3 \times P_4$$

Note: T-2 tankers — gross weight 10,000 tons, displacement 24,400 tons — were the standard World War II American tankers. About 700 were built and quite a few put into service after World War II.

where P_1 is collision probability in the case of this ship passes by other at the distance of the closest point of approach of 1 sea mile, P_2 is ratio of over 10,000 ton vessels in coastal shipping ship, P_3 is the probability of total loss due to collision against over 10,000 ton ships, and P_4 is the probability of sinking when ship is damaged to total loss. Here, this obability is fixed to 1. The expectation of sinking per one voyage, ${}_iE_j$ is,

$$ {}_iE_j = P \times {}_iN_j $$

here ${}_iN_j$ is the number of ships which are passed by this ship at the distance of the closest point of approach of 1 mile, along the assumed routes from Wakasa Bay to Tokai. The resulting expectations of sinking per one voyage are: on a course 5 nautical miles from the promontory, 6×10^{-7} (norhtern circuit) and 1.4×10^{-6} (southern circuit), and a course 10 nautical miles off the promontory, 1.5×10^{-7} (northern circuit) and 3.5×10^{-7} (southern circuit). This would mean that the expectation of sinking at sea would be 1 time in about 70,000~170,000 years for the course 10 nautical miles off the promontory, when the exclusive ship voyages 10 times per year.

(5) Fire accident

It is considered that the most severe fire takes place in the case of collision against an oil tanker. However, the probability for the collision of an oil tanker is far lower than the average collision probability. If a tanker crashes into this ship, the fire may not take place. There is dangerous of fire if this ship crashes into the tanker, but in this kind of collision, the extent of damage of this ship is limited to the bow store-room and the ship usually does not sink.

There are few cases of any effect on the cask store-room of a fire within this ship itself. In the time of fire the casks are sufficiently protected by the Cargo-hold flooding system. In this case too, sinking is highly unlikely.

2.2 Investigation of the Behaviour of Radioactive Material and Evaluation of Environmental Effects during Spent Fuel Shipping Accident at Sea

To determine the release of radioactive materials from sinking cask into sea water and radioactive impact on environment, the model is established as shown in Fig. 2 and is analysed as follow.

(1) Failure of cask and fuel rod

If fuel rod is damaged before the sinking, the ratio of failure of rod is assumed to be upto 1 percent prior to loss at sea. It is considered that two types of failure will be taken place to the cask and sound fuel rods. The first is the loss of sealing function by corrosion. For the cask, corrosion in the crevice between the O-ring packing lid flange and the flange surface leads to the loss of the hermetically sealing function of cask in a relative short time. For the fuel rods, it has been estimated that the fuel cladding tubes are resistant to corrosion for a period of several handred years according to data for the corrosion of zirconium. The second failure is by hydrostatic pressure. If hydrostatic pressure exceeds the limit pressure, this will occur and as a result, radionuclides will leak from cask or fuel rods. Here, the limit depth or the limit pressure is defined as the sea depth to cause failure of cask or fuel rod by hydrostatic pressure, these are determined by the finite element, elastic plastic program and computer.

It has been deduced that Plastic deformation on the evaluated 80 Ton scale casks occurs at critical depth of 2000~4000 meter as a result of finite element elastic-plastic analyses of their resistance to external hydrostatic pressure. It is found that the critical depths, at which hydrostatic pressure causes failure in the fuel cladding tubes, which have been determined for the BWR (7 x 7 Type, 8 x 8 Type) and the PWR (14 x 14 Type, 17 x 17 Type), ry somewhat depending on the type of fuel, and are estimated to be over 5000 years.

(2) Release of radioactive material into sea water

The aforementioned data used as the basis for the calculation of corrosion, etc. is not completely verifiable. Therefore, it is used that the very conservative values of 20 and 100 years as the periods of resistance to corrosion of respectively the cask and the fuel cladding tube in orj postulated release model. The processes and conditions of the release of radioactive material used are given in Table 1.

Using the ORIGEN computing code, the amount and type of radioactive material have been determined within the fuel rods as the function of time.

(3) Ocean diffusion and concentration of radioactivity in sea water

Based on the recent finding on characteristics of ocean environment around the Japan islands and the knowledge on oceanography, the model of ocean diffusion of radioactivity has been established in 3 dimensions, postulating and considering the following conditions.

a) The depth of the sea is uniform and the ocean floor is even.
b) There is a horizontal coastline on one side only.
c) The current parallel to the coastline is predominant.
d) The current has been made as coastline shearing flow which has the velocity distribution in both offshore and underwater directions.
e) Radionuclides are neither sedimented onto nor absorbed by the ocean floor.

For the position of skinking, 3 oceanic regions along the shipping route are considered and the value of parameters to calculate the concentration have been used to estimate the safety side, and decided upon as follows.

Cask sinking points: X_0 = 10 km, Y_0 = 0, and Z_0 = 0 (on ocean floor off 10 km from coastline)

Representative ocean depths: Corresponding to each release case, either 500 or 2,000m depth.

Currents: The three types of currents are used.

Diffusion coefficients: $D_x = D_y = 10^7$ cm^2/sec (Horizontal direction)
D_z = 200 cm^2/sec (Vertical direction)

The concentration of radioactivity in sea is determined as a function of time, situation and nuclide. The computation of concentration is carried out by using computer and three dimensional diffusion analysis program which is prepared for this purpose. The reasons why the depths of 500 and 2,000m were selected are the following. In the former case, the cask will be not damaged at the depth of less than 500m and was evaluated as salvageable. In the latter case, it is assumed that critical depth for the cask's crush by hydrostatic pressure is to be 2000 meter, and this value is very conservative.

(4) Estimation of radiation doses of individual and population

To determine the concentration of radionuclides in sea water used in the exposure estimation, for the individual radiation, the average concentration of radionuclides of sea water is computed for the period of a year at the location with the highest value for the concentration of radionuclides from the surface layer to the ocean bottom and for the population, the average concentration of radionuclides is computed from shore out to sea area of 20 km width and 2,000 km north-south length along the Japan islands. The average concentration is calculated for 1 year at the time with the highest average concentration of radionuclides. From the concentration of a particular radionuclide, the population and individual doses from contaminated fish or seafood consumption are estimated and also the effect of the external radiation from contaminated seashore is considered. The total Japanese fish haul for the year 1974 is used for the seafood consumption.

Table 1 shows the results of the estimates of doses. We have considered the whole body, muscles, kindney, spleen, liver, bones, and lower colon in connection with internal exposure. As the very important matters in this report, we have shown the doses of exposure to the whole body, and to the bones and lower colon which have the most dose among all internal organs. Table 1 gives the values for the doses to children, whose group is expected to be most expose.. The dose of exposure is the highest in release case 2 in Table 1 where the cladding tubes

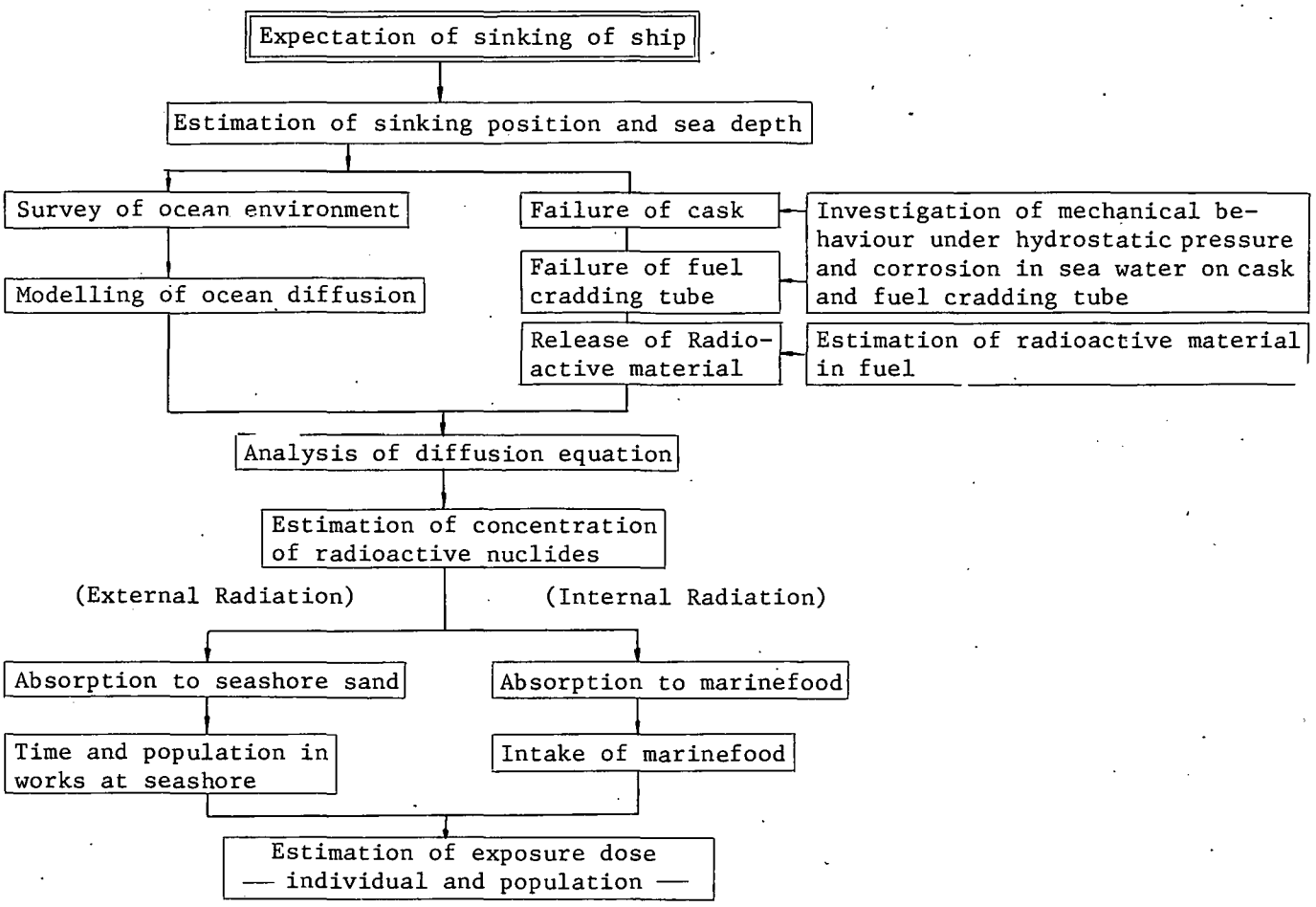


Fig. 2 Model of Release of Radioactive Material and Estimation of Exposure Dose of Individual and Population

corrode 100 years after sinking and there is release from 99% of the fuel rods. Even in the aforementioned release case, if the individual is exposed to the maximum dose of bone radiation with the population, the individual dose amounts to about 10 mrem/year, while population dose amounts to about 6×10^4 man-rem/year.

(5) Evaluation

If the above level of the dose of exposure is considered, the level of individual internal exposure is found to be below 1/100 of the dose of the radioactivity limit 500 mrem/year recommended as safe for the public at ICRP. The population dose of radioactivity, is at the level of 1/100 $\sim$ 1/1,000 of the 10^7 man-rem/year national dose of radioactivity which is due to natural radiation. The external dose of exposure is estimated to be at most less than 1/10 of the internal dose, thus almost negligible. The level of the dose of exposure which served as basis under the conditions investigated in this report is not high and also it is considered that the probability of a sinking accident of the ship is extremely low.

Table 1 Processes of Release of Nuclear Material and Estimated Dose of Individual and Population in Japan

Release conditions: Depth (m)	Damage to: Cask	Damage to: Cladding tube	Leaker[3]	Processes of release, after the time of cask's sinking		Individual dose[4] (mrem/y): Internal radiation: Body	Bone	Lower colon	External radiation: Body	Population dose (man rem/y): Internal radiation: Body	Bone	Lower colon	External radiation: Body
500	no	no	yes	←20 years→	leakage of sea water into cask; release from 1% of fuel rods G[1] / L[2]	0.02	0.2	0.04	0.009	2×10^2	2×10^3	4×10^2	0.4
				←100 years→	cladding tube corrosion; release from 99% of fuel rods G / L	0.9	10	1	0.2	5×10^3	6×10^4	1×10^4	7
2000	yes	no	no	←100 years→	cladding tube corrosion; release from all fuel rods G / L	0.08	1	0.1	0.01	1×10^3	2×10^4	2×10^3	1
2000	yes	no	yes		after cask damage, release from 1% of fuel rods G / L	0.004	0.02	0.02	0.006	9×10^1	5×10^2	3×10^2	0.4
				←100 years→	cladding tube corrosion, release from 99% of fuel rods G / L	0.08	1	0.1	0.01	1×10^3	2×10^4	2×10^3	1

Notes:

1) FP immediately released from fuel rod gap
2) Radionuclides released from pellets by leakage
3) Damage due to pinhole, etc. in fuel cladding tube before sinking at sea
4) Value is for children, as among adults, infants and children, because the latter get the most exposure.

3. Test Program on Full Scale Cask

In support of the aforementioned evaluation and for the safe transport of the cask, the tests of full scale and half scale model of approximately 100 ton cask will be carried out from 1977 to 1982 fiscal year. The test item and schedule are shown in Table 2.

Table 2 Cask Test Item and Schedule

Test item	Fiscal year					Remarks
	1977	1978	1979	1980	1981	
Drop Impact	—	—	—	—		9m falling drop test according to IAEA Transport Regulation
Heat Dissipation		—	—	—		according to IAEA Transport Regulation
Immersion			—	—	—	"
Shield			—	—	—	"
Thermal						"
High Hydrostatic Pressure		—	—	—	—	Maximum pressure will be 300 ∿ 500 kg/cm^2

"MOTOR VESSEL HINOURA MARU"

- A Specialized Carrier for Spent Fuel -

Yukio Mori
Managing Director
Nuclear Transport Service Co., Ltd.
Tokyo, Japan

In Japan, nuclear spent fuels go on piling up and a sharp increase is expected in the near future.

Under the circumstances, their reprocessing has become an urgent matter.

We have been having this reprocessing done abroad, and meanwhile a reprocessing plant in Japan has been completed. Means of transportation from the power stations to the reprocessing plant were investigated and studied from many angles before the implementation.

We examined both land transportation and marine transportation at the first stage taking into consideration road capacity, load capacity of bridges, location of atomic power stations, the fact that Japan is an island country, etc.

Firmly, we had concluded that spent fuel transportation would be compelled to depend on marine transportation.

Meanwhile, CASKs should be in conformity with IAEA standards, but in this standard we cannot ascertain any design criterion for safety in case of sinking in deep water.

Therefore, on inland coastal voyages, to minimize the probability of submersion should be remembered to get the comprehension of the Japanese people.

Under the circumstances, the design of a specialized carrier for spent fuels has been earnestly discussed. Therefore, a special committee of the authorities was set up on the initiative of the Ship Bureau of the Ministry of Transport in February, 1974.

As a result, the committee decided their own criteria on October, 1974 taking into consideration design criteria which were fundamantally based on Rule of Type I SHIP of IMCO and SOLAS Convention concerning an atomic power ship.

Based on the design criteria, reconstruction work of "HINOURA MARU" was started in April, 1977, and completed in July 1, 1977 taking three months, and then she was approved by the Japanese government on July 1, 1977, as the only ship in Japan exclusively used for spent fuel transportation.

1. Main specifications of "Hinoura Maru" are as follows:

Qualified Operating Area		JAPAN SEAS (Great Coast)
Gross Tonnage		1,290 Tons
Length Overall		77.55 M
Length (between perpendiculars)		72.0 M
Depth (to upper deck)		5.85 M
Depth (to erection deck)		7.75
Designed Full Load Draft		4.20 M
Full load Service Speed		abt. 11 Knots
Main Engine		2,500 P.S.
Loaded CASKs		4 Casks (Excellox - 3A or HZ-75T)
Complement	Officers and Crew	... 17
	Owners	.. 2
	Radiation Supervisor	.. 1
	Total:	20

2. Special Structures and Special equipment of "Hinoura Maru"

(1) Special Hull protection structure against collision

In case another ship collided with "Hinoura Maru", the ship's sides are provided with horizontal reinforcing steel plate and the desk is reinforced with steel plates in order that ship's sides can absorb the force of the collision. Simultanously the ship's sides are provided with longitudinal bulkheads to minimize flooding.

This ship is provided with a double hull for safety against collision with another ship.

Also the ship is provided with transversals for safety against foundering even if the longitudinal bulkhead should be damaged and the center should be flooded.

(2) Structure against Grounding

This ship is provided with a double bottom with enough depth (1,460 m/m) to secure the casks from any damage and to minimize flooding.

(3) Strength of Stability

Not only during usual navigation, but even if water should flood the two subdivisions partitioned with transversals due to damage to the outer side-plates, this ship is stable enough not to exceed 25 degree of heel angle.

In addition to the above, this ship is designed so as not capsize on careen of 20 degree more.

(4) Structure of radiation shield

Each cargo hold has radiation shield tanks on the four sides and a radiation shield hatch on the top.

These radiation shield tanks and hatches are installed to limit the crew's exposure level to not more than 0.5 rem a year. We installed about 52 tons of polyethylene and 235 tons of fresh water as the radiation shield materials.

Especially, the fresh water can be automatically supplied from sub-tanks to the main tanks in case the water level is lower than normal. Moreover each tank is installed with a low-level-alarm-device with the alarm in the radiation supervisor's room.

(5) Cargo Cooling Equipment

This ship is provided with two units each of refrigerate equipment and aircooling equipment to keep the temperature under 38°C in the Cargo holds, and with two units of cooling equipment composed of recirculation pump and heat exchanger for the casks water cooling system.

(6) CASK securing devices

To prevent sliding and tumbling of the casks in the cargo holds, the Excellox-3A casks are secured by bolts, and Hz-75T casks are secured by wedges and hooks. These devices are strong enough to withstand rolling of 45° and pitching of 10° and added acceleration of 1G for any distance.

(7) Cargo hold bilge tanks

Two units each of 54 m^3 capacity bilge tank, exclusively used for keeping the unusual bilge water if there should be, are installed.

A special pump of 5 m/h x 37 KW is fitted to be exclusively used for the bilge water tanks.

(8) Navigation Aids

Hinoura Maru is equipped with two radars, an echo-sounding device, an electronic position-fixing device, a position-recording device and a facsimile device for convenience and safety during navigation.

3. "Hinoura Maru's" capital and operation cost

1) Capital

Ship's purchasing cost are Japanese yen 500 millions and reconstruction expense are Japanese yen 1,600 millions and accordingly in total are Japanese yen 2,100 millions.

2) Operation Cost

Annual operation cost are Japanese yen 1,400 millions but this ship is to be operated for 10 years and is to be serviced transportation approximately 120 MTU per year.

4. Radiation Supervisor

As you well know, international rules, for instance, IMCO marine transportation code for dangerous cargo, IAEA rules, etc., require that immediately countermeasures must be taken by a qualified person at the time of accident.

On board it may well be impossible to get any help from shore, and only the people who have professional knowledge of radioactive materials, can prevent radiation injury or hazard. This is the reason why we carry a qualified person, higher than the junior chief technician of radiation on board, which is the rule under the "acts on prevention of radiation injury due to radioactive isotope".

The said qualified person, the so-called radiation supervisor, performs the following services.

1) He gives the necessary advice to the captain for radiation control.

2) He performs radiation monitoring to support the captain.

3) He gives the crew suitable instruction and advice directly regarding the items assigned by the captain.

He performs those services on board using the following equipment.

Film Badge
Pocket dosimeter
Alarm meter
γ-ray Survey meter
Radiation Protection Suites
Decontamination equipment (shower, etc.)

5. Implementation of Spent Fuel Marine Transportation

After such careful investigation, study and construction of the ship, we performed several trial voyages last year and at last we enforced the first voyage, January of this year, loaded with two casks filled up with 2.34 MTU each from Tokyo Electric Power Company, Fukushima Prefecture to Reprocessing Plant, Tokai, Ibaragi-Prefecture.

In 1977 fiscal year, we plan to transport a total of 30.1 MTU of spent fuels by "Hinoura Maru" with seven voyages and there of this ship is to be in service of transportation from nine electric power companies to Tokai reprocessing plant for next ten years.

Thus, we have just started spent fuel marine transportation using our own ship which was built by our hands and know-how. This was an epoch-making for Japanese ship builders as well as nuclear related industries, and we will continue to study and develop safer marine transportation for the future.

STRENGTH AND WATER-TIGHTNESS OF THE CLOSURE HEAD AND VALVES OF A MODEL CASK UNDER HIGH EXTERNAL PRESSURE

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ABSTRACT

This paper describes experimental research on the strength and water-tightness of the closure head and attached valves of a model cask under high external pressure, in simulation of its having been accidentally lost in the deep sea. Both the external pressure tests and the corrosion tests were carried out using scale models of the closure head of an 80-ton spent-fuel shipping cask, and the full size pressure relief valves and drain valves which were to be attached to the actual cask. Based on the results of the above tests, evaluations were made, and new information was obtained on the pressure-proof strength and water-tightness of the closure head of the cask and the valves. Lastly, research which is being carried on in Japan on the pressure equalizer is also introduced.

INTRODUCTION

In Japan, an experimental research program on the safety of spent-fuel shipping casks accidentally lost in the deep sea has been carried on since 1975 sponsored by the Science and Technology Agency (STA), with the finantial support of the Research Fund for the Peaceful Uses of Atomic Energy. The pressure-proof strength of casks excluding the closure head was studied in 1975, and the results have been reported[1] by one of the authors of this paper.

The study described in this paper was the next step in this program and was carried out in 1976. The study deals with the following items on the supposition that the cask has been dropped onto the sea bottom at a depth of 5000 meters:

(1) Pressure-proof strength of the closure head and water-tightness at the closure head seal

(2) Pressure-proof strength and water-tightness of attached valves

The external pressure test and the corrosion test were carried out using the scale models in item (1), and using the typical pressure relief valves and drain valves in item (2).

The results obtained in this study are considered to be very useful in the practical design of casks, and furthermore, in the discussion of the critical failure mode and threshold when the consequences of shipping cask loss at sea are evaluated.

CLOSURE HEAD AND CLOSURE HEAD SEAL

In the ordinary design of the closure head seal of the cask, a gasket is used for the comparatively low internal pressure[2]. However, when the cask is dropped into the sea, external pressure in the direction opposite to the design pressure is applied to it. In this case, a study of the "overall" water-tightness at the closure head seal which contains a gasket is needed.

In this study, an O-ring designed for the external pressure was used as a gasket. In addition to the sealing performance of the O-ring itself, the effects of the external pressure and the corrosion on the all-around water-tightness of the closure head seal were investigated. At the same time, the pressure-proof strength of the closure head plate was also evaluated by both theoretical calculations and actual tests.

External Pressure Test

Specimen and Test Procedure

The 1/4 scale model used for the test is shown in Fig. 1. The model simulates the sealed and bolted closure head of an 80-ton cask of JSME (Japan Society of Mechanical Engineers.) design A study of the main body of the cask is beyond the scope of this paper.

The closure head has shielding lead underneath the mild steel head plate and is attached to the main body by twelve bolts. The head plate has three kind of holes: Two holes for the displacement meters which are to measure the warping deflection at the position of the O-ring, one hole for the pressure gage and one for the moisture detector lead wire. On the surface of the flange of the main body are two grooves. One is for the O-ring and the other is for the moisture wire. The specifications of the O-ring gasket used are as follows:

Type	:	JIS P300-4C
Material	:	Silicon rubber
Shore Hardness	:	Hs = 70
Thickness	:	8.4 mm
External Diameter	:	315 mm
Internal Diameter	:	300 mm

The external pressure was applied step by step to the specimen by using a pressurized water tank. Ten test trials were carried out. In the eighth trial, which was combined with the pressure-proof test, the external pressure was increased up to 49 MPa. In seven of the ten trials, the test was aimed at evaluating the water-tightness at the O-ring seal. Therefore, a supplementary O-ring was set on the internal cylinder wall of the main body to increase the sensitivity of the gage to the internal pressure Pi in the interval between two O-rings. Tests were carried out varying the compressibility of the O-ring by inserting shim plates (type 304 stainless steel) of

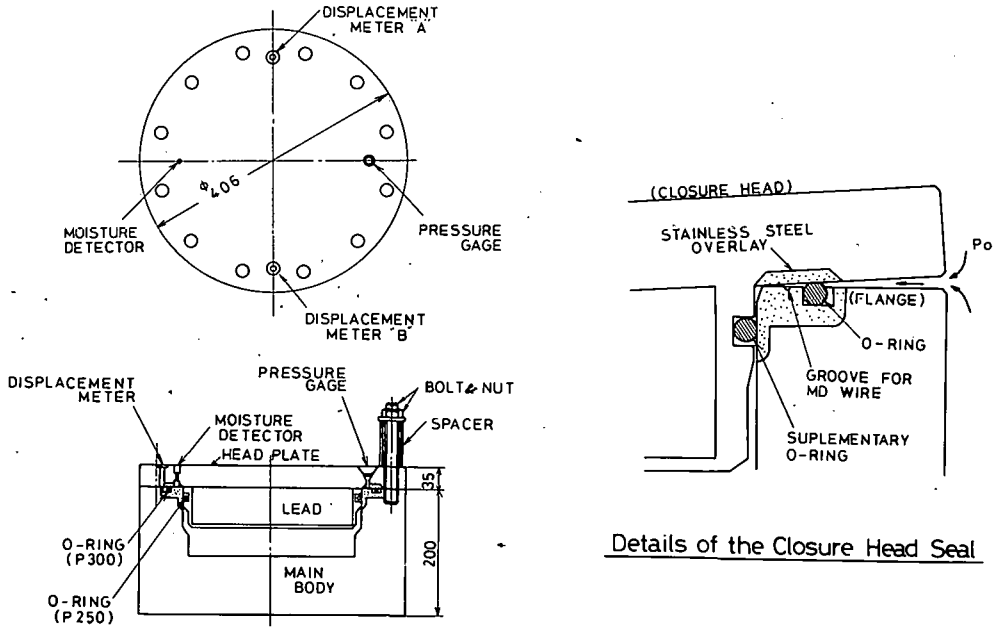


Fig.1. Scale model of the closure head for the external pressure test (dimensions in mm)

various thicknesses between the head plate and the flange.

During the course of the test, the leakage was monitored by the following three techniques:

(1) The Moisture Detector Technique (MDT)
(2) The measurement of the internal pressure Pi
(3) The Acoustic Emission Technique (AET)

As the moisture detector and the internal pressure gage were designed to measure the leakage at the O-ring seal, both sensors were positioned between the O-ring and the internal end of the flange. The acoustic emission sensor was attached to the center of the bottom surface of the main body. These three leakage detectors are sensitive to the following types of leakage respectively:

MD : a small amount of leakage (oozing)
Pi : a large amount of leakage
AE : high velocity leakage

Therefore, the type of leakage at the closure head seal can be known to a certain extent by employing all three of the above techniques. In addition to these techniques, the closure head was carefully disassembled after the test and visual inspection was performed to determine whether or not inleakage of water had occurred inside the main body. The deformation of the head plate was measured by two displacement meters and by plastic strain gages attached at 16 points on the surface.

Water-tightness at the Closure Head Seal

The results regarding the O-ring seal are shown in Table 1. In all trials neither AE nor Pi showed any changes(*), but MD did show significant changes in five trials. This change in MD

(*) In trial X-2, a change in Pi was observed at the stage of decrease in the external pressure.

TABLE I. Test results concerning the water-tightness at the O-ring seal

Sequence number	Test trial	Thickness of shim plates (mm)	Change (corresponding to leakage): MD	AE	Pi	Evaluation of water-tightness [2]	Pct [3] (MPa)	Pmax [4] (MPa)	Traces of damage on O-ring (a) [5]	(b) [6]	(c) [7]	Supplementary O-ring
3	A-1	1.358	O [1]	...	X	oozing	17.7	17.7	0	0	0	used
4	A-2		X	X	X	good	....	17.7	0	0	0	used
5	A-3		O	X	X	oozing	18.0	19.6	0	0	0	none
1	B-1	1.221	O	X	X	oozing	11.2	11.2	X	X	X	used
6	B-2		X	X	X	good	....	19.6	0	0	0	used
10	B-3		X	X	X	good	....	25.5	0	0	0	none
5	C-1	1.129	X	X	X	good	....	19.6	0	0	0	used
2	D-1	0.965	X	X	X	good	....	17.7	0	0	0	used
7	X-1	0	O	X	X	oozing	13.7	13.7	X	X	X	used
8	X-2		O	X	O [8]	oozing	12.3	49.0	X	X	X	none

1) O means "observed" and X "not observed".
2) Evaluation up to Pmax.
3) Critical pressure for oozing.
4) Maximum applied pressure in each trial.
5) Circumferential laceration at the contact point with the edge of the O-ring groove.
6) Damage trace caused by the hole for the pressure gage.
7) Damage trace caused by the hole for the MD lead wire.
8) Observed at the stage of the decrease in the external pressure.

implies that the water-tightness at the O-ring seal has been broken by "oozing". Even in trials X-1 and X-2, in which the compressibility of the O-ring was relatively large, oozing occurred at pressures of 12.3 and 13.7 MPa, respectively. These pressures are equivalent to a depth of about 1200 to 1400 meters.

In the visual inspection of the O-rings after the test, the following three kinds of damage traces were found in many trials:

(1) circumferential laceration at the point of contact with the edge of the O-ring groove
(2) damage traces caused by the hole for the pressure gage
(3) damage traces caused by the hole for the MD lead wire

These traces imply the excessive protrusion of the O-ring. In fact, the MD wire buried inside of the O-ring groove was found to be considerably deformed due to the protrusion of the O-ring.

In all trials, however, the disassembling inspection after the test did not reveal any drops of water inside of the model. Special attention should be paid to the fact that, though oozing was detected in trials A-3 and X-2 out of the three trials (A-3, B-3, X-2) in which the supplementary O-ring was removed, no water inleakage was found. The reason for this sealing action is considered to be the fact that the contact point at the corner of the flange produces the "metallic gasket effect". This metallic gasket effect, rather than the supplementary O-ring, is considered to be responsible for the fact that no drops of water were found inside of the model even in the other three trials in which oozing occurred. According to elastic analysis by means of the contact problem program, the width of the contact area is about 8 mm at an external pressure of 49 MPa. Then, the contact pressure becomes far larger than the yield stress of the material.

Therefore, it became clear that the water-tightness at the closure head seal of the model cask used for the test was sustained due to the sealing performance at the O-ring seal up to about 12 MPa (a depth of 1200 meters), and due to the metallic gasket effect at the internal end of the flange up to 49 MPa (a depth of 5000 meters). This metallic gasket effect is quite normal in the actual cask. Therefore, the overall water-tightness at the closure head seal of this kind of cask is considered to be sufficient up to at least 5000 meters.

Pressure-proof Strength of the Closure Head

The result of the elastic-plastic design analysis on the closure head, neglecting the shielding lead, is given in Fig. 2 (a). Fig. 2 (a) shows the deflection at the center of the head plate vs. the external pressure. It can be seen that the collapse pressure Pcl determined by the ASME method[3] is 49.5 MPa. Fig. 2 (b) shows the maximum strain at the center, which was obtained from the test, vs. the external pressure. It can be seen that the collapse pressure Pcl determined by the ASME method is 48 MPa. The above two values for Pcl are almost the same, so the design procedure in this study is considered to be appropriate.

Fig. 3 shows the warping deflection at the position of the O-ring (measured by two displacement meters) vs. the external pressure and at the same time their calculated relationship according to elastic analysis by means of the contact problem program. The warping deflection becomes an important factor when the compressibility of the O-ring and/or the below mentioned crevice corrosion are discussed. The actual test results and the theoretical calculated ones are roughly the same. The deflection is of the order of about 0.05 mm at 49 MPa.

The theoretical calculations were carried out on the closure head as an axi-symmetrical body by replacing the bolts and nuts with a cylinder which has an equivalent rigidity. This kind of analytical procedure is useful in evaluating the deformation behavior of the closure head of an actual cask.

Corrosion Test

Specimen and Test Procedure

The corrosion test was carried out by using 1/8 scale models similar to the model shown in Fig. 1. A silicon rubber O-ring (JIS P100-4C) was used as a seal. The clearance between the

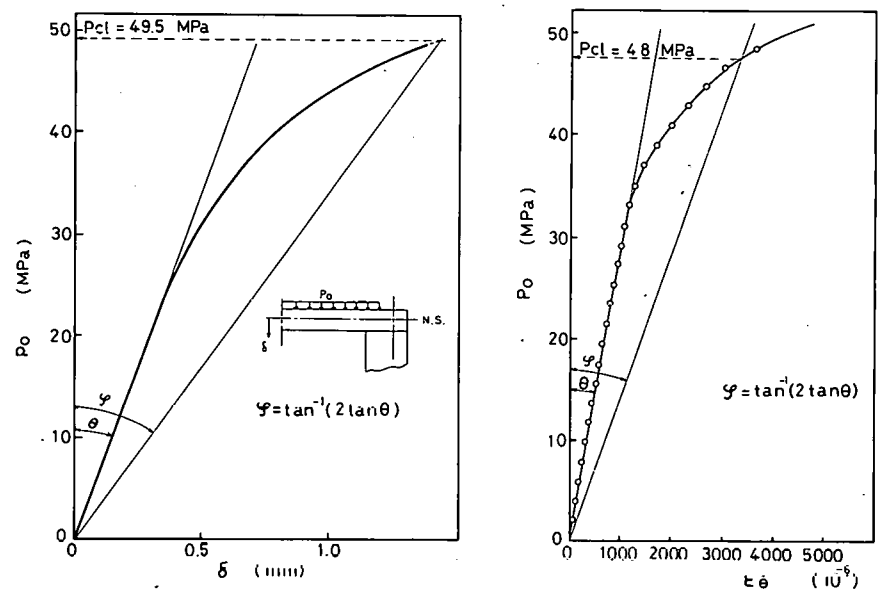


Fig.2. (a) Calculated deflection and (b) Measured maximum strain at the center of the head plate vs. external pressure

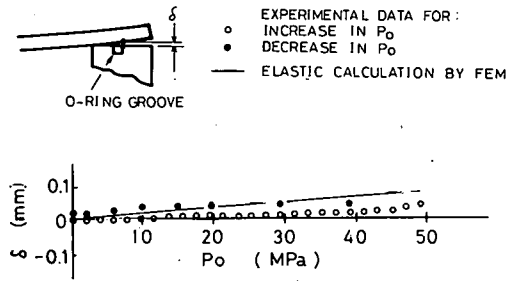


Fig.3. Warping deflection at the position of the O-ring

head plate and the flange were set as 0, 0.2 and 0.8 mm respectively. The head plate and the main body are made of mild steel. Type 304 stainless steel was overlaid around the O-ring seal.

Specimens were immersed in the sea at a depth of 6 meters for about five months and twelve months. After the test, the rust was removed by pickling and the corrosion rate was measured.

Corrosion of Mild Steel

A test result of 0.1 to 0.2 mm/year as the general corrosion rate on the surface of the head plate was obtained. This value is quite similar to the reported corrosion rate of carbon steel in the sea.

Crevice Corrosion at the O-ring Seal

In both the five month and twelve month tests, a kind of groove caused by crevice corrosion was found in the stainless steel overlay around the O-ring seal. Its depth and width increased in proportion to the time of exposure. The corrosion rate Cv in the vertical direction was very small as compared with the rate Ch in the horizontal direction. The corrosion rate Ch in the horizontal direction has a significant influence on the sealing performance at the O-ring seal Fig. 4 shows the average value of Ch which is obtained from the test vs. the clearance. The v

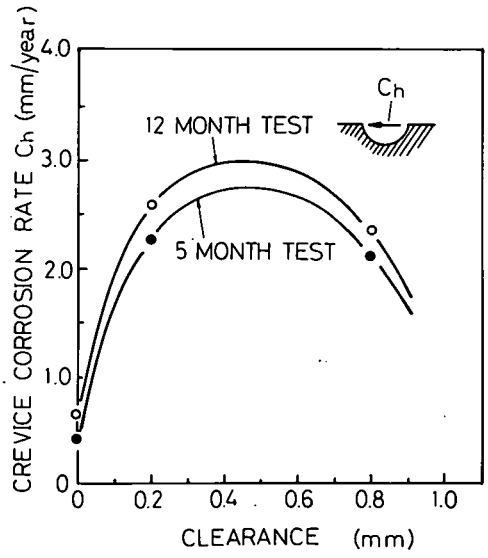


Fig.4. Crevice corrosion rate at the O-ring seal

of Ch is estimated to be of the order of 1 mm/year, when the warping deflection of the closure head at the O-ring seal is far below 0.05 mm as was found in the model cask in this test.

ATTACHED VALVES

When the safety of the cask under high external pressure is evaluated, attached valves must be also taken into account. Therefore, both the external pressure test and the corrosion test were carried out on the pressure relief valve and the drain valve which were to be attached to the actual cask, to evaluate their water-tightness and pressure-proof strength.

External Pressure Test

Specimen and Test Procedure

The specifications of the specimens used for the test are as follows:

(1) Pressure Relief Valve

Manufacturer	:	Circle Seal Products Co., Inc.
Type	:	5120T-2MP-850
Operating temperature	:	-73°C ~ 149°C
Cracking pressure	:	5.9 MPa
Size of connection	:	6.35 mm
Material	:	Type 303 stainless steel
Construction	:	See Fig. 5

(2) Drain Valve

Manufacturer	:	The Worcester Valve Co., Ltd.
Type	:	F44 (Firesafe type ball valve)
Temperature range	:	225°C max.
Valve size	:	12.7 mm
Materials	:	Body - BS 1632 D stainless steel End - BS 1632 GRF stainless steel Ball - BS 970 En 58 J stainless steel Bolt and Nut - Carbon steel
Construction	:	See Fig. 6

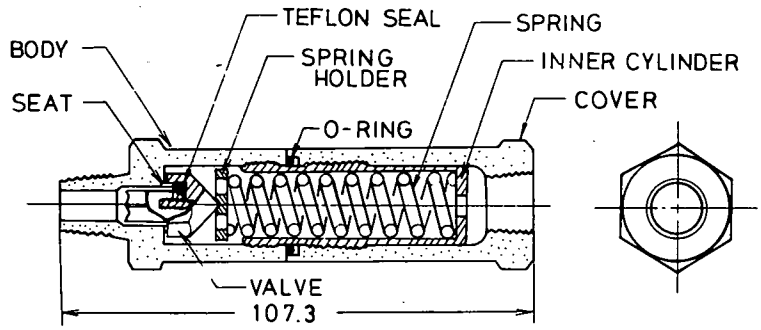


Fig.5. Pressure relief valve

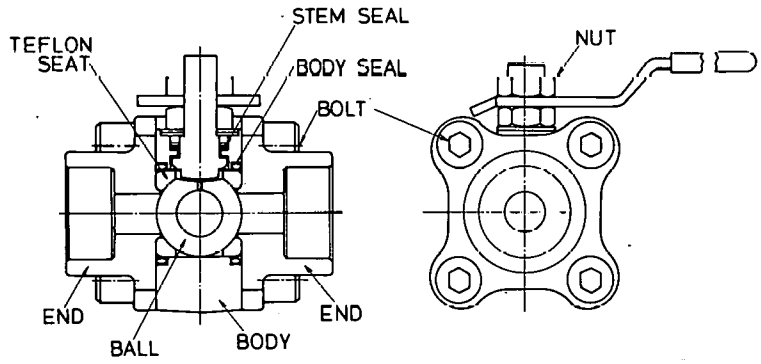


Fig.6. Drain valve

In both the pressure relief valve and the drain valve, teflon is used as the seal. In the former, a teflon gasket is placed between the valve and the seat, and in the latter a teflon seat is placed between the ball and the end.

Nine test trials were carried out by using both new valves purchased from the manufacturers and those valves on which simulated operations had been performed before the test as follows:

		new	used
pressure relief valve		2	1
drain valve	test A	2	1
	test B	2	1

In the drain valve, it is possible for sea water to come through the body seal between the body and the end, or through the stem seal. Therefore, two kinds of external pressure tests, test A and test B, were carried out. Test A was designed to determine the water-tightness against sea water which comes through the ordinary channel. In this case, other possible sources of leakage were completely sealed by plastering with epoxy resin. Test B was designed to determine the water-tightness against sea water from the channels other than the ordinary one. In test B the valve outlet on the sea water side was plugged. In this case, the deformation of the body was also monitored by the strain gages.

External pressure was applied step by step to the specimen up to 49 MPa by using a pressurized water tank. The leakage was monitored by the same three leakage detection techniques as adopted for the closure head seal.

Water-tightness of the Pressure Relief Valve

During the tests, none of the leakage detectors indicated any change. Therefore it was confirmed that the water-tightness at the seat seal of this kind of pressure relief valve is sufficient up to an external pressure of at least 49 MPa, even if the valve has been operated several times. This sealing action is due to the fact that the teflon gasket inserted between the valve the seat produces a sufficient self-sealing effect as a result of its being pushed against seat by the external pressure.

Water-tightness of the Drain Valve

After both test A and test B, water was found inside of the ball as a result of the disassembling inspection. In the course of the test, however, none of the leakage detectors indicated any change. This shows that the sealing performance at the teflon seat on the cask side was satisfactory. As a result, it was confirmed that the water-tightness of this kind of drain valve is sufficient up to at least 49 MPa.

Pressure-proof Strength of the Drain Valve

As the body seal and the stem seal were responsible for the water-tightness to a great extent in one trial out of three in test B, a high differential pressure was applied to the body. As a result, a large strain greater than the yield strain was measured. However, from the evaluation of the strain vs. the external pressure by the ASME method, it was found that the body of the valve would not collapse at 49 MPa. Therefore, it could be said that this kind of drain valve possesses sufficient pressure-proof strength up to at least 5000 meters even if the body seal and the stem seal unexpectedly produce the sealing effect.

Corrosion Test

Specimen and Test Procedure

The corrosion test was carried out using the same kind of pressure relief valve and drain valve as were used in the external pressure test. The specimens were immersed in the sea at a depth of 6 meters for about five months and twelve months. After the test, the specimens were pickled and washed with water, then the crevice corrosion, especially at the seal, was inspected both visually and through a microscope.

Corrosion of the Pressure Relief Valve

Generally speaking, the type 303 stainless steel used in this pressure relief valve is not so resistant to crevice corrosion in the sea. However, as a result of the five month and twelve month tests, no crevice corrosion was found around the teflon seal.

Corrosion of the Drain Valve

In the five month test, no corrosion was found except on the bolts. In the twelve month test, however, crevice corrosion was found at the point of contact between the ball and the teflon seat as well as on the bolts. This fact indicates the strong probability that the water-tightness at the teflon seat will be broken after a short duration of about one year.

In conclusion, the leakage of sea water into the body must be shut out completely in order to maintain the water-tightness of the ball seal, that is, of the teflon seat. Therefore, it can be said that the valve outlet on the sea water side should be plugged, and furthermore, that satisfactory sealing performance both at the body seal and at the stem seal should be ensured.

FUTURE WORK

The effects of corrosion, which is one of the most important factors, were estimated from the results of the five month and twelve month tests. For a more precise evaluation, corrosion tests of longer duration are necessary.

Meanwhile, parallel with this study, a research and development program is being carried

on in Japan regarding the pressure equalizer which is supposed to be attached to the cask. It was pointed out in previous studies[1] that large deformation occurs in the cylinder part or the bottom plate of a cask when it is dropped into the deep sea. The pressure equalizer is one of the effective methods for avoiding such deformation. In this research program, such problems as the operation of the pressure equalizer, vibration, impact and fire are being considered.

ACKNOWLEDGEMENTS

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"U. S. PERSPECTIVE OF TRANSPORTING RADIOACTIVE

MATERIALS BY SEA"

BY: RICHARD B. CHITWOOD

OFFICE OF NUCLEAR WASTE MANAGEMENT

Today I will discuss briefly the reason for the U. S. interest in transportation of radioactive materials by sea. I will also touch on the national and international institutional considerations related to this subject and finally I will touch on some economic aspects in transporting these materials, particularly spent reactor fuel, by sea.

There are a number of reasons why the United States is interested in transportation of radioactive materials by sea from the nuclear power industry. The most compelling being the potential return to the U. S. of significant amounts of spent fuel from non-U. S. reactors. In the report of the Task Force for Review of Nuclear Waste Management (DOE/ER-0004/D, The "Deutch" Report) it was assumed that 10% of the spent fuel discharged from reactors in countries with non-centrally planned economies would be sent to the U. S. Government for storage after a five-year cooling time. An estimate of the number of cask loads and shipping casks that would be required to transport this non-U. S. spent fuel by sea is shown in Table 1. Adequate sea transportation systems must be available to meet this projected need.

TABLE 1

SPENT-FUEL SHIPPING CASKS REQUIRED TO TRANSPORT NON-U. S. SPENT FUEL BY SEA TO U. S. DESTINATIONS*

YEAR	CASK LOADS/YEAR**	CASK REQUIRED***
1980	45	8
1981	45	8
1982	56	10
1983	60	10
1984	72	12
1985	76	13
1986	83	14
1987	87	15
1988	98	17
1989	107	18
1990	118	20
1991	134	23
1992	147	25
1993	167	28
1994	183	31
1995	200	34

*10% of spent fuel discharged from non-U. S. reactors is returned to U. S. after five year cooling

** Cask capacity = 4.5 MTHM

*** Cask capability = 6 shipments/year = 27.0 MTHM/year per cask

United States proposed off-shore nuclear plants will also require shipping systems to service these facilities. Transportation systems for possible sea-bed nuclear waste disposal may be required in the future.

The U. S. has two options for handling the shipment of these casks:

- normal merchant marine commerce
- create one of our own

Experience within the U. S. in shipping large quantities of commercial nuclear materials by sea is presently quite limited. The existence of land-based transportation alternatives for the U. S. nuclear power program has resulted in minimal use of water transport of radioactive materials. The U. S. regulations are predominately land-transportation-oriented. There is little substantive difference in the existing U. S. regulations between the requirements for sea transportation and the requirements for road or rail transportation. Radioactive materials are not required to be transported in special purpose ships and would currently be treated in a manner similar to other hazardous materials. The DOT regulations do, however, apply to both domestic and foreign vessels when in the navigable waters of the United States.

Currently, the existing regulations do not impose design-basis hazards upon radioactive material transportation systems which are specific to the marine environment. It may be determined that sea accidents are sufficiently different from other transportation accidents to justify different design-based criteria. Even in the marine environment, there may be different requirements for inland and coastal waterway systems and the systems which are used to traverse open seas.

Thus, the potential for changes in U. S. regulations and policy is evident.

Potential operational problems exist in developing a U. S. system for transporting spent fuel by sea. These problems include (1) establishing the precise number of casks, and (2) establishing interface design criteria between land, port, and sea system components.

Several spent fuel cask designs which have Nuclear Regulatory Commission approval have been completed, and a few units have been fabricated and are in use for land transport. However, the existing casks and others planned for completion by industry are more than fully committed to domestic-land-based shipping needs. Furthermore, the present cask fabrication capability in the United States is already limited and may become more limited in the future. Current estimates of time required for the development, licensing, and fabrication of a first prototype unit of a new generation shipping cask are between five and eight years. Consequently, planning which will assure that adequate casks are available for sea transportation systems should be initiated now.

Port facilities provide the interface between land and sea transportation systems. These facilities utilized in the transport of large quantities of radioactive materials will undoubtedly differ from existing dock facilities and may require special handling equipment and techniques at dock-side locations. As was stated, a current problem which exists in utilizing systems for transport of radioactive materials by sea is the lack of standardization of interface features. Insufficient attention has been given to assure adequate crane capacities, docking facilities, shielded handling locations, and cask maintenance facilities for nuclear sites utilizing water transportation systems. Interfaces for alternate transportation modes should be established. The recent examination of the practicability of returning spent fuel from the Tarapur reactors in India demonstrates the need for additional interface standardization with sea shipping systems.

At present the U. K. has the only dedicated ships for the international transportation of spent fuel by sea. The Japanese utilize comparable ships, but only for coastal shipping from their reactors for storage at their reprocessing site. The existing U. S. truck and rail casks could probably be adapted to sea transport but this has not been addressed as yet. Operational requirements and potential hazards which may differ from land-based-transportation need to be characterized and assessed. These may include, but are not limited to the following:

1. Hazards such as capsizing, sinking, collisions, and fires which are specific to the sea transportation environment need to be defined.
2. The effect of the ocean environment on cask design should be evaluated in a manner analogous to land transportation systems.
3. Requirements for special ship features such as double hulls, special fire protection, etc. should be determined.
4. The need for accident resistant communication systems to be utilized for locating sunken casks should be assessed.

Up to this point I have attempted to outline the background and nature of the U. S. need for the transportation of radioactive materials by sea. For the remainder of this talk I will focus on the following:

- o The institutions regulating the transport of radioactive materials by sea.
- o The regulatory issues involved within the present institutional framework.
- o The legal environment governing the settlement of nuclear sea transport incidents and accidents.
- o The issues involved with third party liability.
- o Safeguards enforcement considerations.
- o Some economic aspects of radioactive materials transport.

In place today are several institutions which comprise the organizational framework that governs sea transport activities of radioactive materials to and from the United States. The institutional framework includes U. S. and international institutions.

U. S. Institutional Entities

1. U. S. Department of Transportation (DOT)

Congress has assigned to DOT the responsibility for establishing a uniform body of national regulations for the packaging and transport of all hazardous materials, including all nuclear materials. The DOT has published its regulations in Title 49 CFR 100-189, covering all modes of transport (rail, highway, air, and water). These regulations apply to both shippers and carriers of nuclear materials. Within the DOT, the U. S. Coast Guard has the responsibility for monitoring the transport of hazardous materials in U. S. ports, coastal waters, and inland waterways. Coast guard jurisdiction does not extend past U. S. territorial waters.

2. U. S. Nuclear Regulatory Commission (NRC)

Congress has made the NRC responsible for regulating safety in the use of nuclear materials by the nuclear industry, and for issuing licenses to commercial or some Government activities for possession, use, and transfer (including import and export) of nuclear materials. NRC has issued regulations, supplemental to those of DOT, for the packaging of certain nuclear materials (fissile materials and large amounts of radioactivity). NRC field inspectors monitor the activity of its licensees. NRC jurisdiction does not extend to carriers, but is limited to shippers and receivers. To ensure uniformity and harmony between the regulatory agencies, both NRC and DOT (as well as DOE) are parties to a Memorandum of Understanding on their mutual responsibilities.

3. U. S. Department of Energy (DOE)

The DOE is responsible for setting policy and managing federal programs in energy resource development, conservation, and utilization. One of DOE's programs is the overseeing of the shipment of spent nuclear reactor fuel from outside the U. S. to storage facilities in the U. S.

4. Finally, the Fourth US Institution Involved is the American National Standards Institute

The American National Standards Institute (ANSI) is responsible for developing, on a voluntary basis, standards for the packaging, transport, and handling of radioactive materials. Individuals from industry, professional associations, and the regulatory agencies (NRC and DOT) contribute their expertise to the development of standards which are ultimately reviewed and endorsed by NRC and/or DOT. This research procedure is the usual route for standards setting. When voluntary standards cannot be developed in a timely fashion, NRC and/or DOT steps in with the formulation of self-developed standards as a backup procedure.

The International Institutions Include:

1. The International Atomic Energy Agency

On a world-wide basis, the regulations for the safe transport of radioactive materials are based essentially on the recommendations established by the International Atomic Energy Agency (IAEA). All IAEA member states and many world organizations participate in the preparation of these recommendations and these are used as the guidelines for the preparation of individual national criteria and requirements for the transportation of radioactive materials, including transportation by sea.

2. Inter-Governmental Maritime

In addition to the nuclear safety design criteria set by IAEA, the Inter-Governmental Maritime Consultative Organization (IMCO) under the auspices of the United Nations,

defines the guidelines for the conduct of operations for transoceanic shipment of nuclear materials. The various shipping companies and conferences have adopted these recommendations, nd are on their honor to abide by them.

..umerous Institutional Issues Characterize the Sea Transport of Radioactive Materials

Any discussion of this topic must be divided into two areas, the transport of unirradiated materials and the transport of irradiated materials. Both types have been routinely shipped by sea for several decades, but it is the anticipated great increase in the amount of irradiated material - the commercial power reactor spent fuel - that could present some new problems. Thus, the basic institutional question is whether or not the framework that has already been established for the sea transport of small quantities dealt with to date can still apply to the much greater future quantities.

Issues Related to the Above Questions Include the Following:

1. Scope of Responsibilities

The scope of responsibilities held by both U. S. and international institutions are limited to basically setting standard guidelines and design specifications for carriers and packagings. Yet with the anticipated increase in transoceanic shipments of nuclear fuel, numerous new activities should be considered for incorporation in their charter. Among these:

- Broadening the scope of regulatory enforcement to accommodate the control requirements for complete coverage of the sea transport of spent fuel.
- Relating U. S. laws, regulations, and standards to those of other nations.
- Deciding upon the physical security protection requirements for ocean shipment of strategic quantities of special nuclear materials.
- Assuring the continuing availability of ports and waterways for receiving international spent fuel shipments.
- Criteria covering the recovery of sunken highly radioactive cargo.

2. No Enforcement Capability Exists at the International Level

A serious need exists for an international regulatory institution that has the authority to enforce IAEA safeguards and IMCO standards on the high seas, and to take over the responsibilities where individual national jurisdictions end.

Now I wish to Cover Some of the Legal Aspects of Sea Transport of Radioactive Materials. Four Instruments Could Define the Legal Setting to and From the U. S.

1. The First Instrument is the U. S. Price-Anderson Act

In the United States, the Price-Anderson Act deals with domestic third party liability issues. Enacted in 1954, the two goals of Price-Anderson have been: (1) to provide protection to the general public in the event of a damaging nuclear accident, with private and Government funds available for compensation; (2) to encourage the nuclear industry to participate in the commercial nuclear power program by protecting them from the risks of an accident that might create liabilities greater than the gain from their investments.

The Price-Anderson Act was written with land-based nuclear operations in mind, and uncertainty exists as to how it might apply to sea transport activities, especially beyond U. S. territorial waters. This needs to be cleared up.

2. The Second Instrument is the First Convention on Liability signed in Brussels in 1962

The 1962 Brussels Convention which dealt with the operation of nuclear propelled ships has not yet entered into force.

3. The Third Instrument is the Second Convention on Liability signed in Brussels in 1971

The purpose of the convention is to ensure that the operator of a nuclear installation will be exclusively liable for damage caused by a nuclear incident in the course of sea transport of nuclear material. Thus, the shipowners would be exonerated from liability in maritime law for such nuclear damage. The tenets of the 1971 convention are applied only in France, Spain, Denmark, Sweden, Norway, and the Federal Republic of Germany. The U. S. position needs to be determined.

4. Shipping Contracts Constitute the Last Category of Legal Instruments

Within the legal framework set up by the Price-Anderson Act, and two conventions, most import/export shipments of radioactive materials are handled under contract to international freight agents such as Edlow International, Inc., Transnuclear Inc., and Mitsubishi International. Rates and tariffs are set by conferences of the world shipping companies.

Many Issues Remain Unresolved in the Present Legal Framework

Numerous legal issues that are not directly or appropriately handled by the current legal and international treaty environment characterize the sea transport of radioactive materials. Two categories of questions, however, have potentially significant impacts on sea transport. The first category of questions is, "Who is liable in the event of shipping incidents in open waters"? The second category is "What control regime would treat the safeguards aspects associated with nuclear shipments"?

Considering these two questions, legal instruments which deal with these laws that form the basis for two questions may be placed in two groups: (1) laws pertaining to accidents within territorial limits, and (2) treaties pertaining to accidents on the high seas. As the principal features of the legal framework show, the most salient deficiencies lie within the international regime.

1. Key Issues Related to Third Party Liability Questions Include the Following:

- o The question of whether the shipment of spent fuel to the U. S. is covered by the Price-Anderson Act while on the high seas needs to be looked at. As written, it appears to provide this coverage, but it could be argued that storage of foreign spent fuel by the U. S. was not a consideration in the drafting of the Act.
- o Also regarding the Price-Anderson Act, as currently written, it does not provide coverage for a privately owned away from reactor storage facility. Since the Government has solicited industry interest in providing this service, this matter should also be reviewed.
- o Since the U. S. is not part of the first and and the second convention, what legal regime would govern high sea accidents with U. S. flagships carrying foreign fuel on high seas, or domestic accidents involving foreign fuel?
- o Assuming that the first and second convention will have a broad base membership, what legal implications could be expected for the U. S. and the other parties to the conventions on the premise that: (1) the U. S. ratifies one or both of the two conventions; (2) the U. S. is outside the two conventions. Assuming that (2) prevails, how does the Price-Anderson Act interface with the two conventions?
- o Assuming the ratification of the two conventions by the U. S. and other countries, what legal regime would be applicable if damage was caused by transport of spent fuel of a non-member country?
- o Since the second convention established the principle of giving all liability to the nuclear operator (utilities), and to exonerate the shipowner, is the principle acceptable to the U. S. where utilities are privately owned? State owned utilities (such as in France) can provide collateral governmental guarantees.

2. Key Issues Related to Safety and Safeguards Questions Include the Following:

- o What international controls including enforcement should be instituted for safeguarding high seas shipments of nuclear materials?

- o Assuming that an international safeguards agreements could be reached in the future for transoceanic nuclear shipments, which organizations should be made responsible for implementation and execution and enforcement of the agreement? Assuming that IAEA and IMCO could have a leading role, what would be their respective functions and missions in implementing the agreement?

Now I would like to address some of the economic aspects of radioactive material transportation by ship.

Of the two conceivable modes of transportation for transoceanic shipping of radioactive materials, air and ship, airfreight may be competitive for light packages like radioisotopes, but for the new requirements of spent fuel at up to 100 tons per cask, the ship is the only economically feasible means. Within this constraint, it appears that:

- The ship versus airfreight shipment option is not an issue.
- Economy and safety tradeoffs will have to be made between the utilization of many small ships vs a few large ones.
- Economy and safety tradeoffs will also have to be looked at between the use of general purpose vs specialized ships. Today, because of the Japanese requirement for double-hulled vessels to carry spent fuel in their harbors, one highly specialized ship design is already in service. The U. S. will need to look at the advisability of adopting similar requirements for its harbors.

In conclusion, although the transportation of radioactive materials by sea has been going on routinely around the world for many years, as the U. S. becomes involved in the sea movement of large quantities of reactor spent fuel, some new aspects need to be examined and some old ones reexamined. For the sea-going part of the trip, which has been the main focus of this paper, there are many accepted institutions, designs and procedures that can be drawn on. However, as revealed in this discussion, much remains to complete the task.

A final thought. Most trans-ocean trips entail over-land transportation at one or both ends, and it is in this over-land portion of the trip, that is subject to jurisdictions of every city, county and state along the way. Experience to date suggests that we could encounter the most unpredictable and difficult aspects of continent to continent transportation of radioactive materials in the over-land segment of the trip.

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RAILROAD TRANSPORT OF RADIOACTIVE MATERIALS

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A RAILROAD PERSPECTIVE ON TRANSPORTATION OF SPENT FUEL AND HIGH LEVEL WASTE AND RECENT ICC DECISIONS

James R. Paschall
Assistant General Attorney
Chessie System Railroads

The expansion of the nuclear industry into commercial areas, with potentially increased transportation requirements, has generated some legal and scientific disagreements with other industries and the general public. My purpose here is to inform those concerned with the transportation of radioactive materials, particularly spent fuel and high-level waste, of some of the viewpoints of important participants in the movement of these items, the railroads, as they may differ from those of the nuclear industry. I believe legal developments which will have an impact on the manner and cost of this transportation are worthy of note to the participants here. But I think it is also appropriate, in the interest of providing information for the widest range of study and debate, for me, even though not a scientist, to set forth a railroad perspective on some of the technical aspects of this subject, as shown essentially in the railroad position in the recent cases before the Interstate Commerce Commission (ICC).

The United States railroads are perhaps as heavily regulated by the government as the nuclear industry. It was inevitable, then, that disputes as to the manner of rail transportation of spent fuel would be brought before a government agency for resolution. The commercial nuclear industry and the federal agency responsible for government shipments of spent fuel and high-level waste, the former Energy Research and Development Administration, complained to the ICC in early 1976 about certain existing or proposed railroad practices regarding the transportation of spent fuel and radioactive waste materials, including requirements that these commodities be shipped in special trains with no other cargoes.

Among other things, the ICC regulates railroad rates and practices. However, a myriad of other agencies, including the Department of Transportation (DOT) and its many bureaus, also regulate aspects of the railroad industry. This has not only made the disputes as to the manner of spent fuel and high-level waste transportation less easy to resolve in one forum, but it has caused further arguments as to which agency has what jurisdiction to resolve the disputes and to issue regulations.

Simply put, the Eastern railroads, including the Chessie System lines, Consolidated Rail Corporation, and the Norfolk and Western Railway, desire to continue to arrange the transportation of these materials under contract, outside the scope of the many government general transportation regulations, such as those pertaining to the publication of tariffs. This has been the practice of these railroads since the inception of the transportation of these materials. (The Eastern railroads have accomplished this goal of excluding themselves from offering common carrier rates and services for these materials, as they do for most other commodities, by "flagging out" of the tariff offering such rates and services.) In this manner, these railroads hope to tailor each group of shipments to particular operating conditions on their lines,

to maximize safety, and to preserve maximum protection from financial liability in the event of harm caused by the nuclear materials. At least one Western railroad (the Missouri-Kansas-Texas) attempted to change its common carrier position and adopt the Eastern position. ERDA and nuclear industry representatives have attempted to secure ICC rulings that the railroads taking this stance must instead offer common carrier service under published rates and subjec to the same transportation conditions as any other commodity, such as transport in regular trains along with other cargoes, in No. 36312, U. S. Energy Research and Development Administration v. The Akron, Canton and Youngstown Railroad Company, et al., and related cases (the Eastern case) and No. 36307, Radioactive Materials, Missouri-Kansas-Texas Railroad Company (the MKT case).

Although they had held themselves out as common carriers of spent fuel and radioactive waste, the Southern and Western railroads, in early 1976, proposed a tariff amendment which required special train service for transport of those materials. (The Eastern railroads also believe special train service is required for optimum safety in transporting these commodities. This is a key part of their contracts.) The same complaining parties attacked this proposal in No. 36325, Radioactive Materials, Special Train Service, Nationwide (the special train case). The efficacy of special trains was also argued extensively in the Eastern case. As a result of the ICC's unfavorable decision in No. 36325, the Southern and Western lines withdrew the special train requirement. This does not mean, of course, that they agreed with the decision.

I first wish to summarize quite briefly the ICC decisions, then to mention the railroads' thoughts on certain key questions about transporting these materials.

The ICC did not actually rule on the merits of the railroads' special train plan, although an administrative law judge had recommended that such precautions be found unnecessary. In both the special train and MKT cases, the ultimate decision essentially was that the Department of Transportation had jurisdiction over safety requirements for packaging and transportation and that the ICC would not allow the railroads to impose standards which differed from DOT standards or required additional precautions. No court appeals were taken but these lines may be disposed to explore other alternatives inasmuch as they were not satisfied by the decisions, which really did not consider the merits of their plans.

The same administrative law judge who heard the first two cases ruled against the Eastern lines as well. Appeals are pending before a Commission panel. The judge found he could not consider safety (beyond DOT standards) in an analysis of the railroads' duty to offer common carrier rates and services. He then rejected the railroad legal argument that the unusual nature of the commodities and the Eastern lines' previous refusal to hold themselves out publicly to transport spent fuel and high-level waste allowed them to carry these materials under contract requirements, or perhaps to even refuse to transport them. The judge found that the shippers' requests to transport these materials were reasonable and that the railroads contracting practices produced uncertainty and the possiblity of disparate treatment of shippers. Finally, he stated his belief that the railroads' refusal to maintain tariffs offering common carrier rates and services covering the involved commodities hampered the attainment of the nation's energy goals. Since the proper common carrier duty analysis, as well as the merits of the contract requirements, appear to be more directly involved (at least potentially), I suspect the Eastern lines are more likely to appeal an adverse decision to the courts.

Before addressing some of the technical questions raised in the cases, let me comment on erroneous charges that have circulated in nuclear industry publications. A repeated, but incorrect, assertion is that the railroad position is calculated to preserve their coal traffic. Neither the existence nor the level of the special train rates could do this. Nor are such measures necessary to retain this traffic. It should be obvious that existing coal-fired generating plants cannot be converted to nuclear-powered plants nor are these coal-fired plants likely to be abandoned while still useful. Moreover, between 1974 and the end of last year, only the equivalent of 13 nuclear-powered plants of one million kilowatts capacity were ordered while 28 such plants were cancelled and others deferred. (In the same period, 17 such fossil-fueled plants were ordered and 13 cancelled.) It has been reported that due to licensing delays and environmental legal battles it can take over a decade to put an ordered nuclear-powered plant into operation while a coal-fired plant can go from the drawing board into operation

in as little time as three years (with the cost savings this implies). Under these circumstances, railroads have little influence through freight rates, which are but a small fraction of the cost of power generation, over the type of generating plant utilities will construct, as well as little fear of a decline in coal traffic due to construction of nuclear plants. Moreover, despite the recent prolonged coal miners' strike, stated national policy is to increase vastly the use of coal.

Related to this point is the cost of special train service. The projected annual costs of special train service for shipments of spent fuel from a nuclear-powered generating plant would not even approach a utility's annual cost for shipment of coal to a fossil-fueled plant. Even when the cost of moving fresh fuel to a plant is added to the costs of transporting the spent fuel, these costs would not total the transportation costs of coal to a coal-fired plant. It has been estimated that the six to eight shipments of spent fuel to be expected from a nuclear-powered plant in a year would cost no more than $240,000 in annual freight rates in special trains (about double the regular train rate) and probably less than that since the figure is based on an average haul of between 1,000 and 1,500 miles. Moreover, since special trains avoid delays in yards and at intermediate stops, shippers would save cask rental charges because of reduced transit times. Freight charges for coal shipments to most coal-fired plants could amount to several million dollars per year (depending on the plant's size and location). Thus, the special train charge would not influence the type of generating station a utility would build (unless, perhaps, contrary to the electric power industry's position, nuclear plants are less economical than coal-fired plants).

Nuclear industry spokesmen have said that the cost of an average shipment of spent fuel by special train would be about $16,000 greater than by regular train (a seemingly reasonable estimate for discussion) and could amount to $600,000,000 per year in 1986 dollars by 1986 (a most unreasonable estimate). That the latter figure cannot be supported is clear since it would represent 37,500 shipments at $16,000 and certainly many thousands of shipments at any credibly increased figure. Those familiar with the number of existing and proposed reactors, the number of assemblies a reactor requires, and the number of spent fuel assemblies which can be shipped at one time can immediately see the fallacy here. In the Eastern case, a witness for Allied-General Nuclear Services projected an annual movement (eliminating duplications for interline movements) of 463 shipments of spent fuel to Barnwell, South Carolina annually from 45 nuclear-powered plants, a figure which appears generous and may take into account spent fuel already in storage. Certainly, at the high rate of even 10 shipments per year, all 213 existing and proposed nuclear-powered plants (some 156 not yet completed) would take quite a few years to produce the number of shipments entailed in the higher $600,000,000 estimate. Moreover, the estimate undoubtedly does not include the saving of cask rental charges which the reduced transit times of special trains will allow or the fact that the special train charge is the same if two or more casks are shipped at once.

A nuclear industry witness in the Eastern case stated that the cost of special trains for spent fuel shipments had been estimated at six hundredths of a mill ($.00006) per kilowatt hour at the bus bar, hardly a figure which would add significantly to the cost of electricity to the consumer and a more reasonable indicator of the cost of special trains than the 1986 gross projection mentioned above. Given these facts, it is easy to see why the railroads do not accept the contention that the special train charge would be burdensome.

The railroads' primary concern in transporting spent fuel and high-level waste is thus safety. Safety considerations have been debated in all the ICC proceedings, even if the decisions did not, or will not, turn on them.

The Eastern lines' position is based in part upon legal arguments concerning the common carrier obligation of a railroad to haul unusual commodities which the carrier has never held itself out publicly to haul. They believe that properly evaluating these arguments requires the ICC to examine, among other things, the nature of the commodities and the conditions under which they move, including safety considerations. This examination is considered separate from any commitment to other agencies of jurisdiction to set minimum safety standards or to prescribe regulations. The railroads also believe that the Price-Anderson Act and existing insurance provide inadequate protection from financial liability in the event of an accident where harm is caused by the materials being shipped. (This view is based in part on the District Court

decision, now under review, declaring the liability limitation of the act unconstitutional in Carolina Environmental Study Group v. United States, 431 F.Supp. 203 (W.D.N.C., 1977).) Even though these issues were of some concern in the ICC proceedings, however, it seems more appropriate here to go on to address some of the railroads' other viewpoints, other than legal, as to special trains (beyond cost) and the sturdiness of the spent fuel shipping casks, which were brought out in the ICC cases.

The railroads, with their many problems, financial and otherwise, could not have been expected to participate in cask certification proceedings and multi-million dollar studies, although in retrospect, perhaps they should have paid somewhat more attention to some of these matters. This relatively inactive position seems especially justified for the Eastern railroads, however, who were content with their ability to control the transportation environment under contract conditions and thus felt secure in their conservative stance. It is only as the commercial transportation of spent fuel (and later high-level waste) in any quantity has become imminent, and the railroad position has been called into question, that the railroads have been forced to re-evaluate the situation.

Although the railroad position on special trains for transportation of spent fuel is based on operating expertise as well as certain studies and calculations, it is also based on a healthy skepticism, born of critical examination, of the nuclear industry and government position on accident probabilities and the safety of the casks, especially in a regular train environment. It seems reasonable to me that the nuclear industry, which profits through the use of materials generating spent fuel and high-level waste, should bear the affirmative burden of showing that movement of these latter commodities is as safe in regular trains with other cargoes as it is in special trains by itself. The railroads, I believe, should only need to show that due to weaknesses in the industry arguments, and their own knowledge of transportation conditions, the movement of spent fuel and high-level waste can be made safer in special trains than in regular trains. It would seem, after all, that the railroads and their employees are the ones who stand to lose the most in an accident and are entitled to a conservative position until it is conclusively disproved.

As I at least implied earlier, the cost of special train service should not be considered a barrier to that service if safety is increased since it would add an insubstantial amount to the cost of a kilowatt hour of electricity.

The railroads believe that both the frequency and severity of accidents in the regular railroad operating environment differs from the figures and assumptions relied on in prior estimates of accident probabilities and severity advanced by the Atomic Energy Commission (now the Nuclear Regulatory Commission) and the nuclear industry and that railroad accidents are both more frequent and more severe than the latter parties have assumed. For one thing, it seems that the probability estimates were in fact constructed from assumptions, not actual data. One can be skeptical of such methodology since railroad accidents actually occur and presumably can be counted and studied, although, admittedly, in some cases actual data may be difficult to obtain. One reason for this could be because of the method of reporting accidents to the Federal Railroad Administration (part of DOT). For example, only the first cause of an accident is reported. It can be stated with some accuracy that fires will follow collisions and derailments, not usually cause them. Thus, fires will be underreported to the FRA. However, enough can be said about railroad accidents, and the frequency and severity of fires in particular, to show that prior studies of railroad accidents severely underestimated their frequency and severity.

The conventional estimates on the frequency and severity of accidents, including fires, are essentially from Brobst, "The Probability of Transportation Accidents," which is the basis of WASH-1238 and evidently of more recent SANDIA studies and NRC documents and regulations. It should be pointed out initially that the basis for the statistics on accident probabilities in these studies, the number of railroad car-miles travelled in 1969, was erroneously set at 61 billion car-miles, twice the actual number, an error which has apparently never been corrected in official estimates and publications.

Turning to some actual data on fires, it was discovered that at least three accidents involving severe fires, two of which were quite intense and burned for over 24 hours, occurred

in 1976 on the Chessie System lines alone. These lines represent only 3 percent of the nation's rail mileage. According to figures used in the nuclear industry and by the NRC, these accidents would be expected to occur only once in a period of 100 years. An FRA official informally estimated for a railroad witness that the probability of long (over 4 hours) and severe fires was over ten times that given in the Brobst results. As an aside, there does not seem to be hard data on the relationship of fires to train speeds readily available. What can be said is that fires often occur following yard accidents at relatively low speeds. And a recent accident in Arkansas also demonstrates that storage facilities next to a track, as well as lading in the train, can be set afire for long periods following a railroad accident, another risk evidently not previously considered.

Thus, working with what they consider more reasonable data, the railroads estimate that, given expected cask car-miles, by 1990 at least 2 accidents involving casks where severe fires can be expected in regular trains travelling at 50 miles per hour and 10 such accidents with accompanying fires in regular trains travelling at 30 miles per hour.

The railroads are not comforted by claims that shipping casks will not be breached in certain impact and fire situations since these do not cover the most severe types of accidents. Even if these claims concerning lack of a breach in the cask were true, however, it would still seem possible for radioactive materials otherwise to escape from a cask in such accident situations such as through a pressure relief valve or a crack in the closure or a weld. Railroad observers suggest that an escape of cavity coolant, even without a hole in the cask wall, can carry substantial quantities of radioactive material to the outside environment. Additionally, radiation hazards to those in the immediate area of a cask involved in an accident are possible through loss of shielding in a severe fire situation, even if no materials actually escaped from the cask. It has been admitted that lead shielding will melt completely in an intense engulfing fire in 1½ hours, and in slightly longer periods in less intense fires, creating a void which would produce a severe direct radiation hazard even though loss of cask contents would not necessarily be expected.

These less optimistic observations on the probability of release of radioactive materials in an accident situation, and the amount of material likely to be released in such an accident (or radiation likely to be released through a void in the cask's shielding), are based in part on a consideration of actual accident situations more severe than or differing from those used in cask certification proceedings, or for that matter, in recent tests. The railroads believe that consideration must be given to the potential for burial of the cask, the cask's ability to retain internal pressure after severe deformation, its ability to withstand multiple impacts by moving, unevenly surfaced objects, as well as to withstand long and severe fires involving a variety of fuels. (Such weird accident possibilities do occur. One accident in West Virginia sent 90 cars down a ravine, coming to rest six deep in some places. A number of recent accidents has also heightened awareness of the severity of railroad accidents.)

A nuclear industry witness in the Eastern case had found that the NLI 10/24 cask (in a model test) would deform 12 inches at 5 million pounds of force. Such force is easily possible in a railroad accident situation. The railroads were not sure that these tests showed the actual degree of damage because, among other things, ductile aluminum was substituted for the more brittle uranium shielding in the model. Be that as it may, the railroads are unaware of any tests that show the actual cask would maintain internal pressure or that the integrity of welds would be maintained under these circumstances. Moreover, a 12-inch deformation would appear almost certain to breach the fuel rod cladding, leading to the possibility of escape of additional radioactive materials, if these materials were otherwise escaping, and thus increasing the deleterious effects of an accident.

The reliance on second level containment in the properties of the spent fuel and the intact condition of the cladding is not reassuring to the railroads, not just because of the risk to the cladding in the deformation possibility mentioned above, but because of the uncertainty of the information in the area of fuel rod burnup and cladding stability as well as the absence of analyses of fuel rod integrity in multiple impact situations. Indeed, government shipments of spent fuel have required many special precautions in the past for the very reason that there could be fractures of the cladding of fuel elements from shocks in transit or harder impacts from loose switching.

The recent obsolete cask testing program has not convinced the railroads that the casks would be safe in a regular train accident environment or prolonged, severe fire. Problems with these tests not only include the facts that the casks were obsolete and the railroad environment was not simulated but also that the tests were designed to verify calculation models, not to show the integrity of the cask. We have yet to see information on the effect of the tests on the shielding or the amount of internal deformation caused by the tests, although I cannot say, of course, that such information has not been recently released. But more importantly, in the truck-locomotive grade crosssing test, the cask was subjected to a glancing blow directing its flight upward across the locomotive whereas a more direct hit might have better simulated actual conditions as well as giving a better indication of cask integrity and the degree of damage to the cask to be expected upon severe impacts. The fire test tended to confirm that lead shielding would melt, creating a radioactive "window" in the cask, in a severe fire.

The railroads do not believe that the estimates of damage in the event of an accident have been properly computed and not just because they believe it is possible for greater releases of radioactivity to occur in severe accidents. This is an admittedly difficult and speculative area but the commonly-used damage estimate figure of $1,000 per man rem of released radioactive materials appears unduly low. (Even this figure would produce total losses of at least one-half billion dollars under accident conditions admitted possible by the government and nuclear industry witnesses.) The $1,000 per man rem figure was developed as the cost of cleaning up reactor effluents and was once justified as a figure derived for the expected average increase in national health costs due to the consequences of an accident. Since this figure does not comprehend direct costs of cancer and radiation illness treatment, disability benefits, death benefits, evacuation expenses, damage to and loss of use of property, and direct cleanup costs, it does not appear that it can be given credibility for damage estimates relating to transportation accidents. Given the belief that transportation accidents can be more severe and release more radioactive materials than nuclear industry representatives believe, along with the fact that the $1,000 per man rem figure is unduly low, the railroads consider the possible damage from a severe transportation accident in an urban or suburban area could amount to several billion dollars. The number of lives which would be lost, of course, would depend on a number of factors including the weather and how many people were in the immediate area.

I have already mentioned that the railroads consider the costs of special trains reasonable. In order to avoid unduly prolonging this presentation, let me simply catalog the benefits expected from special train service. Added safety is expected to result from special train service due to: (1) reduced speed, (2) reduced switching and less movement in yards where most collisions occur, (3) increased ability to control speed, acceleration and deceleration, (4) far shorter total transit time, (5) reduced mass of the train, (6) the stopping of one train to permit another to pass, (7) the non-inclusion in special train consists of other hazardous materials, particularly combustibles and explosives, (8) special routing to avoid bad trackage, high population areas and freight yards where cars containing other hazardous materials, including combustibles and explosives, may be present, (9) increased opportunity for observation and control by train crews due to the shorter length of the special train, the lower noise level, and lower dust levels, (10) the general tendency of train crews to be alerted to the special need for caution on special trains, (11) the presence in special trains of the best rolling stock, (12) reduced free slack in the train, which lessens shock loads due to run-in and run-out, (13) reduced tendency to cause failure in track structure, (14) reduced likelihood of coupler failure and trainline air hose failure, (15) reduced time of train in grade crossings, (16) elimination of hazards or derailment possibilities caused by other cars in the train, and (17) reduced exposure of railroad employees to radiation present in even a perfectly operating cask. It is significant that the final draft of the Nuclear Regulatory Commission's environmental impact statement, NUREG-0120, at VI-23-5, also recognized certain benefits of special trains for the transportation of spent nuclear fuel and high-level waste including less damage in the event of an accident, less severe impacts, reduced crush forces, less chance of a prolonged fire, decreased derailment possibilities, better surveillance, faster stopping potential, fewer switching mishaps, reduced direct radiation exposure to railroad employees, faster and easier cleanup, and reduced transit time.

The ICC decisions have not necessarily ended the controversies. The railroads are yet to be convinced that their fears concerning the frequency and severity of potential accidents involving spent fuel and high-level waste transported in regular trains are not justified. (G

experience to date is only a small comfort. Once regular movements of spent fuel and high-level waste begin more rail shipments will probably be made in a single year than have been made in the entire history of transporting these materials.) Thus, it may take until at least the next symposium in this series for a report on the final resolution of the legal arguments nd agreement on mutually satisfactory terms for the transportation of spent fuel and high-evel waste between the railroads and the government and nuclear industry. I think we can probably agree, however, that the goal of all involved is the safest possible transportation of spent fuel and high-level waste, which can only be in furtherance of the development of the nuclear industry and of the public interest.

THE GOVERNMENT'S DUTY, AS SHIPPER AND REGULATOR, TO KEEP SPENT FUEL AND WASTE ON TRACK - THE ICC CASES

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PERSPECTIVE

Among the most significant, but least discussed, challenges faced by nuclear energy today is transportation. It is not the type of technological hurdle, like permanent waste disposal, that is familiar to physicists and engineers, and often studied and debated. Yet, because nuclear power has, from its inception, been dependent on an amalgam of resources, starting at the mine and ending at waste disposal, transportation through the fuel cycle has been as necessary an ingredient to usuable nuclear energy as the atom. And like the atom, we have assumed that the wheel will always be with us. Recent events, however, really unrelated to technological progress, have begun to shake our confidence.

Two observations give a clearer perspective of the dimensions of nuclear transportation difficulties as they appear today. First, until recently, transportation problems were routine and were handled by traffic experts in customary negotiations. Today, protracted and complex litigation is the rule. The fact that this conference is now for the first time hearing from several lawyers is itself an omen.

Second, problems in transportation are not the outgrowth of radical or extreme anti-nuclear feeling, but rather the product of the decisions of state and local governments and of major common carriers.

As further background, let us recall that the courts began their involvement in the transportation of radioactive materials when, three years ago, the State of New York sued unsuccessfully to enjoin the air transportation of plutonium.1/ Following this, the City of New York instituted what is tantamount to a ban on transportation of virtually all radioactive materials through the City.2/ The federal government immediately challenged the City in court.3/ That suit is still pending. There then ensued a series of local and state regulations on radioactive materials transportation. Finally, dramatic tariff actions by certain railroads presented a formidable barrier to consistent and economic rail trans-portation of spent nuclear fuel and radioactive waste. Rail transport, is, of course, often the most expeditious and economic mode for these commodities. This sparked three lengthy administrative proceedings before the U.S. Interstate Commerce Commission, collectively involving the major railroads of the Nation. It is this last manifestation of the recent problems in transportation that I will discuss.

THE CONTROVERSY

The controversy, as it has evolved, had, as its principal stimulus, the safety of transporting spent fuel and radioactive waste. Because it gradually evolved as a tariff matter, however, it had originally taken the misleading complexion of an economic, rate-making issue. For some years negotiations of one sort or another had been underway with the railroads to preserve the availability of reliable rail transport for radioactive materials. The Eastern railroads, although they had transported these commodities for the government from the beginning, had done so under a unique tariff arrangement known as a "Section 22 quotation" which was not available to non-government shippers. These railroads had no published tariffs applicable to shippers at large. As the time neared for industry shipments of spent fuel and waste, this absence of tariffs became a matter of growing concern. But, no informal resolution seemed likely.

The Interstate Commerce Commission attempted to provide the impetus for some reconciliation of the matter a few years ago by calling shippers and carriers to a conference to discuss their differences. This failed.

Then came the railroad tariff action that brought the whole matter to a boil: The Missouri-Kansas-Texas Railroad (the M.K.T.) joined the ranks of the Eastern railroads and cancelled, or (in the jargon) "flagged out" of its tariff on the movement of radioactive materials.4/ Although the M.K.T. was a small carrier in comparison to the many others, and only one of many moving materials in its area of the country, its bold action presented an open threat to the entire network of rail transport. If the M.K.T. could cease moving radioactive materials so could any other railroad. Such a precedent would create an impossible situation. It affected the most important element in the reliability and economy of transportation: consistency. Without a consistent network of rail transportation, reliable and economic rail movement is not possible.

At the time of the M.K.T. action, the Western and Southern railroads were already insisting, behind the scenes, on severe and costly restrictions on the movement of spent fuel and waste. This action was then formalized in a tariff, known as the "special train" tariff,5/ filed shortly after the M.K.T. "flag-out." It required that spent fuel and radioactive waste move only in "special trains," that is, in trains dedicated only to that commodity, with separate engine and crew and with its own speed restrictions and other inhibiting features.

These tariff actions were promptly challenged by the U.S. Energy Research and Development Administration, a predecessor to the Department of Energy, by the U.S. Nuclear Regulatory Commission and by major components of the nuclear power industry, including utilities, such as Commonwealth Edison, Duke Power, Vermont Yankee and others, and companies like General Electric, NL Industries, and Allied General Nuclear Services.6/

The time was now also ripe for a resolution of the long-lived concern over the uncertain posture of the Eastern railroads who, as I indicated, had not participated in any common carrier tariffs for moving the involved commodities. ERDA filed a formal complaint bringing this situation as well to the ICC for resolution. Almost coincidentally with this filing, the representatives of the nuclear industry who had become involved in the M.K.T. case and the "special train" case also filed complaints. All complaints against the Eastern carriers were consolidated and handled as one.7/

These ICC cases presented a number of issues for resolution including: the safety of the packaging and movement of spent fuel and radioactive waste, the adequacy of insurance, the comparative safety of regular train versus special train transport, the costs as compared to the benefits of restrictions on the movement of radioactive materials, and last, but not least in importance, the jurisdictional boundaries of the various governmental participants, specifically the ICC, the U.S. Nuclear Regulatory Commission (NRC), and the U.S. Department of Transportation (DOT), in resolving these matters. Before discussing the other issues, I will dwell on this last point for a moment, because it deserves

additional attention from my perspective in government and because understanding it may go some distance toward appreciating the present situation and dealing with it in the future.

THE GOVERNMENT--A REGULATOR AND SHIPPER

Government agencies, so-called sister agencies, do not and cannot always demonstrate the familial character that that word suggests. The business of government has grown so complex, the need for regularity and due process so all-consuming, that federal agencies must deal with one another in quite a heavy, formal fashion. It is with this background that one must view the unfolding of the ICC cases in the full array of adjudicatory or "trial-like" procedures that were brought to bear on this controversy.

Some of the governmental entities principally affected had already assured that the railroads' concerns were unfounded. For example, NRC had promulgated a comprehensive regulatory scheme that amply provided for safe packaging of spent fuel and waste.8/ DOT, in addition to publishing these same NRC packaging requirements in its own regulations, also provided long-established guidelines for the safe movement of these materials.9/ However, because these controversies arose as tariff questions, the ICC had the lead responsibility for examining the railroads' actions and deciding the controversy.

Sorting out the jurisdictional parameters of the affected government agencies then became at once part of the problem and part of the solution. Should the ICC evaluate the reasonableness of the tariffs or lack of tariffs on the economic considerations that are ordinarily associated with such decisions, or should it delve more deeply into the actual justification offered by the railroads for their actions, that is, safety. This was the threshold dilemma in these cases. It was proposed for resolution as the first issue in each case, but was resolved only at the very end of these proceedings and only after the Commission took a second look at the issue on appeal.

During the hearing stage, the ICC permitted the railroads to offer evidence on the question of safety, over the objections of government and nuclear industry participants. This necessitated extensive rebuttal by the latter. And so, the safety issue became the predominant aspect of the evidentiary part of these cases.

Cutting across both the safety and the economic issues was a question of paramount importance to ERDA, and now the Department of Energy, that is, the effect of all this on the nation's energy programs, and more specifically, the effect on a viable nuclear fuel cycle. Also ERDA was involved in these transportation problems even before the ICC litigation began because, as the principle shipper of radioactive materials in its research and defense programs, it had already felt the effects of the railroads' actions. Also ERDA, exercising its responsibility to develop energy alternatives, was concerned about the impact that unreasonable and burdensome transportation conditions might have on the availability of an energy option as significant to the total picture as nuclear energy.

ERDA also brought to the controversy a good deal of understanding and experience in radioactive materials packaging and accident conditions because of the extensive, original work on the subject that has been done to date by Sandia Laboratories.

The confluence then of these several converging governmental interests--safety, energy, and economy--within the framework of a rate-making proceeding added substantial uncertainty and compexity. It also swelled the significance of these cases to alarming proportions. The potential for mischief in these waters was a function of the many currents that had to be channeled into the narrow contours of an adversary proceeding.

SAFETY AND OTHER ISSUES

Because the ICC permitted the railroads to defend their treatment of radioactive materials transportation on safety grounds, the hearings, as I have indicated, dealt principally with the safety of transporting spent fuel casks. The railroads, although

admitting to little acquaintance with the elaborately engineered fuel casks, testified at length about the severity of rail accidents. Their witnesses narrated episode upon episode of rail accidents involving hazardous materials, with fires and explosions, bent metal and crushed cars, but, curiously, none involving spent fuel or damage to spent fuel containers. The railroad witnesses, most of whom were experienced in railroad operations, discussed their concerns about rail accidents, but had little or no knowledge about the construction, testing and licensing of spent fuel casks and about the physical nature of the commodities being shipped as spent fuel and waste. They attacked the adequacy of the conservative testing requirements provided in the fuel cask regulations without fully understanding their severity. I do not doubt these railroad men expressed sincere concern for the safety of hazardous materials transportation. However, their expertise did not require or presume a thorough understanding of these commodities and their shipment. This made it impossible for them to properly assess the minimal risk these shipments presented and the wastefulness of their approach.

Expert witnesses for those protesting the railroads' action, on the other hand, were thoroughly familiar with fuel cask design, construction, licensing and use. Also many had analyzed rail accident scenarios, had evaluated the forces developed, and had demonstrated that such environments were not sufficient to threaten the spent fuel container. What is more, the timely testing program at Sandia, which I alluded to earlier, was studying the predictability of damage to fuel casks in accidents by using computer modeling, scale modeling and full scale testing. This test program provided compelling proof to support the accuracy of such predictions as well as the impressive soundness of the fuel casks themselves. As you have heard discussed in other papers presented here, these tests included, among other things, impacts into reinforced concrete barriers at 60 and 80 miles per hour and a direct collision with a locomotive at like speed. This work bolstered the position urged before the ICC that these casks were fit for ordinary rail transport, and demonstrated, in a fashion easily understood, that the cask design and test specifications developed long ago were more than ample to meet the severe accident conditions of rail transport which prompted the railroads' concern in these cases.

The railroads also argued that the railroad restrictions, regardless of package integrity, offered a substantial measure of additional protection, though at a cost of more than $20 a mile. But, in fact, when the special train method was evaluated in the testimony, it appeared less safe than regular train movements for a variety of reasons. The movement of fuel casks in regular trains, especially when placed at the end of the train, provided additional protection in derailments and collisions since it would be less likely to be involved in these accidents. Also, the irregularity of movement postulated for special trains, such as stopping when passing other trains and reduced speeds, because it deviated from routine practice, presented greater risk of mishap, as well as reduced efficiency.

The railroads indicated some concern over the adequacy of insurance coverage. The testimony demonstrated that this was prompted chiefly by misunderstanding the somewhat complex and unique system of insurance available under the Price-Anderson Act. In any event, the evidence showed that the coverage available was not only quite extensive but also very economical, especially when compared to rates paid by railroads for standard liability and property insurance.

THE ICC DECISIONS

Although I have noted the potential for confusion generated by the multiplicity of government entities brought into the controversy and the adaptation of an economic forum for safety and energy issues, the results in these cases have been favorable. In all three cases, the initial decisions of the administrative law judge who heard the evidence was to reject the railroads actions as unreasonable, unlawful and unnecessary.10/

In two of the cases, the M.K.T. case and the "special train" case, appeals to the Commission have been completed.11/ The third appeal is pending. The results of these

reviews have been affirmations of the initial findings of the judge and of the orders for revocation of the M.K.T. flag-out and the "special train" tariffs.12/ The Commission, upon review, has also determined that, on matters of safety of the transportation of radioactive materials, the ICC will accept the regulatory scheme established by DOT and NRC. The Commission concluded further that it should not have permitted a contest to that regime in these administrative proceedings, and noted that procedures for challenging these safety regulations were available in the agencies responsible for them through rulemaking. This latest development is, in the long-term perspective, most significant because it deals firmly and objectively with the complexity of the government's transportation responsibility. By honoring the divisions of that responsibility, it guarantees consistency and harmony. It is this type of dispassionate and objective outlook that has been absent in so much of the transportation dilemma to-date. Not only the railroads, but even some State and local governments have too narrowly perceived their own interests and concerns and have chosen courses of action which ignore the overriding need for a uniform, national approach.

CONCLUSION

In reflecting upon what has precipitated this costly and time-consuming exercise before the ICC, it appears these questions could have been resolved more simply and less precariously if some of the fundamental differences aired in these proceedings had been fully and forthrightly discussed off the record, before they became fixed points of view. As I see it in retrospect, the very persuasive evidence produced in the hearings on the safety of radioactive materials transportation should have been just as persuasive if discussed informally across the table or at the test site. To put matters of this magnitude into the complex and uncertain process of adversary litigation can at best be a last resort.

FOOTNOTES

1/ State of New York v. NRC, ERDA et al., (S.D.N.Y.) 75 Civ. 2121. This action is still pending, but the District Court has denied the State's requests for injunctive relief on two occasions, September 9, 1975 and May 7, 1976. The Second Circuit has affirmed the District Court on appeal on February 14, 1977.

2/ New York City Health Code, Section 175.111, as amended through January 15, 1976.

3/ United States v. City of New York, (S.D.N.Y.) 76 Civ. 273.

4/ Item 80768-A, Supplement 3, Uniform Freight Classification 12, which became effective February 18, 1976.

5/ Items 80769-A, 80769.5, 81295-B, 81297-B, and portions of Items 80762-A and 80764-A, Supplement 3, Uniform Freight Classification 12, which became effective on March 19, 1976.

6/ The M.K.T. action became the subject of an ICC proceeding, captioned "Radioactive Materials, Missouri-Kansas-Texas Railroad Company," and was docketed as No. 36307. Later tariff amendments flagging out of the movement of fresh fuel and used empty spent fuel casks were consolidated in the same investigation as sub-numbers 1, 2 and 3.

The "special train" tariff became the subject of a separate ICC proceeding, captioned "Radioactive Materials, Sepcial Train Service, Nationwide," docketed as No. 36325.

7/ U.S. Energy Research and Development Administration, et al. v. Akron, Canton and Youngstown Railroad, et al., ICC Docket Nos. 36312, 36312, 36330, 36335, and 36336.

8/ These regulations can be found at 10 CFR Parts 20, 71 and 73.

9/ These regulations can be found at 49 CFR 170-179.

10/ The Initial Decisions were served on April 28, 1977 in the M.K.T. case, on August 24, 1977 in the "Special Train" case, and on December 28, 1977 in the case involving the Eastern railroads.

11/ The appeal decision in the M.K.T. case was served on November 11, 1977. This was reviewed again by the full Commission as a matter of "general transportation importance." The Order resulting from this special review was issued on February 24, 1978.

The appeal decision in the "Special Train" case was served on March 13, 1978.

12/ The M.K.T. Railroad has withdrawn its flag-out and is again participating in the published tariffs. See Supplement 29, Uniform Freight Classification 12, effective April 27, 1978. The Western and Southern railroads have withdrawn the "special train" tariff and will once again move spent fuel and radioactive waste in regular train service. See Supplement 30, Uniform Freight Classification 12, effective May 22, 1978.

PRIVATE SHIPPERS' EFFORTS TO TRANSPORT SPENT NUCLEAR FUEL AND WASTE BY RAIL - THE INTERSTATE COMMERCE COMMISSION CASES

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INTRODUCTION

Three years ago, at about the time of the last International Symposium on the Packaging and Transportation of Radioactive Materials, a number of shippers concluded that the moment had come to challenge railroads' attempts to place unusual restrictions on their carriage of nuclear materials. Several proceedings were commenced before the Interstate Commerce Commission, a forum more familiar to the Nation's railroads than to our nuclear industry. The results to date have been most favorable to shippers, fifteen of which were represented by our firm.

I shall discuss the significance of rail transportation and the ICC proceedings for private shippers, especially electric utilities. The ICC proceedings not only have resulted in favorable decisions but have demonstrated that various parts of the private nuclear industry and concerned government agencies, such as the Department of Energy, can work together (by litigation, if necessary) to eliminate proliferating obstacles to the nuclear option. Other entities, such as state and local governments, are beginning to involve themselves in the radioactive material transportation area. As this happens, joint efforts similar to the ICC proceedings might be commenced.

DESCRIPTION OF THE THREE TYPES OF ICC PROCEEDINGS

To understand the issues before the ICC, it first is useful to review briefly the procedural background of the proceedings: A number of individual cases commenced by separate parties were consolidated into three separate ICC proceedings each involving different railroads and issues. This was because, over the years, rail carriers in different parts of the United States have taken variant positions as to their carriage of radioactive materials.

Railroad tariffs for radioactive materials first were published in 1962. Since then, the Eastern Territory railroads have asserted that they are not common carriers of spent nuclear fuel or radioactive waste materials. These railroads have refused to participate in the uniform tariffs published by the Nation's railroads and effective in other parts of the United States. In other words, the Eastern railroads "flagged out" of the Uniform Freight Classification, the uniform tariff published by the Uniform Classification Committee on behalf of all the Nation's railroads. At the same time, they have failed to publish an alternative tariff, insisting that shippers sign separate contracts for each shipment. ICC proceedings involving the Eastern railroads were commenced when a number of shippers filed formal complaints.1/

In November 1975, a Western carrier, the Missouri-Kansas-Texas Railroad Company (M-K-T), followed the Eastern railroads' lead announcing in a notice to the ICC (or flag-out) that it would no longer hold itself out as a common carrier of spent nuclear fuel or radioactive waste materials. Shortly thereafter, the M-K-T filed similar flag-outs for non-irradiated nuclear fuel and empty containers or cars if used previously to ship radioactive materials. No other railroad in the United States before or since has refused to publish tariffs for fresh fuel or used empty containers. (In fact, the M-K-T's own expert witness testified that the carriage of these materials presented no safety problem.) The ICC's M-K-T proceedings resulted from the filing of protests by shippers.2/3/

The third type of proceeding before the ICC involved the Commission's investigation of the Southern and Western railroads' tariff changes making special train service mandatory for shipments of spent fuel and waste. (These railroads have acknowledged their common-carrier status by their publication of tariffs for various radioactive materials, including spent fuel and waste.) The ICC's investigation followed the filing of protests by a number of shippers.4/

Separate hearings in each of the three types of proceedings were held before the same ICC Administrative Law Judge during 1976 and 1977. The Judge issued initial decisions favorable to the shippers in each of the proceedings, although his consideration of safety and insurance-indemnity issues went beyond what shippers and the Nuclear Regulatory Commission argued was the ICC's jurisdiction. The M-K-T decision was affirmed by Division 2 of the ICC (a panel of Commissioners) last November and by the full Commission this February. Both appellate rulings stated that the Judge had gone too far in considering safety issues, indicating that such matters have been entrusted by Congress to the NRC and the Department of Transportation. The M-K-T cancelled its flag-outs effective April 27th.

The Special Train Service proceeding decision was affirmed in March by Division 2, whereupon the Southern and Western railroads cancelled the mandatory special train provision effective April 14th. No further railroad appeals are anticipated in the M-K-T or Special Train proceedings. Division 2's decision in the Eastern railroads proceedings is expected within the next two months.

PRIVATE INDUSTRY'S NEED FOR RAIL SERVICE

Meeting the Nation's energy needs plainly is one of the most crucial challenges facing us. To lessen our dependence on foreign sources of energy for the production of electricity we must rely more and more on two indigenous resources, coal and uranium. The railroads have more of a stake in carrying the enormous quantities of coal used by power plants.5/ Yet, even they have not contended that rail transportation of radioactive materials is not necessary. What the railroads have attempted to do, however, is to place unusual, if not unique, conditions on the carriage by rail of radioactive materials. If allowed to stand, these restrictions would present significant obstacles to essential transportation links in the nuclear fuel cycle.

Nuclear power plants produce two types of radioactive materials which at some point must be transported from the plant site: irradiated, or spent, nuclear fuel and low-level waste. The latter includes such items as overalls, wiping cloths, tools, resins, filters and miscellaneous compacted trash which have become incidentally contaminated in the operation of the reactor. Most low-level waste now is shipped by truck as it is produced instead of being accumulated at plant sites in train-car size quantities. However, if Eastern low-level burial grounds are forced to curtail their intakes or close, utilities may begin to accumulate larger quantities of low-level waste. Rail shipments might become more practical, especially if the projected shortage of low-level waste shipper containers develops. Rail also would presumably be the preferable mode for shipments of high-level waste, because of the expected weights of casks for such material.

The ICC cases have been concerned principally with the carriage of spent fuel. Whether spent fuel from power plants is to be reprocessed or not, it still needs to be transported at some point from plant sites to temporary or permanent storage facilities. Otherwise, the plants eventually must shut down. The amount of spent fuel needed to be shipped from a power plant in a given year is dependent upon a number of variables including the plant's capacity factor. Spent fuel typically is removed from a power reactor at approximately annual intervals and replaced by fresh fuel. When this occurs about one-fourth to one-third of the core is removed and replaced. For a typical 1000-megawatt reactor, this amounts to about 27-31 metric tonnes of fuel per year. Spent fuel is transferred to storage pools whose capacities vary from site to site.

In March, the NRC released its Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel.6/ This statement contains a more complete description of the Nation's present and projected storage capacity and details about anticipated shipping schedules. Most power plants will have sufficient storage capacity to avoid shipments of spent fuel (other than to nearby reactor storage pools) before the mid-1980's. Many utilities are planning to expand the capacity of their storage pools. But, as reactor storage pools begin to become filled, spent fuel will have to be transferred. Shipments are being planned for now. If transfers are to temporary storage, additional transportation ultimately will be required.

The vast superiority of rail transportation for spent power reactor fuel was totally confirmed by the record in the ICC proceedings. A legal-weight truck container holds less that 0.5 metric tonnes of spent fuel, whereas a rail container can hold over 3 to 5 metric tonnes. In consequence, a utility would need 7 to 10 times more shipments by truck than by rail. Permits sometimes can be obtained for overweight truck shipments, but most states allow such shipments only for commodities which cannot be readily separated. Frost laws in some states restrict overweight shipments to warm-weather months.

Recognizing the advantages of shipping by rail, many utilities already have included large cask handling facilities and railroad spurs in their designs. Other utilities have planned to ship by overweight truck to nearby rail heads. Enormous amounts have been spent for the development and manufacture of rail casks. Witness after witness emphasized the economics - in energy consumption as well as dollars - of using rail casks at both ends of spent fuel shipments. For example, one shipper witness showed that 18 BWR fuel assemblies could be loaded in a rail cask in 1,755 minutes whereas it would take 12,555 minutes to load the same 18 assemblies in the 9 truck casks that would be required.

Limitation to truck transportation would greatly increase the number of miles traveled and the number of people involved in handling the cask. State and local restrictions on highway shipments, such as those which the U.S. Department of Transportation recently has allowed New York City to impose, further increase the desirability of shipping by rail.7/ At this point, the outcome of the DOT's projected rulemaking on routing requirements for radioactive materials is unclear.

THE SIGNIFICANCE OF COMMON CARRIAGE

Private shippers need not only rail transportation but common-carrier service. The Eastern railroads - and more recently the M-K-T - have asserted the right to furnish transportation only on their terms and with the prerogative to refuse to supply transportation at any time for any reason or for no reason. Since 1962, the Eastern railroads have insisted that private shippers sign separate contracts for each shipment of spent fuel. The contracts, which some shippers reluctantly signed after several years of negotiations, contained a number of onerous conditions, such as a complete hold-harmless clause in favor of the carrier and shipper assumption of

the obligation to obtain connecting-line rail service. More recently, these railroads have insisted upon mandatory special train service.

The record in the ICC proceedings demonstrated that shippers cannot rely upon separately negotiated contracts for the rail carriage of spent fuel. Without tariffs filed with the Interstate Commerce Commission and subject to its independent review, shippers are subject to the whims of individual carriers. Under contract carriage, there is no reliability. Advance planning is impossible. One can only speculate about costs, shipping schedules, and how long contract negotiations might take. Special deals with favored shippers would be possible. Such unequal treatment led to the creation of the ICC over ninety years ago. The ICC was established to provide a forum for shipper grievances and to prevent carrier abuses. There should be no unique exception for radioactive materials.

After considering all the evidence presented, the Administrative Law Judge ordered the Eastern railroads to publish tariffs. He found that the railroads "...as common carriers are not free to pick and choose which commodities they will carry in common carrier service and which commodities they will transport as private carriers." He said that he was "inclined to agree...that...insistance on private contracts for each shipment is tantamount to an embargo against the commodities included." If this decision is upheld, shippers will be able to place more reliance upon their transportation projections.

THE BURDEN OF MANDATORY SPECIAL TRAIN SERVICE

While admitting their common-carrier status, the Southern and Western railroads attempted to impose, as an alleged safety precaution, a mandatory special train service requirement upon shippers of spent fuel and radioactive wastes. The Administrative Law Judge and Division 2 of the ICC have rejected this railroad proposal. Had it been allowed to remain in effect, the special train requirement would have amounted to the imposition of a levy on electric utilities and government agencies in amounts measured in hundreds of millions of dollars. Special freight train service tariffs also assure that each participating carrier will be paid for a minimum haul of 110 miles even though the distance carried may be less.

The added costs of special trains to utilities, when applied to both spent fuel and wastes and when escalated to 1986 dollars, could amount to more than $600 million annually by 1986. One witness for an electric utility testified that the special train requirement alone would add one dollar to every customer bill. While the direct effect of these added costs on a utility's decision to construct a nuclear power plant may be small, that hardly could be a justification for wasting millions of dollars especially during a period of increasing energy costs.8/

Railroad witnesses displayed no concern about the enormous added costs of special train service. The primary justification for the increased costs suggested by the railroads were "savings" which might result from "expedited" service. Any such savings are purely speculative because special trains under the applicable tariff provisions are operated by the railroads "at their convenience." Furthermore, if special trains are to be operated with restrictions, such as speed limits and stopping, it is problematical at best that "expedited" service could be achieved.

On cross-examination, the railroads did concede, however, that costs must be considered when evaluating risk. Yet, the railroads spent as little effort attempting to quantify any benefits which might be derived from special trains as they did evaluating the enormous added costs they would have had shippers bear. No railroad witness testified as to the extent by which special trains might reduce the possible consequences, probabilities and risks associated with the transportation of radioactive materials.

In contrast to the railroads' subjective approach, government and private shipper witnesses presented quantitative evidence developed by qualified experts concerning the risks associated with transporting spent fuel by rail. The evidence presented demonstrated that the calculated risk to the public from spent fuel transportation is "thoroughly negligible." The same analysis applied to the transport of radioactive wastes resulted in even smaller risk values.

The Administrative Law Judge concluded that the railroads had "not been persuasive that special trains are safer than regular trains." He therefore found that special train service would be a "wasteful transportation service" not worth the additional costs. The Judge further stated that the record failed to demonstrate that the transportation of radioactive materials in regular train service involves any greater risk than the transportation of other hazardous materials for which no special train service is required.

Division 2 of the ICC affirmed the initial decision but stated that the Judge's inquiry should have been limited to a determination that the shipments involved meet the safety requirements established by the Nuclear Regulatory Commission and the Department of Transportation. Throughout each of the proceedings, the shippers and the NRC had argued that the ICC lacks jurisdiction to sanction the railroads' actions on safety or insurance-indemnity grounds. The Federal regulatory framework governing the safe shipment and carriage of radioactive materials and the insurance-indemnity aspects thereof have been prescribed by Congress. And, Congress has vested exclusive control over those activities in the NRC and/or DOT.

Neither the NRC nor the DOT has found that special trains or changes in the transportation coverage afforded by the Price-Anderson insurance-indemnity system are necessary. The NRC in fact recently has found just the opposite. In 1976, the NRC conducted a rulemaking to consider possible amendments to its insurance-indemnity regulations. The Commission specifically asked for comments as to the adequacy of existing transportation coverage.9/ No comments were filed by any railroad. The NRC then concluded that previously expressed concerns about so-called "gaps" did not warrant changing the existing system by initiating separate transportation coverage.10/

Similarly, the NRC's Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,11/ published in December concludes that special train service "does not appear to be a cost-effective alternative." Consequently, the Statement suggested no changes in the existing NRC transportation regulations. The soundness of this conclusion has been demonstrated by the series of full-scale cask tests conducted by Sandia Laboratories and the Oak Ridge National Laboratory. A recent report prepared by the Congressional Research Service also found that Congressional action concerning transportation of nuclear materials by rail "does not appear to be required at the present time."12/

CONCLUSION

Apparently unconvinced by overwhelming evidence to the contrary, the railroads even now are continuing to attempt imposing unusual restrictions on the transportation of radioactive materials. We understand that the railroads are contemplating filing a rulemaking petition with the Department of Transportation in the near future. The railroads still would subject radioactive materials to treatment much different from other commodities they carry in regular train service, such as propane and chlorine. How they will justify their position remains to be seen.

The ICC cases were commenced only after the railroads refused to settle their differences with shippers of radioactive materials. It was unfortunate that some

agreement could not have been reached. Yet, the adjudicatory proceedings are nearing a conclusion. The shippers thus far have been successful. If other roadblocks appear in the paths of radioactive materials shipments, similar cooperative efforts should be undertaken. We must have a rational and stable regulatory environment for the transportation of radioactive materials. Our Nation cannot afford unreasonable burdens on interstate commerce or on our attempts to achieve energy independence.

Selected References

1/ United States Energy Research and Development Administration v. Akron, Canton & Youngstown Railroad Company, et al., Docket 36312 (I.C.C.); Allied-General Nuclear Services, et al. v. Akron, Canton & Youngstown Railroad Company, et al., Docket 36313 (I.C.C.); GPU Service Corporation, et al. v. Akron, Canton & Youngstown Railroad Company, et al., Docket 36330 (I.C.C.); Commonwealth Edison Company, et al. v. Akron, Canton & Youngstown Railroad Company, et al., Docket 36335 (I.C.C.); and General Electric Company v. Akron, Canton & Youngstown Railroad Company, et al., Docket 36336 (I.C.C.).

2/ Radioactive Materials, Missouri-Kansas-Texas Railroad Company, Docket 36307 (I.C.C.); Empty Containers for Radioactive Materials, Missouri-Kansas-Texas Railroad Company, Docket 36307 (Sub. 1) (I.C.C.); Restricted Usage of Containers and Cars, Non-irradiated Cores, Docket 36307 (Sub. 2) (I.C.C.); Restricted Usage of Cars, Radioactive Materials, Docket 36307 (Sub. 3) (I.C.C.).

3/ In July 1976, another Western railroad, the Illinois Terminal, published its own flag-outs covering spent nuclear fuel and radioactive waste materials. Radioactive Materials, Illinois Terminal Railroad Company, Docket 36307 (Sub. 4). This might have signaled a spate of flag-outs, but the Illinois Terminal cancelled its notices after the ICC refused to defer hearings on that railroad's action pending a determination in the M-K-T proceedings.

4/ Radioactive Materials, Special Train Service, Nationwide, Docket 36325 (I.C.C.).

5/ One study suggests that the railroads' positions have been motivated by their interest in coal transportation: "It would appear that the railroads' efforts to roadblock rail transport of radioactive materials are primarily directed towards increasing coal usage by adding another impediment to the use of nuclear power." Nuclear Assurance Corp., Capabilities of U.S. Domestic Transportation Systems for the Shipment of Radioactive Wastes. (Prepared for U.S.E.R.D.A.), Y/OWI/SUB-77/22330; NAC C-7715 (Sept. 1977) at 33.

6/ NUREG-0404 (March 1978).

7/ The U.S. Department of Transportation Materials Transportation Bureau on April 4, 1978 issued an Inconsistency Ruling (IR-1) which determined that a certain section of the New York City Health Code is not inconsistent with the requirements of the Hazardous Materials Transportation Act or with the regulations issued thereunder. 43 Fed. Reg. 16953 (1978). Similar restrictions have been enacted in Connecticut, 1976 Conn. Pub. Acts 76-231; and elsewhere. See generally W. England, Recent Regulatory Developments Concerning the Transportation of Nuclear Fuel and Other Radioactive Materials, 7 Envt'l L. 203 (1977); and Presentation by W. Brobst, Nuclear Transportation: Problems and Proposed State Regulations, Atomic Industrial Forum Dialogue for State Regulators, Detroit, Michigan (Oct. 19, 1976).

8/ See generally W. Luscutoff and R. Hall, A Safety and Economic Study of Special Trains, Battelle-Pacific Northwest Laboratories (1976).

9/ 41 Fed. Reg. 11327 (1976). See generally Talk by L. Trosten, Transportation Issues, Atomic Industrial Forum Workshop on Insurance, Atlanta, Georgia (May 10, 1977).

10/ 41 Fed. Reg. 40511, 40515 (1976). See 42 Fed. Reg. 46 (1977).

11/ NUREG-0170 (December 1977). See also U.S.A.E.C., Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants, WASH-1238 (Dec. 1972); and, U.S.E.P.A., Transportation Accident Risks in the Nuclear Power Industry, 1975-2000, NSS 8191.1 (Nov. 1974). Cf. U.S.I.C.C., Final Environmental Impact Statement-Transportation of Radioactive Materials by Rail (Aug. 23, 1977).

12/ Cong. Res. Serv., National Energy Transportation, S. PUB. 95-15, 95th Cong., 2d Sess., Vol. III at 53 (Mar. 1978).

PLACING THE SPECIAL TRAINS ISSUE IN PERSPECTIVE

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BACKGROUND

Spent fuel from nuclear reactors has been transported in routine commerce in the U.S. for over 25 years. During this period, no accident has occurred that resulted in the release of radioactive material from a spent fuel cask. It is estimated that about 535 rail shipments of spent fuel totaling 800,000 km (500,000 mi) could be made annually by the mid 1980's. This represents an increase of about a factor of 15 over 1976 levels. Because of these increases, the railroad industry contends that the transportation environment for spent fuel shipments must be controlled more stringently to maintain this excellent safety record. To provide this control over the transportation environment, the Association of American Railroads (AAR) has recommended to member railroads that all spent fuel be handled in special train service. A proposed set of operating practices includes the following restrictions for special trains carrying radioactive materials:

- Speed - Special trains would not move faster than 35 mph.
- Passing - When a special train meets, passes, or is passed by another train, one train would stand while the other moves past at a speed of less than 35 mph.
- No-Other-Freight - Special trains can carry no nonradioactive freight.

On February 18, 1976, the railroads published new rules in their tariffs implementing these requirements. The former Energy Research and Development Administration (now the Department of Energy) and the nuclear industry filed protests with the Interstate Commerce Commission (ICC) seeking cancellation of these new safety oriented rules. In March 1976, the ICC instituted an investigation into the tariff filing. The burden of proof under the Interstate Commerce Act rests with the railroads to show that the tariff changes are just and reasonable.

Testimony was presented to the ICC by the Government/nuclear industry protestants and the railroads. Three separate week-long hearings were held for the purpose of cross-examining shipper and railroad witnesses. The last hearing took place in November 1976. Legal briefs were filed shortly thereafter, and in August 1977 the ICC Administrative Law Judge issued his initial decision in the processing.

In trying to prove their case, the railroads claimed that shippers' studies of cask strength and integrity did not take the very severe railroad accident conditions which the casks might encounter adequately into account. They questioned the adequacy of the cask design criteria in the Department of Transportation (DOT) and Nuclear Regulatory Commission (NRC) regulations, and contended that the cask testing environment was not representative of the real transportation world. Moreover, they argued that any increase in costs due to the use of

special train service, would be offset by lower cask use charges (about $3000/day) as a result of expedited handling and shorter transit times enroute.

For these reasons, and in light of their concerns about adverse publicity and severe economic liability, the railroads argued that the AAR's recommended operating restrictions would reduce their risk in transporting spent fuel and radioactive waste.

The former ERDA/industry position was that shippers would be forced to accept an extremely costly service which was not warranted, was not necessary for safety, and which resulted in unnecessary and wasteful transportation. They claimed that use of special train service would result in large increases in shipping costs relative to normal freight service. The government presented evidence to show that the cost of shipping low-level waste could be as much as 1,600% higher if special train service were required.

The ERDA and the nuclear industry contended that the cask/container systems currently in use and being designed were the best and strongest containers used to transport any hazardous material; that they were designed to maintain their containment integrity even if they were involved in rail accidents. It was argued that the railroads had not shown that there was any significant degree of additional safety in the use of special train service. In fact, it was felt that there may be less safety in some cases, for example, in the case of a head-on collision there would be less cars in a special train to absorb some of the accident impacts.

The ICC's staff investigated the potential environmental impacts of mandatory use of special train service for spent fuel and radioactive waste. They concluded that, approval of the special train tariff would result in 1,160 additional train movements annually. Other conclusions were that the nonradiological impacts of this action would include: (1) the additional consumption of 873,000 gallons of fuel; (2) additional emission of 85 tons of carbon monoxide, 61 tons of hydrocarbons, and 169 tons of nitrogen oxides; and (3) an additional 17 accidents and 1.6 fatalities each year. The ICC went on to say that,

> "With respect to radiological impacts, special train service will, under normal transport conditions, cause an additional 0.81×10^{-3} latent cancer fatalities (LCF) annually (on the average one additional fatality every 1,235 years). Special train service will decrease accident fatalities by 4.4×10^{-4} LCF (on the average one death every 2,272 years). Special trains will increase the environmental risk from the nonradiological standpoint and from the radiological standpoint under normal transport conditions. However, special trains will decrease the environmental risk from the radiological standpoint under accident conditions. In all cases, however, the incremental environmental impact of special trains as opposed to regular trains is very small."(1)

In his initial decision, the Administrative Law Judge stated that he found that the tariffs should be canceled because the railroads had not satisfied their burden of proof, as required by the Interstate Commerce Act, to show that the tariff changes were just and reasonable. He stated also that,

> "The record will support a finding that special trains for the carriage of spent fuel is unnecessary and wasteful transportation, and, therefore an uneconomic burden on shippers and the energy using public."

Understandably, the railroads did not agree with the Judge's findings. They felt that the Law Judge erred in finding that the tariff changes were unjust and unreasonable, and have appealed to the Commission to consider and reverse the Judge's findings. These appeals are largely safety oriented and restatements of their initial position. The former ERDA and other interested shippers have replied to these railroad appeals, contending that the railroads have failed to demonstrate any error in the Judge's decision.

The matter is now in the hands of the Commission and the ICC decision is expected soon. The positions of both the shippers and the railroads are polarized, and follow-on appeals in the courts can probably be expected.

SAFETY AND ECONOMICS OF SPECIAL TRAIN SHIPMENTS OF SPENT FUEL

Pacific Northwest Laboratory (PNL) has undertaken a study sponsored by the DOE to evaluate the potential effects of the special train operating restrictions on the safety and cost of pping spent fuel by train.[2] Since data were not available to compare the safety of special ins and conventional freight trains directly, the potential effects of the operating restrictions were examined using data from the Federal Railroad Administration for freight train accidents that occurred from 1972-1974. Based on this data, evaluations were made on the relative safety of special trains and conventional trains using accident frequency and severity as comparative measures.

Dollar damage to railroad equipment during an accident was selected as the best available measure of accident severity. Increased cask and equipment damage (expressed in terms of dollar damage) does not necessarily represent decreased safety to the general public. Threshold levels of accident stresses (e.g., impact and puncture forces and fire temperature and duration) are required to compromise the integrity of spent fuel casks before any of the contained radioactive materials could be released. Analytical studies, scale model tests and full scale tests of spent fuel casks will survive severe accident conditions.[3] Analyses of accident severity using dollar damage as a basis can provide some insights into the potential safety benefits of the proposed special train requirements. However, the absolute significance of the results should be interpreted with caution because there is no direct correlation between dollar damage and cask integrity.

The potential effect of the speed restriction was investigated by analyzing data on the frequencies of train accidents as a function of train speed and accident severity. Data for all train accidents and for head-on, read-end and side or raking collision accidents only were analyzed. It was found that about 80% of all freight train accidents occur at speeds less than 35 mph. It was also estimated that the accident rate (accidents/mile traveled) for travel at speeds below 35 mph was more than double the accident rate for travel at speeds about 35 mph. This suggests that accident- producing factors are greatest in those portions of the railroad system that have speed limits below 35 mph. Thus, it appears that speed restrictions may not significantly affect the safety of the current shipping system.

Three levels of dollar damage were investigated to determine the effect of the speed restriction on the frequency of involvement in severe accidents. The three levels of dollar damage used - $75,000, $150,000 and $250,000 - represent, respectively, 7%, 3% and 1% of all freight train accidents. It was determined that about 58% of accidents resulting in $75,000 or more damage; about 70% of all accidents resulting in $150,000 damage or more; and 77% of all accidents resulting in $250,000 damage or more, occur at speeds above 35 mph. This indicates that accidents at speeds above 35 mph are more likely to result in high dollar damage than accidents at lower speeds. The 35 mph speed restriction does, therefore, appear to have some potential for reducing the frequency of involvement of a spent fuel cask in high dollar damage accidents. Much of the apparent benefit of the speed restriction in reducing the frequency of severe accidents may lie in the fact that, on the average, more cars are involved per accident in accidents at higher speeds. It was found that the average dollar damage per car is not strongly dependent on the speed at which the accident occurs. This may be partly explained by the fact that only the first few cars involved in an accident experience the impact forces. After initial contact, the relative speed of other cars involved in the accident is reduced.

Data for head-on, rear-end and side-or-raking collision accidents also show that the speed restriction has only a limited potential for improving the safety of spent fuel transportation. It was found that 20% of collision accidents resulting in $75,000 damage or more, 18% of collision accidents resulting in $150,000 damage or more and 23% of collision accidents resulting in $250,000 damage or more occur at speeds greater than 35 mph. Based on this data, imposition of the 35 mph speed restriction could have the potential to eliminate a maximum of about 20% of the most severe head-on, rear-on, and side-or-raking collision accidents.

The effect of the passing restriction was investigated by analyzing data for side or raking accidents. This restriction may be redundant, since on many rail lines, one train is stopped or ving at low speeds on a siding when a passing situation occurs. The restriction appears to ve a potential for improving safety primarily on railroad lines with double main tracks.

These lines presently comprise about 11% of the total length of roadway for Class I line haul railroads. It was found that currently about 71% of side or raking accidents occur with one train standing. The passing restriction, therefore, appears to have a small effect on the safety of spent fuel transportation.

The effect of the no-other-freight restriction was investigated by examining the expected frequency of exposure of a spent fuel cask to fires lasting longer than 30 minutes. This fire duration was selected because as part of the licensing process spent fuel casks must be shown to be capable of maintaining their containment integrity after exposure to a 1475°F fire for 30 minutes. Although exposure to fires of longer duration does not necessarily imply a release of radioactive materials, the licensing standard does provide a basis for comparing the fire hazard in regular train service with special train service. It was estimated that for 1986 shipping levels, a spent fuel cask carried in regular train service might be exposed to a fire lasting longer than 30 minutes once in 1300 years. For casks carried in special train service many of the potential fire sources present in a regular train are eliminated. The casks may still be exposed to locomotive fires and other fires, however. It was estimated that for 1986 shipping levels, a spent fuel cask carried in special trains might be exposed to a fire lasting longer than 30 minutes once in 1600 years. The no-other-freight restriction appears to have only a small effect on the safety of shipping spent fuel by rail.

The results of the safety comparison of the shipment of spent fuel in special trains and in regular train service are summarized in Table 1. The table compares the estimated frequency of involvement of a spent fuel cask in accidents of various types and severities for shipments in special trains and in regular train service. A spent fuel transportation system shipping about 535 rail shipments a total of 800,000 km (500,000 mi) per year in 1986 was used to prepare the table. The basic accident data used in the analysis are summarized in another paper being presented at this conference.(4)

The economics analysis examined the differences in costs between conventional freight and special train shipments of spent fuel. Average shipper costs were calculated for a set of reference shipping parameters. Average shipping costs for conventional freight train service were estimated to be $15.50 per kg U (kg of uranium in the fuel) and $23.50 per kg U if special trains were used. A sensitivity analysis was performed to determine the relative importance of the reference shipping parameters and to evaluate the effects of changes in the basic cost assumptions. It was found that the assumption regarding cask use charges, cask turn-around

TABLE 1. Summary of Comparison of Safety of Spent Fuel Shipments in Regular Trains and Special Train Service

Operating Restrictions	Accident Type	Severity Level	Accident Involvement Interval (Average Years Between Events for 1986 Shipping Levels) Regular Trains	Special Trains
Speed <35 mph	All Types	Damage >$ 75,000	20	40
		Damage >$150,000	40	120
		Damage >$250,000	85	320
Speed <35 mph	Collisions	Damage >$ 75,000	360	400
		Damage >$150,000	470	500
		Damage >$250,000	2000	2100
Passing	Side or Raking	Damage >$ 75,000	870	1100
		Damage >$150,000	3000	3800
		Damage >$250,000	6000	7700
No-Other-Freight	Fire	Duration >30 min	1300	1600

time, average train speed and special train cost charges were the most sensitive in the analysis. Analysis of possible variations in these parameters showed that the cost of shipping spent fuel in special train service could range from 20% lower to 200% higher than the cost of shipping the fuel in regular train service. It was also found that for the reference shipping parameters, the cost of shipping spent fuel in special train service is minimized if three spent fuel casks are shipped in each special train.

The results of the study indicate that transportation of spent fuel in special trains under the operating restrictions proposed by the AAR has the potential to substantially increase the costs of spent fuel transportation. Although special trains could offer economic or logistic advantages for some shippers, analysis of available data does not indicate any substantial advantage of the special train operating practices from a safety standpoint. Although the estimated accident frequencies summarized in Table 1 indicate a small decrease in the frequency of involvement of spent fuel casks in accidents if special trains are used, the accident frequencies for regular train service are already quite low and the small improvement indicated for special train service does not appear to justify the potential increase in shipping costs.

POSSIBLE ALTERNATIVES TO THE CURRENT SITUATION

The special train issue has placed the railroads and shippers of radioactive materials in an adversary relationship. Since movement of radioactive materials by rail is necessary for the efficient functioning of the nuclear fuel cycle, it is imperative that possible methods to resolve the special train issue be explored. It does not seem likely that continuation of the current posture of legal and quasi-legal confrontations between the railroads and shippers will produce the efficient transportation system that the nuclear industry and federal government require. A variety of actions could be taken by the government, the nuclear industry and the railroads to improve the current situation. Several of these actions are discussed in the remainder of this paper.

The railroads have raised many questions about the safety of radioactive material shipments by train. To nuclear industry people experienced in the design, testing and operation of radioactive material shipping packages, it is clear that the packages being used provide a significant margin of safety. Although the safety issues raised by the railroads appear to represent an honest concern for protecting their equipment and employees, some of their questions reflect a lack of understanding of the basic safety concepts involved in cask design and licensing. More participation by the railroads in planning and performing safety studies conducted by DOE and other governmental organizations should clear up some of these misunderstandings and give DOE and the nuclear industry a clearer picture of the specific problems involved in rail transport.

Before significant improvements can be made in the relationship between the railroads and the shippers, it is necessary to open channels of communication to develop a mutual awareness of the issues of primary concern to each group. A forum for discussions between the railroads and shippers could be established by the Department of Energy (DOE) and industry trade groups. The concerns of each side must be understood by the other before meaningful solutions to the controversy can be developed. There are also misconceptions on both sides of the issues that could be identified and dealt with in such a forum. The need, desirability and scope of a cooperative DOE/nuclear industry/railroad research and study program could be explored.

Individual shippers or groups of shippers should be encouraged to examine the economic and logistic parameters associated with their shipments of radioactive material by rail. This analysis should include consultations with railroad personnel and the receivers of the shipments. These studies would identify the type of railroad service that would provide the best utilization of nuclear industry and railroad equipment, minimize the cost of transportation services and optimize other logistic considerations of the shipper and receiver.

Utilities, owners of fuel cycle facilities and architect-engineers should involve railroad operations specialists during the siting and design phases of a new nuclear facility. Advanced logistics planning that includes participation by the railroads can improve the facility operation and promote cooperation between shipper and carrier, thereby improving the quality of service.

Mutual benefits may result from shippers and railroads working together to innovate a range of service options and appropriate rate structures for the shipment of different types of nuclear material. Such a variable service and rate structure may require some changes in the present tariff structure, but it could provide the shipper with additional flexibility to optimize system logistics and furnish the railroads with opportunity to employ operational procedures and practices which could alleviate some of their perceived concerns about the safety of nuclear shipments.

For example, the shipper's service needs for spent fuel are quite different from his needs for low level waste shipments. For spent fuel the shipper needs rail service that provides minimum delays in pickup, transit and delivery of the shipment to minimize his total shipping costs. Some form of expedited service at appropriate rates could meet these service needs. The extra attention that the shipment would be afforded by the railroads to provide this expedited service could provide the railroads with the extra control they desire over the shipment during transit. For shipments of low level waste, the most economically and logistically advantageous option for the shippers may be regular freight movement.

In summary, we believe that regardless of the final outcome of the current legal battles, the commonly desired goal of safe and effective rail transportation services for radioactive materials will not be achieved unless more positive approaches to the problem are also pursued. There appears to be a middle ground between the currently polarized views of the railroads and the nuclear industry that could provide a mutually satisfactory solution to this issue.

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ANALYSIS OF FREIGHT TRAIN ACCIDENT STATISTICS FOR 1972-74

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INTRODUCTION

Public concern for the safety of rail shipments of spent fuel and other radioactive materials is widespread. This concern relates to both the frequency and the possible consequences of transportation accidents. An analysis of accident data has been made to assess some of the factors that affect the frequency and severity of freight train accidents. The data are on computer tapes furnished by the Office of Safety of the Federal Railroad Administration, and include information about freight train accidents on Class I line-haul railroads for the three-year period 1972-74. Information on the tapes includes:

- Identification of railroad involved
- Location and date of accident
- Accident class and subclass
- Speed of train
- Reported cause of accident
- Dollar damage to railroad property
- Dollar damage to track or roadbed

For the time period studied (1972-74), freight train accidents are defined to include all occurrences associated with the operation or movement of trains, locomotives, or cars that result in damage to railroad equipment, track or roadbed in excess of $750.[a] Accidents are classified as collision, derailment, and other train accidents. Collision accidents are subclassed by the type of collision (rear-end, head-on, etc.), and derailments are subclassed by the cause of the derailment.

The dollar damage criterion used to define train accidents resulted in many grade crossing accidents being eliminated from consideration in the study. Grade crossing accidents which result in less than $750 damage to railroad property are defined as train-service accidents. They are summarized in an annual publication of the Federal Railroad Administration.[1] Grade crossing accidents that result in at least $750 damage to railroad property are classified as other train accidents unless a derailment occurs, in which case they are classified as derailment accidents.

[a] In 1975 the minimum dollar damage value for a reportable train accident was increased from $750 to $1750.

ACCIDENT FREQUENCIES

Basic freight train accident data for the years 1972-74 are shown in Table 1. Freight trains belonging to Class I line-haul railroads traveled 1.39 billion miles during this period and were involved in 17,153 reported accidents for an accident rate of 12.3 per million freight train miles. Of these accidents, 83.8% were classified as derailments, 8.4% were classified as collisions, and 7.8% were other train accidents.

The number and percentage of freight train accidents as a function of train speed are shown in Table 2. Most accidents occur at low speeds. About 75% of all accidents occur at train speeds below 30 mph and less than 2% occur at speeds in excess of 60 mph. There may be several reasons why this is so. One reason may be related to total train miles traveled at various speeds. Unfortunately, the distribution of freight train speed as a function of track mileage is not available. However, an estimate made for a representative western route indicated that about two-thirds of all freight train miles are traveled at speeds less than 35 mph. Another reason for the relatively high incidence of freight train accidents at low train speeds could be that accident-inducing factors are greatest in those portions of the railroad system that have speed restrictions. Speed limits are normally associated with rail yards, bad track, curves, bridges, grade crossings, populated areas, and with adverse weather conditions.

Table 3 shows the distribution of freight train involvement in collision accidents by sub-class and train speed. Approximately 63% of all freight train collisions were switching accidents. Most switching accidents occur in rail yards with trains traveling at very slow speeds. Seventy-nine percent of freight trains involved in switching accidents were reported to be traveling at speeds in the 0 - 9 mph range. Accidents that are classified as crossing collisions or as collisions with cars not in trains would also likely occur in rail yards at low train speeds.

In terms of damage to rail cars and equipment, most interest in transportation risk studies has centered on rear-end, head-on and side-or-raking collisions. These accidents accounted for 19% of the trains involved in collision accidents during the three-year period, 1972-74. Of the 271 trains involved, 88% were reported to be traveling at speeds less than 30 mph and only 1% were reported to be traveling at speeds greater than 60 mph at the time of collision.

An important accident statistic for which the data were not available on the accident tapes for 1972-74 was the number of times that a train accident resulted in a fire. Regulations require that a fire be reported only if it is the cause of an accident. The inclusion of descriptive information about train fires is optional with the railroad reporting an accident. A recent study on severities of transportation accidents[2] estimates that fire occurs in about 1% of all train accidents.

TABLE 1. Basic Freight Train Accident Data for 1972-74

Year	Freight Train Miles[a]	Number of Freight Train Accidents				Accidents per Million Freight Train Miles
		Collision	Derailment	Other	Total	
1972	451,032,000	456	3,760	457	4,673	10.4
1973	469,122,000	534	5,050	447	6,031	12.9
1974	469,728,000	446	5,571	432	6,449	13.7
Cumulative 1972-74	1,389,882,000	1,436	14,381	1,336	17,153	12.3
Percent		8.4	83.8	7.8	100.0	

[a] Yearbook of Railroad Facts, 1975 Edition, Association of American Railroads, Washington, DC.

TABLE 2. Number of Freight Train Accidents as a Function of Train Speed, 1972-74

Train Speed (mph)	Collision Accidents		Derailment Accidents		Other Accidents		Total Accidents	
	Number	Percent	Number	Percent	Number	Percent	Number	Percent
0 - 9	924	64.4	5,098	35.5	206	15.4	6,228	36.3
10 - 19	285	19.8	3,610	25.1	156	11.7	4,051	23.6
20 - 29	104	7.2	2,380	16.6	237	17.7	2,721	15.9
30 - 39	51	3.6	1,462	10.2	244	18.3	1,756	10.2
40 - 49	39	2.7	1,111	7.7	247	18.5	1,397	8.1
50 - 59	16	1.1	518	3.6	164	12.3	698	4.1
60 - 69	12	0.8	178	1.2	72	5.4	262	1.5
70 - 79	1	0.1	19	0.1	8	0.6	28	0.2
80 - 89	0	-	2	-	0	-	2	-
Unknown	4	0.3	3	-	2	0.1	10	0.1
Totals	1,436	100.0	14,381	100.0	1,336	100.0	17,153	100.0

ACCIDENT SEVERITY

Several studies[3,4] of the risk from transportation of radioactive material have used train speed as a measure of accident severity. Accident forces are expected to increase as speed increases. However, accident severity is not a function of vehicle speed only. Other factors such as the type of accident, the kind of equipment involved, and the location of the accident can have an important bearing on accident severity.

The most direct measure of accident severity is in terms of the physical forces produced in an accident. However, measurements of accident forces are not routinely available. A possible measure of accident severity, and one for which data are available, is total dollar damage to railroad equipment per accident. Property damage is assumed to be a monotonic function of the magnitude of the physical forces generated by an accident. Thus dollar damage becomes a measure of accident severity. It is recognized that the relationship between dollar damage and accident severity is not necessarily linear and the functional relationship is not well-defined.

TABLE 3. Distribution of Freight Train Collision Accidents by Sub-Class and Train Speed, 1972-74

Train Speed (mph)	Rear-End		Head-On		Broken-Train		Side or Raking		Crossing		Trains with Cars Not in Trains		Switching		Not Elsewhere Classifiable	
	Number	Percent	Number	Percent	Number	Percent	Number	Percent	Number	Percent	Number	Percent	Number	Percent	Number	Percent
0 - 9	39	34.8	18	29.0	11	8.2	69	71.2	2	33.3	25	51.0	722	79.2	38	58.5
10 - 19	41	36.6	25	40.4	27	20.1	14	14.5	2	33.3	13	26.5	147	16.2	16	24.6
20 - 29	17	15.2	9	14.5	35	26.1	8	8.2	1	16.7	4	8.2	24	2.6	6	9.2
30 - 39	10	8.9	7	11.3	19	14.2	3	3.1	0		5	10.2	6	0.7	1	1.5
40 - 40	2	1.8	2	3.2	21	15.7	1	1.0	1	16.7	2	4.1	7	0.8	3	4.6
50 - 59	1	0.9	1	1.6	12	9.0	1	1.0							1	1.5
60 - 69	2	1.8			7	5.2							3	0.3		
70 - 79							1	1.0								
80 - 89																
vn					2	1.5							2	0.2		
;	112	100.0	62	100.0	134	100.0	97	100.0	6	100.0	49	100.0	911	100.0	65	100.0

Several factors limit the usefulness of dollar damage information as a measure of accident severity. At present there are not industry-wide criteria for determining dollar damage. Different railroad companies may produce different damage estimates for identical accidents.

Dollar damage to railroad property incurred in a particular accident is partly a function of the type of equipment involved. For freight train accidents for the year 1972-74, the average dollar damage for head-on collisions was 3 times greater than the average dollar damage for rear-end collisions and 8 times greater than the average dollar damage for all collision accidents. In part, this is because several locomotives are involved whenever a head-on collision occurs. The age of railroad equipment can affect dollar damage estimates. There is presently no established procedure for depreciating equipment which is damaged after several years of service.

An additional factor limiting the usefulness of total dollar damage figures in assessing accident severity is the variation in the number of freight cars involved in an accident. The average number of freight cars involved in an accident is a function of train speed and increases in an approximately linear fashion as the speed increases.[5,6] Average dollar damage per car could be used as an indicator of accident severity in lieu of dollar damage per accident. Prior to 1975, however, railroads were not required to report the number of cars involved in an accident and this information was not available on the FRA accident tapes for 1972-74.

For this study, dollar damage to railroad equipment per accident appeared to be the best available indicator of severity. Levels of dollar damage chosen for analysis included $750 (definition of a reportable accident), $75,000 and $250,000 respectively. The last two levels included approximately 7% and 1%, respectively, of all reported freight train accidents for 1972-74. There was no attempt to relate dollar damage to the severity of accidents that might breach a rail cask used for carriage of radioactive material.

Table 4 shows number of accidents as a function of train speed for the three dollar damage categories. The speed distributions for these accidents are shown graphically in Figure 1. Figure 1 shows that some correlation exists between train speed and dollar damage per accident. Accidents at high speed are more likely to result in high dollar damage than accidents at low speeds. About 1.0% of all accidents at speeds below 20 mph resulted in damage to railroad property in excess of $75,000. Almost 22% of all accidents at speeds above 60 mph resulted in more than $75,000 damage. Less than 0.2% of all accidents at speeds below 20 mph resulted in damage to railroad property in excess of $250,000. Almost 9% of all accidents at speeds above 60 mph resulted in more than $250,000 damage.

TABLE 4. Number of Accidents as a Function of Train Speed for Various Dollar Damage Categories (Freight Train Accidents on Class I Line Haul Railroads, 1972-74)

Train Speed (mph)	>$750 Damage		>$75,000 Damage		>$250,000 Damage	
	Number of Accidents	Percent of Accidents	Number of Accidents	Percent of Accidents	Number of Accidents	Percent of Accidents
0 - 9	6,232	36.3	28	2.4	6	2.6
10 - 19	4,052	23.6	81	6.9	9	3.9
20 - 29	2,724	15.9	218	18.5	20	8.7
30 - 39	1,758	10.3	310	26.3	42	18.3
40 - 49	1,397	8.1	312	26.4	79	34.5
50 - 59	698	4.1	165	14.0	47	20.5
60 - 69	262	1.5	60	5.1	24	10.5
70 - 79	28	0.2	3	0.3	1	0.5
> 80	2	-	1	0.1	1	0.5
Totals	17,153	100.0	1,178	100.0	229	100.0

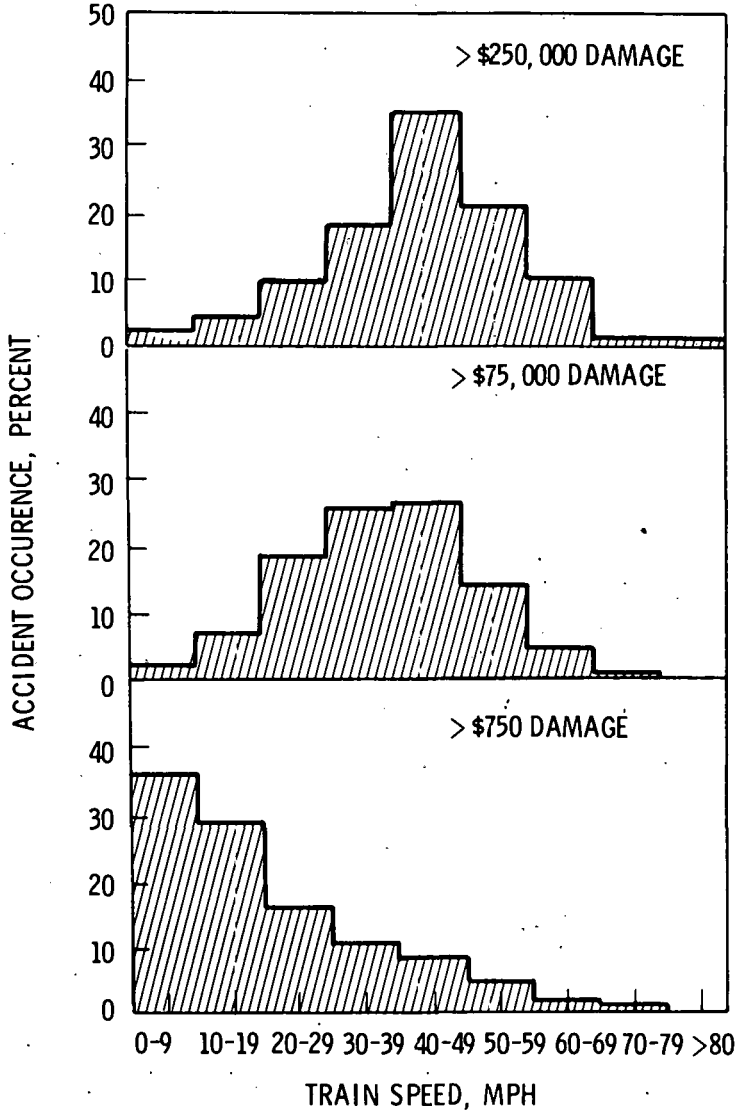


FIGURE 1. Accident Distribution Versus Train Speed for Various Dollar Damage Categories

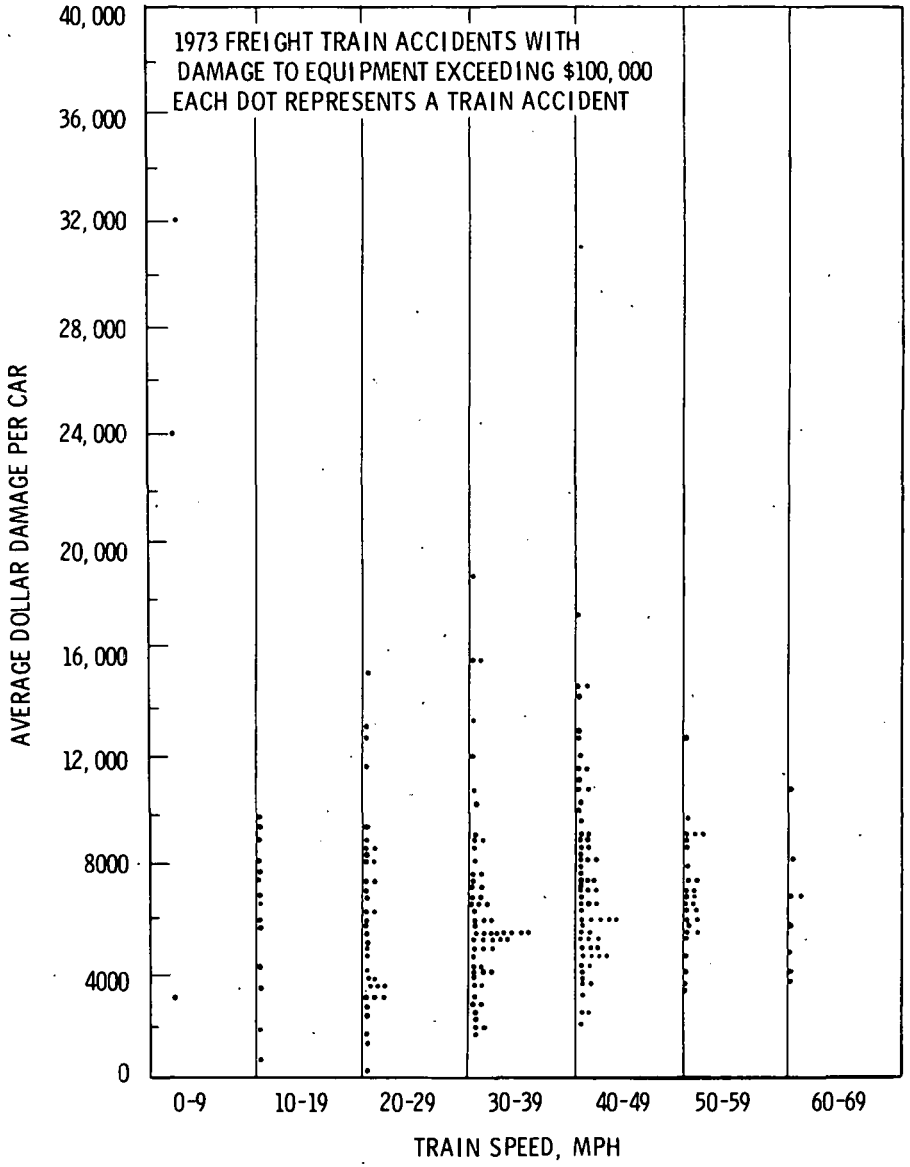


FIGURE 2. Distribution of Dollar Damage per Car as a Function of Train Speed

The observed increase in dollar damage to railroad property with increasing train speed may be due in part to the greater number of railcars involved in accidents at higher speeds. To assess the importance of this factor in accident data, a study was made of the average dollar damage per car for all freight train accidents for the year 1973 for which the total dollar damage exceeded $100,000. Figure 2 shows a plot of average dollar damage per car ranges from less than $1,000 to almost $40,000. Theoretically, high average impact forces (forces related to large magnitude changes in train speed) would be reflected in high dollar damage to an affected railcar. An increase in average dollar damage per car with increasing train speed might, therefore, be expected from the data. Figure 2 shows only a small increase in average dollar damage per car with increasing train speed.

For the accident sample studied, the average dollar damage per car varies over wide limits. Part of the variation for accidents in a given speed range is probably due to the lack of uniformity in damage estimating techniques used by different railroads. However, the absence of a strong dependence of average car damage upon train speed in the data suggests that the severity of impact upon a railcar might not be directly related to train speed. This may be partly explained by the fact that only the first few cars are subjected to the maximum impact of an accident; subsequent cars are cushioned from severe impact by the preceding cars. In a derailment accident, the number of cars derailed has been found to depend on train speed at the time of derailment.[5,6] However, after the first car derails, succeeding cars derail at successively lower speeds and thus at lower levels of severity. An additional factor which affects car damage is the geographical environment of an accident and the availability of objects which have a potential for doing serious damage to an impacting railcar.

CONCLUSION

Both train speed and dollar damage have been used in transportation studies as measures of accident severity. Analysis of freight train accident data for the three year period, 1972-74 showed that, in general, as speed increases dollar damage to railroad property also increases. A greater percentage of high speed than low speed accidents result in high dollar damage. Factors, in addition to speed, that can have an important effect on accident severity include the type of accident, the kinds of railcars and other equipment involved, and the geographical environment of the accident.

Threshold levels of accident stresses (e.g., impact and puncture forces and fire temperature and duration) are required to compromise the integrity of shipping containers used for the transport of radioactive materials. Analyses of accident severity using either speed or dollar damage as a basis can provide some insights into the possible risks involved in transport of radioactive materials. However, care must be taken in the strict use of results since there is no direct correlation between either speed or dollar damage and cask failure threshold levels.

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THE COMPATIBILITY OF RADIOACTIVE MATERIALS TRANSPORTATION WITH RAILROAD OPERATIONS - TWO VIEWS

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INTRODUCTION

The U.S. railroads, especially those in the northeast ("Official Territory", in railroad parlance), have long been concerned about the consequences of a train accident involving spent fuel or other radioactive materials. Responses to this concern have taken two basic forms:

(1) Insistence upon haulage on a contract basis only, thus absolving the railroad of any common carrier obligations related primarily to questions of liability, and

(2) Insistence upon haulage in "special trains" with or without ancillary restrictions such as slow speeds, special routes, day-time operation only, requirements for buffer cars or escorts, etc.

Historically the Northeastern carriers have been willing to handle radioactive traffic only on a contract basis (Item 1 above), coupled in most instances with operating restrictions such as those described in Item 2. The Southern and Western railroads generally handled this traffic as common carriers and without operating restrictions until 1976, at which time many of them began insisting upon use of special trains. One carrier sought to withdraw entirely from this traffic as a common carrier -- thus taking a stance similar to that of the Northeastern roads.

These developments in the South and East, coupled with the continued refusal of the Northeastern roads to act as common carriers, precipitated the filing of three separate complaints with the Interstate Commerce Commission by nuclear industry companies and agencies.

The details of these cases are presented in other papers.

The voluminous record compiled in these three proceedings provides a useful insight into railroad operating practices, and the compatibility of spent fuel traffic * with the railroad operating environment. Of greatest value in this regard

* The term "spent fuel" will be used hereafter to include high-level radioactive wastes which would move in casks similar to those used for spent fuel. Railroad restrictions have generally been directed primarily at these commodity classes.

is the "Special Train Case" (ICC Docket 36325). Most of this paper will draw from the statements, testimony, and briefs submitted in that case. The issues were debated primarily by railroad transportation officers with many years of practical, "on-the-ground" experience (but with no nuclear energy background) and highly trained nuclear scientists with little or limited railroad experience. It is probably fair to say that the railroad witnesses failed to fully comprehend the inherent strength of a spent fuel cask and the nuclear industry witnesses had a less than complete understanding of the forces and consequences of serious railroad accidents. This paper attempts to report the contentions made by both parties without making any judgements about their respective allegations.

HOW SAFE IS "SAFE"

The defendant railroads conceded that the casks and cask cars were sturdy and well-built -- no doubt stronger than any other type of freight car -- and hence that the chance of a breach was admittedly small. But in their view, the cask was not impregnable. Railroad accidents of an intensity capable of rupturing the cask could occur.

The nuclear industry protestants conceded that the cask might not necessarily withstand a "maximum credible accident" without rupture, but the probability was too minute as to be uncalculable. Hence, not only were the railroads' safety concerns unwarranted, but no added safeguards could possibly further reduce the risk by any statistically measurable amount.

The basic disagreement between the nuclear industry and the railroads was thus the extent of the vulnerability of the cask to fail in a serious railroad accident. Actual experience to date has been a near perfect safety record. The rail carriers pointed out, however, that the experience record has been a limited one, and hence is not necessarily a good indication of what may occur in the future.

The design and safety criteria developed by NRC and DOT for the design, construction, and operation of spent fuel casks are exceedingly stringent. This the rail industry conceded. Further, such criteria always have been predicated upon the movement of casks in routine commerce on conventional equipment. Total reliance for the containment of the contents has thus been placed upon the packaging rather than upon any need for special handling while in transit.

It was on this point that the railroads apparently had serious misgivings. It was alleged that the railroad operating environment was one of "multiple impacts, bizarre occurrences, high forces, and lengthy fires". Much of the thrust of the railroad arguments was that nuclear industry criteria and cask certification and licensing tests inadequately portrayed the violence of actual railroad accidents. Further, the full size and computer modelling tests were not representative of actual conditions. One witness testified that "railroad accidents can and do take place in which much more energy is involved in terms of impacts and much longer fires are involved than was assumed in the AEC (now NRC) certification and testing program...." Others expressed similar concern, e.g.:

> "Only obsolete hardware has been used in full scale tests...."
>
> "The actual railroad service environment is so complex that mathematical modelling and analysis cannot correct for gross deviations."

The railroads, thus unconvinced of complete cask impregnability demonstrated by such tests, alleged that it was their collective right and obligation to impose safeguards and precautions beyond and more stringent than those mandated by the packaging regulations of DOT and NRC.

The testimony, especially that in the Special Train Case, contains substantial

description of the kinetic forces and violent impacts which can be generated in a railroad accident. The railroads claimed that even if a small percentage of the total energy content in a train collision were directed against a small area of the cask, the yield strength might be exceeded with a resulting breaching of the cask. Nuclear industry witnesses countered that most of this energy is dissipated and distributed throughout the train, causing buckling, deformation, and collapsing of the cars rather than acting as a single force against the cask.

Nuclear witnesses averred that it is "peak deceleration force", not kinetic energy, which causes damage. This led to disagreements between the parties as to the extent of "slack" or "bunching" of cars. Slack runs in as the head of the train slows. Thus some elapsed time is required to bunch the cars which normally begins to occur only after the initial impact.

The railroads on the other hand claimed that "In the case of most collisions or highway crossing emergencies, crews react and make an emergency brake application prior to impact with the result that (train) slack is not totally in a 'run-out' condition.

> "....As the head of the train slows, the rear cars force in the slack and are 'bunched'. Thus when an object is struck by a train in a bunched slack condition, the kinetic energy of the impacting vehicle approaches that of the entire train. This collision is distinctly different from the one which would occur if each car at a time enters the collision with stretched or run-out slack."

Railroad witnesses expressed concerns about many other types of potential accident consequences -- not only impacts and piercing, but fires, explosions, prolonged "torching", immersion in water, etc. For example, references were made to fires lasting up to 24 to 48 hours and reaching temperatures of as much as 2,800 - 3,000°F. The relatively high likelihood of a cask car ending up in a ditch filled with flammable liquids was pointed out.

One witness claimed, "There have been several railroad accidents involving large numbers of tank cars which were ruptured and burned with the loss of more than 500,000 gallons of fuel oil, gasoline, etc. The ditch along the railroad right-of-way is a perfect trap to hold these flammable fluids and concentrate a very hot fire for an extended period of time and would not be an unlikely place for the cask car to come to rest." Another added that "fire directed against the bolts that seal the head of the cask can seriously impair the efficiency of the holding power of these bolts and furthermore, exceed the 1500°F allowable operating temperature for which the cask's metallic headgasket is designed".

Much of these "what if" type observations were controverted by protestants, summarized in one of the briefs stating that the level, duration, and nature of contact between the fire and the object being studied was necessary for any meaningful evaluation.

In a similar vein, the railroads spoke of the extremely dangerous "BLEVES" (boiling liquid expanding vapor explosion) characterized by explosions and subsequent fireballs. Protestants argued that a BLEVE lacks the duration to pose any danger to a cask. The railroad counter was that a BLEVE or other serious explosion could hurl the cask into a serious fire situation.

According to the railroads, the fire certification test is a poor substitute for "torching", a not uncommon occurrence involving the blowdown of a tank car through the relief valve with a subsequent plume of hot flame. Nuclear witnesses claimed that the torch, while indeed exceeding 1470°F, is so concentrated that it cannot totally engulf the cask. Thus heat can be readily conducted to other parts of the cask not so engulfed and thence dissipated. The rail carriers alleged that failure of the cask cooling system could allow interior cask temperatures to rise

sufficiently to be the cause of a fire. They also pointed to the possibility of the cask being submerged in deep water. Protestants responded by describing the near impossibility of cask rupture due to a cooling system failure, and of its ability to withstand immersion in over 2,000 feet of water.

Included in the railroad testimony were these statements:

"It is not unusual to see track rail penetrating cars of much stronger structure (than trailers) and through traction motors and locomotive frames where metal thicknesses approximate that of a cask water jacket."

"The cask attachment to an overturned carrying vehicle is not sufficient to keep the cask from separating and to roll and tumble down a mountainside into a rock or metal structure below. There are numerous bridges and overpasses, with metal caps on top of concrete pillars or substructures which are more than one hundred feet below the rail."

"There are many commodities which the railroads ship in regular trains, such as steel ingots, steel shafts, heavy-wall pipe, structural beams, and construction machinery that could very well have the mass dimensions and stiffness to puncture or fracture the (cask's) inner core."

Protestants refuted many of these possibilities by referring to Sandia performed fracture tests and other tests involved in slamming vehicles into concrete walls. A broken rail attempting to pierce a cask would fail long before the cask would. Even in the event of a cask impact against a concrete bridge abutment, it was shown that the latter would crush at stress levels well below those of the cask.

In short, the railroads' allegation was "yes, safe, but not safe enough". Their remedy? Special trains.

USE OF SPECIAL TRAINS

In seeking ways of minimizing both the chances of an accident and the consequences thereof, the rail carriers decided that primary reliance should be placed upon the use of special trains.

The events leading to this decision go back a number of years. Following a long period during which the nuclear industry and the eastern railroads failed to reach any accord on the issue of common versus contract carriage, the President and Chief Executive Officer of the Penn Central Railroad asked the Association of American Railroads (the trade association comprising most of the U.S. and Canadian railroads) in early 1973 to study the issue of transporting spent fuel. March 30, 1973 the AAR's Board of Directors assigned the matter to its Research and Test Department. The September 28, 1973 report of the Association's Vice President in charge of Research and Testing to the Board contained this recommendation: "Until we have more assurance that the casks are impregnable, it is my judgement that spent nuclear fuel elements should not be moved in regular train service."

Following this recommendation, a special Task Force of the AAR's Operating-Transportation Division's General Committee fleshed out this general recommendation with operating specifics as follows: "Shipments of casks containing irradiated spent fuel cores should move in special trains containing no other freight, not fastener than 35 MPH. When a train handling these shipments meets, passes or is passed by another train, one train should stand while the other moves past not faster than 35 MPH." This operating practice was approved by the AAR Board on March 29, 1974. Though not a mandatory requirement upon member railroads, virtually all of the industry not already insisting upon special train service began to adopt this stance.

The definitions of a special train differed somewhat among railroads. Most

adopted the three-part AAR recommendation (a train containing no other freight, a maximum speed, and one train stopping when two trains are passing). In various degrees, some respondent railroads added or recommended further qualifications such as these:

- o No humping
- o Minimum or no switching except at origin and destination
- o Use of buffer and/or escort cars
- o Special routings to avoid congested and densely populated areas

Much of the railroad testimony dealt with the reasons why handling this freight in special trains produce a safer operating environment, serving to minimize hazards in two ways: (1) by reducing the probability of accident and (2) the consequences of an accident should one occur.

Let's look at the reasons given by the railroads and some of the protestant's replies:

> "Continued inspection and monitoring of shorter trains was claimed to be much easier and more thorough. Regular trains are frequently a mile or more in length. Because of this and also due to weather, track curves, tunnels, and other obstructions, the train crew can rarely keep the entire train in sight at all times. Since a short train can be kept under constant surveillance, problems such as a "hot box" (overheated wheel bearing) or a shifted load would be readily apparent to a crew."

Carriers briefly alluded to problems of theft, vandalism, and other criminal activities (such as terrorism and sabotage) which carriers continually face, and which could be better controlled in a special train environment.

It is not clear that this would provide a significant increase in the degree of safety. The cask cars are modern, well-built and maintained, with a consequent lesser risk of equipment failure than the average frieght car. However, about 30 percent of all train accidents are caused by equipment, and any reduction in the amount of equipment used in a special train will tend to reduce the possibility of an accident.

Perhaps more to the point is that the same degree of surveillance could be achieved by placing the car in a regular train near the head or rear-end of the train where either the engine crew or caboose crew could monitor it.

The several advantages of shorter trains in reducing both accident probabilities and consequences was stressed by several witnesses:

> "The probability of a train accident (derailment) increases exponentially with the train length, with the number of cars derailed increasing linearly with length....The longer and more massive is the train, the more likely is there to be a failure in some component of the draft system....In addition, the amount of free slack in a train is proportional to the number of cars in the train....These effects combine to cause the probability of derailment to increase faster than the train length increases."

The short, special train would substantially reduce slack action and hence lessening of slack forces due to draft-gear run-in and run-out. Moreover, the presence of only the best rolling stock would lessen the chances of equipment failure.

Trains consisting of but one or a few cars can be stopped in a substantially shorter distance than longer, heavier trains. This in turn reduce the likelihood of

collisions and highway crossing accidents and permits nearly or complete stopping before the cars jack-knife, pile up, and overturn.

However, nuclear industry witnesses claimed that the shorter train length places the cask car in a more vulnerable position in the event of either a collision or crossing accident inasmuch as the cask car necessarily would be located near the point of impact. It was further claimed that in a short train a cask car is more subject to derailment whereas the last few cars of a regular train are rarely derailed.

While a short train may be less likely to be involved in a crossing accident than a longer one, the overall incidence of crossing accidents may increase due to the greater number of total trains operated.

Since a special train would carry no other commodities, the risks of fire or explosion caused by other materials would be lessened. Counterpoint: a reduction in risk could be accomplished by placing cask cars some distance away from hazardous or combustible loads in regular trains. Furthermore, in the case of a collision, there is no assurance that the other train carries no such materials.

A special train would receive minimum terminal handling and switching. It would not be switched by "humping" or without a switch engine attached. While the low speed impacts experienced in freight switching yards (generally no more than 6-7 MPH) would rarely cause a serious accident, incidents have occurred with resulting fires and explosions. A special train is subject to fewer yard handlings. But restrictions on humping and switching without an attached locomotive can be equally (and frequently are) placed on cars of hazardous and sensitive materials moving in regular train service.

With regard to speed restrictions and the requirement that one train stop while another passes, there was considerable disagreement between the parties as to whether a slower (35 MPH) speed provided much more safety. The railroads stated that since the amount of kinetic energy available varies with the square of the velocity, and reduction in speed would correspondingly reduce the severity of an accident. But there is no apparent need to handle cask cars in "hot-shot" trains travelling as much as 70 MPH on some railroads. In most cases, regular trains seldom exceed 50 MPH.

The requirement of stopping one train while another passes is usually reserved for those cases where one train contains a car wider than normal, thus reducing clearances between trains to such an extent that a slight protrusion or excessive swaying could produce a side-swipe. Freight and passenger trains frequently pass each other at combined speeds well above 100 MPH. A 35 MPH speed limit, coupled with the train stopping when another passes, need not be a unique special train restriction. It could be imposed as well upon regular trains handling spent fuel, albeit with some delays imposed upon other traffic moving in the same train.

Other witnesses alluded to other advantages of special trains, many of them variations of the factors listed above. Special trains have greater flexibility to utilize special routings to avoid poor trackage, heavily populated areas and intermediate freight yards. By running through directly from origin to destination, transit time would be reduced, and so would the possible radiation exposure of personnel. Protestants pointed out that radiation levels are so low that no need exists to take added precautions to protect employees.

Nearly all the railroad operating officers testified that in their experience they could not remember a serious accident involving a special train. However, the number of special trains actually operated is a miniscule percentage of total trains, and lack of data precluded a finding as to whether or not special trains are safer in terms of accidents per train mile or car mile. Most special trains are operated primarily to accommodate overweight or over-dimensional loads -- in many cases, at very restricted speeds and hence not representative.

Apart from the safety issue, the railroads repeatedly pointed out that special trains, despite speed and stopping restrictions, would provide a faster, more dependable service than regular train service. The resulting improvement in utilization of expensive casks and casks would more than offset the added costs of special train service (currently costing about $20.00 per mile).

The nuclear industry representatives countered that even if true, this was an economic decision best left to the discretion of the shipper. For no other commodity -- hazardous or not -- do the railroads unilaterally impose a requirement of special train service. Rail tariffs providing for such service have heretofore existed for the benefit of those wishing a special or expedited service, and willing to pay the extra charges for it.

This was one of the significant ironies pointed out by the protestants. Why discriminate against spent fuel when thousands of carloads of potentially dangerous gases and chemicals move daily in regular train service? Most of these commodities move in tank cars which are far less sturdily constructed than cask cars. Many of these are subject to restrictions related to switching and placement in the train which could be applied equally to spent fuel without imposing special train service.

The rail industry's reply was that under the press of time or other conditions, such as human error, there are sometimes lapses in the application of these restrictions. Cars do get humped or are placed in a train next to other hazardous materials. And with a commodity as potentially dangerous as spent fuel, a special train is the only way to assure these lapses don't occur.

Protestants pointed to the loss of national transport efficiency due to the need for extra locomotives, cabooses, crews and fuel to handle cars individually rather than as a part of larger trains. In the context of total railroad operations, however, such increased resource needs are not substantial.

ACCIDENT CONSEQUENCES AND CLEAN-UP OPERATIONS

In the event of a train accident of any kind, it is of extreme importance to the railroads that clean-up operations proceed expeditiously. In many cases, train service is interrupted and must be restored as quickly as possible. Even when it is possible to detour or re-route traffic over alternate routes, it can be costly, particularly if another railroad is involved. Use of a bypass route usually results in significant delays. Attendant to such service are all the problems associated with non-routine operations, e.g., crews and equipment out of position, etc.

Therefore, the first objective at a derailment is to re-open the line. This requires immediate mobilization of railroad and specialized wrecking contractors, personnel and derricks, and other emergency equipment. The line must be cleared, equipment re-railed, and the track structure rebuilt. The time required to restore service is dependent upon many factors, and may vary from a few hours to perhaps days.

Because of the urgency of line restoration, the railroads expressed concern about the added difficulties and delays which could be associated with clearing an accident involving spent fuel. The railroads stated that clearing operations could not commence until the area was declared safe by radiological personnel. The carriers are not equipped to do this. Trained personnel with proper survey and monitoring equipment would be needed. Hazardous material accidents require that Federal Railroad Administration, Environmental Protection Agency, AAR's Bureau of Explosives and various other federal, state, and local officials be notified. The large number of agencies asserting jurisdiction would almost insure conflict and confusion in the opinion of one railroad witness. The result would probably be unacceptably long response times.

One of the protestant witnesses described the national and regional agencies responsible for taking emergency action in case of an accident involving radioactive material, pointing out that an emergency team is usually on its way within an hour or two after the proper agency is contacted.

The carriers alluded to the potential difficulties of moving and re-railing heavy, three-axle cask cars. Special, high capacity cranes would be required, but which might not otherwise be required except for locomotives. The seriousness depends upon the location and terrain. Even more serious would be those cases in which fire resulted from the accident and venting of the casks might hinder fire fighting efforts.

In the event of actual breaching of the cask, decontamination procedures could tie up a line for long periods. And because rail lines frequently parallel or cross major highways and pass near airports, the result could be a long-lasting interference to interstate commerce.

Several railroad witnesses also spoke of possible public reactions to a spent fuel accident. Following is a listing of some of these remarks:

"The public cannot be relied upon to act reasonably in a radioactive emergency....panic could result."

"One could foresee panic caused by exaggerated and overzealous news media."

"As a result of an accident....immediate consideration would be given by government agencies to subjecting the railroads to burdensome regulations to control the transportation environment."

"Carriers don't need more unfavorable publicity and laws."

"A possible spill would occasion a mass evacuation that might initiate panic."

"One mishap....could have a very serious bearing upon the confidence of employees and public, and on the future of transporting radioactive materials even in special trains."

CONCLUSION

These latter comments, as well as those quoted earlier in this paper, typify the concerns - real or imagined - which the railroads have about handling spent fuel. Some concerns may be exaggerated - generated in the heat of adversary proceedings. Others are genuine, but stem from a lack of knowledge about nuclear technology in general and cask construction in particular.

The transportation of spent fuel is, of course, not totally risk-free. The issue is whether special trains or other operating restrictions would provide any meaningful reduction in risk, and do so at a cost reasonably proportionate to such reduction. As of this writing, the Interstate Commerce Commission has ruled that such operating practices are not warranted.

Dup

REALISTIC CHARACTERIZATION OF SEVERE RAILROAD ACCIDENTS CASE STUDY - TANK CARS

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I. INTRODUCTION

This paper presents the results of studies aimed at characterizing the nature of the railroad accident environment. These studies were focused on the way this environment affects proposed shipping of spent nuclear fuel and radioactive wastes. They were developed because of skepticism voiced by railroad personnel in recent years concerning the ability of nuclear packaging to safely withstand severe railroad accident conditions. These concerns were the subject of a major portion of the recent ICC hearings held concerning transport of radioactive spent fuel and wastes by rail.

It has generally been recognized by even critics that such nuclear packages are structurally better than most other containers moved by railroads. The critics objections focused on the following concerns, that:

(1) The package designers do not have a clear perception of design requirements to withstand railroad accidents.

(2) The NRC/DOT accident scenarios which function as design criteria for package design inadequately describe the severity of these accidents. In particular, the forces, fire temperatures, and fire durations specified by NRC/DOT were thought to be insufficient.

During the course of the ICC hearings, these charges were addressed and countered by the parties representing government and industry shippers.

The objective of this paper is not to state that one can accurately define the exact nature of all railroad accidents, nor to state that accident data can easily be translated into regulations and design criteria. History has shown this to be a difficult task for even those who have frequently been involved with railroad accidents. Rather, the intent is to show that upper limits for accident frequencies, physical forces, and fire effects, etc., can be established. These limits can be based on analysis of past accidents and the equipment involved. In simple language, no force is infinite no matter how long the train is and how fast it is going. Similarly, flame temperatures and fire durations are finite. Boundaries can be placed on the loadings imposed on a package. A direct comparison will be made with the programs and regulations established by the Federal Railroad Administration and the railroad industry to make tank car movement of hazardous materials safer.[1] These are compared with the regulations and design criteria used for radioactive material packages.

II. ACCIDENT FREQUENCY

A first line of reasoning is that the probability of a major nuclear shipping accident occurring is very small when viewed on a car-mile basis. The annual rail shipment volume of radioactive material is not projected to exceed over a few million car-miles per year. One difficulty in trying to convince the average person that nuclear transport is safe lies in the lack of data. There have been no major accidents in 25 years with nuclear shipments, but there have been relatively few shipments compared to railroad shipping of other hazardous materials. Recent studies have taken the opposite tack and used the accident history of the entire rail industry. However, there are also difficulties in this approach. First, this is too broad a spectrum. It involves all types of cars and ladings. The relative propensity of this equipment to cause or aggravate a major accident will vary widely. An additional problem is how to define a major accident which would have any possibility of significantly effecting a spent fuel or waste package.

In this study, an "in-between" approach was taken. A case study was made of railroad tank cars carrying all types of hazardous material. This permitted a comprehensive study to be made since there is an adequate statistical basis (in excess of 75-80,000 possible tank cars, many years of experience, and billions of car-miles travelled). A large amount of government and industry attention has been devoted to the study of tank car transport. This is a direct result of the number and severity of tank-car accidents that have occurred within the last 10 years and the adverse publicity resulting. There is also a direct comparison with nuclear shipping, since tank cars are also regulated by the DOT, including the approval of package design. Also, the size and weight of these shipments are comparable to spent fuel and waste cars.

The statistical data concerning tank car accidents was obtained directly from unpublished information compiled by the Federal Railroad Administration (FRA)[2] within the Department of Transportation. A number of technical studies have been published concerning proposed design criteria, accident and safety testing, etc. These, too, were reviewed in detail. The crux of the study on accident frequency was to determine how many accidents in a given year could be classified as severe. Five criterion were postulated and individually evaluated for classification purposes. These were:

(1) Damage cost - A reported damage cost in excess of $300,000.

(2) Public impact - Either any loss of life, five or more injuries reported, or evacuation of 1000 or more people.

(3) Total number of cars involved - Seven cars or more involved in an accident event.

(4) Mechanical or thermal damage to car - Five cars or more destroyed by either collision, puncture, fire, or explosion.

(5) Both mechanical and thermal damage to car - Four cars or more destroyed by both collision and fire damage.

The results of this study are shown in Table 1. Examination of this data would indicate that hazardous material laden tank cars have a probability of being in a serious accident on the order of $1.0 - 3 \times 10^{-8}$ per car-mile. The range of severe train accidents is from 3 to 14 per year with a mean of about six to seven accidents. Category (5) is of greatest interest since it yields the joint probability of a car receiving both mechanical and thermal damage. This is about 1.0×10^{-8} per car-mile.

Applying this data to the transport of nuclear fuel, a typical case can be made based on the projected fuel receipt at the Barnwell Nuclear Fuel Plant (assuming all fuel received by rail).

1 car load = 4.5 MTU
Average trip distance = 1100 miles
Projected fuel receipt rate = 1500 MTU/year

Probability of a loaded nuclear cask being involved in a major accident = (1.0×10^{-8})/car-mile $\times \frac{1500 \text{ MTU/yr}}{4.5 \text{ MTU/car trip}} \times \frac{1100 \text{ miles}}{\text{trip}} = .0036$ per year or once every 270 years. The range for all movement of spent fuel is probably more like once per 75-150 years.

This is in good agreement with estimates made using other statistical bases. It illustrates the unlikelihood of even being involved with a serious accident. Minor accidents are about 10-15 times more frequent, but these minor accidents would have no significant effect on spent fuel cask integrity.

III. COLLISION FORCES

A major point of contention in the ICC hearings was the amount of force which could be applied to an object during an accident. Considerable confusion resulted from attempts to correlate the kinetic energy involved in a typical 70- to 80-car train accident with maximum localized forces. Clearly the salient criterion for evaluating container integrity is the force required to breach the container wall. Also of interest is the shape of the impacting object. The following relevant observations were made after examining both rail car structural data and accident reports.

(1) Trains do not behave as a column structure - examination of numerous photos indicated that the cars were scattered, stacked, and jackknifed in a number of different directions.

(2) The force required to indicate crushing or buckling of the framework of a typical car is about one million pounds.[3, 4] A well designed heavy-duty car with a cushioned underframe may be able to withstand two to three million pounds, but there are comparatively few of these cars.[5]

(3) The rail car coupler itself is a major cause of punctures, particularly in tank car accidents. Typically, couplers will begin to buckle under a loading of about 1-1.5 million pounds.[6] Examination of supported lengths of steel rail, also a major cause of puncture, indicate that they, too, will buckle at or below about 1.3 million pounds.[6]

(4) Studies on the location of coupler punctures on tank cars show that the highest percentage occurred on the elliptical heads.[7] Approximately two-thirds of those occurred in the lower half of the head.

(5) The actual collision forces in an accident typically occur with the individual cars traveling at relative velocities of 15 to 20 mph. The train speed may have been considerably higher at the initiation of the accident, but as the accident progresses, the speed of the cars decrease.

(6) Much of the kinetic energy in a large train, high-speed accident is dissipated in car-to-car collisions including stacking and jackknifing. As a result, no point on a single car would receive even a major portion of the kinetic energy

dissipated in a large train accident. Many accidents involve multiple sequential impacts with lower forces than occurs with a single severe impact.

A study performed for the Federal Railroad Administration (FRA)[8] which attempted to analytically recreate a number of devastating tank car accidents indicate a range of localized forces extending from about one to 3.5 million pounds[7] could have been applied to a particular tank car. The accident evaluations reviewed indicated that even in a long train, forces are higher, but not limitless. An upper limit of localized forces can be estimated for hypothetical situations, of about four to five million pounds.

The FRA has proposed the following legislation requirements[1] to upgrade the capability of certain tank cars to withstand mechanical loadings and minimize the probability of car puncture:

(1) Shelf couplers - E or F type shelf couplers are required to prevent vertical disengagement of the coupler.

(2) Tank head puncture resistance system - typically a steel head shield about 5/8-inch thick to minimize the probability of puncturing the tank car shell.

(3) Performance testing - An 18-mph impact test of a tank car (brakes locked) by a 263,000-pound ram car. The ram car will be backed up by additional cars with a total weight of at least 480,000 pounds. The test car contains pressurized water. The coupler will hit the test car at the thinnest section of the tank. There is to be no loss of contents for one hour after the test.

Recent testimony has indicated that merely changing to shelf couplers and adding head shields will significantly reduce the number of tank car accidents.

IV. FIRE ACCIDENT CONDITIONS

Fire accident parameters like those involving collision forces are very complex. They include: average fire temperature, fire duration, emissivity of the fire and absorptivity of receiving body, the area of the receiving body exposed to flame, and the effective heat capacity of the receiving body. These parameters must be examined jointly since their interrelationships are complex. Isolating any individual parameter such as the temperature or duration can lead to erroneous conclusions.

There are two general types of petroleum-based fuel fires that occur in major rail accidents.

- Pool Fires - This could occur as the result of a tank car spilling its contents into a ditch along side railroad trackage. The flame temperature may vary from about 1200-2000°F locally with a bulk average of about 1400-1600°F.[8] The combustion rate is about 10 gallons per square foot of surface per hour.

- Torching - This might occur when a tank car carrying a flammable gas such as propane is, itself, in a fire. After a period of time, the safety valve opens expelling propane which ignites. The flame temperature will typically be in the range of 2100-2300°F.[1] The resulting flame is highly localized and concentrated.

Examination of the above types of fires indicate that the flame temperature is much lower than the theoretical combustion temperatures. This is due to incomplete combustion of the fuel and is evidenced by observation of a great deal of black smoke.

The following observations were developed from data from accident reports concerning fire consequences.

(1) Fires of several days duration were reported. However, duration is not necessarily representative of the amount and the intensity of the heat from the fire. Frequently, long duration fires are characterized by a number of small, scattered fires of low heat intensity or torching from a tank car valve failure or split in tank shell.

(2) The total time required to burn the contents of one entire 30,000-gallon carload of petroleum in a pool fire is about 1-1/2 hours.

(3) A hot fire that can totally envelop a large body must be large. The fuel must burn with an intense flame. Grass and wood fires, which occur due to burning of rail car components and ladings, are an insignificant source of heat.

(4) The most dramatic type of fire, the BLEVE (Boiling Liquid Expanding Vapor Explosion), is, in effect, a high-intensity fireball of short duration. The total heat input into a nuclear cask would be very small due to the mass of the cask.

The above observations all focus upon the ability of a large, heavy walled, metallic container to withstand a major fire. Namely, (1) the fire must burn in one location for the specified period of time (at least one to two hours), (2) the flame intensity must be at least 1300-1500°F, and (3) must be of sufficient size to substantially envelope the body for the entire period of time. What is of importance is the capability of the fire to transfer heat to the package and the temperature rise of the package material, not how spectacular the fire seems to an observer.

Referring to the tank car case study, the FRA has proposed legislation concerning fire resistance of pressurized tank cars.[1] A thermal protection system (typically insulation) is required. The protection system limits tank car pressure during the fire by lowering the heat input into the car and assuring that the tank safety valve is adequately sized to expell sufficient vapor. A performance specification for this system protects against the following fires.

(1) Pool Fire Test - 1600 ± 100°F hydrocarbon fire for a period of 100 minutes.

(2) Torch Fire Test - 2200 ± 100°F hydrocarbon fire for a period of 30 minutes. Torch velocity of 40 ± 10 mph.

A calibration test for the flame temperature involving the temperature rise in a 4-foot by 4-foot by 5/8-inch thick steel plate is further specified.

V. COMPARISON WITH NUCLEAR PACKAGING CRITERION

Previous portions of this paper have presented data for tank cars related to accident frequencies, collision forces, and fire conditions. To a considerable degree, this data should be representative of the actual railroad environment and in particular severe accident conditions. A frequent concern during the ICC hearings was the comparison between the so-called "real" accident environment and the Nuclear Packaging hypothetical accident conditions. The only way to make a comparison is by evaluation of the conditions on a specific package. This is particularly true since the federal regulations are performance specifications or tests which are generally applicable to all modes of transportation--not just rail. The package selected for comparison was the NLI 10/24 spent fuel rail cask. This 100-ton cask is of all welded construction and is fabricated primarily from steel. Lead is used as the

gamma shield. The relationship between the derived accident environment and the federal hypothetical accidents was found to be as follows:

(1) Fire - The DOT conditions recognize both a 1600°F pool fire (100 minutes) and a 2200°F localized torch fire (30 minutes). The NRC fire is 1475°F for 30 minutes. The DOT fires are test oriented with calibration by temperature rise in a steel plate. The NRC criterion is analysis oriented with stringent definition of the fire emissivity and cask absorptivity. The various fire conditions were calibrated using the TRUMP computer program to a common basis. The simulated fires were then applied analytically to the NL 10/24 cask wall for a period of one hour. Ranking in terms of severity was:

(a) 1475°F enveloping fire (most severe)
(b) 1600°F pool fire
(c) 2200°F torch fire (least severe).

The 2200°F fire was of little concern due to the localized area of the cask affected. Definition of the emissivities in the 1475°F fire made this fire most severe for this time period. These results illustrate that fire temperatures, by themselves, do not totally define a fire severity.

(2) Puncture - The major source of tank car puncture (and loss of lading) results directly from overriding couplers. The NRC criterion is a 40-inch drop of the package on a 6-inch pin. Relating the force at which the coupler buckles to an experimental test of the pin drop reveals:

	FORCE (million pounds)	TYPICAL LOCAL STRESS (psi)
Coupler Buckling	1.0 - 1.5	12,500 (E-type coupler face)
Pin-Drop Test	2.1 - 2.8	53,000

In this situation, the pin drop is clearly 3-4 times more severe.

(3) Major Collision - The NRC accident condition is a 30-foot drop onto an unyielding surface. The only possible comparison is the tank car head shield impact at 18 mph. There is obviously no basis for comparison since neither the impacting or target car is unyielding. The 30-foot drop test is far more severe. The crucial condition is the unyielding surface which assures that all of the impact load kinetic energy is expended in the package. For this reason, the 30-foot drop cannot be equated to a 30-mph collision. There is no object located along railroad trackage that even approximates an unyielding body. Test studies have revealed that a 30-foot drop of a cask onto an unyielding surface is more severe than a 2000-foot drop onto hard-pan soil.

VI. CONCLUSIONS

The railroad accident environment can be very severe. However, it is possible to develop worst case or limiting situations by analyses of past accidents. These conditions can be compared with federal regulations which specify hypothetical accident criteria. The NRC licensing criteria (when applied in its totality) is readily shown to be extremely severe. The major questionable area is fire duration. A duration of one to two hours (rather than 30 minutes) seems possible. However, the NRC fire accident condition is based on completely enveloping the package. From a practical standpoint, this is impossible. The heat input and subsequent temperature

rise to a package, which is partially protected by the car or the ground, is much less than for the idealistic "fully enveloping" situation.

ACKNOWLEDGEMENTS

The thermal evaluations were by Dr. John Ridihalgh of Ridihalgh, Eggers, and Associates in Columbus, Ohio. The structural calculations were by Dr. Thomas Branca of John Henry and Associates in Waltham, Massachusetts.

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TABLE 1

FREQUENCY OF MAJOR TANK CAR ACCIDENTS

Criterion	Annual Number of Events			Average Probability* ($\times 10^{-8}$/car-mile)
	Maximum	Minimum	Average	
1. Cost	11	3	6.0	1.5
2. Public Impact	9	2	6.5	1.3
3. Number of Cars	10	2	5.5	2.6
4. Mechanical or Thermal Damage	6	1	3.5	1.3
5. Both Mechanical and Thermal Damage	6	1	2.7	1.0

* Based on total number of cars involved.

Dup

REALISTIC ASSESSMENT OF HYPOTHETICAL RAILROAD ACCIDENTS INVOLVING A NUCLEAR CASK

CASE STUDY - NL 10/24 SPENT FUEL CASK

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I. INTRODUCTION AND CONCLUSION

A number of safety studies have been made concerning the transportation of radioactive materials (RAM). These studies have examined shipments of spent nuclear fuel by railroad. However, the studies were generic and as a result very general. The assumptions which were used concerning the strength and reliability of the packaging systems as well as the nature of radioactive material released were very conservative. As a result, it is difficult to relate these studies to the real world to answer the obvious question - what would happen in an accident.

The study summarized in this paper has tried a different approach. In order to obtain a measure of realism a case study was made. A specific packaging system, the NL 10/24 rail cask, was used as the basis. Licensing of this cask was completed shortly before the recent ICC hearings on railroad transportation of nuclear material were held. As a result the design-licensing team was still intact. The analytical formulations, computer and scale models of the cask were available for expanded studies. Hence, it was possible to study the response of the entire packaging system to intense mechanical and thermal loads. These loads were based on accident situations far worse than that explored during package licensing. An additional objective of this case study was to defend the actual equipment design rather than falling back on the cliche - "the cask meets or exceeds all federal licensing accident conditions." The initial purpose of this study was to develop data for the ICC hearings. An additional objective was to scrutinize and check out all of the assumptions used in making a study of the radiological consequences of a nuclear spent fuel shipping accident. In checking these assumptions, questions such as the following were asked:

- How strong is the fuel cladding?
- What kind of fission products are in the fuel rod plenum and in what quantity?
- What mechanisms would allow escape of fission products from the cask?
- What kind of protection is offered the package by the rail car?
- What kind of accident situation would result in a release of RAM?
- How do the accident events affect the release of RAM?

The study results indicate that blindly selecting the "worst-possible-case assumptions," and then postulating a RAM release without identifiable mechanisms compounds unrealism and adds confusion. In order for a RAM release to occur, an unlikely series of events must first occur which breach the multiple containment barriers surrounding the fuel. The overall safety margin provided by the packaging equipment increases geometrically beyond the already adequate margin provided by each containment barrier. A case evaluation of the NL 10/24 packaging system illustrates this contention by showing that:

(1) The accident events which must occur before a release of RAM, other than gases, is possible are in themselves incredible.

(2) The biological effects of a release of fission gases will in all likelihood be nil.

Using "worst-case" assumptions and generalized package data is an easy way out. However, this approach can boomerang when subjected to intense questioning. Things seem far worse than they could possibly be. "Mountains are made from molehills." A realistic case can be made, that if there are any hazards involved with transport of this material, that these hazards are not radiological in nature.

II. HOW STRONG IS THE PACKAGE?

A - Package Desciption

The NL 10/24 rail cask is the largest spent fuel cask currently licensed by the Nuclear Regulatory Commission (NRC). This 100-ton cask can carry either 10 pressurized water (PWR) or 24 boiling water (BWR) fuel assemblies. This represents a payload of about 4.5 tonnes of uranium. A measure of the value placed on safety in this design is that only about 7% of total package weight is payload.

Figure 1 is a photo of the cask mounted on the rail car. Figure 2 is a cross section of the cask body. This 100-ton cask is of all-welded construction. The steel thickness ranges from 3 to 13 inches in any direction. The regions with only 3 inches of steel are backed by 6 inches of lead. The neutron shield jacket adds an additional 9 inches of water. There are two removable closure heads (in series) in the upper portion of the cask. Each closure is gasketed, bolted, and preloaded with a number of large-high strength studs. Each closure is individually gas leak tested for tightness prior to shipment. The cask structure is further protected on each end by enormous impact limiters rigidly bolted to the cask. These limiters weigh 12,000 pounds (front) and 8,000 pounds (rear). They assure that impact from the 30-foot cask body drop test accident is totally absorbed prior to any damage occurring to the cask. They consist of balsa and pine wood canned with aluminum and further strengthened by stainless steel. Extensive finning on the exterior of the cask not only serves to increase the cask heat rejection ability but strengthens the neutron shield jacket.

The cask cavity is filled with helium at atmospheric pressure. Under routine operations, the internal fuel cavity pressure will be 10-15 psig or less. There is no shipping requirement for cask mechanical cooling systems. Cooling is passive, in that all of the thermal energy from the fuel will be rejected by combined natural convection and thermal radiation from the outer surface of the cask to the atmosphere.

The cask is rigidly attached by two 5.5-inch diameter pins to the rail car floor. The cask body rests on support saddles at the two cask ends. The cask is centrally mounted on the car about 15 feet from the end. The rail flat car is

designed to be "extra strong." Analyses of the car indicate that it would take at least 3.3 million pounds of force to begin to buckle the frame[1] (several times more than a typical rail car). The following design features enhance car safety:

- The cask is partially recessed into the car body
- Shelf type - F couplers are used
- Roller bearing wheels are used
- The car has hydraulic-center sill cushioning gear.

Note that the NRC licensing process takes no credit for the car structure protecting the cask.

B - Structural Testing of a Scale Model

Many man-years of sophisticated structural analyses including hundreds of computer hours were expended in evaluating the strength of this cask. This was used to verify packaging containment integrity during the hypothetical accident scenarios posed by federal regulations. The analyses were augmented by model testing (1/4 scale) of the cask body. Tests were performed for the pin drop accident and to simulate the broadside impact of a rail coupler. The coupler test is of particular interest. A rail coupler will typically buckle under a 1-1.5 million pound load.[2] However, the face of an F-type coupler represents a puncture hazard during an impact. A short, rigid length of steel, simulating an F-coupler face, was forced into the scale model (see Figure 3). This bar was more rigid than any coupler or rail section. It took in excess of 10 million pounds (scaled) before there was any major deformation of the cask inner cavity. It took about a 14 million pounds force before fracture was observed in the outer jacket and shell of the cask. At this point, the test bar had also failed and the test was concluded. The inner container had not failed. There seems to be no plausible way to hypothesize an accident in the railroad environment that could even approach this loading. To verify this, the response of the cask wall to these test loadings was analyzed and used to predict the effect of the following types of accidents:

1. The 130-mph impact of a 20-inch deep, 40-foot long I-beam weighing 3840 lbs
2. The impact of a 400-pound missile traveling 300 mph
3. The impact of the tank of a high velocity rocketing tank car
4. The piercing action of a broken rail section.

None of these accidents would result in breaching of the outer shell of the cask. The maximum loading developed in any of these accidents is less than 3 million pounds.

C - Prolonged Exposure of Enveloping and Torch Fires

A detailed computer model (TRUMP)[4] of the cask was used to evaluate various fire accident scenarios. Intense fires with durations of up to 50 hours were modeled. Fully enveloping fires with temperatures of 1400-1800°F were studied, as were localized torch fires in the 2200-2600°F range. A combination high ambient from adjacent fire (400°F) and torch fire (2200°F) was also studied. The temperatures were calibrated with DOT-FRA fire test criteria.[3] The cask was assumed to contain nuclear fuel at a decay heat level of 70 kwt. The following results were noted:

(1) Fuel temperature - It would take a minimum of 4 hours of a 1600°F all enveloping fire to raise all of the fuel clad temperatures to 1250°F (incipient failure temperature). The fuel rods would fail individually over the preceding 1-1/2 hour period.

(2) Seal temperature - In the same, all enveloping fire environment, it would take 12 hours to raise the inner seal temperature to 800°F (incipient seal failure temperature). No other portion of the cask would fail and lead to a breach of the fuel cavity wall.

(3) Torch flame effect - Locally heating one-fourth of the cask area with a 2200°F torch for about 1-1.5 hours will insignificantly damage the cask body. Only the cask outer surfaces would be affected. Sustaining the torch flame for many hours will have no effect since the cask body temperatures reach an equilibrium temperature by conduction of heat within this cask body and transferring heat to ambient air in a cooler region of the cask body. A tank car safety valve, which is the source of such torches, will typically empty the car of its lading in about 1.5 hours (depending on the orientation of the car and valve).

(4) BLEVE or fireball effect - Completely enveloping the cask body in a fireball for 10 seconds will only increase the entire cask body temperature by about 32°F and a maximum of 300°F on the outer surface.

(5) Size and duration of the fire - An all enveloping fire surrounding the rail cask would have to be at least 60 feet by 30 feet wide to sustain consistent, high-fire temperatures. A typical "jumbo" tank car load of petroleum would be entirely consumed in about 1-1/2 hours in this size fire. It is unrealistic to assume that more than one or two tank car loads could be emptied into a single constricted pool.

Evaluations of actual accidents show that there is no realistic way in which a fire could completely envelop a cask. Some portion of the cask body would always be shielded from the flame. This considerably reduces the heat-up rate of the cask body and results in lower body temperatures. The conclusion reached after reviewing these analyses is that no conceivable fire will cause the cask containment wall to fail.

III. SHIPPING ACCIDENT ASSUMPTIONS

A - What Can be Released and How?

Obviously a dry cask can not release a liquid. Testing of the cask indicates that it is strong enough to prevent solids from being released. The only substance which could possibly be released is a gas. Specifically the fission gases which are contained within the fuel pins. The most likely route for release would be through the closure seals. These seals could conceivably fail after a prolonged fire. Internal pressure within the cask cavity could force these gases out past the seals. The cavity gas is pressurized to 15-120 psig[4] by the external heat from the fire and the decay heat assuming failure of most of the fuel pins.

B - How Strong is the Cladding?

A fuel pin is a miniature pressure vessel. The fission gas is collected in a plenum at the end of the fuel pin. Hence, in order to release the fission gas to the fuel cavity, the fuel pin cladding must first fail. There is a feeling that irradiated fuel zircaloy cladding is as brittle as glass. This is not true. Of course, irradiation will produce some brittleness in zircaloy. However, the high temperature of the clad during an accident will increase its ductility. The fuel will still behave as a metal. Bend tests performed on fuel irradiated to 5-10 x 10^{21} n/cm^2-sec (3 years of full operation) show that this fuel could take up to 5.1% strain before failure with yield stresses in excess of 85,000 psi. Analyses and experiments indicated that it would not fracture during accident loadings.[5] Fuel is cushioned during an accident by the cask internals. Reference (6) states that:

"The only mechanism that could result in a significant release of nongaseous fission products involves the creep rupture of a large number of rods as the fuel temperature approaches the maximum postulated upper limit of between 1200 and 1500°F. The current generation of reactor fuel probably would not perforate at the lower temperature; however, at the upper limit, particularly if the fuel rods are prepressurized, a significant fraction would rupture."

Fuel clad temperatures during normal shipment will be 800°F or less. Hence, a long, hot fire is necessary for a number of rods to fail and release fission gas to the cask cavity. On a relative basis, the cladding is quite strong and should not fail during normal shipment or due solely to an impact of the cask body.

C - How Much Fission Gas Could be Released?

This is an area where there is considerable conservatism. The NL 10/24 cask carries about 2000 nuclear fuel pins; each one having experienced different burnup and thermal conditions during its operating lifetime. Safety studies of spent fuel casks have used the isotope inventory guidelines of NRC Regulatory Guide 1.25.[7] This guide was written for storage of spent fuel at reactors - not shipping of spent fuel. The physical means of releasing fission gases trapped within the fuel are extremely sensitive to UO_2 temperature during operation in the reactor. This is related to the fuel specific power during the operating liftime and which varies considerably. Taking Kr-85 as an example; if the fuel were operated at maximum specific power continuously, about 30% of this gas would be released from the UO_2 ceramic matrix to the fission gas plenum. This is rarely the case. Typical or average fuel-operating conditions result in only 0.3-3% of the fission gas being released. The same is true, in a relative sense, of the other volatile fission products, (I-129, I-131, and H-3). The release fractions have been experimentally verified using highly irradiated fuel. Use of Regulatory Guide 1.25[7] would predict 10-15,000 curies of fission gas in the NL 10/24. More realistically only 500-1500 curies would be present (see Table 1).[8] If fuel were not shipped for five years or more (versus a four-month period assumed in this estimate), the fission gas quantity would be far lower due to isotope decay.

Much attention has been focused on the possible release of Cs-134 and Cs-137.[6] Cesium is normally a solid and is volatized at 1240°F. It would take 2-4 hours of an intense enveloping fire (considered impossible by our previous studies) to convert this solid to a gas. Cesium is also highly reactive and would undoubtedly compound with the enormous metal area present within the cask or alternatively condense on the cooler metallic surfaces of the cask prior to being released.[8] The probability of cesium gas escaping from a cask during a fire accident is extremely low. In conclusion, without a fire for dispersion, the amount of cesium released as a gas is essentially zero.

D - What Would Have to Happen to the Cask?

It is easy to state that somehow, in some way, the cask could conceivably be in an accident where the fission gas was released. Looking at the NL 10/24, what events must first happen to make even a gas release plausible? As a minimum, all of the following key events would be required prior to postulating a gas release.

(1) Cladding Failure - As noted, this would require an intense all enveloping fire for a sustained period.

(2) Closure Seal Failure - The seals are well recessed within the body of the cask and protected by solid steel and the impact limiters. The most likely cause of failure would be heat from a fire or a combination of fire and high mechanical force affecting the seal region.

(3) Loss of Impact Limiter - The impact limiter protects the head and seal region from any mechanical damage. It also serves as an insulating jacket during a fire. A high mechanical force would be required to shear the impact limiter from the cask.

(4) Puncture of the Neutron Shield - The 9-inch thickness of water in the neutron shield will absorb an enormous amount of heat during a fire accident. The jacket also serves as an impact absorber. It, like the impact limiter, would have to be in a failed condition in order for fire or mechanical loads on the cask to have any substantive effect. Loss of the neutron shield fluid would require either a high-level puncture loading or an extended (1-2 hour) fire.

This list indicates that a number of key events must occur or multiple barriers be breached before a release is possible. Examining the protected location of the cask on the rail car, it is difficult to postulate how this would occur. However, studies of past railroad accidents do show many strange, random-type events. One thing is clear, the accident must involve both a major collision and a hot, long duration, all-enveloping fire in order for a release of RAM to occur. This type of accident is unlikely to take place in a rail classifying yard or near a major population center. The severe forces assumed require a high-speed accident which is unlikely in these areas.

IV. POSSIBLE ACCIDENT SCENARIOS

Generic accident scenarios were constructed based upon events which have been observed in an actual railroad environment. These key events were correlated to the types of package damage required to lead to the assumption that a RAM release was even possible. Even under these situations, one must ask an additional question. Do the accident conditions themselves affect the dispersion of the fission gas, and how does this dispersion affect the accident consequences? For example, fire combustion gases and currents will rapidly spread and dilute the fission gas. Since the fuel pins fail over a time period, the actual release may be either instantaneous or gradual; and the accident will take place over a several-hour period. Also, consider that during a major railroad accident, there should be sufficient time to move bystanders 2000-3000 feet away from the accident. Typically, none of these considerations are adequately considered in accident studies. It is generally assumed that an instantaneous, cold air release (which is more severe) occurs.

The biological consequences, of course, relate to the exposure of the population downwind and the possibility of condemning land, crops, animals, due to long-term food-chain ingestion considerations.

The scenarios considered are the following:

- Is the fission gas release instantaneous or gradual?
- Do the fire convective currents disperse the gases or does a cold-air release occur?

Three scenarios, embodying all of these considerations, were examined:

Case 1 - A major collision, followed by a long duration (5 hours) all enveloping fire. The gas release is assumed to be instantaneous.

Case 2 - A major collision followed by a long duration (1-2 hours) enveloping fire. In addition, a torch fire is localized in the seal region. The gas release is gradual (the release occurs over a two-hour period).

Case 3 - A major collision and fire occurs. After the fire dies out, there is a cold-air release of the fission gas.

Based on knowledge of cask characteristics and railroad accidents, the Case 1 scenario is more likely than Case 3. In fact, Case 3 seems impossible. The relative consequences of Case 1 are generally less than Case 3. This is due to the diluting effect of the fire in spreading the gas. However, in Case 1 there is a small possibility of about 200 Ci of volatile cesium being released. This possibility was included in Table 2. There is no possibility of cesium being released in Cases 2 and 3 due to the cooler temperatures of the cask. Table 2 presents the immediate dose to someone 3000 feet directly downwind from the accident for the three cases. In all cases, the relative dose is seen to be very small. Other studies show if land condemnation were required, the area affected would be less than a few acres.

ACKNOWLEDGEMENTS

Structural analyses and the coupler puncture test were directed by Dr. T. Branca and Mr. J. Henry of John Henry Associates of Waltham, Massachusetts. Thermal analyses were performed by Ridihalgh, Eggers, and Associates of Columbus, Ohio. The meteorological and dose evaluations were performed by Mr. W. Boone of Allied-General Nuclear Services. Discussions on fission gases were held with Mr. R. Lorenz of Oak Ridge National Laboratories.

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(1) Unpublished Analyses of NL Rail Cars by Dr. T. Branca of John Henry Associates - February 1977.

(2) Letter from C. W. Smith (General Electric Company) to W. J. Harris (AAR) dated September 26, 1973.

(3) Amendment to 49 CFR Parts 177, 179 - Docket HM-144, Notice 76-12 dated November 29, 1976.

(4) Safety Analysis Report for the NLI 10/24 Spent Fuel Rail Package - Docket No. 71-9023.

(5) General Electric Submittal to AEC dated August 13, 1973, on Pin Behavior of Fuel in GE IF-300 cask.

(6) NuReg-5069 - Supplement II to WASH-1238 dated July 1976 - Environmental Survey of Transportation of Radioactive Materials.

(7) U. S. Nuclear Regulatory Commission - Regulatory Guide 1.25 - Assumptions Used for Regulatory the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling in Storage Section of Boiling and Pressurized Water Reactors.

(8) Per discussions with R. Lorenz, Chemical Technology Division, Oak Ridge National Laboratory, based on Fuel Scale Research on Fission Gas Release from Irradiated LWR Fuel.

TABLE 1

FISSION GAS QUANTITY IN NL 10/24 CASK(1)

ISOTOPE	ESTIMATE PER REGULATORY GUIDE 1.25	ESTIMATE(2) FOR TYPICAL FUEL
Kr-85	13,060 Ci	1306 Ci
H-3	19	1.9
I-129	.64	.0064
I-131	.04	.0001
TOTAL	13,080 Ci	1308 Ci

(1) Based on 10 PWR fuel assemblies -- shipping after six months.

(2) Dose could include volatile-cesium 135 Ci of Cs-135 and 67 Ci of Cs-137 -- however, it is highly unlikely that it would be released from the cask.

TABLE 2

ACCIDENT DOSE DUE TO FISSION GAS RELEASE FROM CASK ACCIDENT

Case	ORGAN DOSE (excluding food chain)	ORGAN DOSE (including food chain)
1*	< 1 mrem	7 mrem
2	< 1 mrem	2 mrem
3	< 1 mrem	2 mrem

NOTES:
1. Whole Body Dose.
2. Dose measured 3000 feet from accident origin.
3. Dose based on realistic estimate.

* Case 1 - massive fire assumes 200 Ci of volatile cesium 135, 137 is released. No cesium is released in Cases 2 and 3.

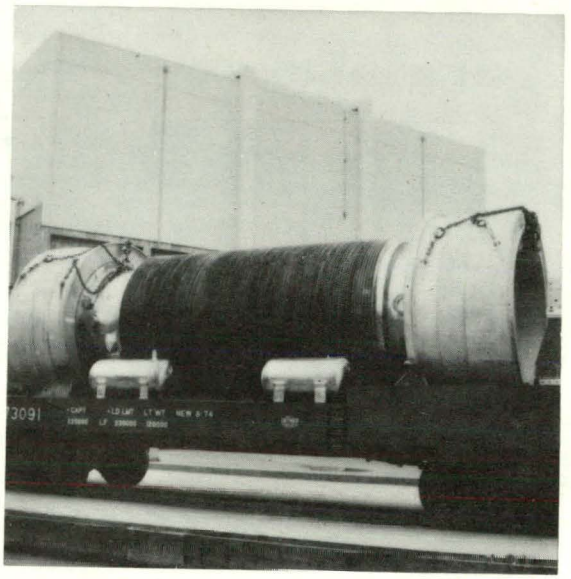
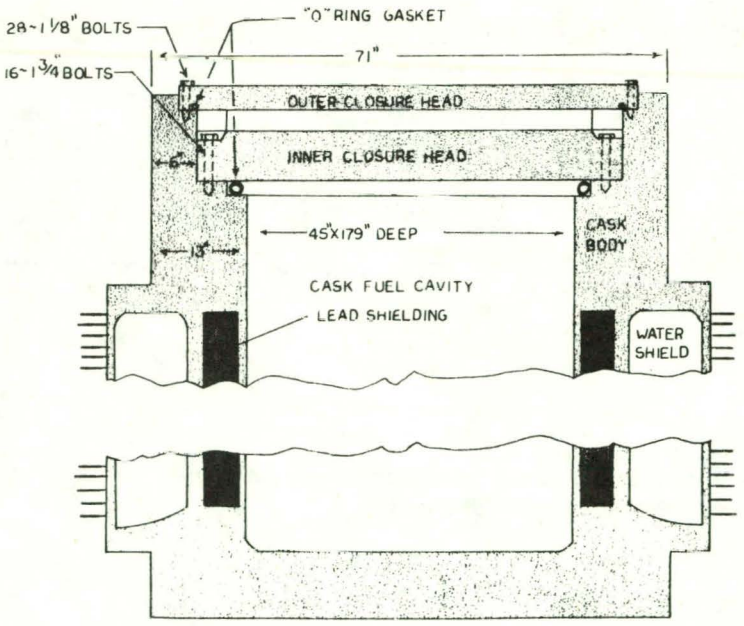


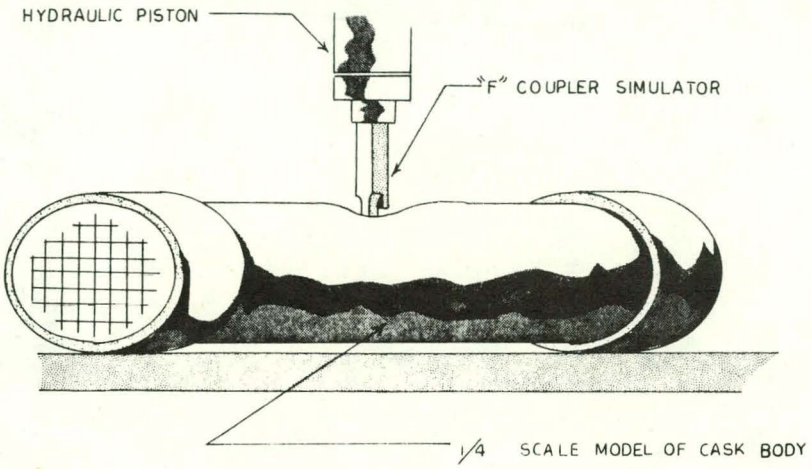
FIGURE 1 - NL 10/24 SPENT FUEL RAIL CASK



NL 10/24 RAIL CASK BODY
FIG. 2

FIG. 3

NL 10/24 QUARTER SCALE CRUSH TEST



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TRANSPORTATION

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EXTENDED TESTING OF PLUTONIUM SHIPPING CONTAINERS

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Initiated by the worldwide discussion concerning the shipment of Plutonium and its compounds, the German Department of the Interior gave a research program to the "Bundesanstalt für Materialprüfung (BAM)" Berlin, to make investigations on the behavior of Pu shipping containers when being exposed to testing criteria which lie beyond the Type B requirements, as given by the 1973 IAEA regulations. These tests were performed to evaluate the margins of safety of these types of packagings, which are normally light ones.

In the Federal Republic of Germany, BAM is the competent authority for the estimation of the design and the testing of prototypes of Type B and packagings of Fissile Classes. The certificate of approval due to the 1973 IAEA regulations will be issued by the "Physikalisch-Technische Bundesanstalt (PTB)" Braunschweig, based on the BAM official test report.

On behalf of the research program, three problems had to be solved at first:

- choice of packagings, which are representative for the shipment of Plutonium compounds (Pu nitrate solution, Pu dioxide) in FR Germany,
- to define how to extend and perhaps modify the test requirements as given by the 1973 IAEA regulations, and
- to establish testing criteria for further accident stresses not yet covered by the regulations, which might endanger these types of packagings.

It was our intention to determine the margins of safety of Pu shipping containers which already are in use, to give proposals to improve design and construction and to think about making worthwhile contributions to the IAEA regulations.

The following types of packagings were chosen and had to pass the extended testings:

- for the shipment of Pu nitrate solutions (liquid) the L-10 and the FL-10 container from the US and the German Packaging 18B,

- for the shipment of Pu dioxide (pulverous, solid, granulates) the 6M2, 6M3, 6M4 and 6M5 container and the LLD-1 container from the US.

The extended testing criteria were carried out as follows:

1. Extended IAEA test requirements

   - impact test from a height from which the packaging can reach its free fall speed onto a hard surface (2oo m)

   - punch test from a height of 2 m

   - open fire test with a duration of at least 75 minutes

2. Testing criteria set up by BAM (also see paper by Mr. Schulz-Forberg)

   - crush test by using a crush load of 2.000 kg ($1 m^2$ base) from a height of 9 m onto the packaging

   - ripping test for packagings with a thin wall construction

   - 9 m impact test while the packaging is cooled down to 233 K ($-40^{\circ}C$).

Extended Fire Tests

Investigations started with the performance of open fire tests on packagings for the shipment of Pu nitrate solutions. Failure threshold for containers with liquid contents seems to be the bursting of the inner receptacle. Therefore we built an open fire test facility (picture 1) to avoid probable damaging of a furnace test facility.

First test object was the 18B container, where the maximum allowable contents of 10 l Pu nitrate solution (350 g Pu/l) is shipped in a titanium bottle. Insulating material is phenolic foam. For the extended fire test, the liquid contents was simulated by 10 l of water, heated up to 408 K ($135^{\circ}C$), representing the source temperature. Instrumentation was done by 11 steel-shrouded thermo-couples fixed to different measuring points in and at the container (fig.1). The container was put in a vertical position onto two steel girders and exposed to a fuel fire for about 75 minutes (mean temperatures about 1123 K ($850^{\circ}C$). Fig. 1 shows the measured and registered temperature curves. Notice the decrease of the inner container's temperature (measuring point 1). Due to test conditions, we had to stop the heating during the test. Picture 2 shows a cut of the container. The phenolic foam has been charred to one third to one half, so that the packaging certainly would have resisted the stress of 180 min fire.

Next testing object was the well-known L-10 Class II container, where the Pu nitrate solution is shipped in a polyethylene (PE)-bottle set into a massive steel tube with a flange cover and an elastomere O-ring gasket (optional). Here we have vermiculite as insulating material. We left out the PE-bottle, for the steel tube with flange cover is to be seen as containment system.

The container was put in a horizontal position onto the two steel girders and exposed to the fuel fire for 90 minutes. The other test parameters were the same as before. The plotted temperature curves are shown in fig. 2. The maximum inner temperature got up to 523 K (250°C) after 4 hours. After complete cooling we found some water dripping out of the flange connection.

Since we had no correct statement of failure, we had to do a second test, where we put a waterfilled PE-bottle into the tube. The tube was then heated by heating wires following the temperature curve we registered from the open fire test (measuring points 2 and 11). Leakage of the PE-bottle could be observed at 453 K (180°C) and leakage of the tube at 468 K (195°C) due to the failure of the elastomere O-ring gasket. The latter failure must be due to the positioning of the O-ring gasket at a shoulder in the flange-ring and not in a groove, as former BAM investigations found out to be advantageous.

On behalf of the phenolic foam insulation of the FL-10 container, we could leave out the extended fire test of the FL-10 packaging.

To verify our test results and to look for a good comparison, we did some computer runs to calculate the respective temperature curves of the 18B, the L-10 and the FL-10 container for a thermal stress of 90 minutes (see fig. 3). Here you see the good correlation between measured and computed data, the satisfying behavior of the 18B and the FL-10 container against such thermal stress, as well as the failure of the L-10 container. The termal conductivity of the insulating materials has been determined by BAM before.

Extended Drop Tests

To determine the packagings' resistance against higher impact forces, we wanted to perform drop tests from a height from which the packaging could reach its free fall speed. Due to calculations of the cylinder-shaped container's drag, all drop tests were carried out from a height of 200 m.

Depending on the political and geographical situation of Berlin, BAM thankfully appreciated the friendly assistance of 7th Flight, British Army Air Corps, to perform the drop tests on an abandoned airbase area at Berlin-Gatow. Here we have a surface consisting of 10 cm asphalt on a 30 cm thick layer of broken bricks. Each packaging was dropped in a vertical position hanging below the helicopter (see picture 3). Documentation was done by taking photographs before and after the respective drop as well as taking pictures by a 16 mm handcamera and a high-speed camera.

We started performing the extended drop tests on the 18B container. The load has been simulated by a dummy cylinder, welded of 2 mm sheet metal corresponding to the original's size. Payload has been 10 l of water. Shortly after releasing, the packaging got into a horizontal position, so that the impact happened on the surface line. We had a rebound of about 1,5 m up and 2 m aside. Besides a flattening of the whole surface line of about 100 mm, no further deformation could be detected. The lid bolts were loose, so that they could be removed by hand, but the lid cover has been drawn into the packaging and could only be opended mechanically. No leakage could be observed.

We did the same testing on another 18B packaging, because we wanted to expose it to a succeeding 30 minutes open fire test. And it survived satisfactory.

After analysing the pictures taken by the high-speed camera, we had free fall speed of about 55 m/s. This is a good correlation to our prior

calculations as well as to the time we measured by stop watches.

We had one impact onto the surface and one onto frozen soil. This could be assumed as sufficiently representative. Next drop tests we performed were on the FL-10, the 6 M packaging series and the LLD-1. Test parameters remained the same, each packaging hit the bitumen surface. We used a dummy payload of 10 l of water in the PE-bottle for the FL-10 container. Impact performance was analogous to the 18B, but here we had the well-known cracking off of the clamping ring closure. This would lead to the packaging's failure, if thermal stress would follow the mechanical one, for flames could directly attack the flange cover (see picture 4). After cutting the packaging in half lengthwise, we observed the strong compression of the tube near the flange due to the wooden spacer (see picture 5). When loosing the flange bolts, we had an outlet of water.

The loading of the Pu-dioxide shipping containers, the 6 M packaging series and the LLD-1, was simulated by a respective dummy payload of metal scrap. The 2 R inner containers were screwed handtight. For the 6 M 5 and the 6 M 4 container, we got the lid closure popping off again, so that the inner container could be stressed by a following accident fire (see pictures 6 and 7). The 6 M 3 and 6 M 2 containers, having a better length-to-diameter ratio, had a better performance (see picture 8 and 9), for they did not take in a horizontal drop position. That is the reason why we only had slight damaging of the bottom plates.

The wooden transport box of the LLD-1 container was completely destroyed. Bottom and lid cover of the bird-cage framed cylinder sheered off and the 2 R inner container was able to get out, and so be accessible to a succeeding thermal stress (picture 10).

Extended Punch Test

The 18 B container was exposed to a 2 m punch test, i.e. doubling the drop height prescribed by the regulations. The packaging was dropped in a horizontal position. It got a rebound of about 30 cm, then rolled aside. There was only a minor dent in the outer steel wall of about 5 cm. Pictures of the high-speed camera show plastic and elastic behavior of the phenolic foam. We had no bending of the inner tube.

Additional investigations of the insulating material's ability of energy absorption proved, that none of the packagings we wanted to pass this kind of test would have failed. Summary of data can be seen from fig. 4 , so we left out further test performances.

If we think of further extension of the punch test, we should either go to a drop height which might be two or three times higher (4 m to 6 m) or performing a kind of reverse test by dropping a punch-shaped drop weight onto the packaging.

Crush Tests

To determine the resistance of light packagings against stresses caused by crush forces, BAM performed crush tests on the 18 B, the L-10 and the FL-10 container. The 2 R inner container put in the 6 M packaging series and the LLD-1 in addition to the outer container seemed to be strong enough to withstand high mechanical stresses.

At the BAM drop test facility at Berlin-Grunewald we dropped a crush load of 2.000 kg from a height of 9 m onto the packaging lying flat on the unyielding target. The crush load consists of two massive cast steel ingots fixed on a 20 mm steel plate of 1 m^2 (picture 11). Documentation

was done by taking photographs before and after the test, as well as taking pictures with a 16 mm hand-camera and a high-speed camera.

The first crush test was performed on the 18 B container. We did not e a dummy payload. The test performance lead to the compression of the ntainer from a diameter of 600 mm to 230 mm at the impact area, the inner tube was pressed from 145 mm to 32 mm. Besides the sharp kink (picture 12), the integrity of the container remained, so we would not get the spreading of the liquid contents at once, if such an accident happens.

Next container to be tested was the L-10. Test parameters were the same as before. Here we had a compression of the drum from 570 mm to 300 mm at the impact area, the clamping lid closure popped off and loose vermiculite poured out. The birdcage frame was pressed down to the tube as well as to the flange ring and there was a bending of the tube, so that the PE-bottle could not be removed by hand (picture 13), but there was no leakage. Looking at a succeeding fire, there will be failure too.

The FL-10 container did a similar performance (picture 14). It survived the crush stress, too, but failure will occur, if there is a thermal stress following. But we learnt, that the concept of having a massive steel flange is sufficient to withstand a crush stress. So we are now going to modify the design of the 18 B container by using a reinforced inner tube with at least three massive steel rings around. We expect this improvement, together with the concept of a drawn-in closure to lead to a good performing packaging.

Summary

BAM investigations are still going on by performing ripping tests and drop tests with packagings cooled down, but the test results we got up to now proved that phenolic foam ist a very good insulation against fire and absorbing mechanical energy. It is superior to fluid insulation materials, felt or Celotex. Local damage of the thin wall construction does not lead to the loss of thermal insulation. Sufficient thermal resistance of at least 180 minutes could be calculated by computer program and verified by experiment meanwhile. Even after the mechanical stress of a 200 m drop, we find good resistance against a 30 minutes fire. A nearly plastic behavior implies good protection against mechanical stress. While exposed to crush stresses, there will be a good performance by some additional improvements, such as stiffening rings. For the lid construction, there should be a bolted version of the closure, similar to the 18 B.

Concerning all types of Pu shipping packagings, which we exposed to the extended testing criteria, the margins of safety lie far beyond the limits as given by the IAEA adopted regulations. Though the behavior against thermal stress is much better than the one against mechanical stress, we think that with a little constructive improvement, it will be possible to build a packaging for the shipment of Plutonium compounds, which will be safe enough to withstand at least the above mentioned testing criteria.

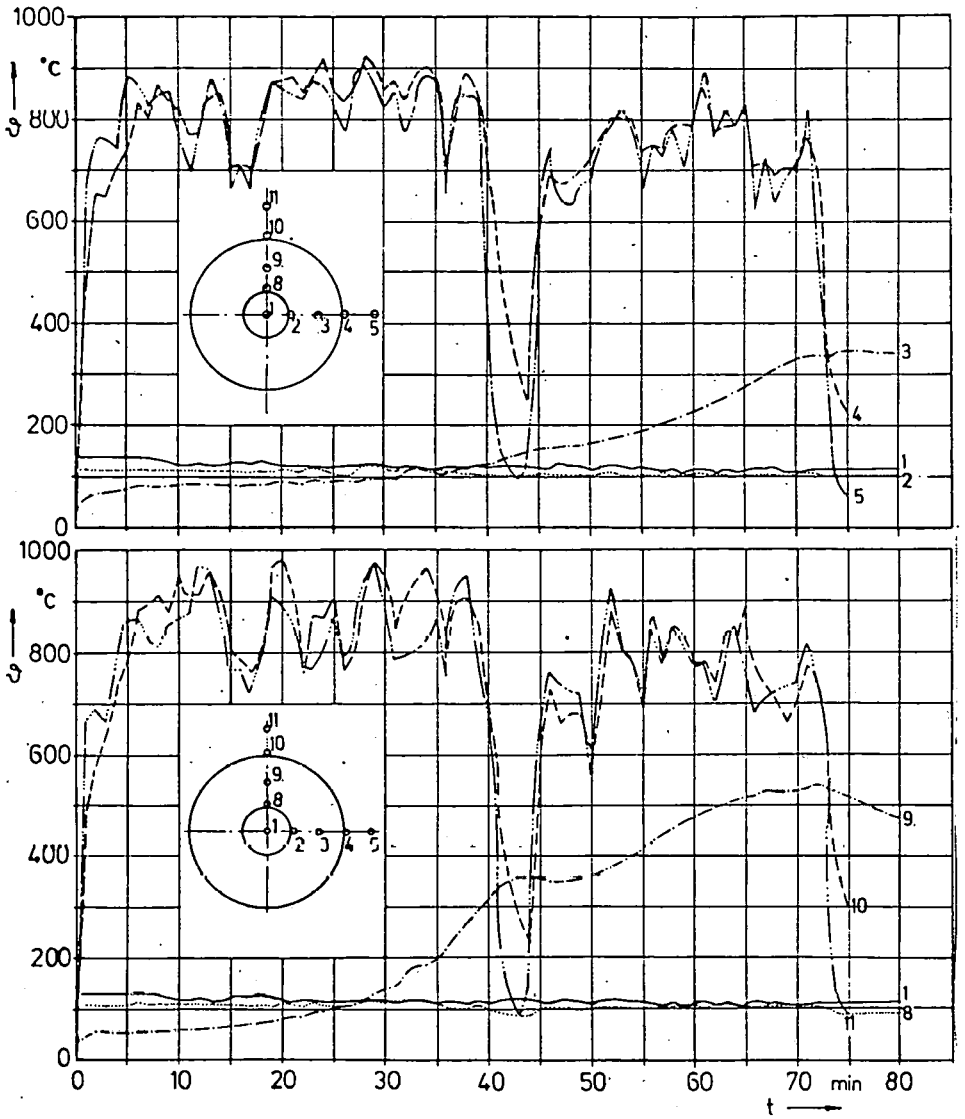


FIGURE 1 18 B Container in a 75 min Open Fire

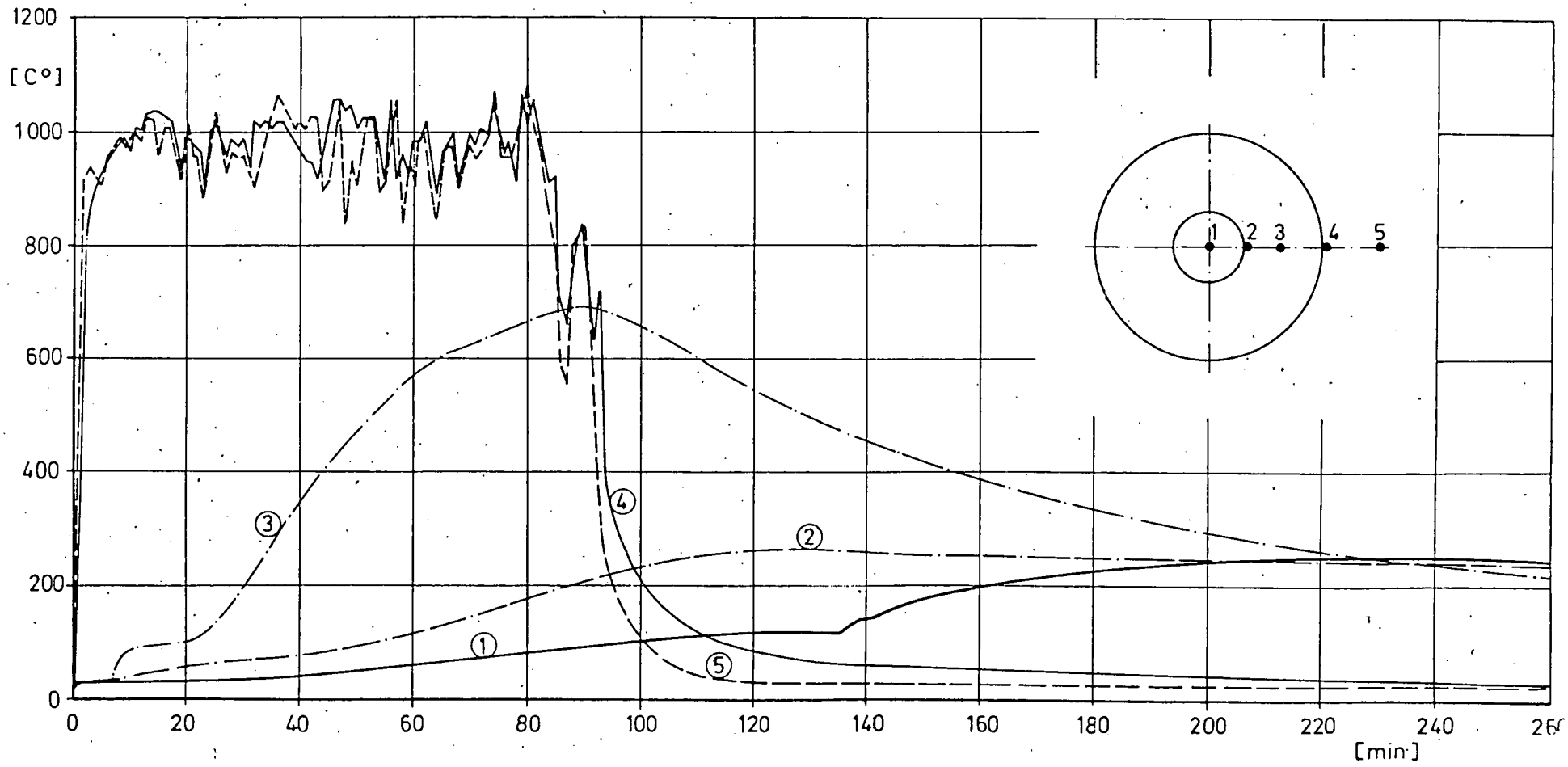


FIGURE 2 L-10 Container in a 90 min Open Fire Test

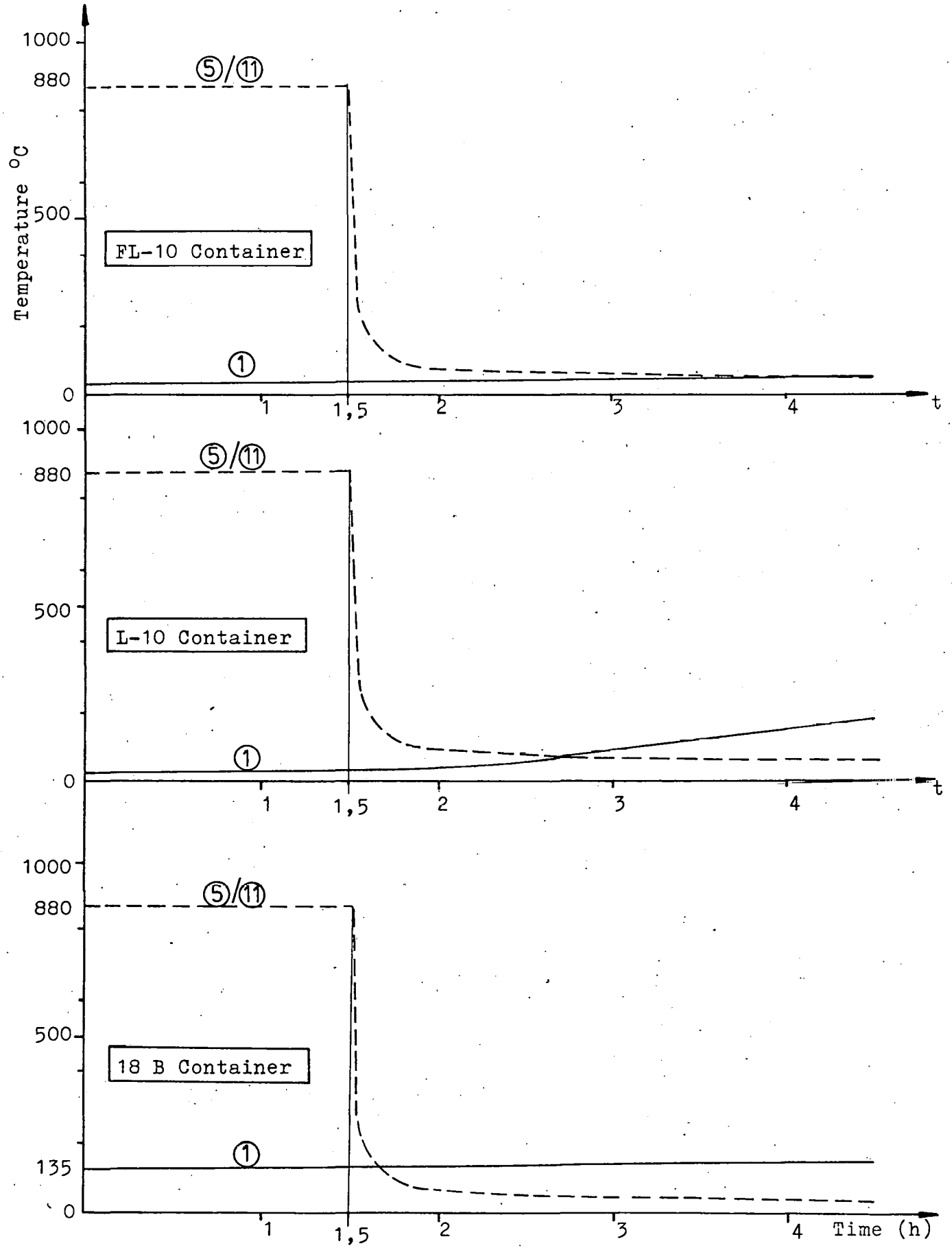
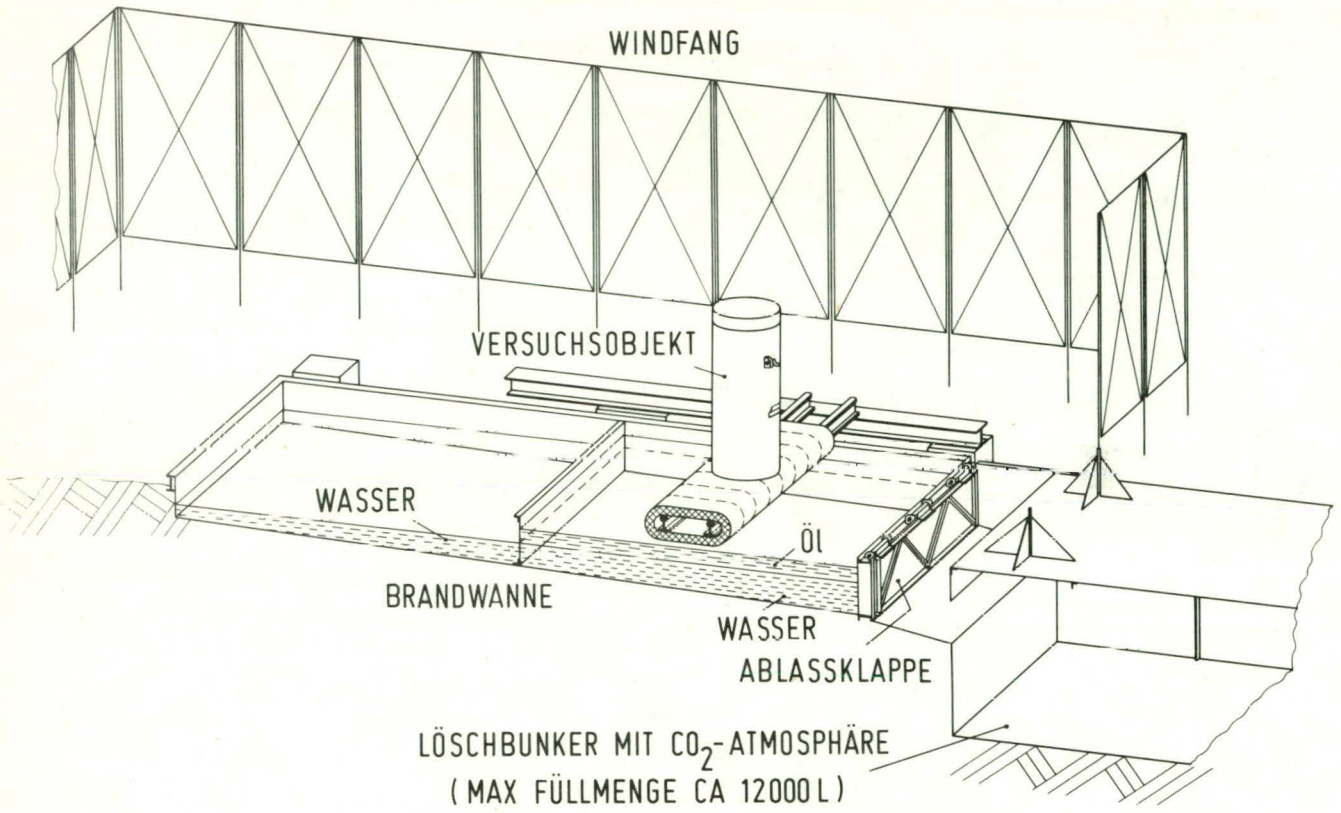


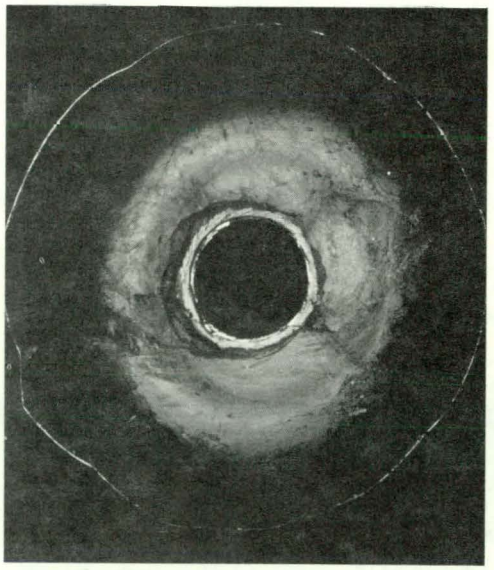
FIGURE 3 Packagings for the Transport of Pu-Nitrate Solutions in a 90 Minutes Fire, 1153 K (880°C)

TYPE OF PACKAGING	DIAMETER [mm]	LENGTH [mm]	WEIGHT [kg]	NATURE OF CONTENTS (max) [l,kg]	INSULTING MATERIAL	ABILITY OF ENERGY ABSORPTION [Nm]
18 B	600	1700	150	10; Pu-nitrate	Phenolic Foam	273
L-10	565	1776	230	10; -"-	Vermiculite	99
FL-10	565	1776	230	10; -"-	Phenolic Foam	135
6M5	580	1790	216	16; Pu-dioxide	Felt,Celotex	160 - 300
6M4	580	900	108	13,9; -"-	Felt,Celotex	-"-
6M3	580	600	59	9,6; -"-	Felt,Celotex	-"-
6M2	410	540	43	7,6; -"-	Felt,Celotex	-"-
LLD-1	□ 410	640	50	7,0; -"-	None	None

FIGURE 4



Picture 1. BAM facility for open fire tests.

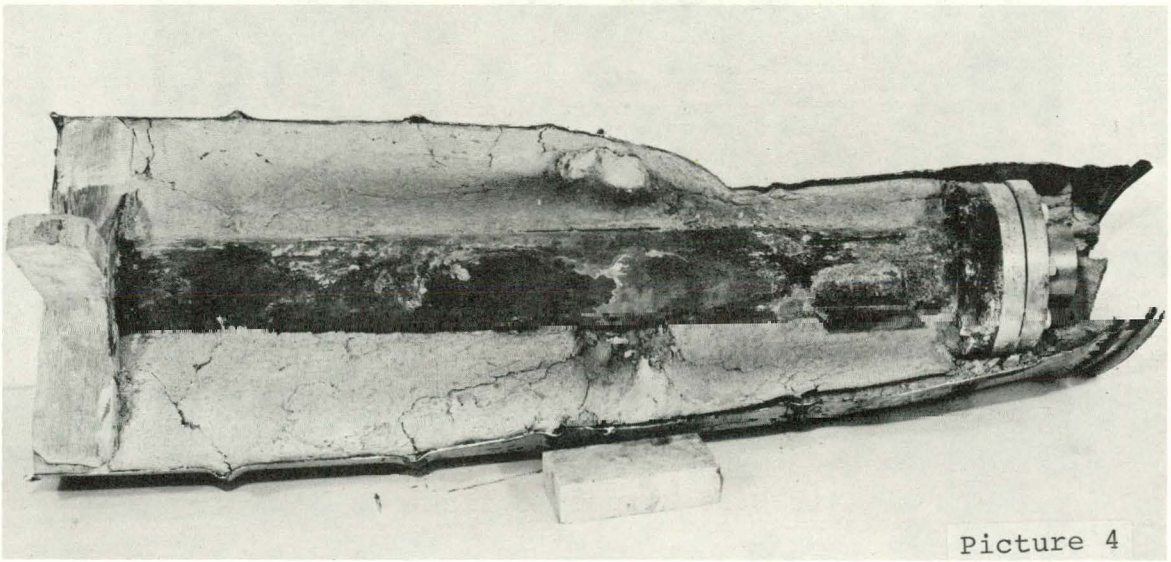


Picture 2. Cross section of 18 B container after 75 minute open fire test.



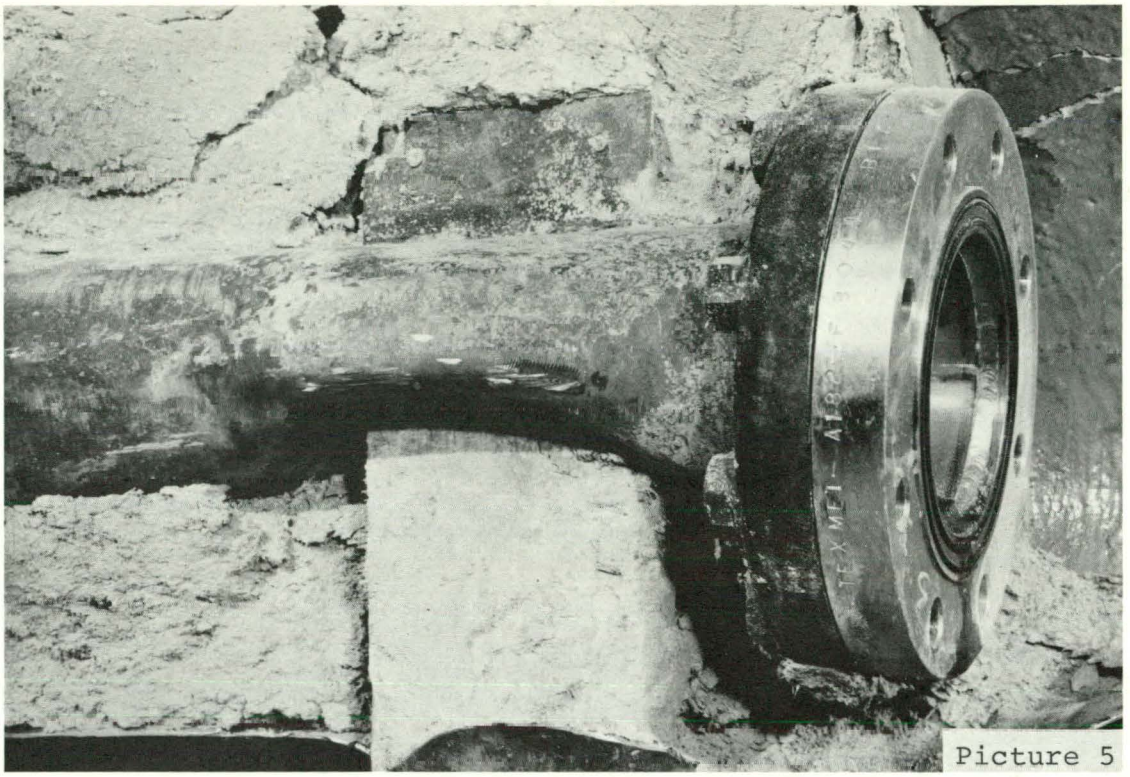
Picture 3

200m Drop Test on Light Pu-Packagings by Gazelle Helicopter (British Army Air Corps)



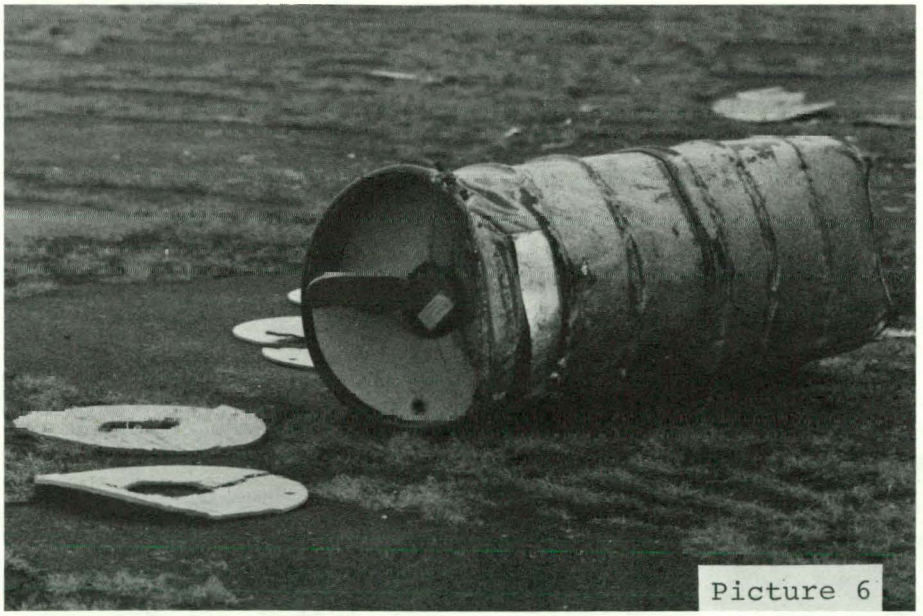
Picture 4

Cut of FL-10 Container after 2t Crush Test



Picture 5

FL-10 Container after 200m Drop;
Failure of Inner Tube



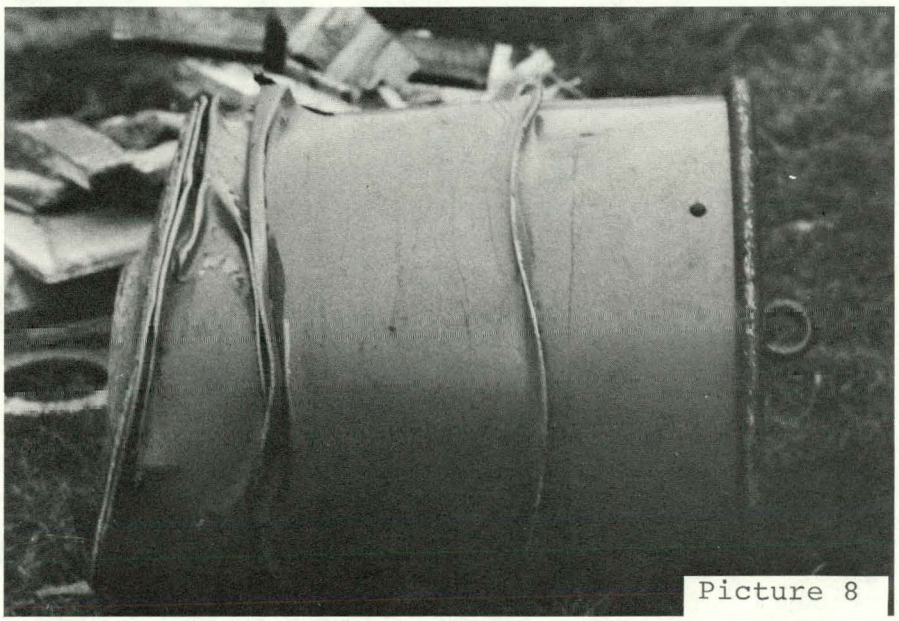
Picture 6

6M5 Container after 200m Drop onto an
Air Base Bitumen Surface



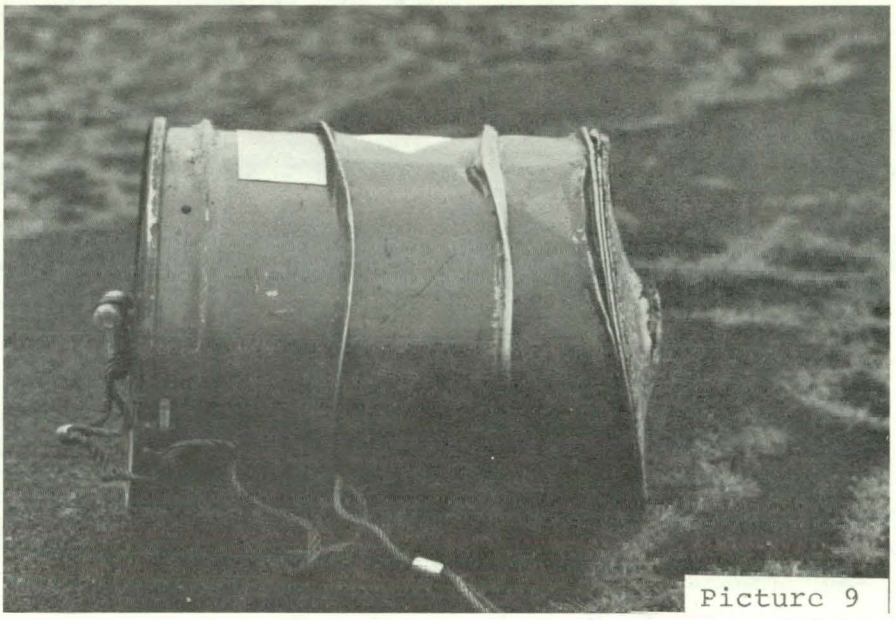
Picture 7

FL-10 Container after 200m Drop onto an Air Base Bitumen Surface



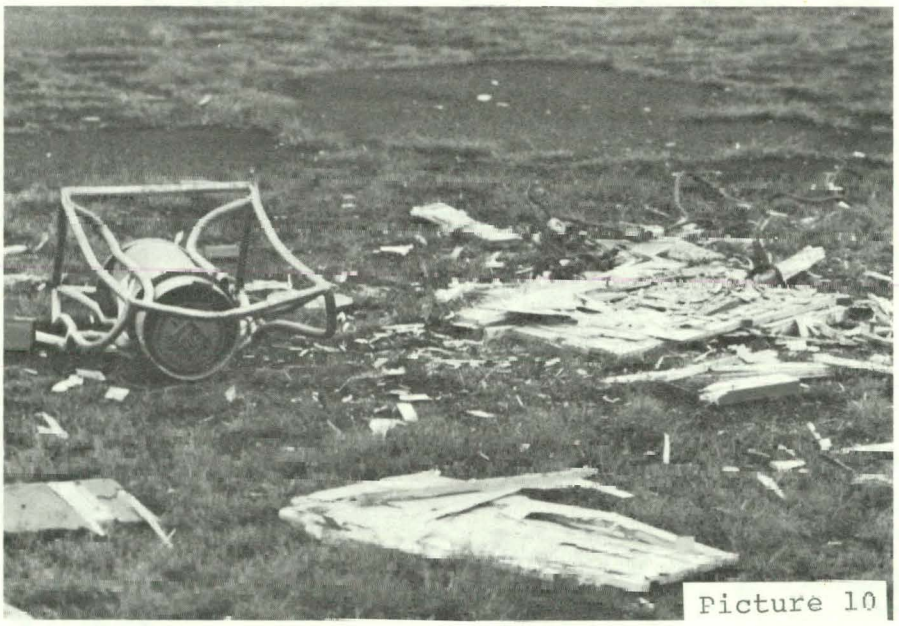
Picture 8

6M3 Container after 200m Drop onto an Air Base Bitumen Surface



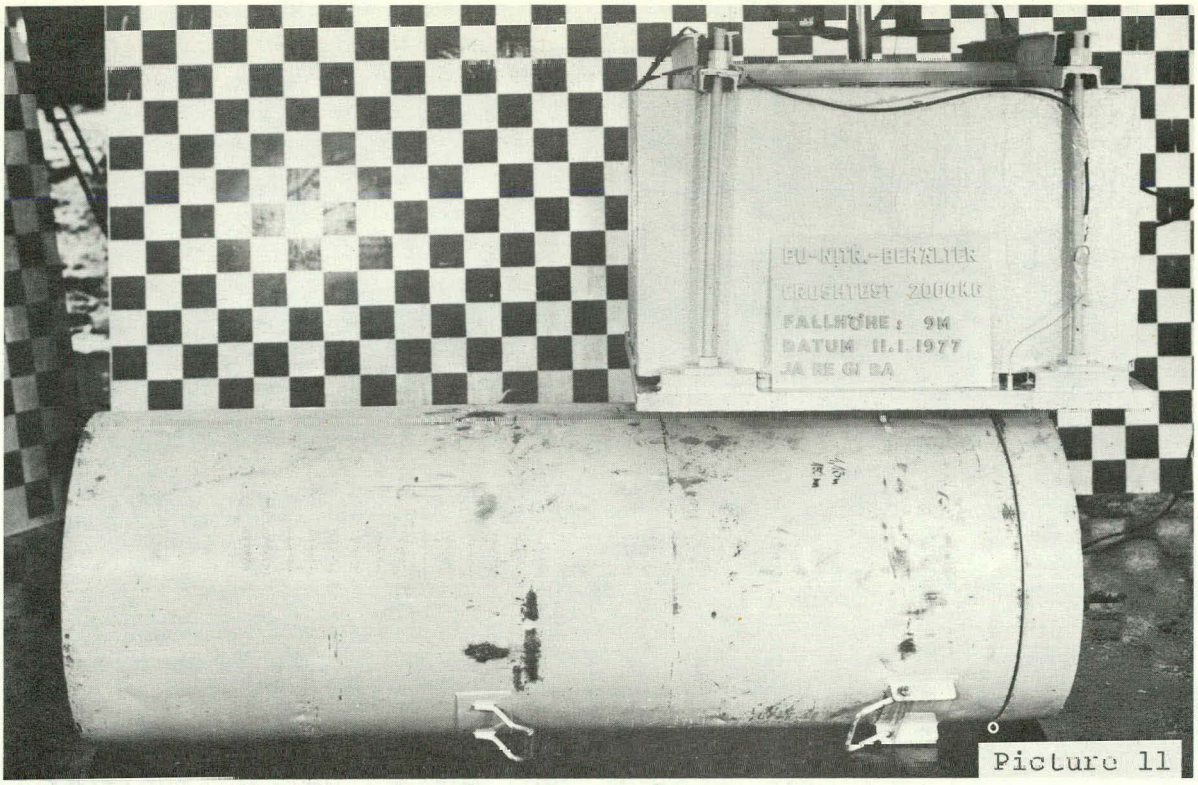
Picture 9

6M2 Container after 200m Drop onto an Air Base Bitumen Surface



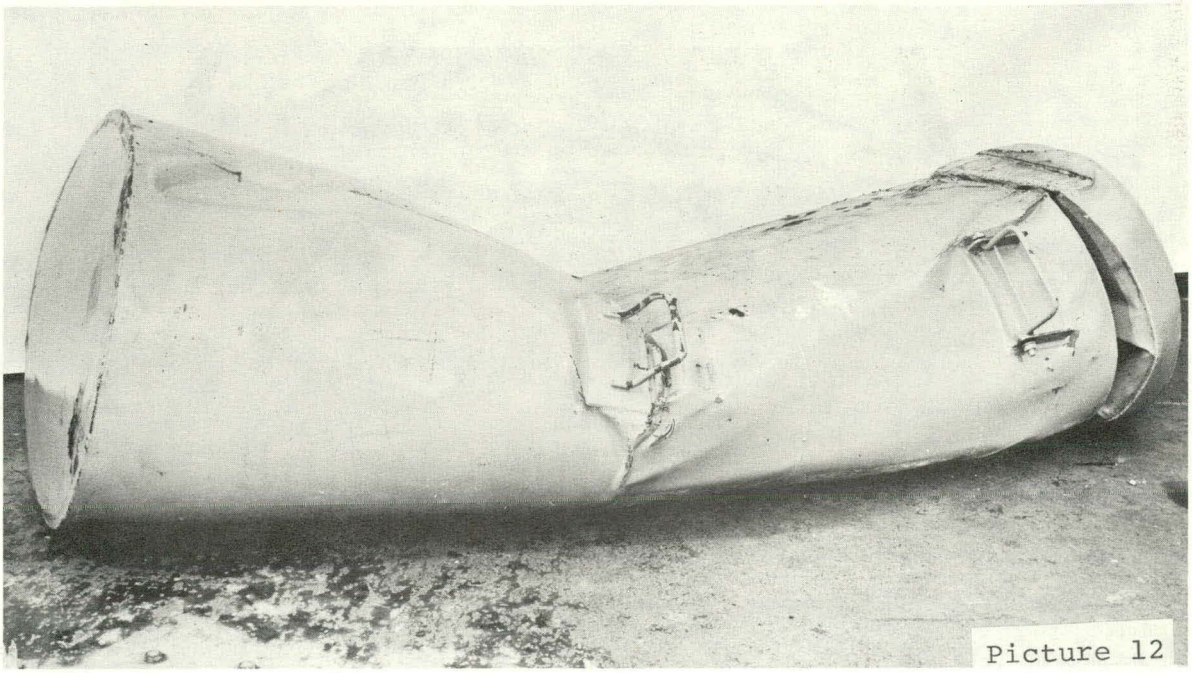
Picture 10

LLD-1 Container after 200m Drop onto an Air Base Bitumen Surface



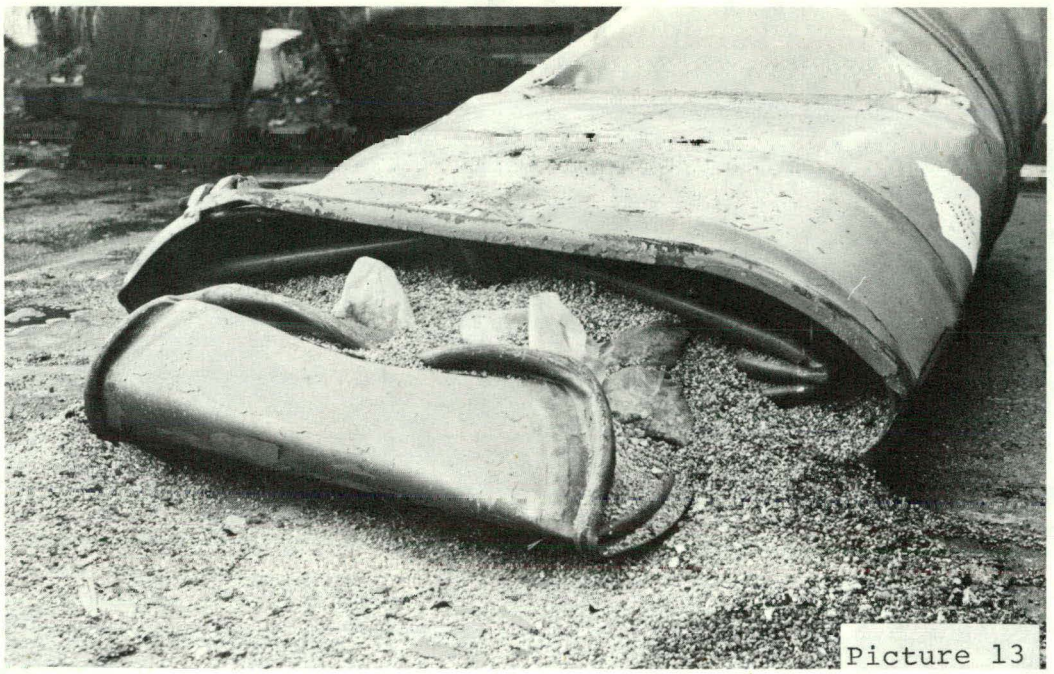
Picture 11

Test Set-Up for BAM Crush Test (2t from 9m onto 18B Container)



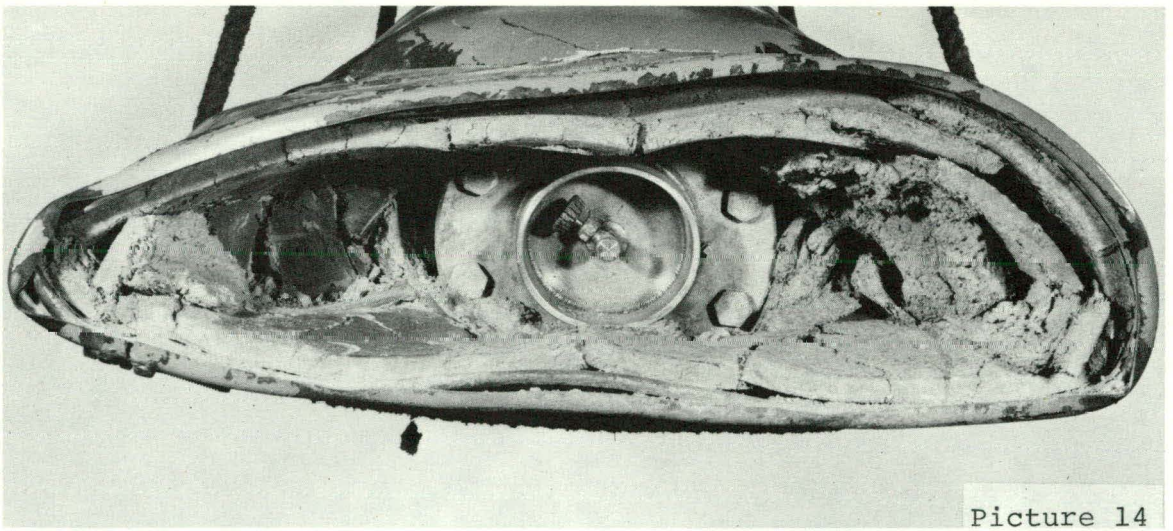
Picture 12

18B Container after 2t Crush Test



Picture 13

L-10 Container after 2t Crush Test



Picture 14

Front View of FL-10 Container after 2t Crush Test

PLUTONIUM ACCIDENT RESISTANT CONTAINER PROJECT*

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A recent United States Public Law (94-79; August 9, 1975) restricts the air shipment of plutonium. This law reads, in part, that no plutonium (except for very small quantities of material in medical devices) may be air transported until the NRC (U. S. Nuclear Regulatory Commission) "...has certified that a safe container has been developed and tested which will not rupture under crash and blast testing equivalent to the crash and explosion of a high-flying aircraft...." Although there were problems of translating the general language of the law to technically meaningful definitions, a very severe accident-modeling criteria was developed by the NRC, who also engaged Sandia Laboratories in the development of a transportation package that would acceptably survive the new criteria. The initial tasks were to promulgate criteria and then to design a package to satisfy those criteria. Both the new criteria and the new package have been presented for approval to the Advisory Committee for Reactor Safeguards and the National Academy of Engineering's Ad-hoc Committee on the Air Transport of Plutonium. Results are being, or have been, presented to the U.S. Congress, as originally mandated.

The NRC criteria became a document defining those measures necessary to qualify and certify a package for the air transport of plutonium; hereafter, this will be referred to as the Qualification Criteria. The Qualification Criteria essentially consists of a test program with supporting rationale and stringent acceptance standards. The rationale embodies a maximum credible accident approach, with very severe single-event accident elements all applied sequentially to the same package. Also, certain individual tests are included as well as a requirement to conform to existing regulations.

That portion of the program done at Sandia was dubbed "PARC" for Plutonium Accident Resistant Container. The PARC project resulted in the development of the PAT-1 (Plutonium Air Transportable) Package. The package was designed concurrently with and in response to the Qualification Criteria and survives the sequential and individual tests of both the new and old criteria and meets the applicable acceptance standards in each case.

The Qualification Criteria are summarized in the next two tables; Table 1 defines the test program of new sequential and individual tests, and also summarizes the tests of the existing regulations, 10 CFR 71.

*Project funded by the United States Nuclear Regulatory Commission.

TABLE 1

QUALIFICATION CRITERIA TO CERTIFY
A PACKAGE FOR AIR TRANSPORTATION OF PLUTONIUM
(NUREG-0360)

Sequential Tests		
Impact	--	129 m/s (422 fps; 250 KTS) perpendicular to flat unyielding target; most severe orientation
Crush	--	310 kN (70,000 lb) through 5.1 cm (2 in.) wide steel bar; most severe location
Puncture	--	227 kg (500 lb) steel probe dropped 3 m (10 ft); most severe location
Slash	--	45 kg (100 lb) steel angle dropped 46 m (150 ft); twice onto package tilted at 45°
Fire	--	Engulfed in large JP-4 fire for one hour; left to self-extinguish
Submersion	--	Under 1 m (3 ft) water for 8 hours
Individual Tests		
Hydrostatic	--	4.1 MPa (600 psi) for 8 hours [411 m (1350 ft) depth]
Terminal Velocity Free Fall	--	Test required if terminal velocity is more than 250 KTS
10 CFR 71 Tests		
Normal	--	Heat, cold, pressure, vibration, water spray, 1.2 m (4 ft) drop, penetration, compression
Accident	--	9 m (30 ft) drop, puncture, fire, submersion

Table 2 summarizes the acceptance criteria, essentially comprising three requirements: containment, shielding, and criticality.

TABLE 2

ACCEPTANCE CRITERIA

Containment of Plutonium

-- Release must be < IAEA A2 weekly quantity following test sequence of new criteria

-- "No release" from double containment following 10 CFR 71 normal or accident conditions measured as a leak rate -- ANSI N 14.5: 10^{-7} cm^3/s -- or as actual loss of surrogate: less than 10^{-8} g, by fluorimetry

Shielding

-- Normal transport - 49 CFR 173 requires that external radiation be limited to:

10 mrem/hr @ 1 m (3 ft), and
200 mrem/hr @ surface

-- Post-Accident - 10 CFR 71 requires that external radiation, following the more severe tests of the new criteria, be limited to:

1000 mrem/hr @ 1 m (3 ft)

Criticality

-- Undamaged single packages and large arrays must be subcritical per 10 CFR 71

-- Arrays of damaged packages must be subcritical per 10 CFR 71, following the more severe tests of the new criteria

In response to the Qualification Criteria a package was designed, analyzed, and developed. The resulting PAT-1 (Plutonium Air Transportable Model 1) package is shown in Fig. 1. The package is 62 cm (24-1/2 in.) O.D., 108 cm (42-1/2 in.) in length, and weighs approximately 227 kg (500 lb) when loaded. Externally, it resembles a 65 gallon commercial stainless steel process vessel. The PAT-1 package comprises an AQ-1 (Air Qualified Model 1) overpack, TB-1 containment vessel, and PC-1 product can, as shown in Fig. 2.

Air Qualified Overpack (AQ-1)

The AQ-1 configuration consists of an outside drum, with at least two layers and in some locations five layers of 16 gauge, 304 stainless steel, an outer and inner grain-oriented redwood assembly with an interstitial load spreader assembly, and a heat conductor element.

The outside drum ends, both top and bottom, are secured in a unique manner, shown in Fig. 3, to permit relatively simple access to the container and yet retain the removable parts in a violent crash.

One factor leading to the selection of redwood as the shock mitigator/ thermal barrier is its high specific energy absorption capability parallel with the grain, as indicated in Fig. 4; redwood outranks most shock mitigators such as elastomeric and rigid foams, aluminum and stainless steel honeycombs, foam-filled honeycombs, and other natural products. This

capability was examined relative to density so as to constrain the final package size and weight while maintaining utility as an industrial air transportable package with a practical (although small) internal payload.

Thermal barrier or char performance was the other principal characteristic leading the choice of redwood. As indicated in Fig. 5, redwood char performance compares favora with other high energy absorbers, even outperforming some fire-retardant-treated materials which were not as efficient, and only slightly outperformed by balsa (too bulky for the air transportable design) and some rigid foams (not as efficient in energy absorption).

New redwood strength data, Fig. 6, were generated to support design analysis across an appropriate broad temperature range. An impact energy balance analysis, Fig. 7, was iterated to optimize the design application of the inner and outer regions of redwood and the interstitial aluminum load spreaders.

Aluminum was chosen for the load spreader assembly because of its favorable mass, strength, modulus, thermal conductivity, cost and availability. Load spreaders were found to be essential to distribute dynamic inertial compressive loading from the relatively small area loading of the containment vessel within the package, to a larger area of the shock-absorbing material. In a side or lateral impact, the tube is the principal load spreader; in an end or longitudinal impact, the discs are the principal load spreaders. Referring to Fig. 2, the extended region of the tube, beyond the discs, deforms inward in a severe corner impact, constricting possible passage of the discs and containment vessel in an outward direction.

This deformation/constriction action also occurs in side and end impacts. Additionally, as shown in Fig. 8, load spreader deformational energy is also accounted for in the impact energy balance, again for design optimization to restrict package weight and size while retaining great toughness.

A photograph, Fig. 9, of the resulting package cross-section shows the double-walled outside drum, the radial grain orientation of the outer and inner annulii of redwood, the load spreader tube, and the containment vessel. The non-removable elements of this assembly are permanently bonded together with a polyester-flexibilized epoxy adhesive which has resilience over a wide temperature range. When impact forces cause deformations, this bond acts in unison with the wooden elements and their adjacent metal elements.

Containment Vessel (TB-1)

The containment vessel, TB-1, Fig. 10, consists of a body, a lid secured by bolts, a copper gasket, and an O-ring. The vessel body and lid are fabricated from PH13-8Mo precipitation hardened stainless steel. The H1075 temper enhances ductility while preserving high strength from low to high temperatures. The TB body and lid are designed with approximately hemispherical end shapes and cylindrical side wall shapes to resist deformation from either external or internal loads or pressures. The lid is hermetically sealed to the body by the use of a ductile copper gasket in conjunction with knife edge sealing beads on both the body and lid, and a pattern of bolts, as shown in Fig. 11. The sealing surfaces are arranged to afford handling protection to the knife edge sealing beads. The lid has a pilot diameter region of great structural shear strength which fits closely into the mating internal diameter of the body. This closely limits any possible radial motion between these parts, especially motion that would be induced from deformations resulting from accidental crash, crush, or puncture loads. This pilot diameter is also equipped with an O-ring in a groove, as a secondary seal to supplement the upper copper gasket and double knife edges, for containment of contents within the TB-1 containment vessel.

The twelve TB-1 closure bolts, 1/2-in. in diameter, shown in Fig. 10, are forged from A-286 stainless steel, with over 30,000 lb ultimate tensile strength per bolt. This material resists corrosion in conjunction with the stainless steel TB-1 body and lid, and provides high temperature strength to maintain the TB-1 seal at elevated temperatures. The bolts are silver plated to prevent galling of the stainless steel bolt in the stainless steel vessel.

A shock mitigation spacer within the TB-1 containment vessel is fabricated from aluminum honeycomb (see Fig. 10) with axial cell orientation. This honeycomb spacer prevents the flat end of the PC-1 product can from entering into the hollow hemispheric lid in the case of severe impact loads in the axial directions. This spacer also serves as a thermal conductor for heat generated by certain PuO_2 contents.

Finite element stress analysis (Fig. 12) was used to define the containment vessel design, leading to optimization of vessel mass, itself a threat to the surrounding shock mitigation material. The vessel was designed to directly accept impact and puncture threat, including armor piercing ballistic attack.

The TB-1 inner container is highly resistant to sea water corrosion and will withstand the hydrostatic pressures specified in the NRC Qualfication Criteria.

Product Can (PC-1)

The PC-1 product can, right side of Fig. 10, is fabricated from stainless steel. It is closed by crimping in a canning machine and is also sealed with an epoxy material. The close fit of the TB-1 containment vessel limits product can deflection or permanent change of shape under severe impact loads. The product can provides double containment under the normal and accident conditions of transport performance tests as specified by 10 CFR 71.42. This product can may be loaded to a maximum weight of 2 kg PuO_2 contents, not to exceed a maximum of 25 watts thermal activity, of PuO_2 of various isotopic composition.

Thermal Analysis

Thermal analyses of the PAT-1 package included finite difference modeling for internal heat load capacity, leading to the 25 watt limitation on thermally active radioisotopic contents; the limiting consideration here was long-term protection of the redwood for preservation of known performance factors. Finite difference thermal modeling was also used in the case of externally applied heat such as the standard hot day conditions, and to predict package performance in the 1010°C (1850°F) large JP-4 jet fuel fire environment.

Test Program

Test activities are best shown by a motion picture, "Plutonium Air Transportable Project", Sandia Laboratories, March 1978. Table 3 summarizes these tests and also indicates results. This table indicates that five PAT-1 packages were subjected to a similar sequential test series, with the initial impact test oriented so as to encompass the five different principle threat orientations of top, top corner, side, bottom corner, and bottom. The crush, puncture, slash, fire, and immersion tests that follow are essentially identical for all packages, with the application point of each test being chosen to produce the most damaging cumulative effect on each package.

Table 3 includes the individual hydrostatic test required by the Qualification Criteria, and high and low temperature engineering development impact tests, applied as the first test in a sequential series.

Before the tests, each package had been loaded with a finely divided surrogate UO_2 powder and helium gas. The results show that no uranium oxide escaped as indicated by a fluorimetric test with a detection capability of 10^{-8} grams. Also, only very small helium leak rates were induced through the containment vessel seals.

Experimental work with actual PuO_2 was conducted under NRC sponsorship at another laboratory (Ref. 10), to correlate the observed helium leak rates with conservative bounding estimates of worst-case possible plutonium loss (again, no surrogate powder escaped). These conservative bounding assessments of plutonium loss were compared to IAEA "A2" quantities, Table 4, by the NRC staff. These assessments demonstrated successful performance of the PAT-1 package, satisfying the criteria for plutonium containment.

Other

Other analyses conducted to determine package conformance to the Qualification Criteria and to 10 CFR 71 and 49 CFR 173 involve calculations for shielding and criticality. These results are summarized in Table 5.

Table 6 summarizes the results from testing the PAT-1 package to the requirements of the present regulations, 10 CFR 71.

The integrity of the inner containment vessel, determined by both analysis and test, is summarized in Table 7.

Conclusion

The PAT-1 package, developed in the PARC program, survives the severe accident-modeling test threats of the Qualification Criteria, meets the acceptance criteria for containment, shielding, and criticality, and provides a safe means for air transport of plutonium.

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6. Transportation, Title 49, Code of Federal Regulations, Part 173, January 1, 1975.
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9. Von Riesemann, W. A., and Guess, T. R., The Effects of Temperature on the Energy-Absorbing Characteristics of Redwood, SAND77-1589, Sandia Laboratories, Albuquerque, NM, 1978.
10. Schwendiman, L. C., et al., Quarterly Progress Report, October 1, 1977-December 31, 1977, Study of Plutonium Oxide Leak Rates from Shipping Containers, BNWL-2260-5/NRC-12, Batelle Pacific Northwest Laboratories, Richland, WA, January, 1978.

TABLE 3

SUMMARY OF NRC QUALIFICATION TESTS, PAT-1 PACKAGE

Impact Orientation	Impact Vel. ⊥ to Unyielding Target, fps	Crush 70,000 lb	Puncture 5000 ft-lb	Slash 15,000 ft-lb	Fire 2200°F 60 Minutes	Immersion	Uranium Detection $\geq 10^{-8}$ g	Post-Test Air Leakage (cm^3 /s)
Top 0°	442	✓	✓	✓	✓	✓	None	$<4.6 \times 10^{-6}$
Top Corner 30°	451	✓	✓	✓	✓	✓	None	$<4.5 \times 10^{-5}$ probably $\sim 1.7 \times 10^{-7}$
Side 90°	445	✓	✓	✓	✓	✓	None	1.4×10^{-6}
Bottom Corner 150°	443	✓	✓	✓	✓	✓	None	$<5.5 \times 10^{-6}$
End 180°	446	✓	✓	✓	✓	✓	None	1.9×10^{-6}
Side* 90°(-40°C;-40°F)	443	✓	✓	--	✓	✓	--	2.4×10^{-6}
Side* 90°(+93°C;200°F)	424	✓	✓	--	✓	✓	--	7×10^{-8}

Individual Test: 600 psig hydrostatic; 8 hours - No detectable water leakage; $\dot{q}$ $<10^{-10}$ cm^3/s

*Engineering development tests

TABLE 4

EXAMPLES OF ALLOWABLE MASS RELEASE RATES -- IAEA "A2", SAFETY SERIES NO. 6

Isotope	Normal Conditions of Transport, μg/hr	Accident Conditions of Transport, μg/wk
Pu^{238}	0.000176	0.176
Pu^{239}	0.032	32.2
Pu^{240}	0.0087	8.7
Pu^{241}	0.0009	0.9
Pu^{242}	0.77	770
Typical Mixture	0.00255	2.55

TABLE 5

SHIELDING and CRITICALITY

SHIELDING

Normal Transport -- PAT-1 Package Provides Sufficient Shielding (49 CFR 173)

Required -- <10 mrem/hr 3 ft from surface
Calculated -- 4 mrem/hr 3 ft from AQ-1* -- horizontally
1 mrem/hr 3 ft from AQ-1 -- vertically

Required -- <200 mrem/hr at surface
Calculated -- 30 mrem/hr at surface of AQ-1* -- horizontally
15 mrem/hr at surface of AQ-1 -- vertically

Post-Accident -- Containment Vessel (TB-1) Provides Sufficient Shielding (10 CFR 71) -- This permits AQ-1 overpack to be discounted

Required -- <1000 mrem/hr 3 ft from surface
Calculated -- 8 mrem/hr 3 ft from surface of TB-1* -- horizontal or vertical

*Using most radioactive recycled plutonium as conservative source model.

CRITICALITY

Normal Transport -- Undamaged Infinite Array
K_{eff} ~0.22

Post-Accident -- Damaged Infinite Array (full consequence of Qualification Criteria)
K_{eff} ~0.34

Single Water-Flooded and Reflected TB-1
K_{eff} ~0.62

(K_{eff} = effective neutron multiplication factor)

TABLE 6

RESULTS OF 10 CFR 71 QUALIFICATION TESTS, PAT-1 PACKAGE

Normal Conditions of Transport: Heat, cold, internal pressure, vibration water spray, drop (4 ft), penetration, compression

-- No Effect on Shielding

-- No Effect on Criticality

-- No Release: Leaktight (leakrate $<10^{-10}$ cm^3/s) Containment Vessel
No Release ($<10^{-8}$ g) of UO_2 Surrogate from Product Can

-- Double Containment (product can and containment vessel both meet requirements)

Accident Conditions of Transport -- Drop (30 ft), Puncture, Fire, Immersion

-- No Effect on Shielding

-- No Effect on Criticality

-- No Release: Leaktight (leakrate $<10^{-10}$ cm^3/s) Containment Vessel
No Release ($<10^{-8}$ g) of UO_2 Surrogate from Product Can

-- Double Containment (product can and containment vessel both meet requirements)

TABLE 7

CONTAINMENT VESSEL INTEGRITY

Internal Pressure

-- Maximum Credible Accident Environment -- 1080°F, 1253 psi (Bounding Assessment)

Tested to: 1000°F, 3330 psi, 18 hrs
Many tests @~1080°F, ~1253 psi

Analysis 18,300 psi stress
93,000 psi strength
At 1080°F and 1253 psi, Margin of Safety $\simeq 4$

-- Maximum Normal Operating Pressure -- 215°F, 34.3 psi

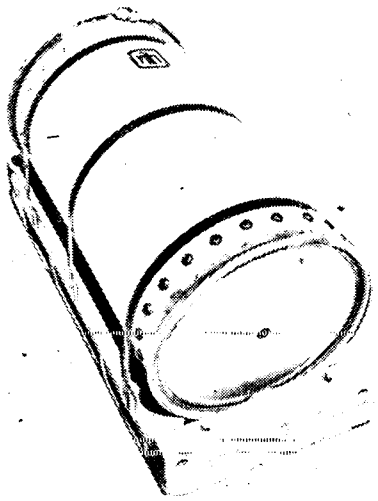
Analysis 455 psi stress
140,000 psi strength
At 215°F and 34.3 psi, Margin of Safety $\simeq 306$

External Pressure

-- Hydrostatic Requirement -- 600 psi

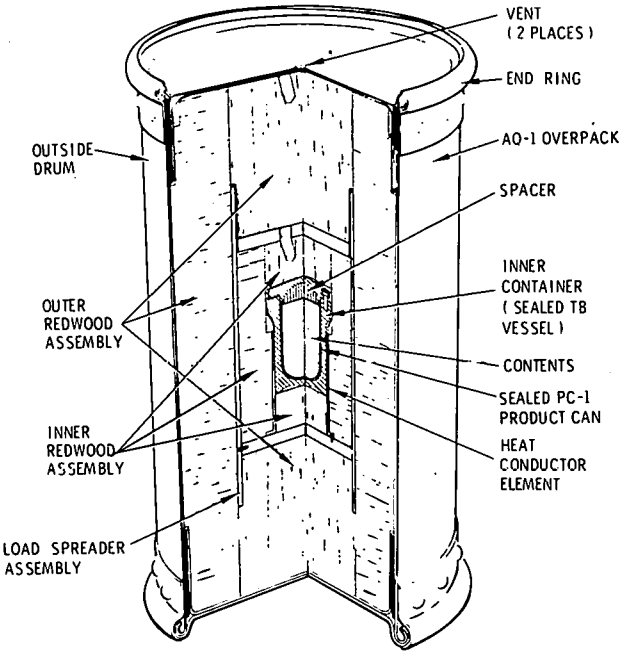
Tested to: 5,000 psi: No leak

Analysis 5,000 psi load produces -43,000 psi stress
150,000 psi strength gives Margin of Safety $\simeq 2.5$
Margin of Safety $\simeq 20$ @ 600 psi



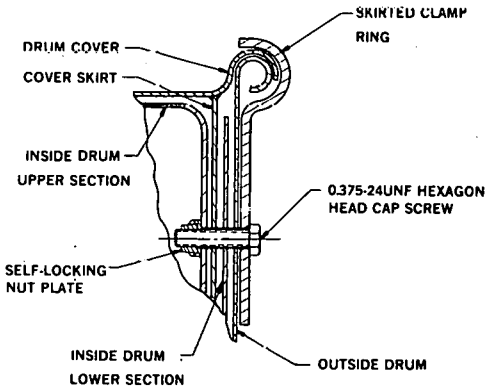
PAT-I PACKAGE

FIG. 1



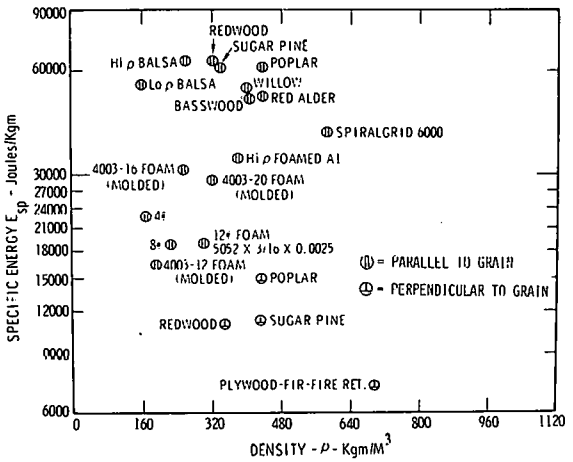
PAT-1 PLUTONIUM AIR TRANSPORTABLE PACKAGE

FIG. 2



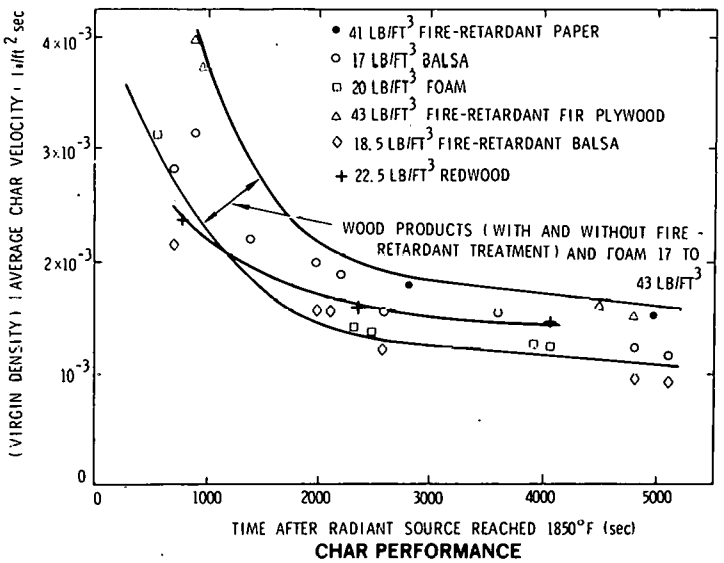
AQ-1 CLOSURE

FIG. 3



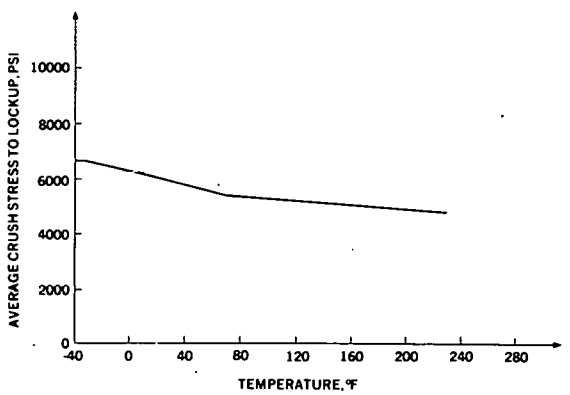
ACCIDENT RESISTANT CONTAINER MATERIALS EVALUATION

FIG. 4



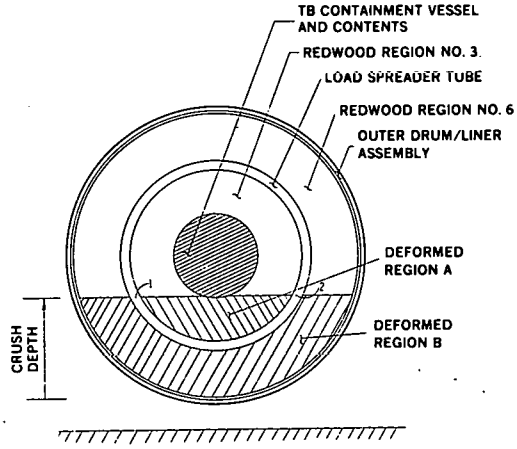
CHAR PERFORMANCE

FIG. 5



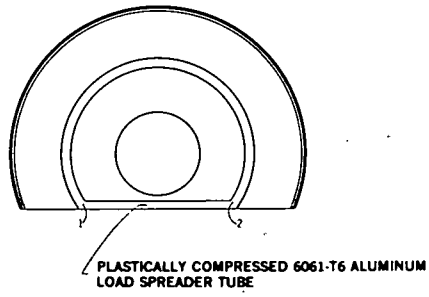
AVERAGE CRUSH STRESS FOR REDWOOD AS A FUNCTION OF TEMPERATURE

FIG. 6



IMPACT ENERGY BALANCE

FIG. 7



IMPACT ENERGY BALANCE

FIG. 8

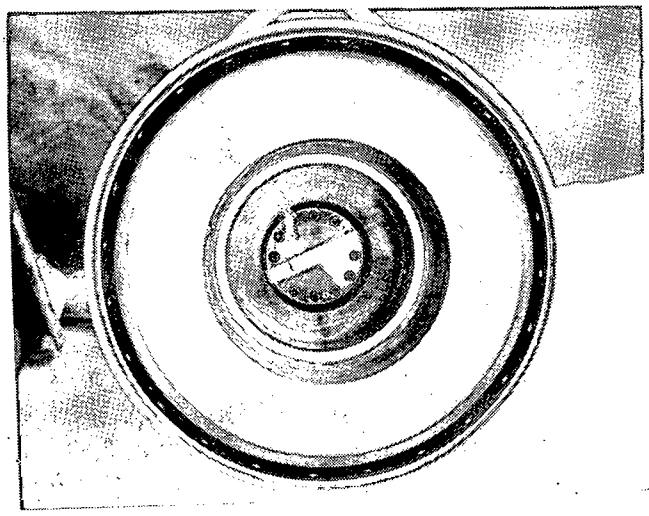


FIG. 9

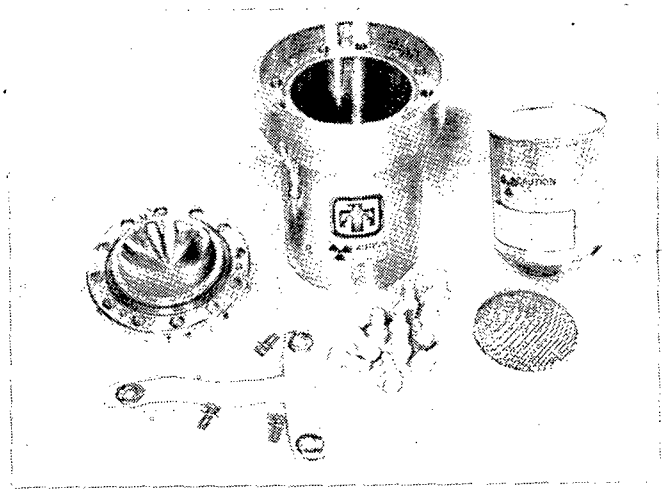
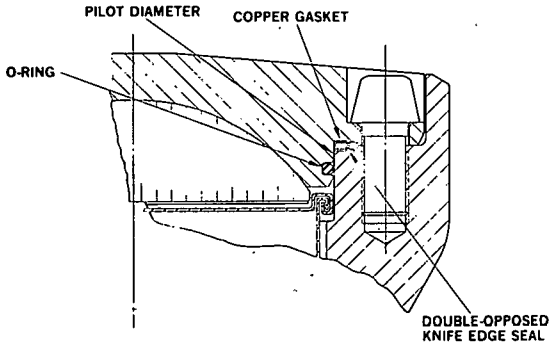
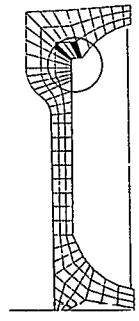


FIG. 10



CONTAINMENT VESSEL SEAL

FIG. 11



CONTAINMENT VESSEL FINITE ELEMENT STRESS ANALYSIS

FIG. 12

NUCLEAR CRITICALITY AND RADIATION SHIELDING CALCULATIONS FOR DAMAGED AND UNDAMAGED ARRAYS OF PLUTONIUM AIR TRANSPORTABLE (PAT-1) PACKAGES

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Sandia Laboratories, Albuquerque, New Mexico

INTRODUCTION

The Plutonium Air Transportable (PAT-1) package is a shock- and fire-resistant assembly designed for safe air transport of new (non-spent) powdered PuO_2 reactor fuel. The package (Figure 1) consists of a leak-sealed central fuel vessel called a TB, with 1.4 cm (0.550 in.) thick stainless steel walls. This vessel is protected from possible shock loads and fire damage by layers of redwood within a double-walled stainless steel drum. Metal heat conductors can dissipate up to 25 watts of radioactive decay power through the redwood. These steel and redwood components also provide a shield against nuclear radiation from radioactive decay of the fuel.

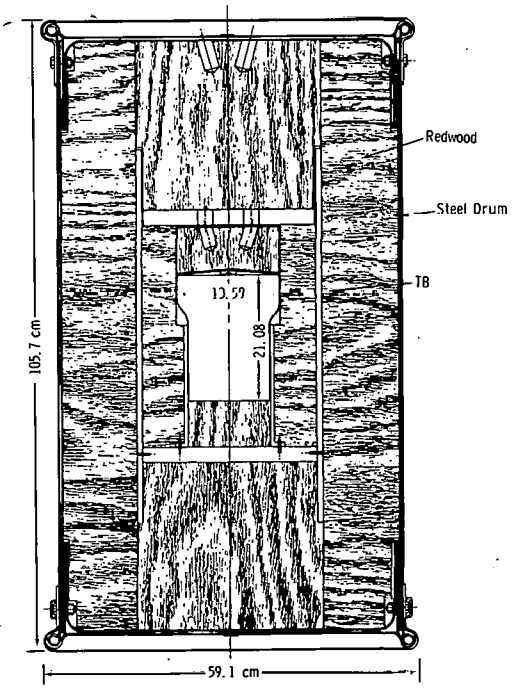


Figure 1. PAT-1 Package

This paper defines the radiation source terms, and presents the methods and results of shielding calculations as well as the nuclear criticality calculations for stacked arrays of PAT-1 packages. The accident model for nuclear criticality and radiation dose rate calculations is based on the results of field tests conducted earlier to verify the integrity of package under a variety of postulated accident conditions [1]. The impact test criteria were a minimum impact speed of 250 knots, perpendicular to an unyielding target, and in the most damaging orientation (sideways) [2].

The PAT-1 package is intended to hold a maximum of 2 kg of new Hanford PuO_2 powder. To be conservative, however, the criticality calculations are based on a previous 3.15 kg specification. Because of the radioactive decay of ^{241}Pu in the fuel, the decay power and radiation dose rate increase with storage time. The specification of a maximum decay power of 25 W limits the initial PuO_2 load in the package to an amount which depends on the age of the fuel (or storage time). An initial 1.84 kg PuO_2 loaded in the package at zero time* decays to 1.65 kg PuO_2 at a conservatively long storage time of 13-1/2 years** at which time the decay power is 25 W. To be conservative, the shielding calculations are based on the radiation source strength of the fuel at this time (13-1/2 years).

CRITICALITY CALCULATIONS

Criticality calculations were made with the KENO three-dimensional Monte Carlo computer program [3], using the Hansen and Roach 16 energy group neutron cross sections [4] with a potential scattering cross section, σ_p, of 20 barns/atom for Pu^{239}. The KENO program is particularly well suited for studying these fuel packages in stacked arrays because each TB vessel of fuel can be discretely located within an array.

Fuel Description

For criticality calculations the 25 W radioactive decay power limitation was ignored, and the amount of fuel in the TB vessel was considered to be a maximum 3.15 kg of PuO_2. To be even more conservative, the plutonium was considered to be all isotope ^{239}Pu with no other actinides present. Neutron absorption by the other plutonium isotopes, which are actually present in the fuel at any given time, more than compensates for the substitution of ^{239}Pu for ^{241}Pu atoms which are also fissionable.

Model for Criticality Calculations

Cylindrical geometrical models were used to represent both damaged and undamaged packages. A single undamaged package model is illustrated in Figure 2. For stacked package array calculations this single model was assumed to be stacked in various array sizes up to 12 x 12 x 12 for a total of 1728 packages. The arrays are assumed to be immersed in, and reflected by, water.

Dimensions for the accident-damaged package model were obtained from field test data [1]. Figure 3 shows the displacement of the TB vessel within the assembly due to a very severe and very unlikely accident. Several different models were used for damaged package array criticality calculations in order to concentrate the fuel in the array as much as possible; this required arrangements in which the damaged sides of packages were in contact with each other. The model regional material composition and atomic densities are given in Table I. For criticality calculations the package regions outside of the TB vessel were homogenized into the redwood region.

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*Zero time is the start of decay time which begins with the chemical separation of plutonium from the Hanford nuclear reactor charge.

**13-1/2 years is approximately the half-life of ^{241}Pu, for which values in the literature range from 13-1/2 to 15 years.

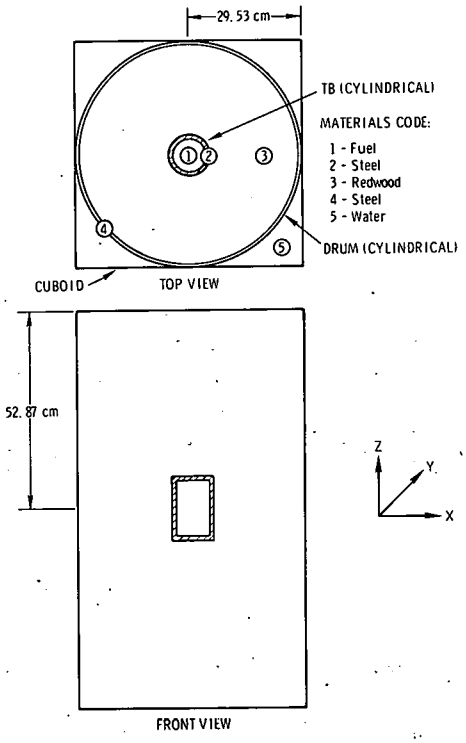


Figure 2. Undamaged Package Model

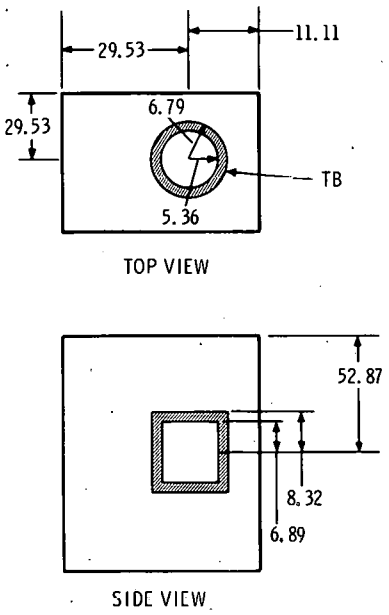


Figure 3. Damaged Package Model

TABLE I

Regional Material Compsitions
for Cylindrical Criticality Model

Region Material	Element	Atomic Density (atoms/b·cm)	Region Material	Element	Atomic Density (atoms/b·cm)
Fuel	^{239}Pu	5.63×10^{-3}	Redwood	N	2.65×10^{-5}
	O	1.13×10^{-2}	(homogenized)	O	5.46×10^{-2}
				C	8.21×10^{-3}
PH13-8Mo	Cr	1.15×10^{-2}		H	1.23×10^{-2}
(TB)	Ni	6.38×10^{-3}		Mg	2.26×10^{-5}
	Al	1.91×10^{-3}		Fe	1.44×10^{-3}
	Fe	6.36×10^{-2}		Si	5.06×10^{-5}
				Cu	3.23×10^{-4}
				Cr	4.17×10^{-4}
Water	H	6.68×10^{-2}		Al	1.74×10^{-3}
	O	3.34×10^{-2}		Ni	1.84×10^{-4}
Air	N	3.29×10^{-5}			
	O	8.86×10^{-6}			

Results from Criticality Calculations

Results show that large arrays of damaged or undamaged packages are subcritical, and the neutronic interaction between packages is small. Table II lists the calculated k-effective values.

TABLE II

Calculated Nuclear Criticality Results
(models fully reflected with 15.24 cm H_2O)

Model Description	k_{eff}* (with standard deviations) Undamaged	Damaged
Single package	.310 ± .004	.305 ± .004
Undamaged 6x6x10 array, 360 pkgs	.308 ± .004	
Damaged Package array, 12x6x4		
288 pkgs, crushed redwood		.362 ± .003
288 pkgs, air only between TB-1's (burned-out model)		.338 ± .003
Damaged Package array, 20x10x5		
1000 pkgs, redwood replaced by air Al and SS structure remain (burned-out model)		.372 ± .004
12x12x12 array, 1728 pkgs	.311 ± .007	.356 ± .004
Infinite array	.315 ± .005	.361 ± .003
Infinite array with water instead of redwood	.319 ± .003	

*k_{eff} is the effective neutron multiplication factor.

SHIELDING CALCULATIONS

Radiation dose rate calculations were made with the ANISN one-dimensional discrete ordinates multigroup computer program [5]. The Hansen and Roach 16 group neutron cross sections and an 11 energy group gamma cross section library, generated by GAMLEG [6], were used with the program. The first-order Legendre polynomial anisotropic scattering approximation was used with fourth-order direction cosines and weighting value quadratures.

The gamma and neutron source strengths and spectra were calculated, using the ORIGEN program [7], by first assuming irradiation of a charge mixture in a Hanford-like nuclear reactor containing 1.25 wt% ^{235}U and 98.75 wt% ^{238}U in a thermal reactor neutron flux of 10^{14} n/cm^2·s for one year.† At this time the plutonium isotopes are assumed to be chemically extracted from all the other elements. The oxide form of these plutonium isotopes constitutes the reactor fuel which is to be packaged. Then decay chain calculations with the ORIGEN program provide the isotopic inventory for fuel at 13-1/2 years from which radiation source data are calculated.

Radiation Source Description

The radioactive decay power of the fuel, due primarily to alpha particles, increases as a function of time (see Figure 4) because of the beta decay of ^{241}Pu to produce ^{241}Am. The major decay power isotopes in 13-1/2 year old fuel are listed in Table III. Due to the 25 W limit, even doubling the weight fraction of ^{238}Pu, a major alpha particle producer, would have only a minor effect on the total dose rate from the package because the other actinides would be correspondingly reduced. Any fission product contamination in the fuel would likewise reduce the amount of the actinide isotopic mixture which could be loaded into the container because of the decay power and radiation dose rate limitations.

†This long irradiation time is conservative and produces about five times more ^{241}Pu than in resent Hanford PuO_2 fuel to give a conservative radiation source model for dose rate calcu-ations. However, source strength is governed by the 25 W decay power limit.

Both the gamma and neutron source strength of the fuel also increase with time as shown in Figure 4. The total radiation source strength is closely associated with alpha decay of the actinides. The gamma source spectrum and dose rate response for the fuel at 13-1/2 years are given in Table IV. About half of the gamma dose rate for the undamaged package comes from the decay of ^{241}Am and ^{208}Tl (from ^{236}Pu chain), with most of the remainder from other isotopes in the decay chains which include ^{237}U, ^{212}Bi, ^{239}Np, ^{240}Pu, and ^{238}Pu.

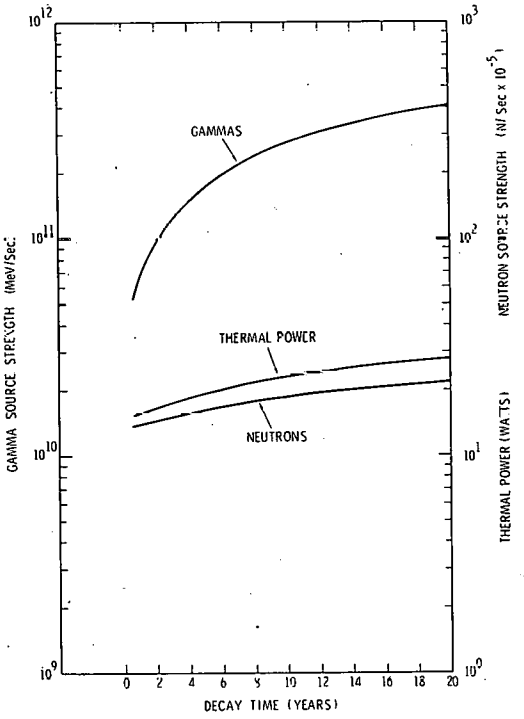


Figure 4. Package Fuel Source Strength vs. Time

TABLE III

Major Decay Power Contributors
(13-1/2 Year Old Fuel, 25 W Decay Power)

Isotope	Power (W)	Power Fraction	Fraction of Total Fuel Weight
^{238}Pu	8.10	0.324	8.17×10^{-3}
^{239}Pu	1.48	0.0591	5.04×10^{-1}
^{240}Pu	2.52	0.101	2.09×10^{-1}
^{241}Pu	1.55*	0.0622	6.37×10^{-2}
^{241}Am	11.4	0.454	5.66×10^{-2}
	25.0	1.000	8.42×10^{-1} ✓

*Due primarily to beta decay.

✓The remainder consists of minor actinide isotopes plus about 11 wt% oxygen.

TABLE IV

Gamma Source Spectrum
(13-1/2 Year Old Fuel, 25 W Decay Power)

Energy Group	Mean Energy (MeV)	Upper Energy (MeV)	Source Strength (photon/s)	Response [8] mrem/h / photon/cm^2·s
1	3.25	3.50	2.17×10^{4}	4.36×10^{-3}
2	2.75	3.00	1.56×10^{7}	4.00×10^{-3}
3	2.38	2.60	8.17×10^{4}	3.71×10^{-3}
4	1.99	2.20	4.45×10^{5}	3.24×10^{-3}
5	1.55	1.80	9.26×10^{5}	2.77×10^{-3}
6	1.10	1.35	1.60×10^{8}	2.30×10^{-3}
7	6.30×10^{-1}	9.00×10^{-1}	7.87×10^{8}	1.51×10^{-3}
8	3.00×10^{-1}	4.00×10^{-1}	5.06×10^{8}	8.30×10^{-4}
9	1.50×10^{-1}	2.00×10^{-1}	1.29×10^{10}	3.60×10^{-4}
10	5.00×10^{-2}	1.00×10^{-1}	5.69×10^{12}	3.70×10^{-4}
11	5.00×10^{-3}	1.00×10^{-2}		3.70×10^{-4}

The neutron source strength in the packaged fuel is primarily due to spontaneous fission and α,n reactions. The latter are generated by the interaction of energetic alpha particles, from decay of the actinide isotopes, with lighter nuclides in the fuel--primarily oxygen. The α,n neutron source strength was calculated by the formula [7],

$$\frac{\text{neutron}}{\text{alpha disintegration}} = 1.0 \times 10^{-10} E_{\alpha}^{3.65}$$

where E_{α} is the alpha energy in MeV.

Table V lists the actinide isotopes primarily responsible for the neutron source strength at 13-1/2 years. Table VI breaks down the calculated total neutron spectrum between the spontaneous fission neutrons, fitted to a ^{239}Pu fission spectrum, and the α,n neutrons, fitted to an experimental ^{238}Pu and boron α,n source Spectrum. It should be noted that ^{241}Am, which is not present in the sample at time zero when the Pu is assumed to be chemically clean, is responsible for both the largest source of neutrons and the largest isotopic source of decay power in 13-1/2 year old fuel.

Model for Shielding Calculations

Dose rate calculations for both the damaged and undamaged fuel packages were made with the one-dimensional computer program ANISN using a spherical geometry approximation for simplicity. The thicknesses of the various material regions in the spherical model correspond to the design thicknesses, and the diameter of the fuel region corresponds to the inside diameter of the cylindrical fuel container.

TABLE V

Neutron Source Strength by Isotope
(13-1/2 Year Old Fuel, 25 W Decay Power)

Isotope	Spont. Fiss (n/s)	α,n (n/s)	Total (n/s)	Fraction of Total	Strength Ranking
^{238}Pu	4.33×10^{4}	4.97×10^{5}	5.40×10^{5}	2.89×10^{-1}	2
^{239}Pu	2.57×10^{1}	7.91×10^{4}	7.91×10^{4}	4.23×10^{-2}	5
^{240}Pu	3.73×10^{5}	1.09×10^{5}	4.82×10^{5}	2.58×10^{-1}	3
^{241}Pu		3.64×10^{2}	3.67×10^{2}	1.96×10^{-4}	6
^{242}Pu	1.41×10^{5}	3.79×10^{2}	1.41×10^{5}	7.54×10^{-2}	4
^{241}Am		6.29×10^{5}	6.28×10^{5}	3.36×10^{-1}	1
^{242}Am		3.46×10^{1}	3.46×10^{1}	1.85×10^{-5}	7

The regional compositions for the damaged fuel package model, using the worst case of the bare TB steel container without any external shielding, are the same as given in Table I except for the fuel region which represents the actinide composition of 13-1/2 year old fuel. The regions for both damaged and undamaged models are defined in Table VII. The redwood is a separate region and not homogenized with the aluminum and stainless steel regions for the shielding calculations.

Results from Shielding Calculations

A summary of calculated dose rates for the PAT-1 package is given in Table VIII for neutron, primary gamma, and secondary or activation gamma radiation. These results show that the package as designed meets the radiation dose rate specificiations in the Code of Federal Regulations [9] for packaging and shipping radioactive materials: the external surface dose rate is less than 200 mrem/h and, at a distance of 3 ft (91.4 cm) less than 10 mrem/h to give a transportation index* less than 10; from a damaged package the dose rate is less than 1 rem/h at 3 ft [10].

TABLE VI

Neutron Spectrum
(13-1/2 Year Old Fuel, 25 W Decay Power)

Energy Group	Upper Energy (MeV)	Spont. Fiss. (n/s)	α,n (n/s)	Total (n/s)	Fraction of Total
1	10.00	1.25×10^5	1.38×10^5	2.63×10^5	0.140
2	3.012	7.41×10^4	5.21×10^5	5.95×10^5	0.317
3	2.300	1.16×10^5	3.14×10^5	4.30×10^5	0.229
4	1.496	9.30×10^4	2.05×10^5	2.98×10^5	0.159
5	0.9072	7.90×10^4	1.41×10^5	2.20×10^5	0.117
6	0.4979	6.08×10^4		6.08×10^4	3.24×10^{-2}
7	0.1111	7.97×10^3		7.97×10^3	4.25×10^{-3}
8	3.183×10^{-2}	9.47×10^2		9.47×10^2	5.05×10^{-4}
9-16	9.119×10^{-3}	0.	0.	0.	0.
Total		5.57×10^5	1.32×10^6	1.88×10^6	

TABLE VII

Spherical Geometry Configurations
for Dose Rate Calculations

	Undamaged Package			Damaged Package (Bare TB)		
Zone	Material	Thickness (cm)	Outer Radius (cm)	Material	Thickness (cm)	Outer Radius (cm)
1	Fuel	5.351	5.351	Fuel	5.351	5.351
2	304 SS	0.0330	5.384	304 SS	0.0330	5.384
3	PH13-8Mo	1.397	6.781	PH13-8Mo	1.397	6.781
4	Air Gap	2.032	8.813	Air	91.44	98.22
5	Redwood	5.105	13.92			
6	6061 Al	1.270	15.18			
7	Redwood	12.62	27.81			
8	304 SS	0.3048	28.11			
9	Air	91.44	119.6			

*The transportation index is the highest dose rate in mrem/h at 3 feet (91.4 cm) from the su e of the package [9].

TABLE VIII

Radiation Dose Rates
(13-1/2 Year Old Fuel, 25 W Decay Power)

Radiation	Outer Surface (mrem/h)	91.4 cm (3 ft) Away (mrem/hr)
A. Redwood, as designed		
Primary γ's	11.	0.5
Secondary γ's	0.2	$9. \times 10^{-3}$
subtotal	11.	0.5
Neutrons	22.	1.
Total	33.	2.
B. Water instead of Redwood		
Primary γ's	6.	0.3
Secondary γ's	0.6	$3. \times 10^{-2}$
subtotal	7.	0.3
Neutrons	2.	$9. \times 10^{-2}$
Total	9.	0.4
C. Air instead of Redwood		
Primary γ's	13.	.7
Secondary γ's	0.1	$6. \times 10^{-3}$
	13.	.7
Neutron	33.	2.
Total	46.	3.
D. Damaged Package (bare TB)		
Primary γ's	3.2×10^{2}	1.
Secondary γ's	2.	$8. \times 10^{-3}$
	3.2×10^{2}	1.
Neutrons	6.8×10^{2}	3.
	1.0×10^{3}	4.

CONCLUSIONS

The fuel load in the PAT-1 package is restricted primarily by the 25 W radioactive decay power limitation. Because of the radioactive decay of ^{241}Pu, the decay power and the radiation dose rate both increase with time, and this must be considered when the fuel is first loaded into the package. Due to the 25 W limitation the maximum amount of new PuO_2 allowed in the package for a very long and conservative storage period of 13-1/2 years is 1.84 kg. This becomes 1.65 kg of PuO_2 at 13-1/2 years due to radioactive decay.

Therefore, criticality calculations based on 3.15 kg PuO_2 are extremely conservative at the outset and additional conservative assumptions are made in the method and models. To qualify as a Fissile Class I package [10], any number of undamaged packages must be subcritical with optimum moderation, and 250 damaged packages must be subcritical. Calculations show that an infinite number of damaged or undamaged packages immersed in and reflected by water are far from critical.

Even if radiation backscatter from surroundings is conservatively estimated at 100%, the total calculated radiation dose rate at the surface of the undamaged package is less than the maximum 200 mrem/h, and the dose rate at a distance of 3 ft (91.4 cm) is less than 10 mrem/h which qualifies the package for a transportation index less than 10. The calculated dose rate at a distance of 3 ft (91.4 cm) from a damaged package is much less than the maximum limit of 1 rem/h.

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THE EFFECT OF COMPRESSION ON REACTIVITY OF PLUTONIUM BASED MATERIALS

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SUMMARY

A question has been raised as to whether or not a criticality accident could occur, due to the compression of plutonium materials, during a high speed impact accident of a Plutonium Air Transportable Package (PAT). At the request of the Nuclear Regulatory Commission, this potential accident has been investigated. The approach taken in this investigation was to provide results which were quite general and not limited to any particular shipping package design or arrangement, fuel composition or quantity of fissionable material. In order to achieve this goal of generality, the reactivity change due to compression of bare (unreflected) spheres was investigated. It will be shown that a bare system will experience the largest reactivity change during compression. The investigation utilized the one-dimensional transport theory code, DTF[1] for the bare sphere calculations. In addition, a simple formula was derived, using the Wigner Rational Approximation, which provides quantitative predictions of the reactivity change due to compression. A recommended maximum permissible initial multiplication factor for a plutonium package can be computed from this formula and a knowledge of the maximum density change during a high speed impact accident. It is argued that the formula used to predict the maximum initial multiplication factor will give conservative results. Calculations were also performed using the KENO[2] Monte Carlo computer code, in which a typical air transportable package was explicitly modeled. The results of these "explicit" calculations support the assertion that the formula used to obtain the maximum permissible initial multiplication factor yields conservative results.

DETERMINATION OF LIMITING CASE

In this analysis, limiting or "worst case" results were obtained by calculating the change in reactivity after compression of bare (unreflected) spheres of plutonium based materials. The actual PAT package (Fig. 1 shows a typical package design) will contain materials that will reflect, absorb and moderate neutrons. In many cases there will be more than one package which could result in a degree of neutron coupling between the various packages. After densification, the actual PAT packages will show a smaller reactivity change than the bare spheres for the following reason.

A bare fissionable mass will always have a higher fraction of neutron leakage than the same mass surrounded by other materials, since all materials will reflect some of the leakage neutrons back to the mass. Due to the increase in density during compression, the leakage out of the reflected and unreflected systems will

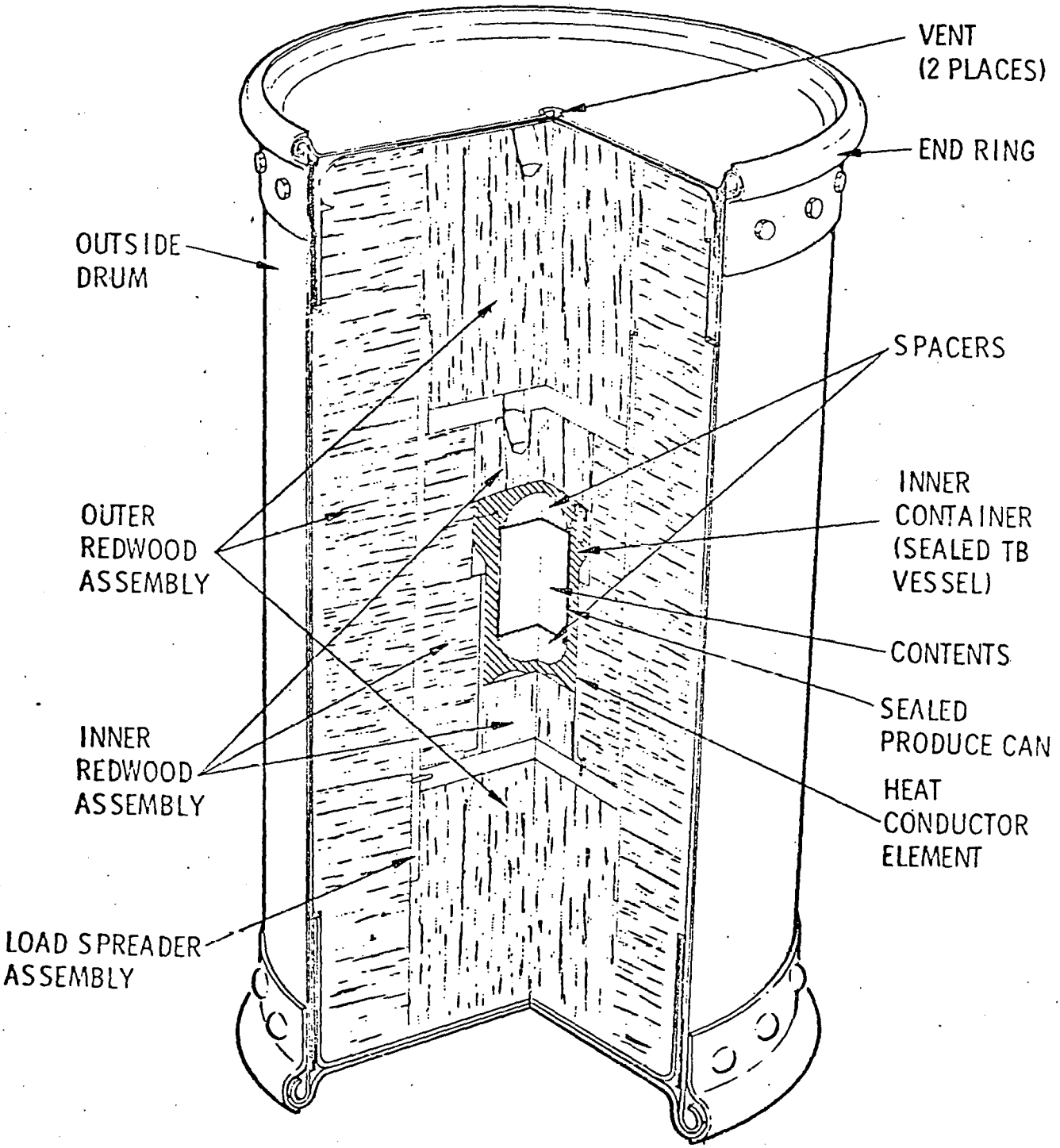


Figure 1. Schematic diagram of a Plutonium Air Transportable (PAT) package.

be reduced, resulting in an increase in the multiplication factor for both systems. Since fewer leakage neutrons will reach the reflector of the reflected system during compression, fewer neutrons will be reflected back into the fissionable mass. Hence, the component of reactivity due to reflection will be reduced during compression. The net result is that the reflected system will experience a smaller reactivity increase due to densification than will the unreflected system. Similar arguments can be made for the case of coupled systems (an array).

The claim that a bare system will represent the limiting case can be demonstrated quantitatively. A simple derivation yields:

$$\frac{\Delta k^{R}_{eff}}{\Delta k_{eff}} = 1 - R \qquad (1)$$

Here Δk^{R}_{eff} and Δk_{eff} are the changes in reactivity due to compression for the reflected and unreflected systems respectively. R is the neutron reflection probability per neutron incident on the reflector material. Since R varies between zero and one, Δk_{eff} is always greater or equal to Δk^{R}_{eff}. Thus, the bare system is demonstrated to be the limiting case. It must be pointed out that some changes in the neutron spectrum will also occur during densification, and these changes can also affect the reactivity. The spectral effects are, however, of less importance than the leakage effects. Nevertheless, these leakage effects are implicitly considered in the DTF and KENO calculations.

ONE-DIMENSIONAL CALCULATIONS

A series of one-dimensional transport theory calculations was run using the DTF computer code. These calculations used sixteen energy group Hansen-Roach cross sections. The S_4, P_1 approximation was assumed. It was also assumed that plutonium consisted entirely of the Pu^{239} isotope. The cases investigated included bare spheres of plutonium metal and plutonium dioxide powder with densities of 1.8, 3.0, and 5.0 Mg/m^3. The mass of these systems ranged from 0.65 Kg to about 6.7 Kg. The percent compression varied from 11 to 100%. Here, percent compression is defined as $\frac{(d^c-d)(100)}{d}$. d and d^c are the densities of the plutonium prior to and after compression. The various cases investigated and the calculated reactivity change are presented in Table I. The calculated reactivity change as a function of percent compression is also presented in Fig. 2. It is seen from Fig. 2 that there is almost a linear relationship between the reactivity change, for bare spheres, and the percent compression. For comparison, results are also given for water reflected spheres. As predicted, the reactivity change for the reflected spheres is less than for the bare sphere case. For the sake of comparison, some bare sphere KENO Monte Carlo calculations were made. As can be seen, KENO and DTF agree within the uncertainty limits for KENO.

REACTIVITY FORMULA FOR COMPRESSION

A simple formula, which predicts the change in reactivity due to compression, can be derived using the Wigner Rational Approximation. According to this approximation,

$$P_{esc} \approx \frac{1}{1 + \Sigma \ell} \tag{2}$$

P_{esc} is the escape probability from a body, Σ is the macroscopic absorption cross section, and ℓ is the mean chord length of the body. For a sphere of radius r, $\ell = \frac{4}{3} r$. Therefore

$$P_{esc} = \frac{3}{3 + 4\Sigma r} \tag{3}$$

Now the nonleakage probability, P_{NL}, is

$$P_{NL} = 1 - P_{esc} \tag{4}$$

For a bare sphere

$$k^{o}_{eff} = k_{\infty} P_{NL} \approx \frac{k_{\infty} 4\Sigma r}{3 + 4\Sigma r} \tag{5}$$

where k^{o}_{eff} is the multiplication factor for the uncompressed sphere. Now for the macroscopic absorption cross section,

$$\Sigma = \frac{d N \sigma}{W} , \tag{6}$$

where N is Avogadro's number, d is the density, σ is the microscopic cross section, and W is the atomic weight. During compression, the density is increased by a factor F, and the radius is reduced by $F^{1/3}$. If $\Sigma^{c} r^{c}$ is the value of Σr after compression, then

$$\Sigma^{c} r^{c} = \frac{dFN\sigma}{W} \cdot r F^{-1/3} = \Sigma r F^{2/3} . \tag{7}$$

Defining K^{c}_{eff} and P^{c}_{NL} as the multiplication factor and nonleakage probability after compression, then

$$k^{c}_{eff} = k_{\infty} P_{NL}{}^{c} = k_{\infty} \frac{4\Sigma^{c} r^{c}}{3 + 4\Sigma^{c} r^{c}} . \tag{8}$$

Using Eq. (7) in Eg. (8), the multiplication factor for the compressed condition is

TABLE I

Reactivity Change During Compression of Plutonium Spheres Predicted by DTF

Material	Initial Density (Mg/m^3)	Mass (Kg)	Reflected	Percent Compression	Reactivity Change ($\Delta\rho$)
Pu metal	19.5	6.67	No	33%	0.176
		2.21	No		0.190
		1.28	Yes		0.136
		3.5	Yes		0.130
		0.65	No		0.199
		1.28	Yes (infinite array)		0.132
		1.0	No	11%	0.069
		2.0		11%	0.067
PuO_2 Powder	11.5	2.21		33%	0.214
	1.8	3.0		18%	0.117
		3.0		68%	0.418
		6.0		18%	0.117
		6.0		68%	0.418
	3.0	3.0		17%	0.116
		3.0		100%	0.605
		6.0		17%	0.116
		6.0		100%	0.604
	5.0	2.0		20%	0.133
		2.0		75%	0.467
		4.0		20%	0.133
		4.0		75%	0.465

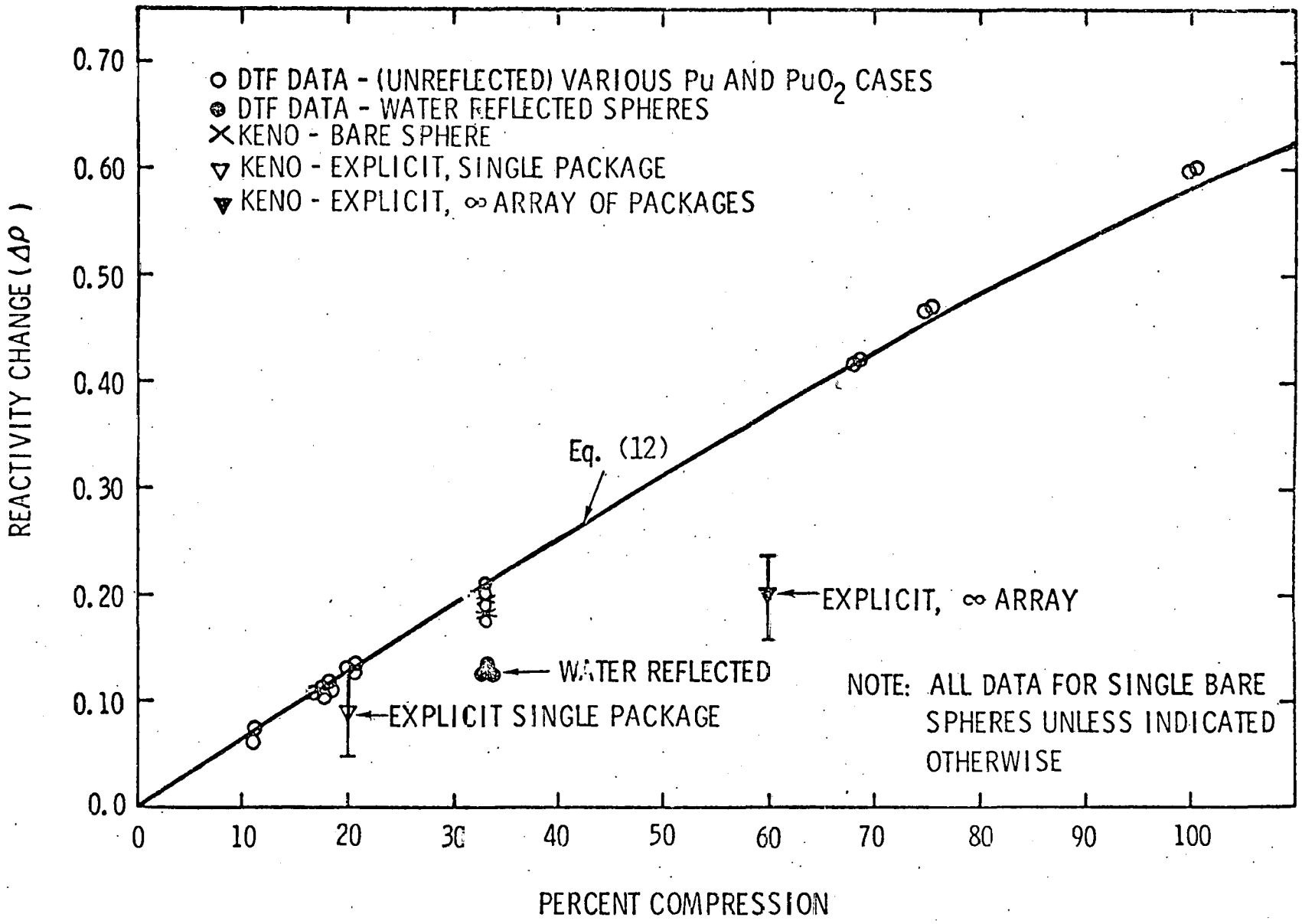


Figure 2. Fractional change in reactivity as a function of compression.

$$k_{eff}^{c} = \frac{k_{\infty}\, 4\Sigma r F^{2/3}}{3 + 4\Sigma r F^{2/3}} \quad . \tag{9}$$

Finally, taking the ratio of Eq. (9) to Eq. (5) yields

$$k_{eff}^{c} = k_{eff}^{o}\, \frac{(3+4\Sigma r)F^{2/3}}{3 + 4\Sigma r F^{2/3}} \tag{10}$$

When Σr is a small value, the terms containing Σr can be neglected, and Eq. (10) reduces to

$$k_{eff}^{c} = k_{eff}^{o}\, F^{2/3} \quad . \tag{11}$$

*

Using the expression for reactivity change

$$\Delta\rho = \frac{k_{eff}^{c} - k_{eff}^{o}}{k_{eff}^{o}} = \frac{\Delta k_{eff}}{k_{eff}^{o}} = F^{2/3} - 1 \quad . \tag{12}$$

The reactivity change due to compression predicted by Eq. (12) is plotted in Fig. 2. It is seen that the formula given by Eq. (12) is in excellent agreement with the bare sphere calculational results. Since the bare sphere calculations are conservative relative to the reactivity change in an actual PAT package, the formula given by Eq. (12) should overpredict the reactivity change due to compression.

Maximum k_{eff} for a PAT Package

The maximum permitted initial k_{eff} for a package or an array of packages can be computed from Eq. (12) and a knowledge of the largest expected compression during a high speed impact accident. R. J. Lawrence has computed the maximum compressions for solid plutonium, solid plutonium dioxide, and plutonium dioxide powders with initial densities of 1.8, 3.0 and 5.0 Mg/m^3. This analysis was conservative in that the calculations did not account for the cushioning effect of a PAT package. The maximum densification of plutonium based materials due to high speed impact was calculated to occur for PuO_2 powder with an initial density of about 5.0 Mg/m^3. An impact velocity of 422 ft/sec is the proposed NRC qualification criteria for a plutonium package. The maximum compression for an impact velocity of 422 ft/sec was calculated by Lawrence to be 17%. From Eq. (12), the maximum permitted k_{eff} for the uncompressed system is 0.90 if this impact velocity is assumed. Naturally, this k_{eff}^{c} limitation applies

*This formulation was proposed in Ref. 4.

only to densification considerations and is not intended to replace more restrictive criticality limitations.

Explicit Package Calculations

Calculations were made for a typical PAT package using the KENO Monte Carlo computer code. In these calculations, a package similar to the one shown in Fig. 1 was explicitly modeled. The container was assumed to contain 3 Kg of PuO_2 powder with a density of 5 Mg/m^3. The reactivity change for the explicit package calculations are presented in Fig. 2. The results are for a single package after 20% compression and for an infinite array of packages after 60% compression. It is seen that the reactivity change for these "explicit packages" is less than the change predicted by Eq. (12) or the bare sphere calculations. Equation (12) predicts values of k_{eff}^{c} which are too high by 4% for the single package case and by 17% for for the infinite array case.

Conclusion

The maximum permitted initial neutron multiplication factor for a PAT package is 0.90 for a maximum impact velocity of 422 ft/sec. Since this limit for the initial neutron multiplication factor is quite large and, in addition, based on conservative analysis, it can be concluded finally that these types of PAT packages do not represent a criticality hazard during a high speed impact accident.

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Dup

RESEARCH PROGRESS ON PARTICLE TRANSMISSION THROUGH VERY SMALL OPENINGS

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ABSTRACT

A research study is described whose objective is to provide experimental data permitting rational estimation of plutonium oxide leak rates through small gas leaks in shipping containers. Analysis methodology is to be developed based on experimental data and theoretical concepts for determining the possible release of PuO_2 to the environment following mechanical and thermal stress to containers in transportation accidents. Significant progress has been reported in review documents, and quarterly reports which are referenced. Apparatus and methods are described and the more important data obtained to date are reported.

INTRODUCTION

Certification of shipping containers for radioactive material requires a thorough analysis of the proposed design under auspices of the Nuclear Regulatory Commission. Criteria for integrity during the maximum accident scenario must be developed and survival of the container in its packaging must be demonstrated in crash tests and fire tests when called for by the criteria. Particularly strenuous integrity requirements were necessary for containers for air transport of plutonium oxide to insure public safety. Highly desirable would be a container with such strength and fire resistance that "no leakage" under any circumstances could be demonstrated. Since this ideal may be approached,but not realized, acceptably small leak rates are permitted. Acceptable leak rate are universally specified in terms of a gas leak rate, usually helium, because of the sensitivity of the helium detection instrument. These leak rates have little meaning, however, when the material being shipped is not a gas, but very fine particulate material, such as plutonium oxide. An acceptable particle leak rate for such a powder might be such that the same passageway would permit a fairly large gas leak. Relatively few studies had been conducted to relate powder leaks with pressure and orifice dimensions, hence the study to be discussed was sponsored to investigate relationships of gas flow and fine particle transmission through characterized leak paths.

OBJECTIVE

The purpose of this paper is to describe a research program sponsored by the U.S. Nuclear Regulatory Commission, whose objective is to relate particle transmission rates with leak rates

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through very small openings. To this end tasks, milestones, and progress to date are presented.

BRIEF SUMMARY OF WORK OBJECTIVES

The research has the objective of correlating plutonium oxide powder leak rates with gas escape rates which are used to measure shipping container integrity following stress testing. Several interrelated tasks must be undertaken to provide the required data to arrive at the objective. Ultimately, experiments are to be undertaken designed to measure typical PuO_2 powder leak rates under a range of leak conditions. Prior to carrying out these direct leak experiments tasks will be addressed which are intended to provide analysis methodology for estimating the attenuation as a function of leak geometry and the pressure-temperature cycle imposed during an accident. These tasks include literature review of related past work, methods for calculatin gas leak rates for an accident scenario based on measured gas leak rates at the test conditions, and experiments with uranium oxide or other particles to provide support for the attenuation calculational methods.

WORK PLAN

Major tasks are the following.

Task A

Review literature on experimental and theoretical work relating to transmission of particles through very small channels.

Milestone 1 -- Draft of document for technical review.

Milestone 2 -- Submission of Final Report.

Task B

Investigate the relationships of gas flow rates, leak geometries, pressures and temperatures. (This task will be a combination of technical literature review and experiments using well-characterized leak geometries. Pressures appropriate to the range considered will be used. Tracer gas may be used, but in many cases may not be required.)

Milestone 1 -- Review literature on topic. Draft summary report.

Milestone 2 -- Final report on technical literature.

Milestone 3 -- Select methods and design apparatus for flow experiments.

Milestone 4 -- Fabricate and assemble apparatus.

Milestone 5 -- Shakedown tests for apparatus.

Milestone 6 -- First flow test conducted.

Milestone 7 -- Complete series of tests.

Milestone 8 -- Draft report of tests.

Milestone 9 -- Final report of tests.

Task C

Measure transmission of a well-characterized simulant (UO_2 powder) through several leaks characterized in Task B under two separate conditions: (1) leak location above the powder level, (2) leak location is below powder level. (This investigation will establish limiting conditions for "blow-out" of particles for leaks of diameter somewhat smaller or larger than the largest particles present. The results will also support or refute attenuation "models" for relatively simple leak geometries. It will also help establish limiting conditions for the PuO_2 experiments which will be more time-consuming to do extensive parametric studies.)

Milestone 1 -- Prepare experimental apparatus (will probably use same setup for Task B).

Milestone 2 -- Perform first experiment with one leak geometry (a circular aperture of 50 μm and 100 μm).

Milestone 3 -- Complete series of experiments with 2 or more leak paths.

Milestone 4 -- Submit draft report.

Milestone 5 -- Submit final report.

Task D

Measure fuel grade PuO_2 leak rates through a "standard leak" incorporated into a suitable container. (This task is to determine the transmission of PuO_2 particles through an aperture of simple geometry. The aperture would be characterized dimensionally and also by gas leak rate under one or more conditions. Selective temperature and pressure regimes would be employed.)

Sub-task 1 -- Design of Experimental System

a. Define criteria for accomplishing proposed measurements.

b. Define specifications for equipment to achieve above criteria.

c. Prepare conceptual drawing for experimental system.

d. Evaluate system for operability in gloved box.

Sub-task 2 -- Assembly of Experimental System

a. Procure components.

b. Fabricate components including gloved box.

c. Check and calibrate components.

d. Assembly and check out operability of system.

Sub-task 3 -- Execution of Simulant Experiments

a. Select and analyze simulant material for particle-size distribution.

b. Carry out series of checkout runs to establish characteristics of operating system.

c. Carry out extended series of runs with simulant covering all extremes in parameters.

Sub-task 4 -- Transfer of Experimental System to Gloved Box

a. Disassemble system and reassemble in gloved box.

b. Check out operation of all compenents.

c. Carry out simulant runs under various conditions for comparison with bench-top results.

Sub-task 5 -- Execution of Phase 1 Hot Experiments

a. Characterize PuO_2 for particle size distribution.

b. Check operation of experimental system with PuO_2.

c. Carry out leak rate experiments with PuO_2 under full range of parametric variations.

Milestone 1 -- Complete experimental system design.

Milestone 2 -- Complete fabrication and check of experimental system.

Milestone 3 -- Complete simulant run and establishment of the experimental system.

Milestone 4 -- Confirm experimental system operability in gloved box.

Milestone 5 -- Complete leak rate data collection for "standard" leak.

Milestone 6 -- Draft report of Task D.

Milestone 7 -- Issue report for Task D.

Task E

Measure fuel grade PuO_2 leak rates from a simulated leaking container. (In this task a leak would be fabricated in such a way to represent a shipping container failure. A failed gasket, a defective weld, or a crack in the wall, etc. The failure would be characterized as to gas leak rate and dimensionally insofar as practical.)

Sub-task 1 -- Development of Simulated Leaking Container

a. Develop process for fabricating defective weld type of leak.

b. Develop process for fabricating cracked wall type of leak.

c. Fabricate several leaky containers of each of above types for experimental run.

d. Characterize the containers for the leak rate as a function of temperature and pressure.

e. Characterize sample of each type metallographically.

Sub-task 2 -- Execution of Simulant Experiments with Simulated Leaking Container

a. Carry out series of checkout runs to establish characteristics of operating system with simulated cans.

b. Carry out extended series of runs with simulant powder and simulant cans covering all extremes in parameters.

Sub-task 3 -- Execution of Phase 2 Hot Experiments

a. Check operation of experimental system with PuO_2.

b. Carry out leak rate experiments with PuO_2 under full range of parametric variations.

Milestone 1 -- Complete fabrication of simulated defected container.

Milestone 2 -- Complete simulant runs and establish characteristics of the experimental system with simulated failed container.

Milestone 3 -- Complete leak rate data collection for "simulated" failed container.

Milestone 4 -- Draft report of Task E.

Milestone 5 -- Issue report of Task E.

Task F

Develop the analysis methodology to correlate PuO_2 leak rates with gas leak at the test conditions. (This task will consolidate the theoretical and experimental aspects of the study and provide insofar as practical the calculational methods to permit rational estimates of PuO_2 escape during an accident scenario. It is anticipated that limiting cases will be identified in which particle leaks will be excessive [blowout case] or insignificant [leak dimension smaller than particle dimension].)

Milestone 1 -- Consolidation of experimental and theoretical data.

Milestone 2 -- Definition of limiting cases.

Milestone 3 -- Submit draft report.

Milestone 4 -- Issue final report.

STATUS OF RESEARCH

- Preliminary Assessment Using Available Data. (Part of Task A)

An initial literature review was completed to provide a preliminary assessment of particle transmission. This work was reported (1). As part of this assessment, a series of experiments was undertaken to demonstrate the rapid clearance by gravitational settling of dense particles (UO_2) from the air space in a container of dimensions similar to the PuO_2 shipping container under study. This study, in which small air samples were extracted periodically from the air spacing following agitation of the powder, showed very rapid clearance of the powder from the air space. Results are shown in Figure 1.

The conclusions from this preliminary assessment was that gas leaks large by leak-tight standards would permit transmission of very small quantities of a dense, though fine, particulate solid.

- Review Literature and Write a Report on Relationship of the Flow Through Small Apertures. (Task B).

Reference (2) written during this task discusses the theory of gas flow through very small openings, both thin plate orifices and capillaries. The theory is well developed; however, limited confirming experimental work is available for very small openings. The relationships

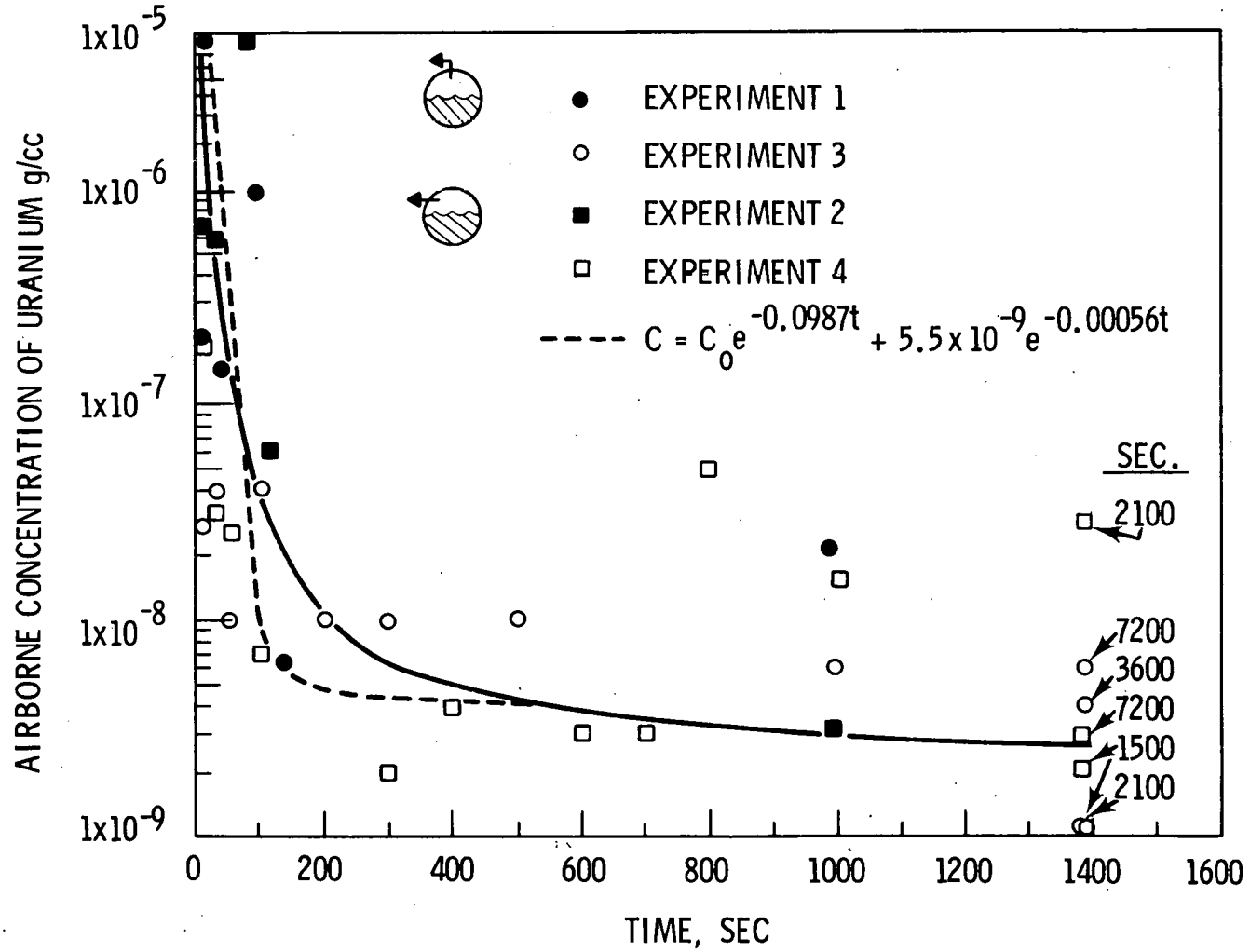


FIGURE 1 -- CONCENTRATION OF UO_2 IN THE AIR SPACE ABOVE THE POWDER LEVEL FOLLOWING TUMBLING OF THE POWDER.

of flow rate to gas density, temperature, pressure, and aperture geometry are reviewed in this report. Sonic flow implications are presented.

- Design and Fabricate an Apparatus and Use to Determine Gas Flow Relationships for Orifices and Capillaries. (Task B)

In Figure 2 is shown a diagram of the apparatus which has been used to determine orifice coefficients and demonstrate the general validity of standard orifice equations for thin plate orifices ranging from 200 μm to 20 μm in diameter. A report is to be issued in the first quarter of 1978 of this work. Work is underway to measure flow as a function of capillary length and diameter.

- Simulant Particle (UO_2) Transmission through Characterized Leaks. (Task C)

Apparatus and orifices calibrated in Task B have been used to make initial measurements of UO_2 powder transmission. Orifices of 200 μm permit unrestricted flow of powder with an AMMD of 2 μm. Blockage occurs at about 50 μm orifice diameter, although some particles are initially carried through orifices of this size. Work is underway with all orifices calibrated in Task B.

An agitated powder pressure vessel has been designed and tested for measuring UO_2 powder leak rates through orifices and capillaries under air fluidizing conditions. This apparatus will assure a particle concentration in the air space which can be controlled from a condition of quiescent settling to continuous bed fluidizing, giving rise to a maximum air concentration. This apparatus is shown schematically in Figure 3.

- Leak Rates of PuO_2 Powder (Tasks D and E)

This work performed by Battelle-Columbus Laboratories has progressed through the sub-tasks of design, assembly, and checkout of the experimental apparatus. Several experiments have been performed using nuclear grade PuO_2 which confirm the low leak rates to be expected. Results to date are reported in this Symposium Proceedings (3).

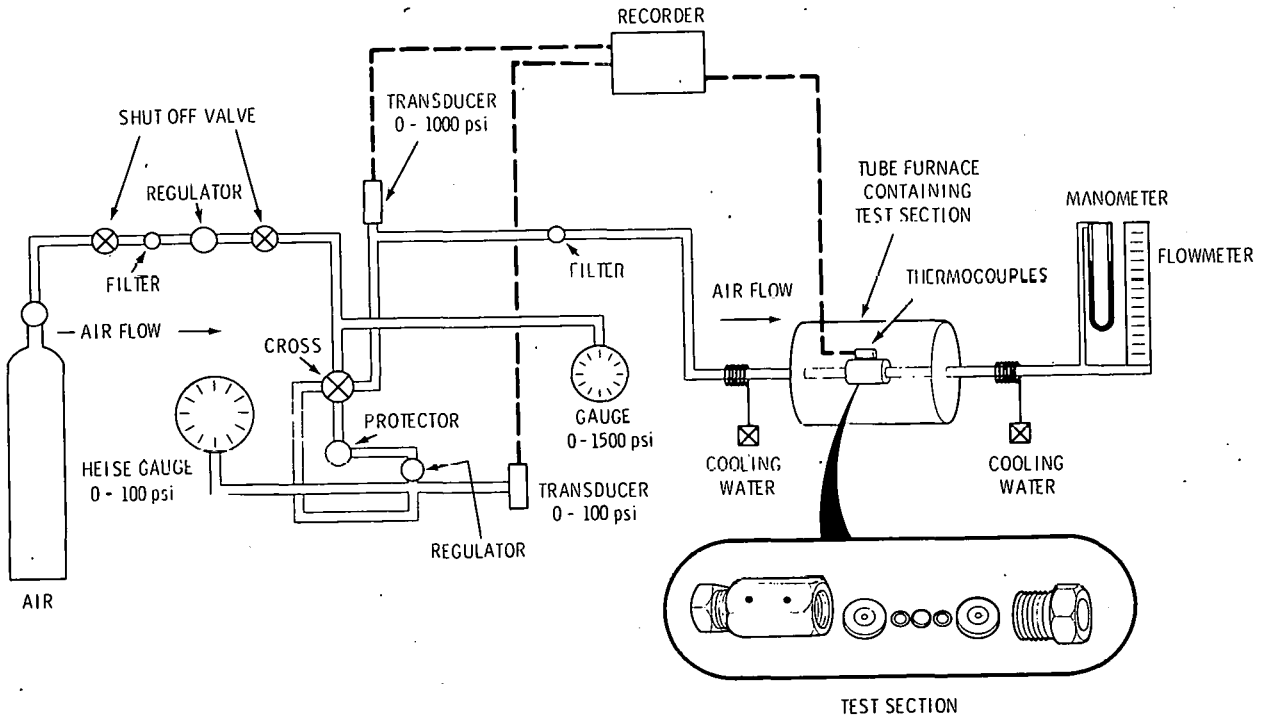


FIGURE 2 -- Schematic Diagram of Apparatus for Flow Calibration of Micro-Orifices and Capillaries

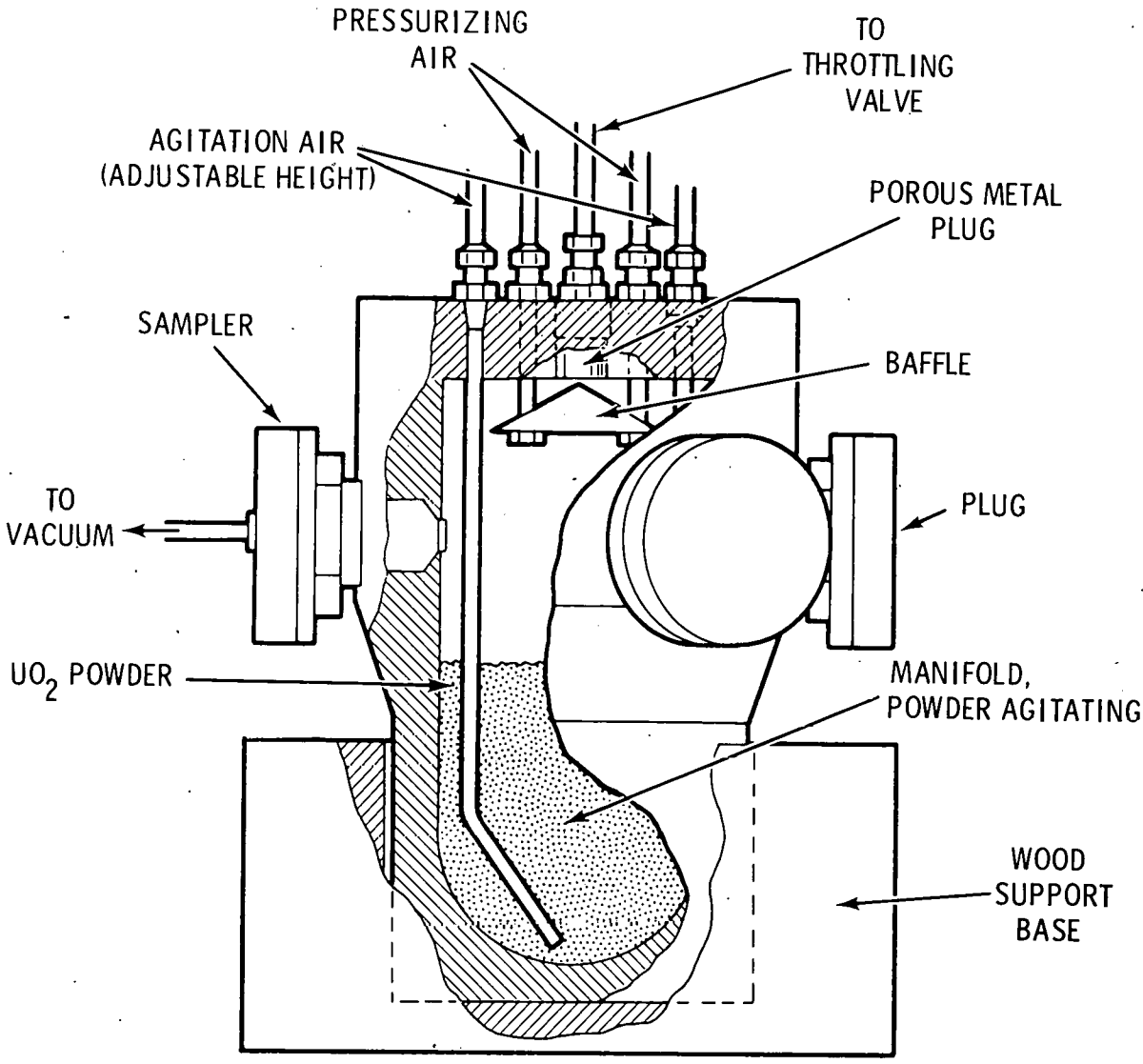


FIGURE 3 -- Agitated Powder Leak Test Vessel

CONCLUSIONS

Research is underway whose objective is to provide information and methods for estimating powder transfer through very small openings. The work is not yet complete, but data obtained to date show, that sizable gas leaks can occur with virtually negligible quantities of finely divided PuO_2. Tasks set forth in the Work Plan when completed, should provide the needed information to permit an objective assessment of powder leaks in PuO_2 in shipping containers

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STUDY OF PLUTONIA LEAK RATES FROM SIMULATED CONTAINER LEAKS

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INTRODUCTION

The leak tightness of containers for shipment of radioactive materials is generally evaluated by measuring the leakage rate of a tracer fluid, usually a gas, from the container.[1,2] The maximum permissible leakage rate is a function of the specific radioactive species to be transported in the container.[3] Application of these criteria to shipping containers for transport of plutonia powder result in highly restrictive specifications with respect to leak tightness. This program is being conducted to establish an experimental correlation between plutonia particulate and helium gas leaks from simulated container leaks, and thus provide a data base for a more realistic plutonia shipping container leakage rate specification. The effects on the leakage rate of various parameters to which a container may be subjected are also being investigated. Leaks through standardized small orifices in thin discs have been studied to date. The parameters investigated include internal gas pressure, orifice size, orifice orientation with respect to the plutonia powder and agitation.

Working with plutonium creates practical handling difficulties. Therefore, the approach taken was to divide the work effort in two phases -- a simulant experiment phase and a "hot" phase using plutonia powder. The initial phase, in which thoria was used as the plutonia simulant, served to check out measurement techniques and experimental system operability. The "hot" experiments then generated data germane to establishing the leak rate correlations and determining the parametric effects.

EXPERIMENTAL SYSTEM

Experimental Setup

The essential components of the experimental system are the orifice, a tube to support the orifice disc and contain the powder, a helium supply, and sampling devices (the Battelle cascade impactor for size discrimination collection and analysis, or a membrane filter for total emission collection). Figure 1 is a schematic representation of the setup. The orifices are in 4-mil-thick platinum discs, 9.5 mm in diameter. The leak tube, depicted in Figure 2, is constructed of Inconel-600 and has a reservoir volume of about 4 cc. For the experiments, the leak tube was loaded with 4 grams of PuO_2 powder, making the tube about 25 percent full. During operation, the

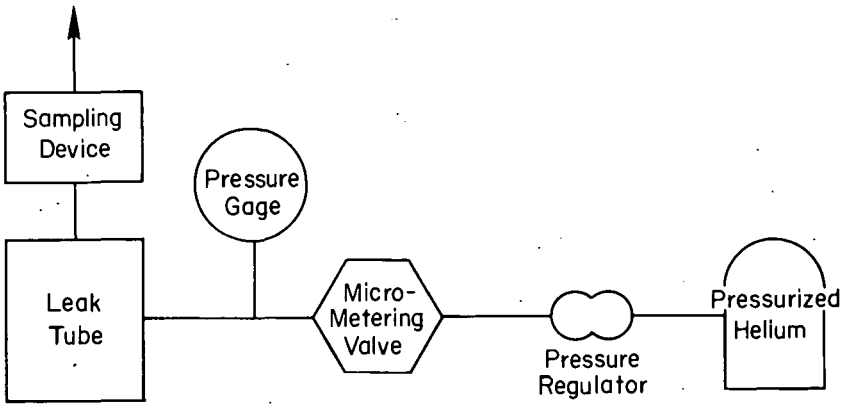


FIGURE 1. EXPERIMENTAL SETUP

orifice is mounted and sealed with stainless steel gaskets in the end of the leak tube. Compressed helium is injected into the tube and regulated to set the internal tube pressure. Agitation, when applied, is accomplished with a mechanical vibrator (Scripto Vibratool, 120 cps, 1/32-in. displacement uncoupled).

The leak tube assembly and sampling device are mounted in a stainless steel, filtered containment box. The box, mounted on two vertical supports, may be rotated about a horizontal axis with the leak tube assembly in place, thus allowing the orifice to assume one of three orientations: (1) up, with powder resting in the opposite end of the vertical tube, (2) down, with powder covering the orifice, and (3) sideways, with the tube horizontal and powder resting along one side. This box also serves to isolate the experiment, thus establishing a contaminant-free working environment within the glove box.

Experimental Procedures

In order to establish the amount of plutonia leaked under any given condition, collections of the emitted particles are made over a 10-minute period and analyzed for mass. One of two types of collection is made: (1) a Battelle cascade impactor collection, or (2) a total membrane filter collection. The Battelle cascade impactor employs inertial deposition to collect emitted particles according to their aerodynamic

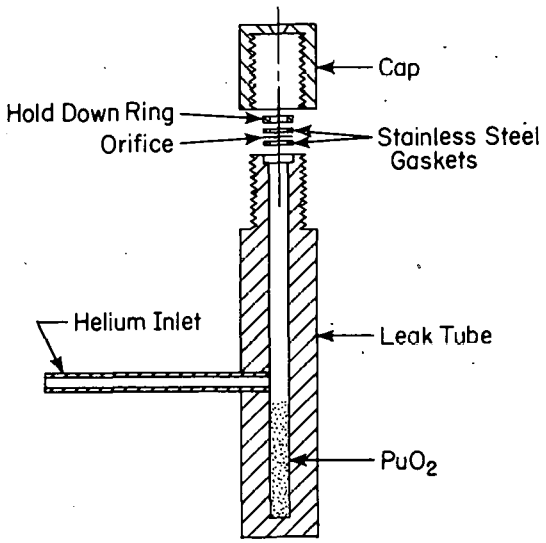


FIGURE 2. SCHEMATIC DRAWING OF INCONEL 600 LEAK TUBE

size. By this technique, particles are divided into several size classifications. A size distribution is established by comparison of the amounts collected in each size range. The more easily implemented membrane filter technique employs a high efficiency filter to collect all particulates emitted, regardless of size.

Such minute amounts of material are emitted that gravimetric analysis is impractical and chemical analysis is costly and time consuming. Therefore, radioassay analysis is used to determine the mass of the collected alpha-emitting powder. The correlation with helium leak rate is established by comparison of the measured particulate leak rate with the corresponding helium leak rate, which is measured by pressure drop or volumetric methods using an empty leak tube. A minimum of two runs was conducted for each experimental condition to check reproducibility of results.

BACKGROUND EXPERIMENTS

Helium Leak Calibration

Specifications for container leaks may be measured by the amount of helium leaked with time. Of the parameters being investigated in this study only pressure and orifice size are likely to have an effect on the helium leak rate. Therefore, in order to provide a basis for ultimate particulate/helium leak rate correlation, the helium leak rates through the 5-μm, 10-μm, and 20-μm-diameter orifices to be used in the experiments were measured at 440 and 920 psig using a bubble flow tube. The magnitude of the flow rates was also calculated assuming sonic flow and an infinitely thin orifice. Due to this second assumption, calculations are expected to be accurate only within a factor of 2.[4] The results of these measurements and the corresponding calculations are shown in Table 1. For several cases the leak rates were further validated by predictive calculations based on the system volume and the rate of pressure decay of a closed (except for the orifice) system assuming an ideal gas. This pressure decay method was used in subsequent experiments to check the flow rate and monitor for plugging of the orifice by the test particulates. This latter method is the proposed ANSI standard for leak testing packages for shipment of radioactive materials, N-14.5 (12th draft).[1] For the case of ambient internal pressure, the approximate leak rate was calculated assuming molecular diffusion of helium into air.[5] These results are also given in Table 1.

Simulant Experiments

Preliminary cold-bench tests were conducted using thoria as a simulant for plutonia. These tests were made to: (1) check out and refine operation

TABLE 1. HELIUM LEAK RATES

Orifice Size, μm	Pressure, psig	Measured Leak Rate, standard cc/sec	Calculated Leak Rate, standard cc/sec	Ratio: Measured Rate/ Calculated Rate
5	920	0.77	1.26	0.61
5	440	0.39	0.63	0.62
10	920	2.89	3.52	0.82
10	440	1.39	1.76	0.79
20	920	8.09	11.42	0.71
20	440	4.21	5.71	0.74
5	14.7	-	1.25×10^{-5}	-
10	14.7	-	5.03×10^{-5}	-
20	14.7	-	2.02×10^{-4}	-

and measurement techniques, (2) check out system design and operability for ultimate transfer to the glove box necessary for the plutonia handling, and (3) establish a preliminary data base.

The thoria powder was sized using an impactor and compared with the size distribution of the plutonia powder to be used in the "hot" phase experiments as reported by ARHCO[6] and verified by Battelle impactor analysis. Although the size distribution was not precisely similar, the thoria powder appeared to be a representative plutonia simulant.

Experiments were conducted to investigate the amount of particulate leaked as affected by the different parameters. Although the system functioned according to design, due to the small amounts of thoria emitted and the relatively low specific activity of Thorium-232, quantitative assessment was limited. However, based on these simulant experiments, it was estimated that the leak rate correlation for a worse case situation was approximately 10^8 atoms helium/1 molecule thoria.

Specific Activity of Plutonia

As stated earlier, all collected mass determinations were made using radioassay counting. Thus, the actual measurements were made of the activity present and not mass. In addition to a correction accounting for counter efficiency, a calibration had to be made relating specific activity to mass for the plutonia powder. In order to obtain an actual specific activity of the powder being used in this study, three pairs of standards were prepared by dissolving the powder in nitric acid and depositing different quantities of the solution on small watch glasses. The solvent was evaporated by heating the watch glasses on a hot plate. The activity of these standards was measured and the specific activity was calculated. The average specific activity of the standards was found to be 0.096 ± 0.004 Ci/g of plutonium as compared to the literature value, 0.06 Ci/g, for Plutonium-239.

EXPERIMENTAL RESULTS AND DISCUSSION

The effects of the following parameters on the emission of plutonia powder through small diameter orifices was investigated.

- Orifice Size: 5-, 10-, and 20-μm diameter (nominal). Actual diameters were within 10 percent of the nominal sizes.
- Helium Pressure: Ambient, 440 and 920 psig
- Tube Position: Down, up, and sideways
- Vibration: With and without vibration of the leak tube.

Effect of Experimental Parameters on the Quantity of Plutonia Emission

Figure 3 shows the effect of the experimental parameters, except vibration, on the average total emission of plutonia powder through orifices. Each of the parametric effects is discussed in the following paragraphs.

- Orifice Size. The data in Figure 3 show that, in general, the total plutonia emission increased, as one would expect, with the size of the orifice.
- Pressure. The effect of helium pressure on the plutonia emission is also shown in Figure 3. With a 5-μm orifice, the average emission did not appear to change with pressure. In the case of the 10-μm orifice, the

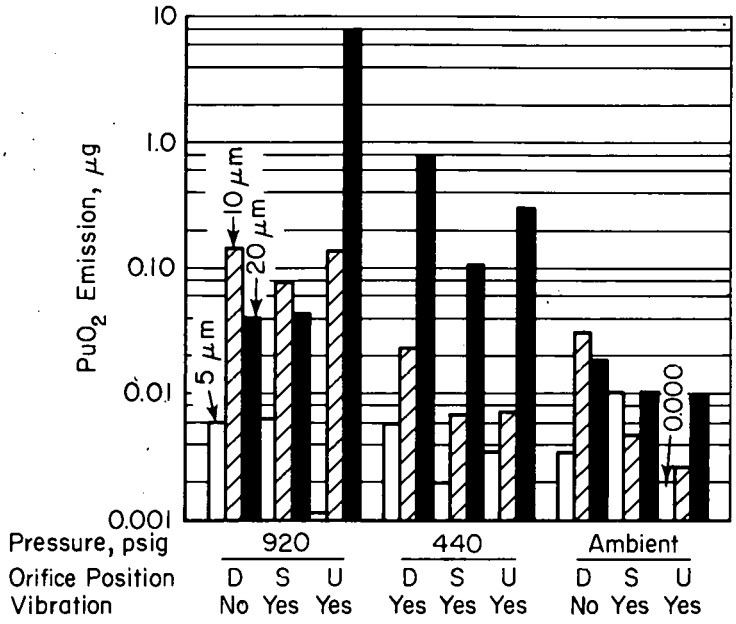


FIGURE 3. EFFECT OF EXPERIMENTAL CONDITIONS ON THE PuO_2 EMISSION THROUGH ORIFICES AT ROOM TEMPERATURE

emission increased with an increase in the helium pressure. For the 20-μm orifice, the average emission was greatest in all but one case at a helium pressure of 440 psig and was lowest at ambient pressure.

- Tube Position. The position of the leak tube during the experiments appeared to have a definite effect on the emission of plutonia. These data indicate that, in general, the average emission was greatest when the leak tube was in the down position, i.e., with the powder piled up on top of the orifice. The smallest emission generally resulted when the leak tube was in the sideways position.

- Vibration. The effect of vibrating the leak tube on the particulate emission was investigated only in the case of the 20-μm orifice. The data in Figure 4 indicate that vibration generally increased the emission of plutonia particles.

The background level of plutonia aerosol inside the containment box was measured periodically during the course of the experiments. The background level collection over a 10-minute period was found to vary from 0.00006 to 0.0050 μg with an average of about 0.0027 μg. This background quantity of plutonia has been subtracted from the total emission data presented above.

The main objective of this program is to correlate the plutonia particulate leak rate with the helium gas leak rate. Based on the highest emission of plutonia powder emitted through the 5-, 10-, and 20-μm orifices, the following He/PuO_2 leak rate correlations on an atom/molecule basis are given in Table 2. These data show that the helium leak rate is greater than the plutonia leak rate by factors varying from 10^3 to 10^8, depending on the experimental conditions.

Effect of Experimental Parameters on the Particle Size of the Plutonia Emission

The effects of orifice size, internal helium pressure, leak tube position, and vibration on the particle size distribution of the plutonia emissions were also

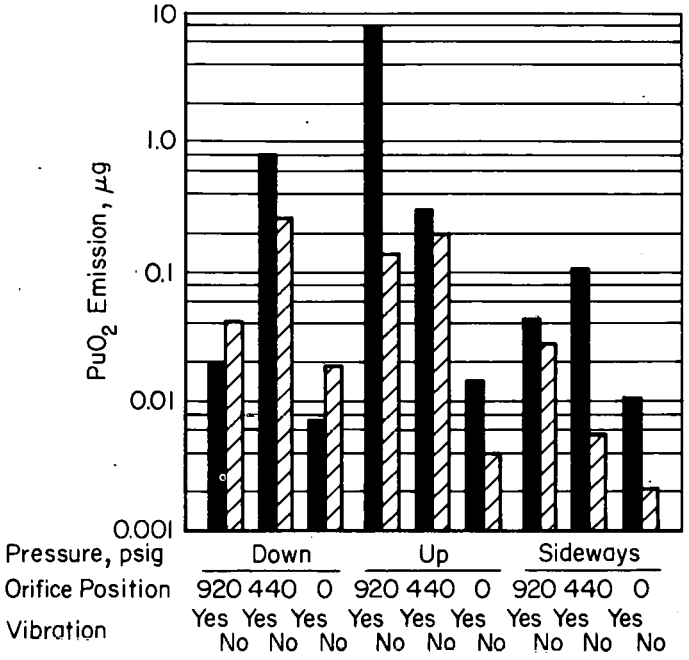


FIGURE 4. EFFECT OF VIBRATION ON THE EMISSION OF PuO_2 POWDER THROUGH A 20-μm ORIFICE AT ROOM TEMPERATURE

evaluated.

In about half the experiments carried out with 10- and 20-μm orifices, the average particle size distribution of the leaked powder paralleled that of the starting powder. In the case of the 5-μm orifice, however, the emission particle size distribution did not parallel that of the starting powder and generally was skewed toward sizes smaller than 1 μm. In most of the other runs with 10- and 20-μm orifices, the particle size distribution was skewed toward sizes greater than 2 μm.

The parametric effects on the emitted particle size distribution are discussed below.

- Orifice Size. The effect of orifice size on the average mean particle size of the plutonia emission is shown in Figure 5. These data indicate that the mean size of the emission decreased with a decrease in orifice

TABLE 2. RELATIONSHIP BETWEEN HELIUM AND PuO_2 LEAKAGE RATES

Orifice Size, μm	He Pressure, psig	He/PuO_2 Leak Rate Correlation
5	Ambient	6.7×10^3
5	440	3.9×10^8
5	920	4.8×10^8
10	Ambient	2.5×10^4
10	440	2.8×10^8
10	920	5.0×10^7
20	Ambient	5.8×10^4
20	440	1.9×10^7
20	920	3.3×10^6

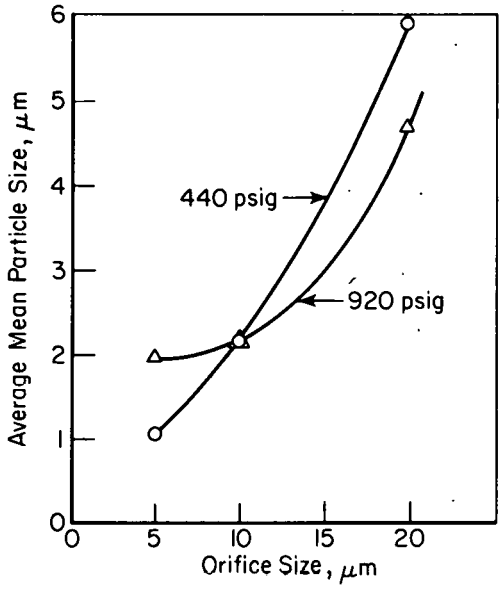


FIGURE 5. EFFECT OF ORIFICE SIZE AND PRESSURE ON THE AVERAGE MEAN PARTICLE SIZE OF EMITTED PuO_2 AT ROOM TEMPERATURE

size at helium pressures of 440 and 920 psig. This may be accounted for by increasing shear stress with decreasing orifice size resulting in breakup of agglomerates. Also, one would expect an increased effect of simple sieving collection with the smaller orifices (i.e., a solid 6-μm particle cannot exit a 5-μm hole).

- Pressure. Increased pressure means increased shear stress on the particles in the flow stream as they are emitted. Thus, agglomerates will break up more readily at a higher pressure. The experimental effect of pressure on the particle size distribution of the plutonia emission appears dependent upon the orifice size, as shown in Figure 5. These data indicate that at 440 psig, the mean particle size was smaller, the same, and larger than that at 920 psig for 5-, 10-, and 20-μm orifices, respectively.

- Tube Position. With the orifice or leak tube in the down position, the powder tends to pack over the orifice, creating agglomerates of larger particles. Experimental results indicate that the mean size of the emitted particles was generally larger in the down position for the 5- and 20-μm orifices and larger in the up position for the 10-μm orifice.

- Vibration. The effect of vibration on the emitted particle size was studied only for the 20-μm orifice. In this case, vibrating the leak tube appeared in the majority of the runs to increase the size of the emitted particles, as shown in Figure 6. It is believed that the vibration may be serving as a mechanism enhancing agglomeration.

DISCUSSION

It should be noted that the duplication of the experimental results was not good. However, this was not unexpected because radioassay counting provides the sensitivity to detect PuO_2 in quantities as small as about 30 nanograms.

The data generated to date are also conservative in regard to the leakage of PuO_2 inasmuch as the types of leaks studied so far are those that provide a straight-through path for the PuO_2 powder, rather than a zig-zag path that a real crack would provide.

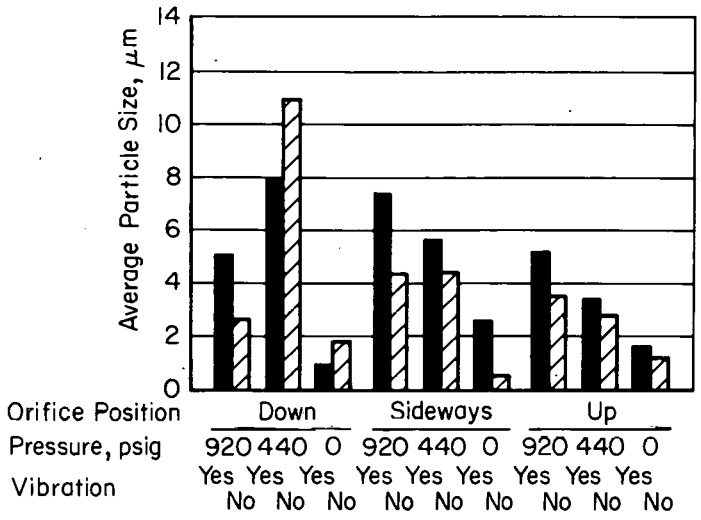


FIGURE 6. EFFECT OF TEST PARAMETERS ON PARTICULATE SIZE OF PuO_2 EMISSION THROUGH A 20-μm ORIFICE AT ROOM TEMPERATURE

We plan to study the emission through a capillary and through a simulated container crack to verify the difference. Also, we plan to investigate the effect of elevated temperatures on the PuO_2 emission through well-defined leaks and simulated cracks.

The results of this study to date lead to two general conclusions. First, the He:PuO_2 leak rate correlation was shown to be rather conservative especially at higher pressures when it ranged from 3.3×10^6 to 4.8×10^8. These pressures are expected under certain accident conditions if a fire is involved. However, one would anticipate that a failed shipping container would be pressurized for only a relatively short period of time when compared to the overall accident plus cleanup time. Therefore, shortly after the fire is extinguished, the container would rapidly depressurize to near ambient conditions. The leak rate correlation at ambient pressure was found to be conservative by three to four orders of magnitude as compared to six to eight at the higher pressures. The second general conclusion deals with the predictable nature of the studied parameter. For the most part, the system behaves as expected. Pressure, orifice size, orientation, and vibration all produced parametric effects that can be explained in terms of relatively simple mechanisms. This leads to a feeling of confidence that the plutonia particulate leak rate from a leaky container, within the range studied, may be highly predictable.

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RADIOACTIVE WASTE MATERIAL TREATMENT AND PACKAGING

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EUROPEAN EXPERIENCE WITH ASPHALT PACKAGING OF RADIOACTIVE WASTES

John E. Stewart
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In Europe, the use of asphalt is synonymous with volume reduction of low and medium level radioactive wastes. It started at Marcoule, France in the early 1960's and soon was adopted by Karlsruhe in West Germany, Eurochemic in Belgium, and in other countries.

The use of asphalt (or bitumen) as a binder, or immobilizing agent, in the VRS (Volume Reduction and Solidification) process proved to be beneficial in many ways. At Karlsruhe, for example, the VRS asphalt system replaced a cement system (non-volume reducing) and resulted in two drums of solidified waste versus ten with cement. This process reduced the transportation problem considerably. Asphalt is an inert, waterproof material and provides significantly improved package integrity under all conditions of on-site storage, transportation and burial.

Low level radioactive wastes are produced in nuclear power plants as a liquid mix, with the solid radsalts comprising anywhere from five to fifty percent of the total volume. The greatest proportion lies in the range of five to twenty percent solids, and 95 to 80 percent water.

The objective in packaging and transportation of these wastes to the disposal site is to eliminate the water and ship only the solid radioactive portion which should be immobilized in some kind of binder.

Historically, the binders used were cement and urea formaldehyde. Neither of these, however, eliminates the water, but rather uses the liquid as the medium in which to accomplish a chemical process of solidification. Europe was the first to investigate a new technology which revolves around the use of a twin-screw extruder-evaporator. The liquid radwastes are fed to the extruder-evaporator simultaneously with the binder, which in this case is asphalt (or bitumen). As the liquid, radsalts, and the asphalt progress from feed to discharge, the following events take place simultaneously. (See Figure 1.)

1. The liquid is evaporated out.
2. The solid radsalts are kneaded into small particles of $\sim$ 20 microns.
3. Each particle of radsalt is coated with asphalt.
4. The entire mix at discharge is homogeneous, free of unwanted water and highly resistant to leaching and other environmental impacts.
5. The volume of the end product is reduced by a factor of from five to 20 times less than the volume fed into the process.

The asphalt VRS system provides considerable cost savings, particularly for the recurring ems such as binder, containers, transportation, and burial. These annual savings may approach

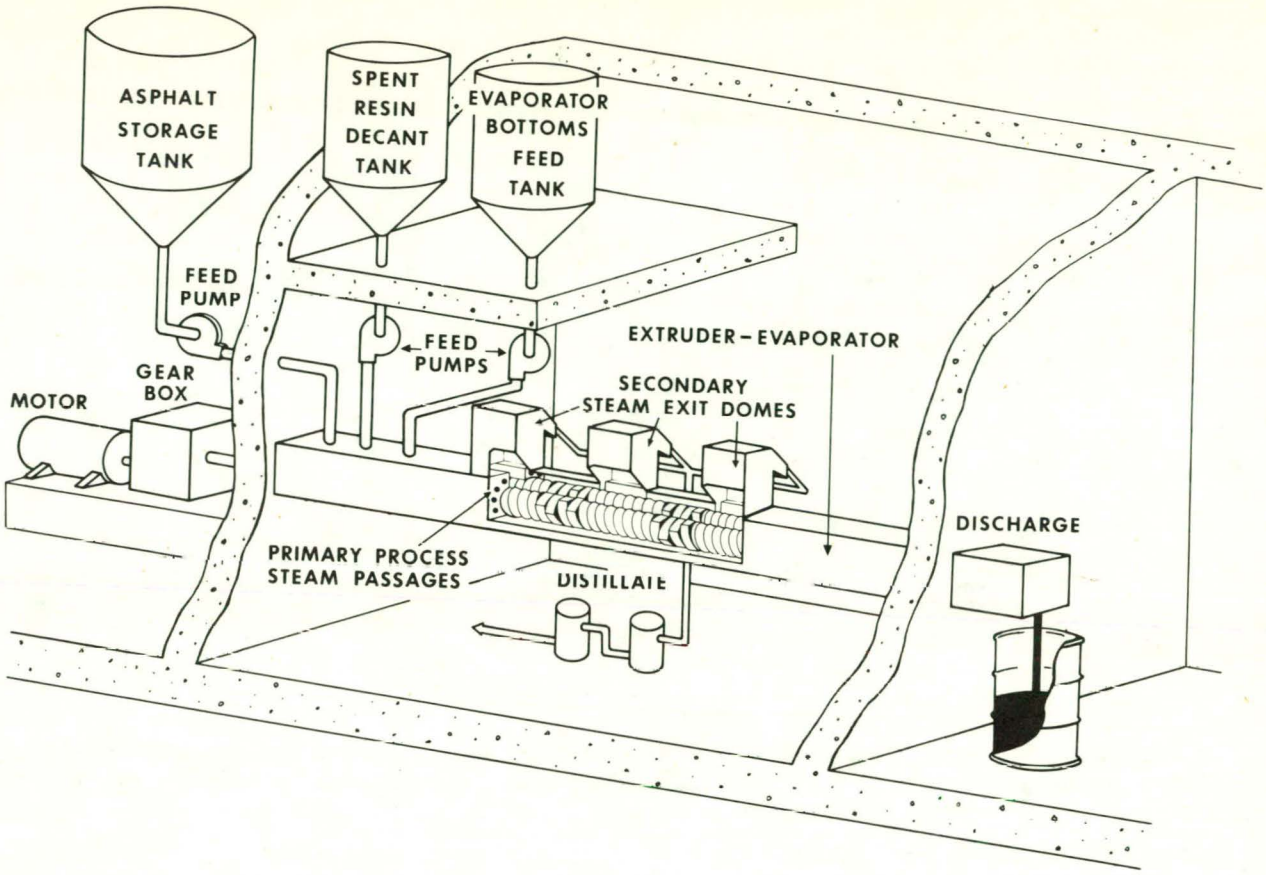


Figure 1. Schematic of the VRS radwaste system.

$500,000 annually for a 1,000 MWe nuclear plant.

End product advantages include higher resistance to leaching and other environmental impacts, as well as less internal corrosion of the drums.

In terms of packaging and transportation, the asphalt volume reduction and solidification system achieves many objectives, all of which were proven in Europe prior to the introduction of the VRS system in the United States. Among the major benefits are the significant reductions in handling, storage, transportation and burial, primarily due to the reduction in volume. Hand-in-hand with those are significant cost savings, plus assurance of compliance with guidelines and regulations.

History

The VRS system was developed in Europe at the Marcoule facility of the French Atomic Energy Commission (CEA). This development took place in the early 1960's, and resulted in the first VRS unit being installed at Marcoule in 1965 to replace the cement system they were using. Complimenting Marcoule's efforts, the Karlsruhe Nuclear Research Center in West Germany also embarked on an R & D program on the same system. Karlsruhe also selected and installed a VRS system in 1972, plus a research VRS system a year later. Since then, over 20 VRS systems have been installed or purchased around the world, and the process has over 50 system years of operation.

KARLSRUHE

A simplified functional flowsheet of the Karlsruhe VRS is shown in Figure 2. Note that the asphalt solidified end product is discharged into 55-gallon drums. The asphalt end product is shipped from Karlsruhe by railroad to the Asse Salt Mine.

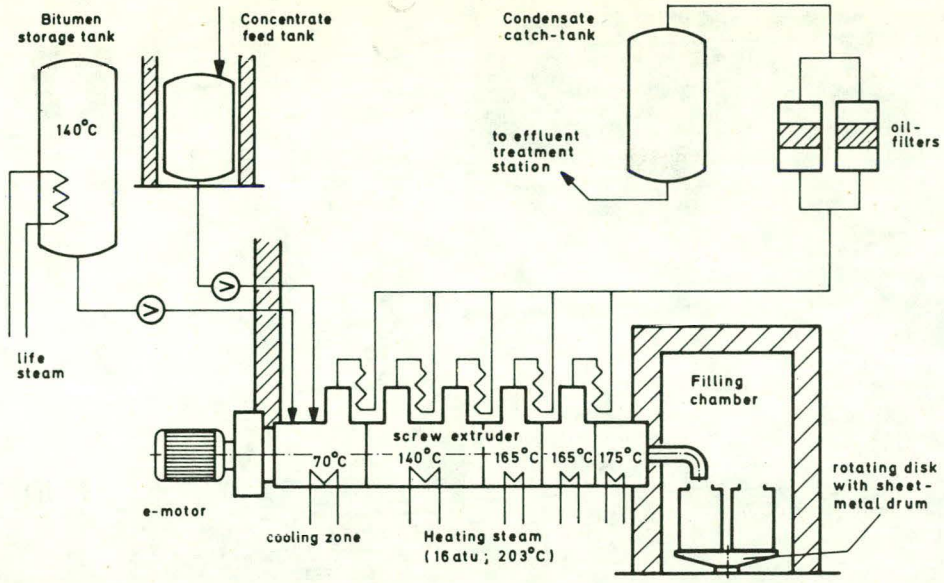


Figure 2. Simplified functional flowsheet of Karlsruhe VRS.

TABLE I

VRS EXTRUDER-EVAPORATOR INSTALLATIONS

PLANT	NUMBER OF UNITS	STARTING DATE
CEA, Marcoule, France	2	1965
CEA, Cadarache, France	1	1969
Karlsruhe, West Germany	2	1972-73
Borssele (PWR), Holland	1	1974
Atucha (PHWR), Argentina	1	1974
Eurochemic, Mol, Belgium	1	1976
Neckarwestheim (PWR), W. Germany	1	1976
AECL, Chalk River, Canada	1	1976
Unterweser (PWR), W. Germany	1	1977
Goesgen (PWR), Switzerland	1	1977
Laguna Verde (BWR), Mexico	2	1978
Japan	2	1978-79
Midland (PWR), U.S.	1	1978
Skagit (BWR), U.S.	1	1980
Iran I & II	2	1980-82

Depending upon surface dose rates, the drums may be shipped in prefabricated concrete containers (Figure 3) or in reusable shielded casks as shown in Figure 4 (one drum) and Figure 5 (7 drums).

It should be noted here that Karlsruhe also solidifies medium level wastes from a fuel reprocessing plant located there. Hence, the surface dose rates are higher than those from a power plant.

Aside from the effects of surface dose rates, what difference does the use of an asphalt binder make in packaging and transportation? A comparison of solidification at Karlsruhe with cement and asphalt is shown in Figure 6. First of all, note the volume reduction achieved. The two drums of asphalt solidified wastes contain the same amount of radsalts as the ten drums previously produced with cement. This has an effect in many areas of operation.

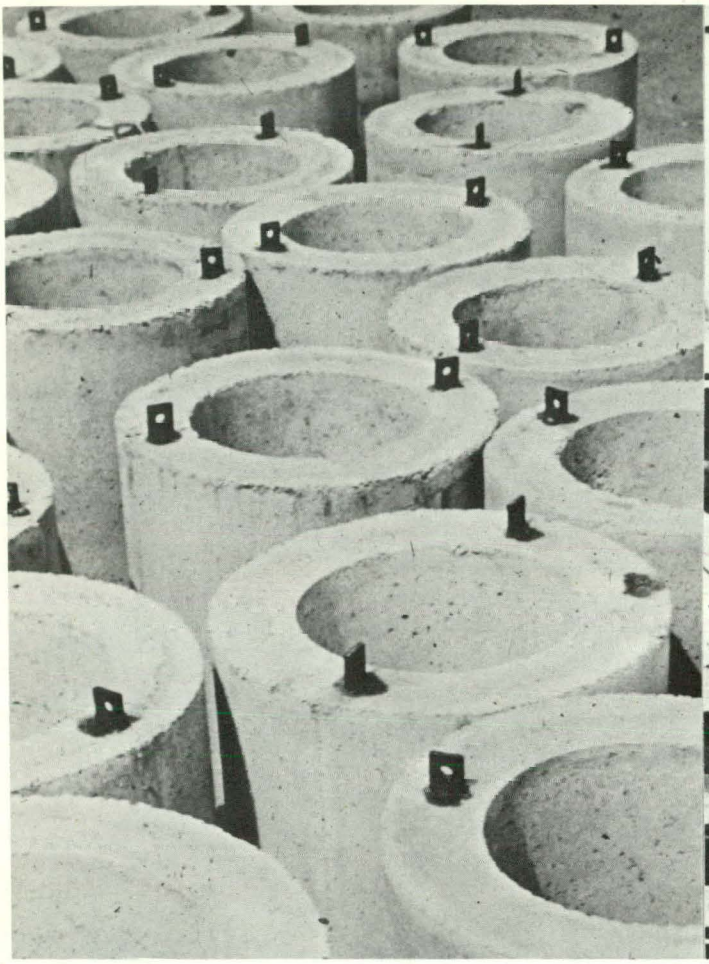


Figure 3. Disposable concrete shielding sleeves.



Figure 4. Reusable shipping cask for one drum.

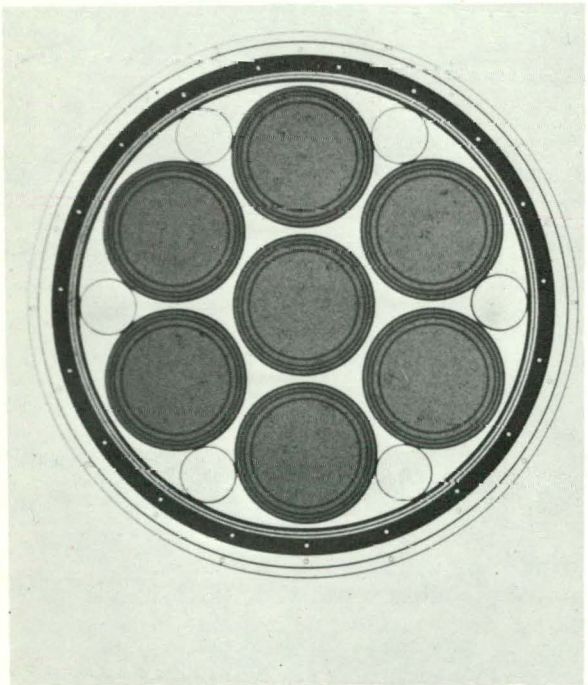
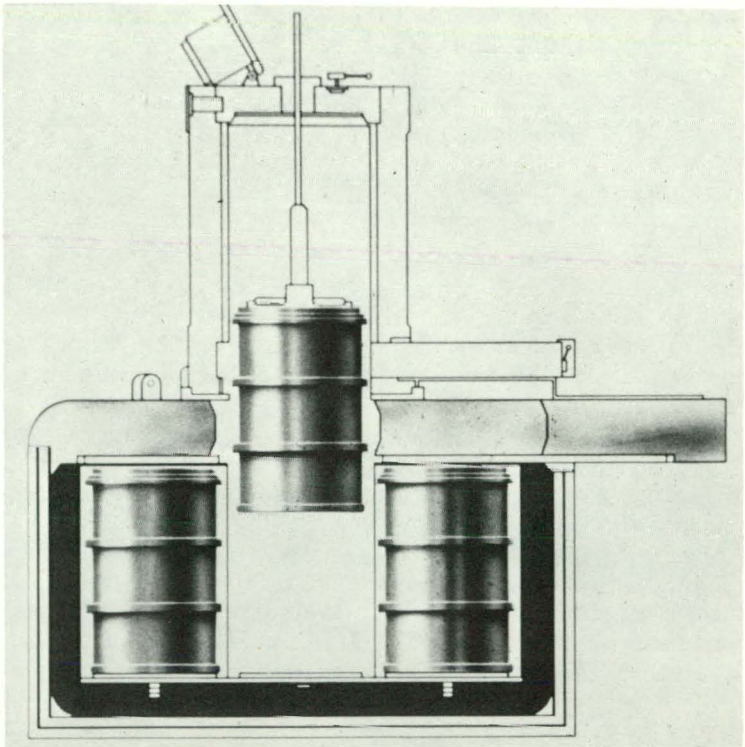


Figure 5. Shipping cask for seven drums.



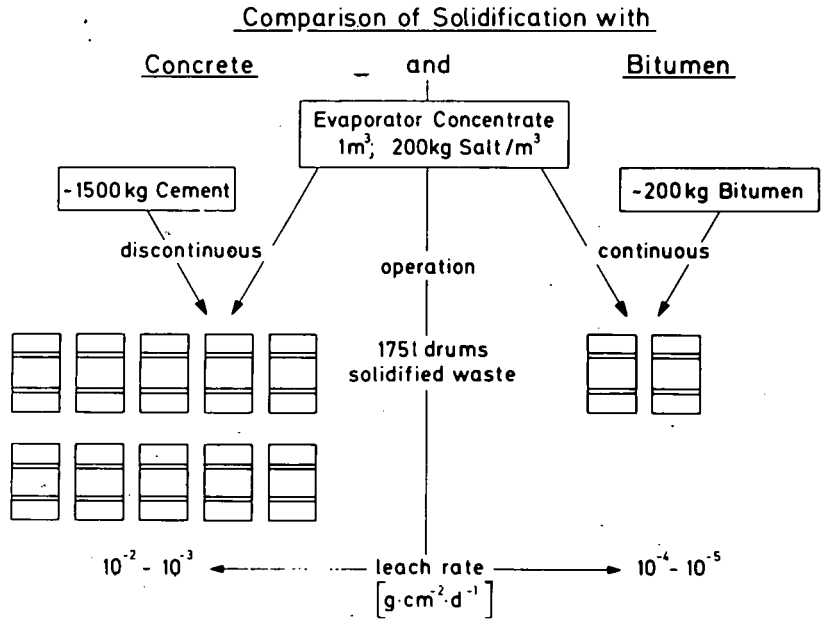


Figure 6. Comparison of concrete and asphalt for volume reduction.

On-Site Storage

Prior to shipment, filled containers are stored on site for variable lengths of time. During this time, as well as later, any free water in the end product will settle out and begin almost immediately to corrode metal drums from the inside out. Asphalt, because it is an inert, waterproofing material, inhibits drum corrosion. Couple this with the fact that the VRS process has no free water and we have a situation where internal drum, or package, corrosion is for all practical purposes eliminated.

Another factor to be considered is decay of the radionuclides during storage. Low level wastes contain mostly short half-life isotopes. Asphalt can withstand, as tested in Europe, up to 10^9 rads integrated dose before degradation of the asphalt begins. Power plants produce wastes on the order of 10^7 rads integrated dose, which is two orders of magnitude lower.

At Eurochemic in Belgium, the water table is too close to the surface to allow any kind of burial of solidified wastes, so they are using only above ground, engineered storage. Eurochemic is strictly a fuel reprocessing center and they are using the asphalt VRS process for many of the reasons already mentioned.

Transportation

The biggest impact of the asphalt VRS process on transportation is the reduction in the number of shipments. Consider a power plant that is 1,000 miles from the nearest burial site. It is not unusual for a power plant to have 150 shipments per year using a non-volume reduction system. With a volume reduction system at a VR factor of five, the shipments drop to 30 per year. In miles it translates as follows.

	SHIPMENTS	ONE-WAY MILES TO BURIAL SITE	ANNUAL MILES	MILES SAVED WITH VR
Non VR	150	1,000	150,000	
VR	30	1,000	30,000	120,000

Most of you are familiar with the cost of transportation, which exceeds $1.00 per mile. Thus, the VR savings exceed $120,000 per year just on the basis of one-way trips.

Another factor in transportation has to do with weight. There would be less savings if the volume reduction factor were five but the binder weighed 10 times more than the binder used in non-volume reduction system. However, asphalt has a density of near one, therefore achieving significant reduction in weight as well as volume.

One of the environmental advantages of asphalt is its higher resistance to leaching, a factor that is important should the container be breached from the outside or corroded from the inside due to free water in the container. At Karlsruhe, as shown in Figure 5, the leach rate of asphalt is generally two orders of magnitude lower than with the previously used cement process.

Cost Benefits

When a system meets the technical requirements, then the economics of investment must be evaluated. Since total evaluated cost is the overwhelming criterion, more attention is focused here. After the waste is packaged, it must be temproarily stored, then transported and buried. The costs increase almost linearly with volume. To compound this trend the cost of burial and transportation is rapidly escalating. The table below shows a cost comparison of a VRS with a conventional cement system.

TABLE II

ITEM	VRS	CEMENT	SAVINGS
INSTALLED COST	$1,450,000	$1,450,000	none
OPERATING COSTS			
Drums and Binder	$14,000	$120,000	$106,000
Transportation	33,000	251,000	218,000
Burial	14,000	134,000	114,000
Steam & Cooling Water	6,000	- -	(6,000)
TOTAL ANNUAL OPERATING SAVINGS WITH VRS ⟶			$438,000

In conclusion, the typical 1000 MWe BWR comparison above indicates that one could afford to pay in excess of $3,000,000 (the present capital value of the operating savings over the 40 year life of the plant at a cost of capital of 12 percent) more for a VRS than for a cement system and still be ahead.

Summary

From a packaging and transportation point of view, asphalt and the VRS system are as beneficial to U.S. utilities as to European utilities. They can be summarized as follows.

1. Reduced volume means reduced storage requirements, reduced quantity of containers, and reduced shipments. The reduced shipments, because they involve movement of radioactive material through towns and cities, also implies improved public acceptance benefits. As all of you are aware, radioactive wastes are of concern to the public and in fact New York City has banned all such shipments over any of its streets.
2. Reduced volume means significant cost savings, particularly with recurring items such as containers, binder, transportation and burial.
3. The use of asphalt improves the integrity of the package while in the plant, during transportation, and for disposal.
4. The VRS process produces an environmentally superior product that has less of an effect on the package in the event of an upset situation. In view of the negative publicity directed by the media at some of the burial sites, this also should improve public acceptance benefits.
5. The VRS has been reviewed by the various regulatory agencies and found to be acceptable method to comply with the regulations and the ALARA concept. The NRC has accepted the VRS Topical Report for utility referencing in license applications.

ACKNOWLEDGMENT

We express our appreciation to the owners and operators of the nuclear power plants in Europe for sharing their knowledge and experience with us. We are indebted especially to the personnel at two European research and reprocessing centers. These include Messrs. Baehr, Kluger, John, and Meier at Karlsruhe, West Germany, and Mr. Eschrich at Eurochemic, Mol, Belgium.

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ADDENDUM

Since this paper was originally prepared, several significant events have occurred which lend more credibility to the foresight of the European approach. Only the Barnwell, South Carolina burial site in the eastern U.S. remains open and on 1 April 78 it limited the waste it will accept to 135,000 cu.ft./mo. Also, effective 1 June 78, Barnwell will increase the burial rates by $1.50/cu.ft. meaning the minimum burial rate for LSA material with surface readings of less than 200 mR/hr. will increase by 71% from $2.10 to $3.60/cu.ft.

Any shift in transporting wastes from the eastern U.S. to the western sites will mean increasing transportation costs. For example, Commonwealth Edison was averaging $700 per shipment to Sheffield and $2,000 to Barnwell. A shipment to Beatty, its nearest western site, would average $5,000 per trip.

The above only emphasizes the need for a cost effective approach in which volume reduction of the solidified end product is a vital part.

LOW LEVEL RADWASTE PACKAGING - WHY NOT CEMENT?

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BACKGROUND

Over the past several years many words have been expended in a quest to define a variety of competing radioactive waste immobilization technologies. With the more recent recognition of the technical pitfalls of urea-formaldehyde (UF) a liquid chemical binder considered as optimum less than two years ago, utilities, architect-engineers and systems vendors find themselves in a technology void, awaiting the inevitable breakthrough which will identify the "perfect" immobilization agent.

As is to be expected, however, time to await such a breakthrough is not available. The continuing need for new radwaste solidification systems to meet plant construction schedules (a need stemming to a large part from regulatory pressures with an urgency dictated primarily by plant licensing schedules) has created strong incentives to expedite technology. Moreover, the very real operational problems being experienced with virtually all solidification methods now in use, have challenged the technical credibility of the traditional techniques and opened the way to nearly any method, as long as it seems different. At the same time, escalating costs for the transportation and burial of low level radioactive waste, as well as higher than originally expected volumes of operating plants wastes, have increased the emphasis on improving radwaste management methods, including finding means for reducing the volume of wastes to be packaged, upgrading the reliability of waste solidification systems, and reducing the radiation exposure to operations personnel which results from radwaste system operation and maintenance.

The culmination of these pressures has brought about the introduction of new immobilization technologies including: one which offers both volume reduction and immobilization in yet another new binder agent; the costly development of highly sophisticated volume reduction systems the highly-concentrated products from which may pose as-yet unknown immobilization problems; and, the marketing of several new more expensive liquid chemical binders which are reputed to have eliminated the kinds of problems associated with urea-formaldehyde. The many claims and counterclaims associated with these new products have generated a growing controversy, with experts in various utilities, architect-engineers and vendors choosing their preferred approach and defending their selection against all adversaries -- possible to the point of losing their objectivity.

At the risk of adding further to this controversy, and with little doubt that the author's affiliation will raise questions as to his objectivity, this paper addresses these issues by coming full circle and arriving back at the initial approach employed for low level radwaste immobil-

ization, the use of cement. Based on an evaluation of the three principal, competing immobilization approaches, liquid chemical, bitumen and cement, the merits and drawbacks of each is examined. As will be described, an objective assessment of these competing technologies has resulted in a somewhat surprising conclusion that, while none of the approaches is without disadvantages, cement be shown to offer the most reliable, versatile long-term solution to today's needs.

By way of establishing credibility in this conclusion, it should be noted that Hittman Nuclear & Development Corporation has been involved in systems design for all three approaches, and has supplied both liquid chemical and cement solidification systems to nuclear power stations. While this factor might imply that the author's purpose is to defend these products, it is emphasized that, quite to the contrary, the real goal is to open the minds of those whose early prejudgement of cement solidification and dogged defense of a competing technology serve as the basis for the controversies and indecision of today.

CEMENT AS AN IMMOBILIZATION BINDER

Since successful, reliable immobilization of low level radioactive wastes is a function of both the particular solidification system design and the binder used, the first basis for comparison of technologies should be the binder characteristics, independent of system designs. This is especially important in making an objective assessment under the current controversy since many of the operating problems being experienced relate more to the system design than to the binder.

Three categories of binders are addressed: (1) cement, which chemically combines with the water component of the waste and retains the salts within its chemical matrix; (2) bitumen, which as commercially used, evaporates the water component and mechanically coats the salts; and (3) liquid chemicals, which encapsulate both the water and salts interstitially within a long chain hydrocarbon matrix. For the latter category, two newly-developed agents are considered: Dow Chemical Company's proprietary vinyl acetate agent and Hittman's POLYPAC™, a modified blend of urea-formaldehydes and additives now being field tested.* A third liquid chemical, the water-extensible polyester developed by Washington State University, is not addressed since this technology has not yet reached the stage of in-plant field testing. Table 1 summarizes the comparisons discussed in the following paragraphs.

Reliability

Of primary importance in evaluating a radwaste binder is its reliability in terms of assuring satisfactory solidification. Under current regulatory requirements, this means the ability to form a dry (i.e., no uncombined or pockets of free liquids), free-standing monolith having the radwaste salts homogeniously distributed throughout. Homogeneity is dependent on the method utilized in mixing the radwaste with the binder and is addressed in the following section. Since all the binders being addressed can form a free-standing monolith in their generic form (i.e., without mixing radwaste with them), their ability to fully satisfy the above criteria relates primarily to the requirement for dryness. Dryness when blended with radwaste has been found to be a function of the binder sensitivity to waste chemistry (pH level and chemical constituents) and temperature.

Bitumen is employed in a mechanical process which evaporates the liquid and volatile elements of the waste streams and is relatively insensitive to the waste chemistry and temperature[1]; however, it is sensitive to the level of oil contaminants in the waste stream, since these can breakdown the asphalt matrix causing soft spots, and it can experience foaming when mixed with decontamination solutions. It is also sensitive to oxidizing agents in the radwaste salts, which can reduce its resistance to irradiation, producing increased gassing and swelling[2].

*No published references - information provided is based on Hittman internal R&D program results.

Table 1

Comparison of Radwaste Immobilization Agents

+ = Superior 0 = Acceptable - = Marginal/Poor

	Cement	Dow	POLYPAC	Bitumen
Reliability				
Product Dryness	+	0	0	+
Waste Chemistry Sensitivity	0	+	0	+
Safety and Handling				
Shelf Life	+	0	0	0
Conveyance/Metering	0	+	+	0
Toxicity/Breathing Hazard	0	0	0	-
Flammability	+	-	0	-
Off-Gassing	0	0	-	-
Longterm Storage				
Leach Resistance	0*	0	0	+
Tensile/Compressive Strength	+	0	0	-
Product Corrosiveness	+	0	0	+
Biodegradability	0	0	0	+

*Can be substantially improved with clay and flyash additives.

The Dow liquid chemical is also relatively insensitive to waste chemistry[3], although Dow representatives have indicated that the binder is blended to suit specific waste types, implying that several blends might be required to handle the spectrum of wastes in a typical nuclear station. The Dow binder is, however, sensitive to waste stream temperature, due to the high exotherm given off during gelling (60-80°F) which limits waste temperatures to about 130°F to prevent boiloff.

As a modified urea-formaldehyde blend, Hittman POLYPAC is similar to generic UF, requiring pretreatment of the waste and the use of a concentrated acid catalyst to obtain a low pH (2-4) for gelling; however, laboratory and prototype testing to date have shown this chemical to be less sensitive than generic UF to a wide range of waste mixtures, and more tolerant of pH variations. POLYPAC is relatively insensitive to waste temperature variations.

By comparison to these alternate binders, cement falls in the middle. Both Portland Type I and Type II cement require a relatively high pH (greater than 10) for adequate setup, especially when mixed with acidic wastes[2]. For pretreated wastes, where pH is adjusted to greater than 8, cement is more tolerant of the waste chemistry, but is sensitive to the presence of borates which retard the setup process, a factor which has been responsible for many of the operating problems associated with early (pre-1976) cement solidification systems. Failure to solidify boric acid concentrates has been overcome in later system designs through the use of chemical pretreatment to adjust pH to the basic range, by increasing the cement:waste ratio and with additives such as liquid sodium silicate injected into the wet cement after mixing with the waste (United Nuclear Industries) or anhydrous Metso Beads* (sodium metasilicate) blended with the dry cement prior to mixing (Hittman). Operating experience has also shown that cement can be sensitive to temperature when it is mixed with hot, low pH boric acid concentrates causing an acid-base exothermic reaction that has deformed and popped drum lids; later system designs have overcome this problem through preadjustment of waste pH as discussed above.

*Metso Beads is a registered trademark of Philadelphia Quartz Co.

Within these qualifications, all of the binders described will produce a dry product which meets the requirements for disposal. Bitumen has the edge in this category, primarily due to the manner in which it is used whereby the liquid waste component is evaporated and the remaining salts are coated by the hot asphalt. However, cement falls just behind bitumen; while more sensitive to te chemistry than bitumen, since it relies on chemical combination with water to setup, cement superior to the liquid chemical encapsulation approach where chemistry and/or temperature variations can seriously impact the gelling action and the ability of the chemical matrix to absorb the water component in the waste. Moreover, recent study results[5,6] have indicated that when exposed to air much of the entrapped liquids tend to evaporate, challenging the long-term viability of all liquid chemicals for radwaste immobilization.

Safety and Handling

A second category of importance in the evaluation of immobilization binders is their safety and handling requirements. This includes such considerations as shelf life, ease of conveyance and metering, toxicity, flammability and off-gassing.

Bitumen is weakest in this category. Although it has a long shelf life, it must be maintained at an elevated temperature (250°F) in order to be pumped. More importantly, as a hydrocarbon it is subject to both toxicity and flammability hazards, requiring an enclosed ventilation system with fire detection and suppression systems to safeguard the facility and operators. In addition, the need for high process temperatures (350°F) to support evaporation of the water component necessitates accurate process controls to prevent an over temperature excursion approaching its ignition temperature (550°F). During cooling, irradiated bitumen gives off hydrogen and related hydrocarbon gases, requiring that the processed waste/bitumen blend be allowed to cool slowly to avoid trapping an explosive concentration of these gases; this requirement limits the process rate achievable and dictates partial filling of several containers in sequence, limiting the system throughout.

Off-gassing due to radiolysis continues after the bituminized product is cooled; irradiation of bitumen specimens has shown that the release of gases, including hydrogen, is about ten times greater than for cement, and can result in a breakdown of the asphalt matrix at high (10^7-10^8 Rad) cumulative doses, especially if accompained by radiolytic oxidation[2]. The off-gassing associated with bitumen necessitates a pressure-relief-type of container cap for bituminized waste. Since bitumen reliquifies when exposed to temperatures above 200°F, and burns at temperatures above the 600°F, it poses a substantially higher risk in the event of an accident during transportation, where the container might be breached and/or subjected to a hot fire.

The Dow binder has a low ignition temperature (100°F) requiring added precautions during storage. Tests have shown neither the Dow binder[3] nor Hittman POLYPAC will support combination in their solidified form. POLYPAC has a limited toxicity hazard, due to its formaldehyde constituent; although no MPC has yet been established, it is expected that it will be somewhat better than for generic UF. Testing to date has shown that POLYPAC is far less sensitive to storage temperatures and has a shelf life which, although not yet fully determined, is in excess of the six month limitation for generic UF. The toxicity and shelf life characteristics of the Dow binder are not yet known. The level of off-gassing due to radiolysis has not yet been established for POLYPAC but should be in the same range as for generic UF, which is approximately the same as for bitumen; a potential for matrix breakdown due to long term irradiation may exist if the total dose exceeds 10^8 Rads. Tests by Dow[3] have indicated that there was essentially no effect on its product for irradiation doses in excess of 10^6 Rads. No information has yet been reported on any off-gassing for the Dow binder.

Cement offers significant advantages in this category. It is inflammable, essentially nontoxic, and has a nearly unlimited shelf life if protected from moisture. Off-gassing due to radiolysis, including hydrogen evolution, is substantially less than for either bitumen or the liquid chemicals.

Longterm Storage

Since the primary purpose of an immobilization binder is the retention of the waste radioactivity over a long period of time, until it decays below a level safe to the environment, it is portant that the effects of long term storage be evaluated. There is little specific data re-

lated to actual radioactive storage, although several programs are underway to recover and study stored waste[7]. As a result, evaluations to date have been based on laboratory studies of simulated waste mixtures and the anticipated storage effects associated with the generic binders[2].

The principal emphasis in this category has been on the potential leachability of specific long-lived isotopes when solidified wastes are exposed to means for their migration from the stored waste location, such as groundwater. Unfortunately, several different leaching test methods exist, resulting in a wide variation in the results reported. In the expectation that the IAEA leachability standards will be implemented in the United States, most tests have focused on the leachability of ^{137}Cs and ^{60}Co isotopes[8]. The varying results [1,2,3] have generally indicated that bitumen offers the highest leach resistance, with the Dow binder a close second. POLYPAC has not yet been tested in this area, but is expected to have results in the same range as generic UF, just below those for the Dow binder. Cement is generally considered to have the poorest leach resistance, primarily due to its porosity which increases the surface area potentially exposed to the leaching medium; however, the addition of flyash and various clays to the cement grout have been shown to vastly improve its leach resistance to most critical isotopes[8, 9].

Laboratory leaching tests can be misleading in that they do not take into account the effects of solidified waste fracture or disintegration when buried; cracks, fissures, or crumbling serve to increase the surface area exposed to the leaching medium, and in the case of encapsulated wastes such as those immobilized in liquid chemicals, tend to release the entrapped liquids. Therefore, the structural strength of the various types of waste, when solidified in an immobilization binder can be as important as the leach resistance of a laboratory sample.

Tensile and compressive strength tests generally show that cement offers the best overall structural qualities, although these results are variable according to the ratio of cement to waste and the type of waste--solids such as bead resins and filter aids typically produce a weaker product. The Dow Binder structural qualities are very close to the typical results for cement (2000-3000 psi compressive strength; 4-5 in-lb/in impact strength)[3]. Bitumen tends to flow when subjected to compressive forces at room temperature, and has an impact strength significantly below the other materials, even when subjected to very low temperatures[10]. No strength data is yet available for POLYPAC.

Other areas of concern for long term storage include the corrosiveness of the solidified waste product and its biodegradability. Bitumen has good resistance to acidic/alkali attack, and to bacteriological attack. Cement is susceptible to galling when exposed to strong acids or sulfates, but is generally considered as resistive to the typical groundwater constituents, and to bacteriological attack. Little is known in this area for the liquid chemicals being considered; however, it should be noted that for all types of binders, leaching of acidic or corrosive waste salts will inevitably accelerate the corrosion of the typically-used mild steel containers.

One alternative to current practices, which could ultimately eliminate most long term storage concerns regardless of the immobilization binder, is the use of nondestructible container materials in place of mild steel or concrete, such as polyethylene or fiberglass[11].

SOLIDIFICATION SYSTEM TECHNOLOGY

The following paragraphs provide a brief overview of the solidification system technologies currently being offered. Table 2 summarizes the highlights of this discussion by comparing these technologies on a subjective basis.

Cement Systems

The majority of the radwaste solidification operational problems which have prompted the evolution of both binding agents and system designs, can be attributed to the technology associated with the method employed for blending the wastes with a binder. The predecessor of today's continuous process systems consisted of little more than a means for manually filling a drum with dry cement and liquid waste, which was followed by placing a weight in the drum, sealing it and

tumbling the contents on a horizontal drum rolling mechanism. This approach, as well as later designs which perform the mixing in the drum with a disposable blade, result in high exposures to the operator and are subject to high cement dust levels in the atmosphere around the operation. In addition, the need for sufficient freeboard to allow either tumbling or propellor mixing significantly reduces the amount of wastes which can be solidified per container (packaging efficiency).

Several more sophisticated cement systems have been introduced, which are designed to handle all types of wastes from a remote location, minimizing operator exposure and increasing both packaging efficiency and the throughput of waste solidification. Unfortunately, the initial production versions of these designs were subject to new problems. Cement dust leakage from the pneumatic transport systems, and from the container opening, coats equipment and clogs floor drains. Inline dynamic mixers tended to clog. Flushwater used to remove residual cement from the process components has managed to feedback into the cement feed train, causing blockage. Flushwater which is processed back into the retention tanks or drain system leaves a residue in the tanks clogging tank nozzles. Thus the savings in operator exposure were more than offset by the increased exposure due to maintenance. Since early system designs had not yet recognized borate retardation of cement setup, there were many problems with the inability to solidify concentrated boric acid wastes; and these have been complicated by system deficiencies which resulted in innaccurate or unreliable cement metering, reducing the cement:waste ratio and causing residual free water in the container.

All of these problems are related more to system design deficiencies than to the cement agent. In an attempt to overcome such deficiencies, Hittman developed its own cement solidification system design, in 1976, building a full scale prototype with which to test out and prove features specifically aimed at resolving early cement system problems. Although the first production version will not be operational until 1980, testing and demonstrations conducted with the RSS(C) prototype have shown that the deficiencies in earlier designs can be overcome. A topical report which fully describes the Hittman RSS(C) and its integrated materials handling systems has been accepted by the USNRC[4].

Liquid Chemical Systems

In spite of efforts by several vendors to improve upon early cement system designs, the widespread introduction of urea-formaldehyde (UF) in 1974 appeared to be the preferable solution. As a liquid chemical, UF could be more easily conveyed and metered. Inexpensive inline static mixers could be used to blend the wastes with it, a seemingly simpler, more reliable approach than the expensive, complicated dynamic mixers. Less facility space was necessary, although the recent trends toward automated, remote-operated materials handling equipment normally dictate the overall space requirements moreso than the process equipment.

Despite some initial break-in difficulties, the advantages of a liquid chemical system design have been generally borne out through experience. However, the problems associated with urea-formaldehyde as an immobilization binder have more than offset such advantages, and many operating radwaste facilities which were designed around or have installed UF systems are now considering retrofitting with an alternate technology. Because of the cost impact of having to retrofit, especially for facilities where the purchased UF systems have not yet been placed in operation, Hittman embarked on a program to develop an immobilization binding agent which would be compatible with its RSS(P) liquid chemical system. Although not yet fully proven, Hittman's POLYPAC agent is the hoped-for answer to this need, at least for the near term.

However, as the previous discussion indicated, all liquid chemicals being considered are subject to serious long term drawbacks. Besides their comparatively high cost (3-5 times the cost of generic UF, and up to 40 times the cost of cement), the fact that these chemicals only encapsulate the waste liquid, as opposed to chemically combining with it, challenges their viability as a long range solution for waste immobilization. It is primarily for this reason that UF has generally not been accepted for radwaste solidification outside of the United States.

Unfortunately for the industry, the apparent advantages of liquid chemical binding agents, even in the face of known drawbacks such as free water and radiolytic gassing, created a "bandwagon" effect in the mid-1970's, with a variety of utility, architect-engineer and vendor experts

vying with each other to promote this technology over cement. Looking back today, it is clear that unless a liquid chemical binder is identified which can truly satisfy the likely long range requirements for immobilization, a great deal of utility money will have been wasted on systems which may not be usable in the future.

Bitumen Systems

With the deflation of the "urea-formaldehyde balloon", many utilities and architect-engineers have looked with new interest to the third immobilization technology. Bitumen has been used for radwaste immobilization since the mid-1960's in Europe, and has been available in the United States since 1974. There are several competing system designs available, but because of their high initial cost (2-3 times higher than cement systems; 3-4 times higher than UF systems) they were considered unacceptable until relatively recently.

From a system design standpoint, bitumen systems have advantages over cement similar to those for liquid chemical systems. The binding agent, asphalt, is metered in a liquid form, making accurate metering easier. Since no catalyst or additives are required, the process needs only two streams, radwaste and the asphalt binder. However, the need to maintain a high process temperature, for both liquefaction of the asphalt and evaporation of radwaste liquids, necessitates a heating medium; and, condensation of the evaporated effluent stream dictates a cooling medium. Because bitumen can be reliquified by heating above 250°F, there is less concern over clogging the evaporator/mixing device--it need only be brought back up to temperature after a failure. And, for the same reason, cleaning the evaporator/mixing device is relatively simple and can normally be accomplished with steam.

Although bitumen systems have been used for over ten years in Europe, most of the older facilities with such systems are either experimental or involved in fuel processing, with wastes significantly different in both kinds and quantities from those created by a power reactor station. The few operating bitumen systems which resemble the designs now being offered in the United States have only limited operational time. Moreover, the generally lower waste volumes generated by European reactor plants are more compatible with the low process rate available in a bitumen system.

The bitumen system design currently being purchased by United States utilities employs an evaporator/extruder as its principal component. A competing design employing a thin film evaporator is available from several sources, but has not yet been purchased in this country. Several operational problems have been associated with the evaporator/extruder design in Europe. Foaming and plugging of the steam domes through which the evaporated liquids are removed and condensed was an early problem, especially with concentrated decontamination wastes. Carryover of oil into the condensate, a problem resulting from volatilization of oils in the bitumen, necessitates the installation of oil screens or filters to reduce the level of oils in the effluent; cleaning or replacement of the screens adds to the maintenance workload and has reportedly been only marginally effective in controlling the oil contamination of the condensate. The condensate effluent may not be recyclable if the level of oils is above a minimal level. The system is subject to much higher energy consumption than either cement or liquid chemical systems, due to the large (50-60 HP) extruder drive and the steam and electricity used for heating the bitumen and the evaporator.

As is to be expected, the vacuum created by the "downfall" of urea-formaldehyde has generated a trend towards more serious consideration of bitumen systems, especially among the various experts who had previously promoted liquid chemicals in general and UF in particular. Just as the advantages of a liquid chemical over dry cement were used to originally justify the adoption of UF over cement systems, the current trend has been to justify bitumen on the basis that it offers both volume reduction and solidification, whereas cement can only solidify wastes--a variety of papers have been presented to show how the savings due to volume reduction can more than offset the higher initial capital cost. While there is no question that reducing the volume of the wastes which must be solidified represents a substantial savings in transportation, burial and materials costs, to use this justification to promote bitumen technology without adequately considering the potential long term problems of bitumen as an immobilization binder, is analgous to the kind of premature reaction which resulted in the UF "debacle".

Table 2

1 = Preferable Design 2 = Average 3 = Weakest Design

	Cement	Liquid Chemical	Bitumen
Facility Design Impact			
Equipment Space Allocation	2	1	3
Waste Storage Space Requirement	2	2	1*
Services/Power Requirement	1	1	2
Fire Detection/Suppression Requirement	1	2	3#
HVAC Requirement	1	2#	3#
Process Design			
Simplicity of Design	2	1*	2
Automation/Programmed Logic	1	1	2
Instrumentation & Controls	1	1	2
Metering Accuracy	2#	1	1
Process Rate	1	2	3#
Operational Considerations			
Facility Throughput	1*	2	3#
Remote Operability	1	1	2
Maintenance/Reliability	3#	2	1
Operator Exposure	2	1	2

* Strongest Advantages
\# Most Serious Disadvantages

VOLUME REDUCTION & ECONOMICS

In this section, the comparison among the three competing technologies is carried to its ultimate, most controversial aspect, economics. Before embarking into a war of numbers, it must be noted that while traditional justifications for "better" technologies, especially in the changing field of radwaste management, have always resorted to economics, the current promotion of bitumen systems over competing immobilization methods is unfairly taking advantage of their evaporation capability. There are other methods for reducing the volume of the wastes which must ultimately be immobilized, which can be interfaced with cement solidification systems. Therefore, using the cost savings associated with bitumen system evaporation of waste liquids, as compared to another method without any volume reduction, is guilty of the proverbial "apples and oranges" syndrome.

For purposes of comparing economics in the following paragraphs, two waste types are considered:

#1 25 w/o Sodium Sulfate - BWR Chemical Regeneration waste (Brookhaven National Laboratory - Type 3a)[10]

#2 12 w/o Boric Acid - PWR Boric Acid Waste (Brookhaven National Laboratory - Type 3c)[10]

Materials Costs

For bitumen in both cases, the wastes are fed directly and combined into 55 gallon drums on basis of 1:1 by weight of anhydrous salts and bitumen. When using liquid chemicals or cement, it is assumed in each case that the waste is first concentrated to the range of 50 w/o in a thin

film or forced circulation evaporator/crystallizer, converting the above waste types to Brookhaven National Laboratory Types 4a and 4c, respectively[10]. These concentrated wastes are then solidified into 55 gallon drums or 80 ft^3 large disposable containers with cement, using the Hittman dry Metso Bead additive (see reference 4), or with liquid chemicals using volumetric mixing ratios (waste: chemical) of 1.5:1 for Dow and 1:1 for POLYPAC.

Assuming the following material costs for the respective agents, the volumes and materials costs for each waste type and immobilization method are summarized in Table 3A:

Bitumen	$.05 per lb
Portland I Cement	$.035 per lb (including proportioned Metso Beads additive)
POLYPAC	$.45 per lb
Dow Agent	$.85 per lb

Transportaion and Burial

Table 3

A. Comparative Materials Costs (55-Gallon Drum Pkg)

Agent	Waste Type	Total Shipping Vol. (Gal)	Raw Waste Vol. per Pkg. (Gal)	Net VR	Cost Per Gallon Raw Waste *
Cement	#1	51	110	2.1	$0.26
	#2	51	173	3.3	$0.19
Bitumen	#1	51	120	2.3	$0.27
	#2	51	246	4.8	$0.12
POLYPAC	#1	51	53	--	$1.55
	#2	51	84	1.6	$2.00
Dow	#1	51	64	1.3	$1.89
	#2	51	100	1.9	$2.55

B. Comparative Annual Shipping Requirements

Typical Number of Shielded Shipments Per Year for Liquid Waste (@ 7 drums per Cask)

	Waste Type #1 (BWR)	Waste Type #2 (PWR)
Bitumen - Drums	43	12
Cement - Drums	47	17
- 80 ft^3 LC	34	26
POLYPAC - Drums	97	34
- 80 ft^3 LC	63	22
Dow - Drums	80	29

*Includes $17.00 per empty drum in all cases.

Since transportation and burial charges vary widely depending on plant location, waste shielding requirements and the total volume shipped, this discussion is limited to a comparison of the approximate number of annual shipments for the waste types previously described.

Assuming an average generation of 36,000 gallons per year of waste type #1 (BWR) and 20,000 gallons per year of waste type #2 (PWR), Table 3B summarizes the number of shipments for each technology.

Conclusion

Figure 1 illustrates the theoretical basis for the conclusion that cement solidification with upstream concentration is the equal to bitumen-VR. As was indicated by the economics in the preceeding paragraphs and can be seen in this curve, the use of up to 50 w/o concentrations with cement yields a relative volume reduction factor of six, as against a factor of ten for solidification of dry salts or bituminization. This equates to one more drum per group of ten drums now being shipped than would be the case with bitumen-VR; improvement in concentrations in this range on the curve nets less overall volume reduction. When considering the substantial increase in capital investment for both equipment and facility design to accrue these final, limited savings, the cost effectiveness of the bitumen alternative becomes all the more debatable.

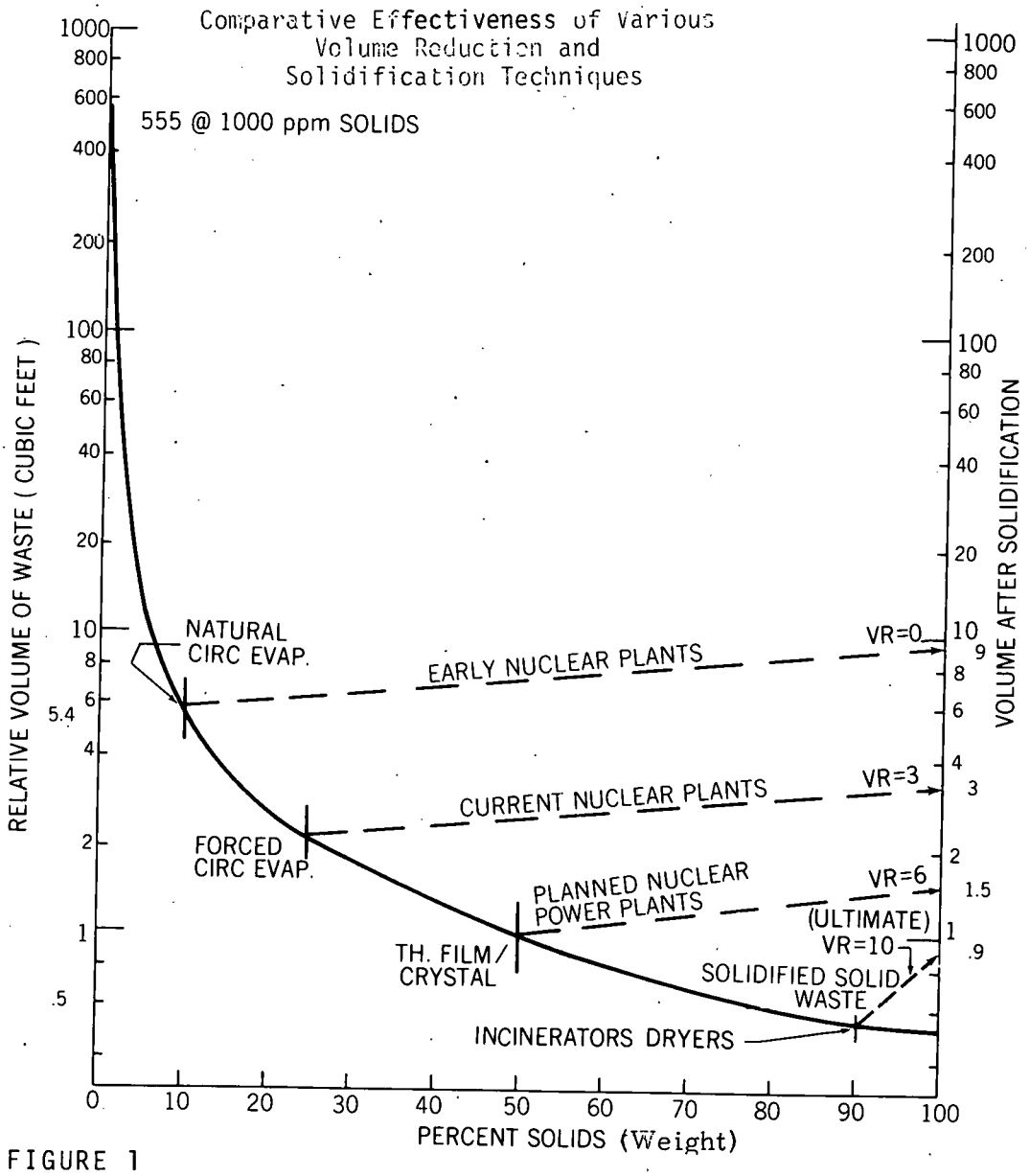


FIGURE 1

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THE DESIGN AND USE OF THE MODEL N-55 AN INDIVIDUAL 55 GALLON DRUM TYPE B OVERPACK

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I. Introduction

On June 1, 1977 Nuclear Packaging, Inc. was granted Certificate of Compliance No. 9070 for the Model N-55, an individual 55 gallon drum Type B overpack. The N-55 was licensed at a time when permits for similar drum overpacks were expiring and remains today as one of the few existing single drum overpacks.

This paper presents a description of the N-55 overpack and outlines some of its major design features. The licensing effort will be discussed including a description of the physical tests and analytical evaluations conducted on the package to demonstrate its integrity against the hypothetical accident conditions. Finally, a comparison will be made between transportation systems based on individual drum overpacks and multiple drum overpacks such as the Super Tiger (DOT 6400). The various packaging and transportation efficiencies of each system will be shown.

II. Package Description.

The following illustrative drawing (Figure 1) shows the main design features of the N-55 overpack. The basic structure consists of an outer steel shell and an inner molded fiberglass shell. The outer shell is fabricated of ductile low carbon galvanized steel which allows it to undergo large deformation without fracturing. Fiberglass provides a tough, high strength inner shell or liner capable of taking abrasive handling and corrosive material. The volume between the two shells is filled with medium density polyurethane foam. This poured in place insulating material bonds to the shells creating a unitized construction. In addition, high strength paper honeycomb is added to the top end of the overpack providing additional protection to the drum rolling hoop area.

Once assembled, the overpack takes the shape of a vertical right angle cylinder with a central separation plane. In use, the lower unit comprises the body or base of the container while the upper unit serves as the lid. The stepped joint between the two halves is sealed with a neoprene gasket. Both halves are secured by four heavy duty toggle clamps. Tidedown and lifting devices at four places equally spaced along the separation plane provide convenient handling points and a means of removing the lid.

The overpack is not intended to be the containment vessel. Its prime function is to duce the severity of the hypothetical accident conditions and assure there will be no loss

MODEL N-55 TYPE B OVERPACK

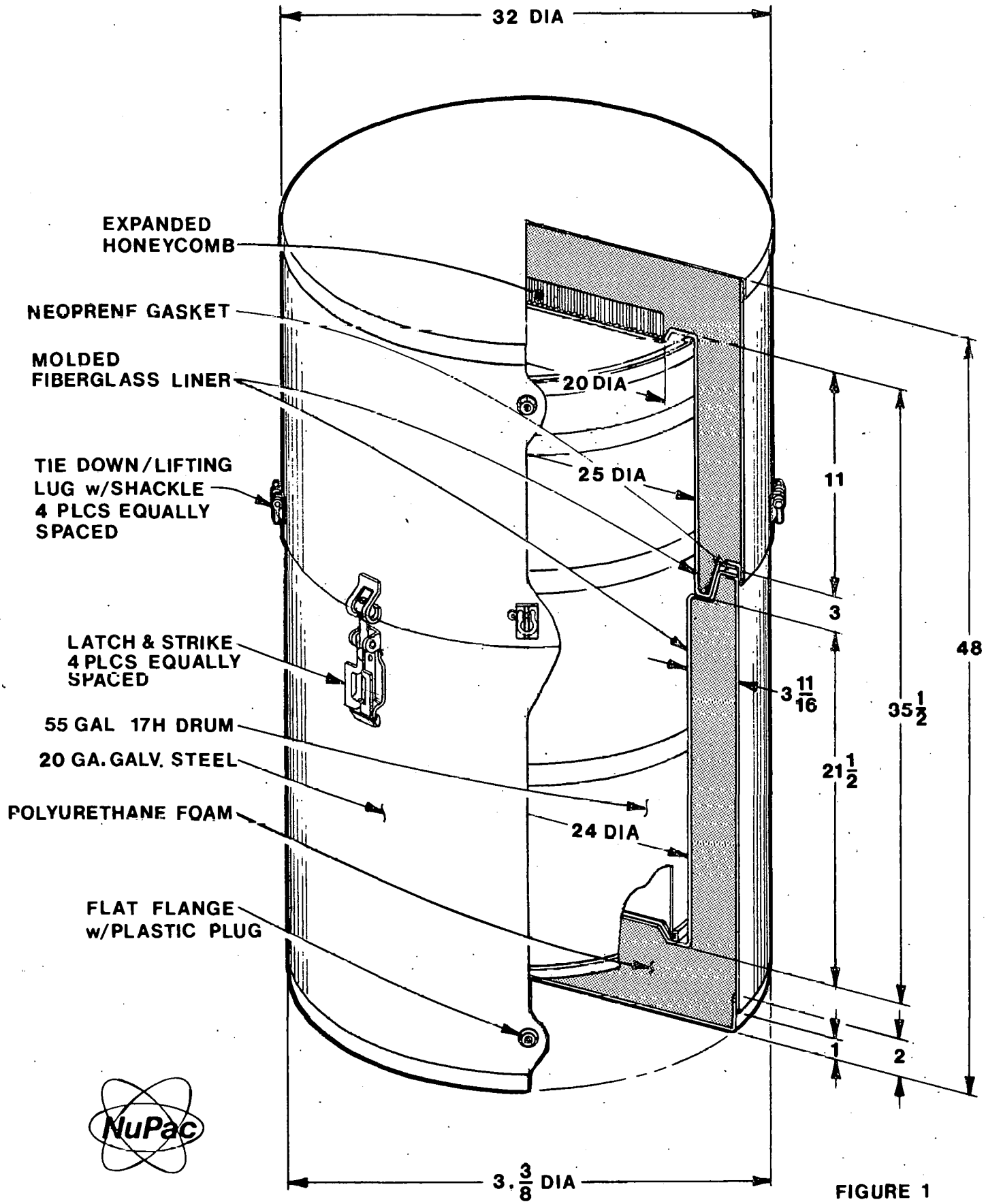


FIGURE 1

of contents to the 55 gallon drum. The containment vessel is a 55 gallon drum meeting the requirements of Specification 17H. This by definition means a gasketed lid and in the case of an open top drum, a heavy duty bolt ring lid attachment.

The N-55 is authorized to handle Type A or Type B quantities of radioactive material in normal or special form as well as "large quantity" material. Fissile quantities are those limited to the amounts as generally licensed under 10CFR71.9. Contents can be in any chemical or physical form except liquids, powders or slurries. (Testing is currently underway to qualify the package contents to include liquids). Shielding is not a part of the packaging although it can be added within the containment vessel. The package gross weight is approximately 750 lbs. Maximum weight of the contents cannot exceed 550 lbs.

III. Licensing

The primary function of the N-55 overpack is to protect a 55 gallon drum from the hypothetical accident conditions. To demonstrate its capabilities, a combination of full scale tests and computerized analysis were run. Full scale drop tests verified the overpack's impact insulating capabilities while an analytic solution was used for the hypothetical fire test condition.

From previous testing on drums alone, it was determined that the most critical area was that of the bolt closure ring. To minimize local deformation in this area and therefore insure its integrity, the drum was surrounded with a tough undeflecting fiberglass shell. This inner shell was stabilized with a high energy absorbing medium density foam in combination with a ductile, low carbon outer steel shell capable of absorbing energy through localized deformation. As a result, concentrated loads such as the 30 foot corner drop could be uniformly distributed over a large portion of the drum. This distribution eliminated localized loads that would result in bolt closure ring deformation and subsequent leakage.

The package was assessed for its ability to withstand the 30 foot free drop and 40 inch puncture by subjecting it to full scale testing. A Specification 17H drum filled with wet sand and weighing approximately 550 pounds was pressure tested to 10 PSI prior to the test. No leaks were noted. The drum was then placed in the N-55 overpack. Both halves of the overpack were secured together by the four heavy duty toggle clamps.

A series of three 30 foot drops were conducted on the overpack in drop orientations predicted to cause the most severe damage. The first drop test produced a corner impact directly over the drum's bolt ring. The second drop impacted on the diagonally opposite corner. The third drop produced an impact on the overpack's side directly over one of the latches. Figure 2 shows the results on the overpack after three 30 foot drops. Subsequently, the package was tested for puncture resistance by dropping it 40 inches onto a 6 inch diameter pin. Damage to the overpack amounted to an 8 inch crush depth to the corner and only a 3/4 inch local flattening as a result of the 30 foot side drop. The 40 inch puncture test produced only minimal local deformation.

The overpack was then opened for inspection. All latches, including the one receiving the direct impact, remained fully locked during the tests and upon opening, worked smoothly. The 17H drum including the lid experienced only minimal deformation and there was no loss of contents. (See Figure 3). The side drop produced a slight flattening of the drum but appeared to have no effect on its integrity. The fiberglass liner showed little or no deformation except for a few small cracks resulting from the side drop. It was concluded that the overpack provided the needed protection to the drum enabling it to maintain its integrity through the drop and puncture tests.

The fire test requirements were verified by a special computer program developed especially for Type B package designs. This method of verification has been used successfully for 8 other Type B container designs using polyurethane foam as the shock and thermal insulating material. (In some cases, actual fire tests were run showing a close correlation

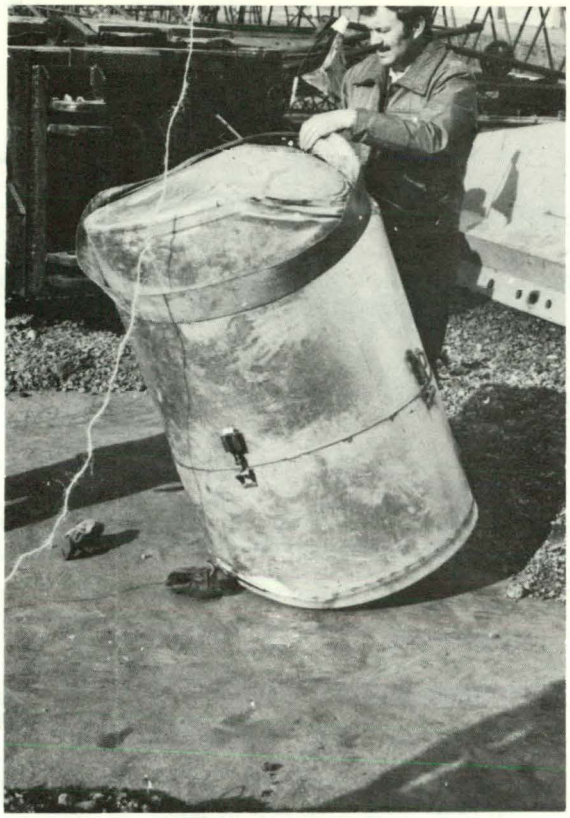


Figure 2. N-55 overpack after three 30 foot drop tests.

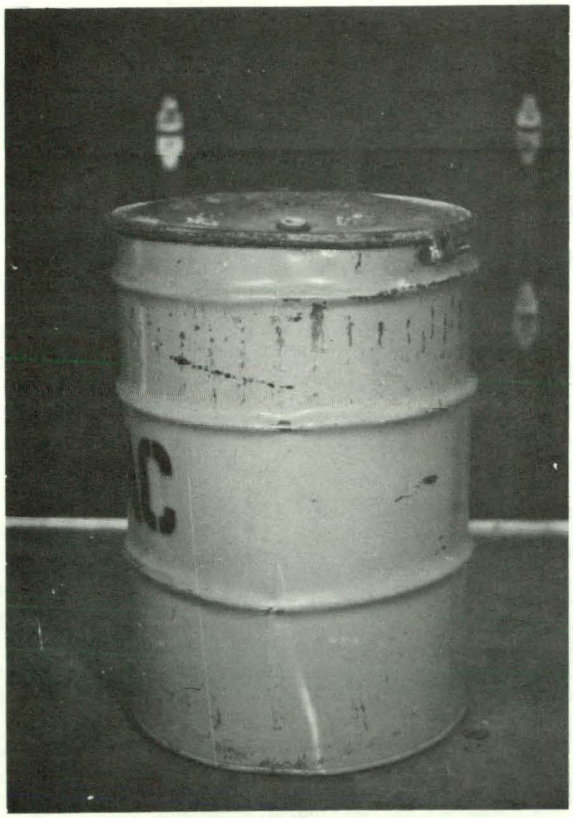


Figure 3. 17H 55 gallon drum after puncture test and three 30 foot drop tests.

to analytic results). In the case of the N-55 overpack, it was the only method available since local pollution regulations ruled out physical tests of this nature.

Figure 4 shows the results of the hypothetical fire test on the N-55 and its contents. illustrated, the 55 gallon drum and inner fiberglass shell experienced temperatures less an 200°F. The payload itself experienced temperatures of only 172°F. Therefore, it was concluded that the exposure to a 1475°F thermal environment would not cause a pressure detrimental to the containment vessel.

IV. Use of Individual Drum Overpacks as a Transportation System

When confronted with the problem of transporting large numbers of unshielded drums (which will likely be the case for future shipments of transuranium (TRU) wastes), use of individual drum overpacks should be considered as a viable alternative to multiple drum overpacks such as the Super Tiger or ATMX Railcar. As a total transportation system, individual drum overpacks offer not only flexibility in handling but also distinct packaging and transportation efficiencies due to effective utilization of cargo space. To illustrate this point, a comparison is drawn below between the "Super Tiger", a popular multiple drum overpack, and the N-55.

Packaging efficiency can best be expressed as the ratio of overpack weight to payload weight. For example, the empty weight of the "Super Tiger" is approximately 17,000 pounds with a payload capacity of 42 drums.

$$\frac{17{,}000 \text{ pounds}}{42 \text{ drums}} = 405 \text{ pounds overpack per drum}$$

Therefore, for every drum transported you must also transport 405 pounds of overpack which can be directly compared to the weight of the N-55 overpack at 150 pounds. Using an average drum weight of 450 pounds, a packaging efficiency can be calculated.

$$\text{Super Tiger} \quad \frac{450 \text{ lbs. drum weight}}{405 \text{ lbs. overpack weight}} = 1.11$$

$$\text{N-55} \quad \frac{450 \text{ lbs. drum weight}}{150 \text{ lbs. overpack weight}} = 3.00$$

This yields an improvement in packaging efficiency of 270%. The obvious conclusion is that the less weight devoted to overpack, the more capacity remains for payload.

Transportation efficiency is measured in the amount of drums capable of being transported in a single load. Multiple drum overpacks are limited to the amount of drums that particular overpack can handle, notwithstanding the fact that additional trailer space may be available for the trip. Should additional trailer space be available, the tare weight alone of a second overpack might easily cause the load to reach or exceed legal highway load limits. This is not the case for an individual drum overpack system. Here, payload capacity is not limited to a fixed "containment volume" but rather, assuming legal weight restrictions are met, to the volume of the transport trailer itself. As Figure 5 illustrates, the volume of a standard 40 foot semi-van would allow shipment of 76 N-55 overpacks assuming the average weight of each 55 gallon drum did not exceed 450 pounds. This is compared with 42 drums for the Super Tiger. In another example, a standard 45 foot van would accomodate 84 N-55's with an average weight of 385 pounds while the Super Tiger would again be limited to only 42 drums. Other examples of increased transportation efficiencies could be made for rail shipments as well. As for highway transport modes, the above examples demonstrate that transportation efficiencies can be increased by as much as 80 to 100% using individual drum overpacks.

Individual drum overpack systems also offer flexibility and efficiency in the areas of handling and loading drums. Figure 6 depicts a possible loading sequence using N-55 erpacks. A conventional 40 foot van is shown backed to a loading dock. The overpacks

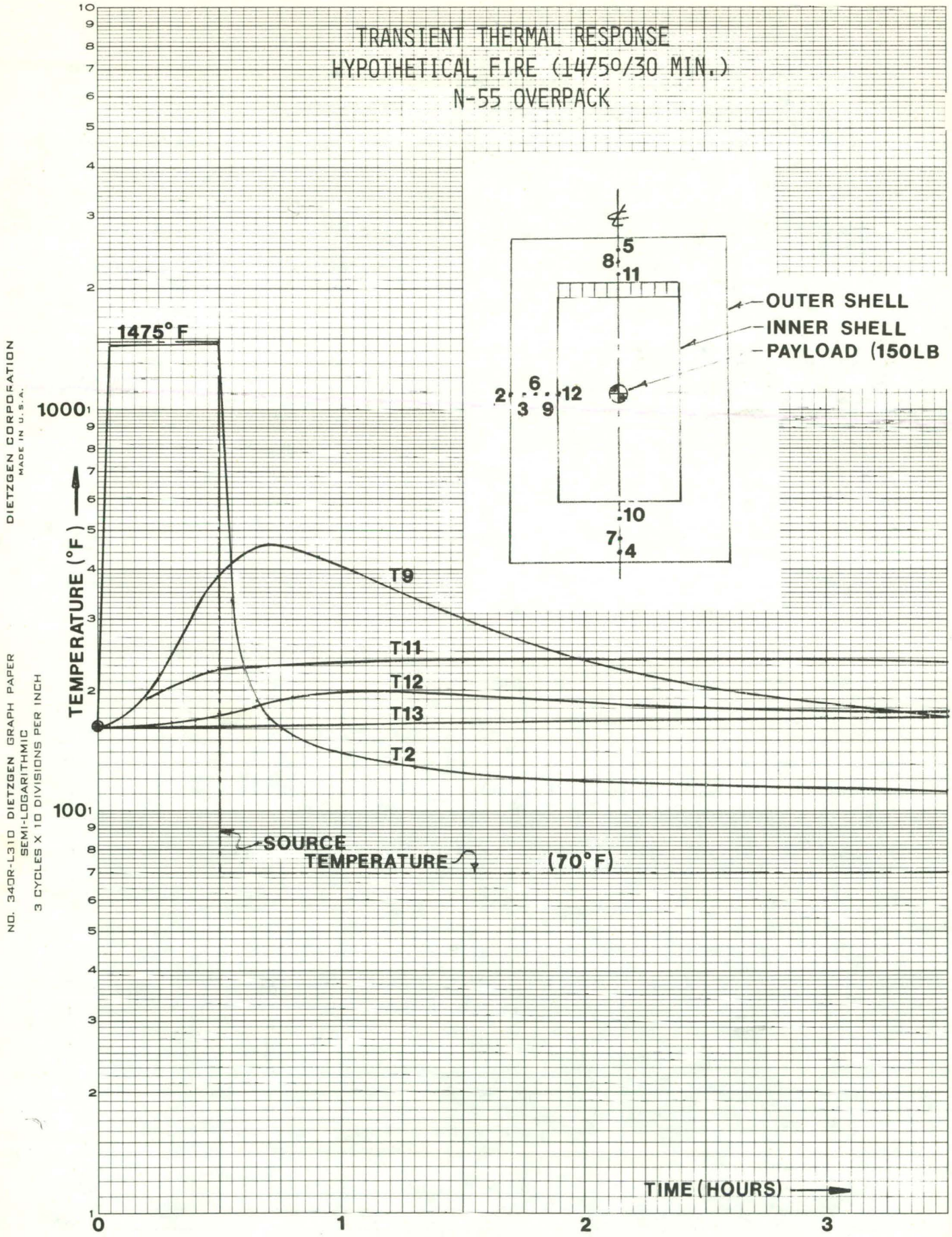

TRANSIENT THERMAL RESPONSE
HYPOTHETICAL FIRE (1475°/30 MIN.)
N-55 OVERPACK
1475° F
1000
100
TEMPERATURE (°F)
T9
T11
T12
T13
T2
SOURCE
TEMPERATURE
(70°F)
TIME (HOURS)
0
1
2
3
OUTER SHELL
INNER SHELL
PAYLOAD (150LB
5
8
11
2
6
12
3
9
10
7
4
DIETZGEN CORPORATION
MADE IN U.S.A.
NO. 340R-L310 DIETZGEN GRAPH PAPER
SEMI-LOGARITHMIC
3 CYCLES X 10 DIVISIONS PER INCH

FIGURE 4

TRANSPORTATION EFFICIENCIES USING N-55 OVERPACKS

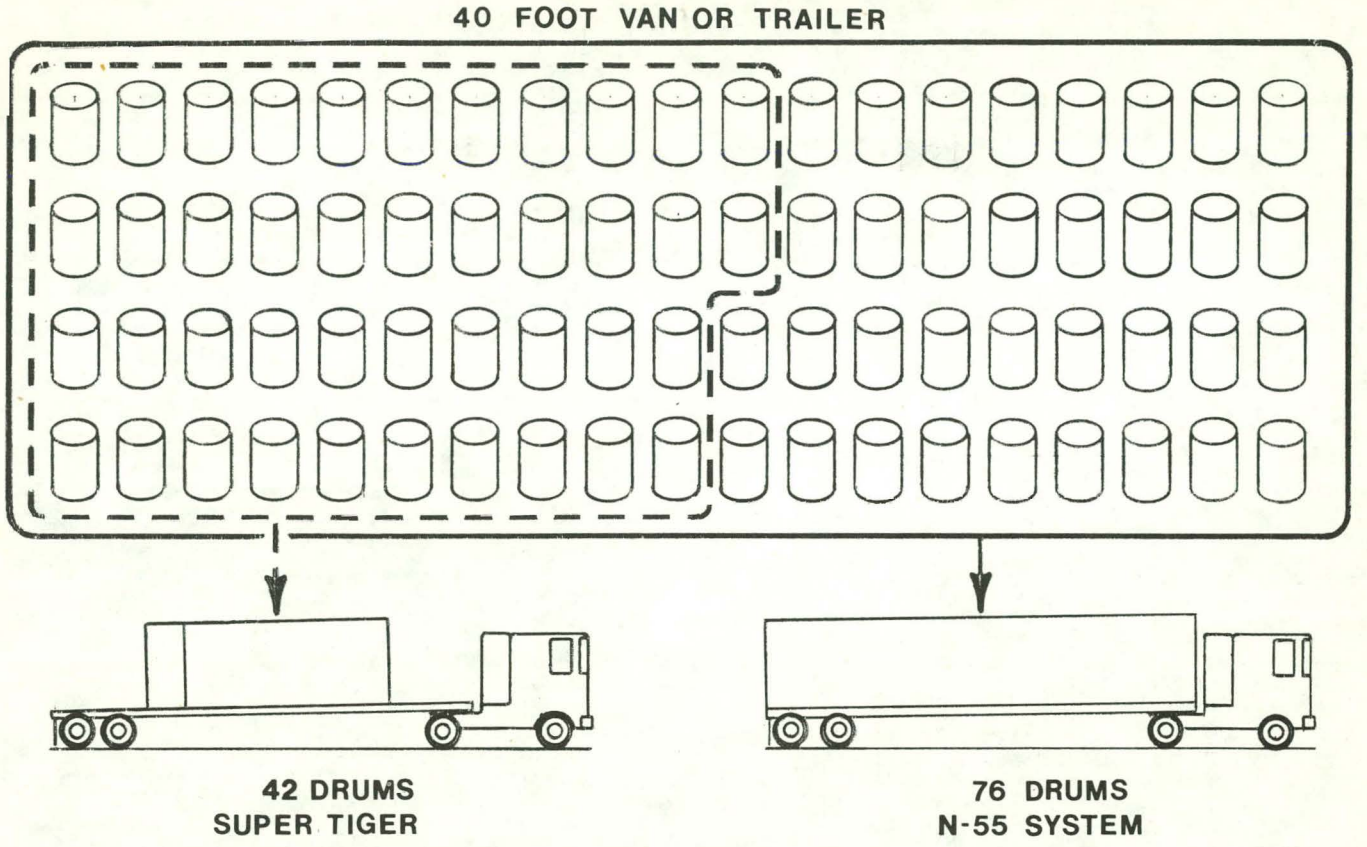
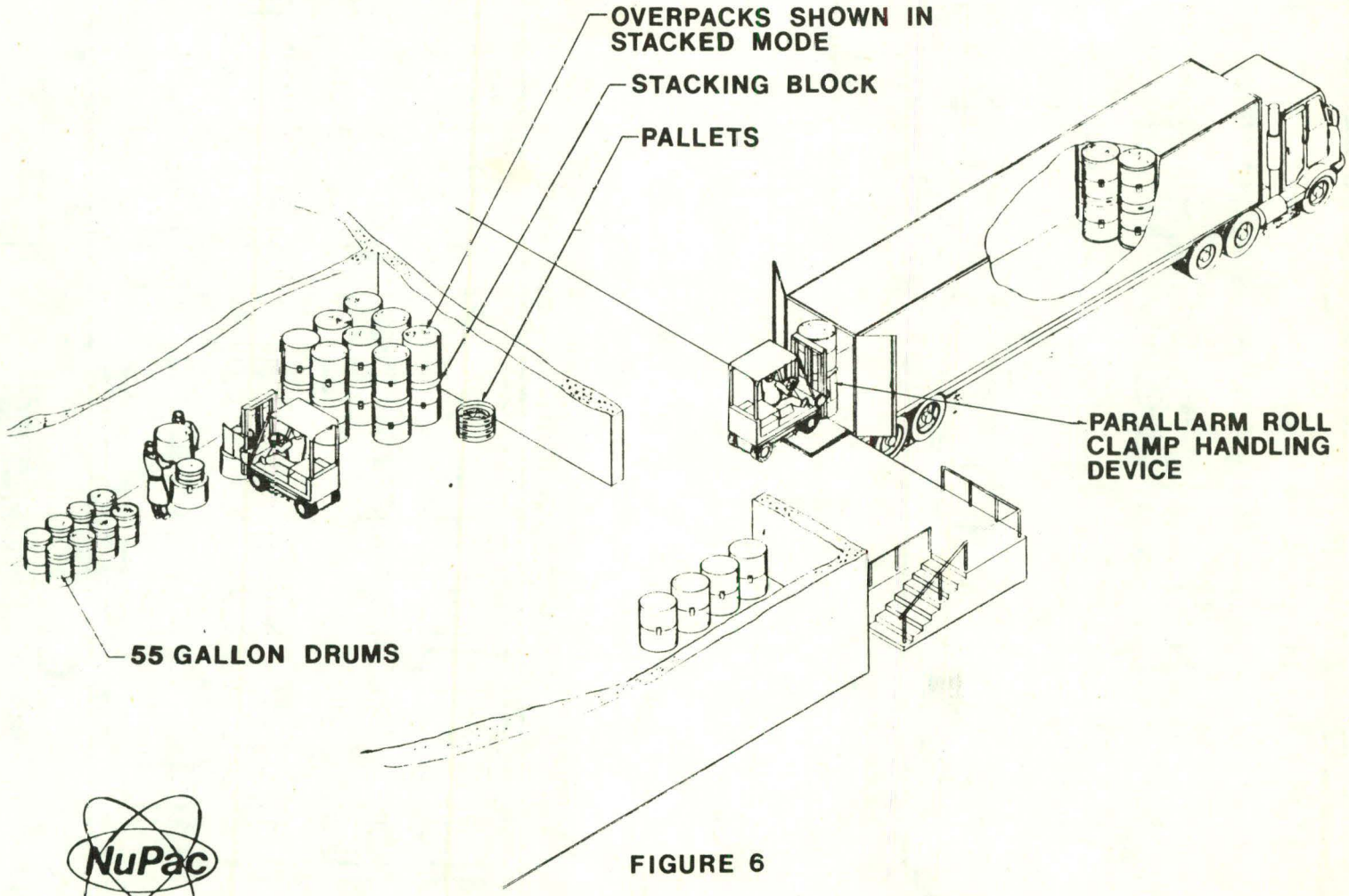


FIGURE 5

LOADING SEQUENCE
N-55 OVERPACKS



NuPac

FIGURE 6

are stacked two high and held in place with the aid of molded stacking pallets which can be reused for the return trip. The bottom of the overpack can accomodate a lightweight molded pallet configured to accept a fork truck loading operation.

The drawing shows the use of a lift truck with standard hydraulic paper roll clamps tached. With this material handling equipment, the operator can (1) remove the overpack lid and set it conveniently nearby, (2) pick up and load the 55 gallon drum, (3) replace the overpack lid, (4) stack the overpacks two high, and (5) load onto a semi-trailer van. This semi-remote operation can be accomplished by one man thus providing minimized exposure and cost.

V. Summary

Inherent in the design of the N-55 overpack is the ability to increase the various packaging and transportation efficiencies when transporting 55 gallon drums of radioactive material. Packaging efficiencies are demonstrated when comparing overpack weight to payload weight. Obviously, the less weight devoted to overpack, the more capacity remains for payload. This can be an important economic consideration when freight rates are factored against total cargo weight.

The amount of drums capable of being transported on a single load is also a prime economic consideration when comparing various transportation systems. It has been shown that transportation efficiencies as high as 100% can be obtained using individual drum overpacks rather than multiple drum overpacks. In other words, it is possible to double the number of drums per truckload in some instances.

This type of comparative analysis should be run whenever a choice exists between various transportation systems. To maximize packaging and transportation efficiencies is to get the most out of the transportation dollar and is in the best interest of the shipper, the public and the nuclear community.

Dup

CONCEPTUAL TRU WASTE CONTAINER DESIGN SPECIFICATIONS

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INTRODUCTION

Monsanto Research Corporation completed a program during FY-1977 at Mound Facility specifying Conceptual Acceptance and Design Criteria for defense transuranic waste packaging for Interim Storage and/or Terminal Isolation. A contractor questionnaire was used to gather pertinent data. Site visits were made to formulate an integrated contractor consensus; a packaging meeting was held to examine, discuss, and integrate packaging philosophies; and data collected from these activities and from Task Force meetings were consolidated to provide input to the Basic Application Checklist and Criteria Checklists. Conceptual Design Criteria have now been developed from an analysis and evaluation of the application data against federal regulations and interim/terminal storage constraints.

Background

The major objective of DOE Manual Chapter 0511 is responsible technical management of radioactive wastes. With the generation of radioactive waste in DOE operations, this responsibility spans the lifetimes of the radionuclides. Close examination of the entire waste management cycle clearly indicates that a totally acceptable waste package containment system is imperative if this objective is to be realized. As presently specified in DOE Manual Chapter 0511-044d(4), solid transuranic waste packaging and storage conditions shall be such that the packages can be readily retrieved in an intact, contamination-free condition for 20 yr.

The retrievable storage site for defense transuranic wastes at the Idaho National Engineering Laboratory (INEL) has been accepting waste since November 1970, and has stored this waste in an area designated the Transuranic Storage Area (TSA). The packaging and storage conditions for the waste stored at the TSA meet the requirements that the containers be readily retrievable in an intact, contamination-free condition for 20 yr.

Current planning of the Division of Waste Management, Production and Reprocessing (DWPR) indicates continuing use of the retrievable storage areas through FY-1987 with the New Mexico Waste Isolation Pilot Plant (WIPP) facility attaining full operational status in FY-1988. According to projections, WIPP will begin receiving transuranic wastes in FY-1983. This waste will be stored so that it can be monitored to evaluate the behavior of the waste types under

*Mound Facility is operated by Monsanto Research Corporation for the Department of Energy under Contract No. EY-76-C-04-0053.

these storage conditions. Projections indicate that the Pilot Plant phase will continue for 3 to 5 yr, after which, with retrieval demonstrated and experimentation successfully completed, the pilot plant will be converted to an operational repository for permanent disposal of wastes.

Data Collection and Analysis

A data collection and review phase involving contractor site plans, federal regulations, and a technical literature search was completed early in the program and provided the project team with all available published data.

A contractor questionnaire was transmitted to appropriate ERDA (DOE) Operations Offices and contractors by ERDA (DOE)/ALO in early February 1977, and information received on transuranic waste packaging from solicited contractors was consolidated, evaluated, and reported [1].

During April 1977, at the request of ERDA (DOE)/ALO, MRC personnel presented the contractor questionnaire data at the TRU Waste Form and Packaging Criteria Meeting at Albuquerque, New Mexico. These data represented current information directly affecting the WIPP Acceptance Criteria. MRC provided a mid-term program review at ERDA (DOE) Headquarters on May 9, 1977. This review addressed program progress to date and emphasized data derived from the contractor questionnaire. At that time, ERDA (DOE)/DWPR directed MRC to concentrate efforts to develop Packaging Acceptance Criteria for the Low-Level Transuranic Wastes, because the volume of low-level TRU waste generation for the period FY-1976 and FY-1976A was 184,144 ft^3 compared to the 4,856 ft^3 of intermediate-level TRU waste generated during the same period.

As previously reported [1], only 57.5% (107,542 ft^3) of the low-level TRU waste generated (187,144 ft^3) was shipped and stored at INEL. These wastes are packaged in containment systems satisfying both DOE Manual 0511 and Interim Storage (INEL) Criteria. A small portion, representing 1.6% of the low-level TRU waste generated (3.034 ft^3), is shipped to the Nevada Test Site for storage. The containment systems for these wastes stored retrievably are different from those used at INEL. The other portion of the waste generated (76,538 ft^3) is stored onsite by contractor generators. Onsite storage containment systems are of varied configurations, primarily because contractors developed container systems to fit their unique environmental storage requirements.

In any information-gathering effort, not all the information provided is complete and, therefore, each reporting contractor site was visited to ensure understanding of the questionnaire data, investigate significant program concerns, and obtain contractor consensus on TRU waste packaging. Throughout the site visits it was evident that varying levels of compliance with Manual Chapter 0511 exist. This is particularly true with reference to separation of combustible and noncombustible waste materials. Although the questionnaire data received reflect that only 21.6% of the low-level TRU waste generated during FY-1976 and FY-1976A was combustible, a reasonable doubt in the accuracy of this composition must be noted. An inaccuracy in the combustible waste volume could affect both the application of future incineration technology as well as the combustion-loading requirement of the WIPP facility during the pilot plant development phase.

An analysis of the information exchanges with the contractors dictates that both a drum configuration and a box geometry (preferably a modular concept) are needed. This analysis and mutual packaging consensus are based on the following contractor requirements and waste generation history:

1. Present Material Handling Systems
2. Current and Future Waste Processing Systems
3. Present Material Assay Systems
4. Available Modes of Transportation
5. 71% of the low-level TRU waste generated in FY-1976 and FY-1976A was packaged in box geometry

Cost of any new packaging system is extremely important to the contractors, and their request was that the new system be lower and/or comparable to the present packaging economics. This cost conservation is not only based on future generation of low-level TRU wastes at the contractor sites,

but also strongly influenced by known and planned decontamination and decommissioning projects at Mound, Rocky Flats, LASL, Atomics International, LBL, and Oak Ridge, where substantial increases in low-level TRU wastes are projected.

The contractor need for a standardized TRU containment system is immediate. If packaging of TRU wastes continues in nonstandard containment systems, it is probably safe to assume, based on the present WIPP Acceptance Criteria and supported by the Final Report of the Task Force on WIPP Criteria Interrelationships, that contractors, along with interim storage sites, will be required to repackage stored wastes prior to placing these wastes in a terminal isolation mode. The repackaging costs, however, would be less expensive if standardized containers are defined, proven, and available as soon as possible.

Following conversations with contractors generating transuranic waste, a meeting was held at Mound Facility on June 2, 1977, for the specific purpose of discussing proposed criteria for an acceptable TRU waste package for application in both interim storage and terminal isolation. Results of this meeting have been previously reported [1]. At the meeting, contractors and the WIPP representatives requested that the project team initiate a survey of container manufacturers to determine whether any commercially available packaging/containers are applicable for the shipment and storage of low-level TRU waste. Information has been received from manufacturers on container types, materials of construction, sizes, shapes, internal volume, weight, closures, DOT certification (if applicable), and unit cost. Performance data relating to materials of construction have also been received and evaluated.

To ensure that acceptable TRU packaging is developed for defense wastes that is consistent with packaging being developed for commercial wastes, representatives from ORNL Office of Waste Isolation (OWI) visited Mound Facility on May 4, 1977. At the meeting, packaging acceptance criteria, package/containment systems standardization, central procurement, and transportation of the wastes for interim and terminal storage were discussed. The OWI position, at that time, was that packaging criteria and TRU packages developed should be conservative and that cost of the packages to users/generators was not important. MRC and OWI agreed that the packaging acceptance criteria should be consistent for both defense and commercial TRU wastes, since both types of generators produce essentially the same types of waste. OWI representatives agreed that standardization of both container quality and design specifications would reduce the environmental risks.

At the request of ERDA (DOE)/ALO, Mound representatives attended the Task Force Meeting on WIPP Criteria at ERDA (DOE)/ALO on August 2, 1977. The specific role of Mound personnel was one of providing technical information on present packaging systems and their compatability to the WIPP Criteria. Information was supplied to the Task Force as requested during the meeting. Based on the combustibility and radiolysis requirements as written in the WIPP Acceptance Criteria, it was concluded that present TRU packaging does not meet the criteria for all types of wastes. In the final report of the Task Force on the WIPP Interrelationships, a recommendation was made that a standardized container system designed to optimize space, provide fire barrier, and limit gas generation during storage should be developed. The Acceptance Criteria for TRU Waste Packaging for Interim and/or Terminal Isolation developed through this program are directed toward accomplishing this recommendation.

CONCEPTUAL TRU WASTE CONTAINER DESIGN SPECIFICATIONS

More than 18 different packaging configurations are being used for interim storage of TRU wastes. Of these configurations only the DOT 17C (17H) steel drum is readily acceptable for direct shipment and isolation in the WIPP. The proliferation of waste packages currently in use is a direct result of efforts of the waste generators to package their unique TRU wastes into containers able to meet the 20-yr retrievability requirements under the differing environmental conditions of onsite storage. Completing the conceptual design for a family of standardized TRU waste containers will be difficult without established parameters with regard to waste form and interim storage environment. In the selection of materials for construction of a particular packaging system, it is important first to consider the characteristics of the system, giving special attention to extraneous factors that may influence decomposition. Since these factors would be peculiar to a particular system, it may be impractical to offer hard and fast rules that would cover all situations. As stated in one of the Sandia early rough drafts of TRU

Table 1 - BASIC APPLICATION CHECKLIST

Package Number	TRU Low-Level Waste Package Description	Volume ft^3/m^3	Approx. Unit Cost
1	55-gallon steel drum, DOT 17H, zinc dipped, polyethylene (PE) bag liner	7.42 0.21	$32.00
2	55-gallon steel drum, DOT 17H interior and exterior painted, PE bag liner	7.42 0.21	$34.00
3	55-gallon stainless steel drum, DOT 17H, PE bag liner	7.42 0.21	$135.00
4	55-gallon steel drum, DOT 17C, interior and exterior painted, PE bag liner	7.42 0.21	$20.00
5	55-gallon steel drum, DOT 17C interior and exterior painted, 90 mil high density polyethylene (HDPE) liner when lead shielded $60.00 additional	7.42 0.21	$60.00
6	55-gallon steel drum, DOT 17C zinc dipped, 90 mil HDPE liner	7.42 0.21	$60.00
7	30-gallon stainless steel drum, PE bag liner	4.01 0.11	$70.00
8	30-gallon steel drum, interior and exterior painted PE bag liner, 2 each inside concrete cask	8.02 0.22	$300.00
9	Corrugated metal pipe, zinc dipped, 2 1/2" diameter x 20' long	98 2.78	$250.00
10	55-gallon steel drum, zinc dipped, 90 mil HDPE liner, inside concrete culvert	7.42 0.21	$60.00
11	Welded steel box, 1/8" thick, PE bag liner, random sizes, will fit inside supertiger	Random	$1000.00
12	Fiberglass reinforced polyester (FRP) plywood box, 4'x4'x7', PE bag liner	112 3.17	$400.00 $800.00
13	FRP plywood box, random sizes, PE bag liner	8-1500 .23-43	$100.00 $1000.00
14	Steel bin, 4'x5'x6', 12-gauge, 55 gallon DOT 17H steel drums or 1/2" thick plywood box placed inside	120 3.40	$500.00
15	FRP plywood box, random, steel drums inside	Random	$2900.00
16	FRP plywood boxes, cold rolled steel boxes, inside concrete culvert	Random	$1320.00 $8000.00

Table 2 - BASIC APPLICATION CHECKLIST

Package Number	Waste Generator													Total Volume Per Waste Package Type
	RI	ARHCO	MRC	LASL	SRP	ORNL	ANL-E	LLL	LBL	BMI	AI	EG&G	ANL-W	
1								X	X					1430 ft^3 40.5 m^3
2									X	X	X			421 ft^3 11.9 m^3
3						X								1925 ft^3 54.5 m^3
4		X		X										14561 ft^3 412.4 m^3
5	X		X	X								X		23314 ft^3 660.3 m^3
6					X									3000 ft^3 85.0 m^3
7						X								1925 ft^3 54.5 m^3
8				X										800 ft^3 22.7 m^3
9				X										4500 ft^3 127.4 m^3
10					X									2000 ft^3 56.6 m^3
11								X						1500 ft^3 42.5 m^3
12	X		X											79700 ft^3 2257.2 m^3
13				X										2000 ft^3 56.6 m^3
14							X							4700 ft^3 133.1 m^3
15		X												43368 ft^3 1228.2 m^3
16					X									2000 ft^3 56.6 m^3
											Total Volume			187,144 ft^3 5,300 m^3

Waste Acceptance Criteria, "It (conceptual design criteria) is in no sense an attempt at unilateral rulemaking." However, through Mound's study of TRU waste packaging and resulting comparisons of the Basic Application Checklists (see Tables 1 and 2) to the transportation and isolation requirements, a set of conceptual design specifications was assembled.

Waste Container Definition

The box or drum, including any associated liner and/or shielding material, that immediately surrounds (and is considered to be an integral, disposable part of) the waste material.

Structural Design

The structural design of all low-level TRU waste containers must meet the requirements for Type A packages as outlined in 49CFR173.398b. Low-level TRU waste is any solid waste material, other than high-level waste, which is contaminated with long-lived alpha emitters to the extent that, under the provisions of DOE Manual chapter 0511, it is not suitable for surface burial, but which exhibits sufficiently low radiation levels ($\leq$500 mrem/hr) that it is amenable to handling by "contact" methods. This minimum structural design requirement shall be required for all TRU waste packages to assure safety to personnel during handling, loading, and unloading operations. During shipment, the Type A containers may be placed inside a reusable Type "B" overpack. The Type "B" container must meet more rigorous structural design requirements and tests than Type A containers to provide for maximum safety during shipment. Cost effective packaging and transportation of TRU waste materials will require the single use Type A packages to be relatively inexpensive but capable of meeting the requirements of contamination control from the time the containers are filled until they are backfilled inside the WIPP facility.

Design Life (Decomposition)

The design life of all TRU low-level waste containers for contamination-free retrieval shall be 10 yr minimum when stored in a noncorrosive atmosphere (pH 7-8), 60% relative humidity, and 100°F. The design life parameters may suggest a change in DOE Manual 0511 from 20-yr intact contamination-free retrievability to a 10-yr intact contamination-free retrievability concept. Life of the shipping container will start from the time the container is manufactured until backfilled in the WIPP. The 10-yr life is based upon the forecast that the WIPP will be fully operational for TRU waste containers in 1988. Life cycle of the container will include manufacturing, delivery, storage, transmittal into the WIPP, analysis, and backfilling. This life cycle should be approximately 5 yr; however, it could approach 10 yr because the backlog of interim stored wastes will be in direct competition with freshly packaged waste for isolation space in the Isolation Facility. All filled waste containers must be protected from environmental conditions that could significantly reduce the design life of the waste containers to less than 10 yr.

Materials of Construction

Materials of construction shall be based on design life and structural design requirements. Ferrous and nonferrous metals, plastics, reinforced plastics, fiberboard, corrugated fibers, wood, and concrete have been considered for container materials. All these materials can meet the requirements for hazardous materials packaging and are acceptable in the WIPP. Therefore, the choice of materials, or combinations thereof, can be made from the above group.

Difficulty of choice will be influenced by the waste form, design, economic interests, and final WIPP TRU Waste Acceptance Criteria.

Maximum Weight of Container and Contents

The weight of a single container filled to 98% capacity is limited to 25,000 lb (11,400 kg) ;ed on a contents density of 125 lb/ft^3 (2000 kg/m^3). This design weight is based on the ,000 lb (11,400 kg) maximum capacity of the WIPP low-level hoist cage.

Shape

The container family shall be modular, having a shape which will provide maximum packing efficiency. The cylindrical container has a packing efficiency of ~0.69 and the void space will be 31 ft^3 for every 100 ft^3 of waste in terminal isolation. The cylindrical container, 1 to 8 ft^3, is readily mass produced and available in metal, plastic, and fiberboard. However, because of the underground location for isolation of TRU wastes, emphasis must be placed on container shape with higher packing efficiencies for the waste materials.

Dimensions

Waste container dimensions shall be based on the criteria given in Table 3 to provide flexibility in mode of transportation.

Handling Appurtenances

All low-level TRU waste containers must be provided with cleats, offsets, or chimes which permit handling by fork lift.

Security Seal

The outside of each waste container must incorporate a feature such as a seal that is not readily breakable and that, while intact, will be evidence that the package has not been illicitly opened.

Cost

Current low-level waste packages which can meet the reqirements of DOE Manual Chapter 0511, WIPP, and DOT Type A have costs ranging from \$3.57/ft^3 (4x4x7 ft fiberglass reinforced polyester resin box) to \$18.19/ft^3 (DOT 17H, 55-gal, stainless steel drum) for the packaging materials. Cost per cubic foot of storage volume for the standardized container family should be toward the lower end of this range to be cost effective.

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Table 3 - MAXIMUM DIMENSIONS THAT MUST BE CONSIDERED FOR WASTE CONTAINER DESIGN

Dimension	ATMX Railcar	Super Tiger	Iso/Cargo Container	WIPP Hoist Cage
Inside Length	600 in. (5.24 m)	172 in. (4.37 m)	228 in. (5.79 m)	144 in. (3.66 m)
Inside Width	108 in. (2.74 m)	76 in. (1.93 m)	93 in. (2.36 m)	96 in. (2.44 m)
Inside Height	108 in. (2.74 m)	76 in. (1.93 m)	91.5 in. (2.32 m)	108 in. (2.74 m)
Door Width	Top Load	76 in. (1.93 m)	90 in. (2.29 m)	96 in. (2.44 m)
Door Height	Top Load	76 in. (1.93 m)	89.5 in. (2.27 m)	108 in. (2.74 m)

A SPENT FUEL SHIPPING AND STORAGE SYSTEM FOR ATUCHA NUCLEAR POWER STATION

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1. SUMMARY

Argentina's Atucha Nuclear Power Station is of the HWR type with pressure vessel.

Its natural UO_2-zircaloy fuel assemblies are bundles of 36 fuel rods, 620 cm in length and 11 cm in diameter. The fuel elements are irradiated up to 7,000 MWD/MT and stored in a conventional decaying pool adjacent to the reactor building.

A project was developed to design and build a shipping cask and a dry storage place for interim storage of spent Atucha's fuel.

This paper describes the cask and storage design, as well as the special container (basket) for the fuel elements.

The operation of the whole system is described, with emphasis on safety aspects, and a brief reference is made to heat transfer, corrosion and costs.

Safety analysis shows that even in accident conditions the radiation dose on workers in the area will be under the maximum allowed levels.

Cask and storage design allows for refrigeration by air convection without surpassing a temperature of 250°C in the hottest fuel rod.

Estimate unit costs for the storage system are under 0.1 mills/KWh.

2. INTRODUCTION

The Atucha Nuclear Power Station has a pressure vessel type reactor, refrigerated and moderated by heavy water. The electrical output of the power station is of 319 MW.

The reactor' core consists of 253 natural UO_2 fuel assemblies cladded in zircaloy 4. Each fuel assemblies is formed by a bundle of 37 rods. One of those rods is a structural supporting one, while the others are full of sintered UO_2 pellets.

The average burn-up of the equilibrium core is of 7,000 MWD/MT, and each fuel assembly spends 250 days in the core.

The fuel assemblies are 620 cm long and have a maximum diameter of 11 cm. Fuel rods have an outer diameter of 1 cm. Due to those structural features, the fuel assemblies must always be either hanging from its upper end or laying supported by its entire length.

The power station decaying pool has sufficient room for 5 cores. In august 1975 a project was initiated to build an interim fuel fuel shipping and storage system of 10 years full power operation capacity. November 1977 was the date fixed for starting the operation of this system.

This paper describes the state of development of the project in april 1976, when the authors' participation in it was discontinued and the project interrupted.

The utilization of foreign technology and suppliers would have rised construction costs and imposed a too long time for project completion. On account of this, all the design was made trying to maximize the use of local industrial experience and capacity.

3. SHIPPING CASK

There would be no other Atucha type power station build in Argentina, so it was considered reasonable selecting the power station site for its spent fuel storage. This greatly simplified the shipping cask design and construction standards, as this cask is only intended for controlled zone utilization. This had the obvious advantage of reducing construction time and costs.

Cask capacity was fixed by means of an optimization study, looking for the minimum cask weight compatible with such a capacity that minimized transport frequency. This minimal transport frequency was important because cask charging operations interfere with fuel handling pool normal operation. Optimal cask capacity was found to be of 7 fuel assemblies with a minimum decaying period of two years. This number also allows for an optimal storage design from the heat transfer and building points of view.

The fuel assemblies are charged in the cask, sealed in a container (basket) described in the next section. The cask is charged by its upper end, and discharged by the bottom. It also has draining holes for the water introduced during the charging operation. Inner and outer shells are made of carbon steel. Lead is used for shielding.

Basket cooling in the cask takes place by radiation to the inner wall, then by conduction to the surface and finally by convection to the environment. In normal conditions, forced cooling is not supposed to be necessary, but the cask has ducts for air circulation on the basket, and the shipping trailer is provided with an air blower for emergency cooling.

4. BASKET FOR FUEL ELEMENTS CONTAINMENT

The basket must provide the second containment for fission products formed in the fuel rods (Fig. 1). It is made of 7 alluminium tubes welded in centered exagonal array to an upper alluminium plate. Tubes are made of Kaiser 6061 T6 extruded alloy, and the upper plate is cast in Kaiser AS 106 alloy. A water collector connects the lower end of the tubes.

These tubes provide the necessary lateral support to the fuel assemblies which are hanging from a specially designed passing clevis, we called "hasp", put between the bars that link the fuel bundle to the fuel assembly-head. This "hasp" is resting on the upper plate.

The basket cover is made of the same alloy used for the plate, and fastened to it by means of bolts. It has a threaded hole where a hanging bar is screwed for lowering the basket in the storage pits. A tight seal is obtained by means of radiation resistant gasket.

There are two ducts for draining and drying the inside with an air current. Both ducts may be tightly sealed.

5. SPENT FUEL STORAGE

An evaluation was made of different underwater and dry storage concepts, and this dry storage design was selected. One of the reasons for this choice was the simplicity of the civil works. This implies shorter building times and a simpler technology. Other reasons were the much

less complex operation and maintenance, that improves the long term safety conditions and the economy of the system. Works performed elsewhere (1,2) supported this choise, and later researches (3) confirm the validity of our evaluation.

The storage facility is formed by a set of four independent modules with 108 silos each one (Fig. 2). Those modules would be built sequentialy according to the storage necessities. This type of modular design allows for future expansions or technical modifications, as for instance the storage of fuel assemblies in tightly sealed silos after a long decaying period that reduces heat generation to a level that may be handled by conduction of the ground.

Storage silos are lined with a carbon steel tube of 45 cm diameter with holes in the lower end. Pits are connected by a lower common plenum formed by two concrete horizontal slabs. There is an upper concrete slab for shielding, with holes for silo's openings that are covered by a concrete lid.

Basket's cooling is performed by air convection. The air enters through empty silos, flows through the plenum and goes up the pits circulating around the basket's tubes. The hot air is released through the labyrinth shaped gap between the pit's mouth and the lid.

The lower openings of the pits are covered with a wire mesh to contain the material used to fill the silo if eventually a basket falls and is broken.

The power station grounds are at 21 meters over the maximum recorded level of river Paraná. The minimum depth of ground water is 22 m and the plenum floor is at a depth of only 7 m. There are no other water streams that might flood the storage silos, and the rain water that might enter them may be easily extracted with pumps.

Each silo is equiped with ducts to measure the air temperature at different levels. Those temperatures and the exhaust air activity would be measured periodically. It is important to note that only non defective fuel would be stored.

6. SYSTEMS OPERATION

The cask is placed on a bracket of the fuel handling pool, where the water layer above it provides enough shielding. The uncovered basket is placed in the cask, and both are lowered to the pool bottom. The fuel elements are carried from the storage pool and hanged from a pole in the handling pool. The "hasp" is then put in place by means of a special tool. The pool building bridge-crane is then used to introduce the fuel elements in the basket, holding them with the manual handling tool attached to the crane hook.

Once the basket is full, cask and basket are raised again to the bracket, and both basket and cask are tightly covered. The cask is extracted from water and drained, the basket inside is dried, all ducts are closed and the cask is decontaminated if necessary.

The cask is carried to the storage facility by the transport vehicle, raised with a 40 tonnes capacity bridge-crane and placed vertically on the ground. Bolts that fasten the lower cover are extracted and the hanging bar is screwed to the basket cover. The basket is then raised a few centimeters with the crane and the hanging bar is locked to the upper cover of cask. The bridge-crane is then free to carry the cask and put it over one of the silos. A telescopic extension tool is added to the hanging bar, the basket is again raised a few centimeters to test all the hanging device and unlock it, and the basket is lowered into the silo. The hanging bar is unscrewed, the cask is taken away and the silo is covered.

7 HEAT TRANSFER - (4)

The spent fuel power generation was calculated with the expression from Untermeyer and Weils (5). For two years decaying time a value of 0.58 KW/fuel assembly was found. The radiation heat transfer was computed with a modified J.S. Watson code (6). The cooling by air convection in the silos was computed with a modification of the code RA3-AIRE (7).

The results were experimentally tested simulating the charged basket with a set of 7 tubes electrically heated. The outer six tubes contained one resistance each one, and the central one contained 36 resistances arranged as the fuel rods of Atucha's fuel bundle. This mock up was one meter long and its diameter was equal to the basket's one. Experimental runs were conducted in the laboratory and at outsides, at clear noon sun, without wind and at 25°C air temperature. Results showed that outsides conditions gave lower temperatures. Table I shows equilibrium temperatures in the cask.

Tab. I. Equilibrium Temperatures in Transport Simulation

	Power	Central Tube	Central Fuel Rod	Outer Shell
	KW/fe	°C	°C	°C
Experimental (in lab. conditions)	0.5	150	195	-
	0.7	193	252	95
Calculated	0.5	150	189 ± 6	
	0.7	193	235 ± 6	

Storage conditions were simulated placing the mock up vertically in a pit whose geometry was equivalent to that of the silos. Table 2 shows experimental and calculated values for this device at a power equivalent to 0.7 KW/FA and 25 °C of inlet air temperature.

Tab. 2. Simulated storage Temperature

	Central Tube	Central Fuel Rod	Exhaust Air
	°C	°C	°C
Experimental	85 ± 4	122 ± 1	52 ± 1.5
Calculated	115 ± 0	180 ± 8	65

It is seen that the code used gives higher temperatures than those measured. This code computes the data shown in Table 3 for real storage conditions.

Table 3. Calculated Storage Temperatures

	Central Tube	Central Fuel Rod
	°C	°C
0.7	220.0	256.0 ± 5.0
Power KW/FE		
0.5	179.0	213.5 ± 5.5

It is supposed that one of the principal reasons for the difference between calculated and measured values is due to the uncertainty of friction magnitude in the silo.

8. SAFETY EVALUATIONS

8.1. Cask

As cask will be used only in controlled area, the operation code will stablish adequate standards to avoid collision risks. An emergency forced draft cooling system is provided for a possible too long stop during transport but, as it was said previously, computed and measured values show that fuel rods would never reach temperatures dangerous for cladding integrity. An accidental fall during cask handling with the bridge-crane would not release an activity greater than that described in section 8.3.

8.2. Storage, Normal Conditions

If we accept that 10% of the fission gases are released from the fuel pellets (8), and we make the pessimistic assumption that 10% of the fuel rods have such a defective cladding that the whole of its gas is lost, we shall have an overpressure of 1 at in a recently stored basket. Supposing the whole capacity of storage is filled in the above conditions (3,000 fuel assemblies)all the radioactive gas may steadily flow out of the baskets in one year without exceeding a dose of 0.25 mR/hour for a worker placed at an average distance of 100 meters. The design standard for the basket was established at 1/4 of that leaking rate, equivalent to a daily drop of 0.07 % of initial pressure. This corresponds to a flow rate of about 3 liters/hour for an overpressure of 1 at.

8.3 Storage, Accident Conditions

The maximum design accident is the fall of a basket in a silo during discharge operation. If this happens, the basket flange will strike the upper end of the steel tube where it normally stands. The kinetic energy will be partially absorbed by the alluminium distortion, and the rest would possibly break the fuel assembly linkage rods. (These linkage rods are so designed to break under an abnormal pull during core discharging operations, to avoid damage to the fuel rods containing UO_2 pellets). If the linkage rods break, the fuel bundle will drop and strike the bottom of the basket. This may break the fuel cladding and the basket bottom, but even if the basket bottom stands, there will probably be a loss of tightness due to flange distortion with a radioactive gas flow to the environment.

The following assumptions were made to calculate the dose received by a person close to the silo :

Kr 85 is the only radioactive gas after two years decaying time.
Activity of Kr 85 = 259 Ci/fuel assembly.
Free Kr 85 in fuel rods = 10 % of total (8)
Fuel rods broken by stroke = 10 %
The activity released to the atmosphere will amount then to :

$$259 \frac{Ci}{FA} \times 0.1 \frac{\text{free } Kr^{85}}{FA} \times 0.1 \frac{\text{broken rods}}{FA} \times 7 \text{ FA} = 18.13 \text{ Ci}$$

We suppose that all this Kr 85 is instantly dispersed in uniform concentration in an hemisphere of 10 meters diameter. Finally, we suppose that the radiation received by a person standing in that hemisphere is equivalent to the radiation produced by an infinite atmosphere with the same Kr 85 of the hemisphere. A person staying 5 minutes in those conditions would receive a dose of 1.2 Rads in his skin, and a wholebody dose of 9.6 mRads.

It is important to note that, in real conditions, the cask remains over the silo, shielding during some time the Kr atmosphere produced and giving people some extra time to leave the place without receiving all that dose.

8.4 Zircaloy Corrosion

All the bibliographic information collected at that moment showed that there should be no problems of zircaloy corrosion in the temperature and humidity conditions to be found in the storage. On the other hand, a possible temperature raise over the calculated limits that might increase the oxidation rate of zircaloy, would deplete the oxigen content of the basket, lowering the oxigen partial pressure and the oxidation reaction rate. A Canadian paper published later (3) shows values that support our evaluations.

9. STATE OF PROGRESS OF THE PROJECT

At the moment it was discontinued, the project had arrived to the civil works bidding stage. The possibility of inserting fuel assemblies in the basket tubes had been tested in the pool using first a dummy fuel assemblies and then an irradiated one. Prototypes of the basket and the "hasps" were under construction to make a general test of the charging operation.

Detailed design was being made and methods and standards were being studied for cask construction.

Irradiation tests were being performed with gasket materials and the safety analysis was finished.

10. ECONOMIC ASPECTS

Costs estimated for the first module of civil works, the first lot of baskets to operate this module and the shipping system, amounted to 924,000 US dollars of 1975 value. The following three modules would cost 254,000 US dollars.

The economic analysis shows that the total cost, present worthed with 10% interest rate in 10 years would be of 1,743,000 US dollars. The incidence of this storage cost in the electricity cost was estimated in 0.08 mills/KwH for a power station capacity factor of 0.80, value that corresponds to about 4 US dollars/Kg U stored.

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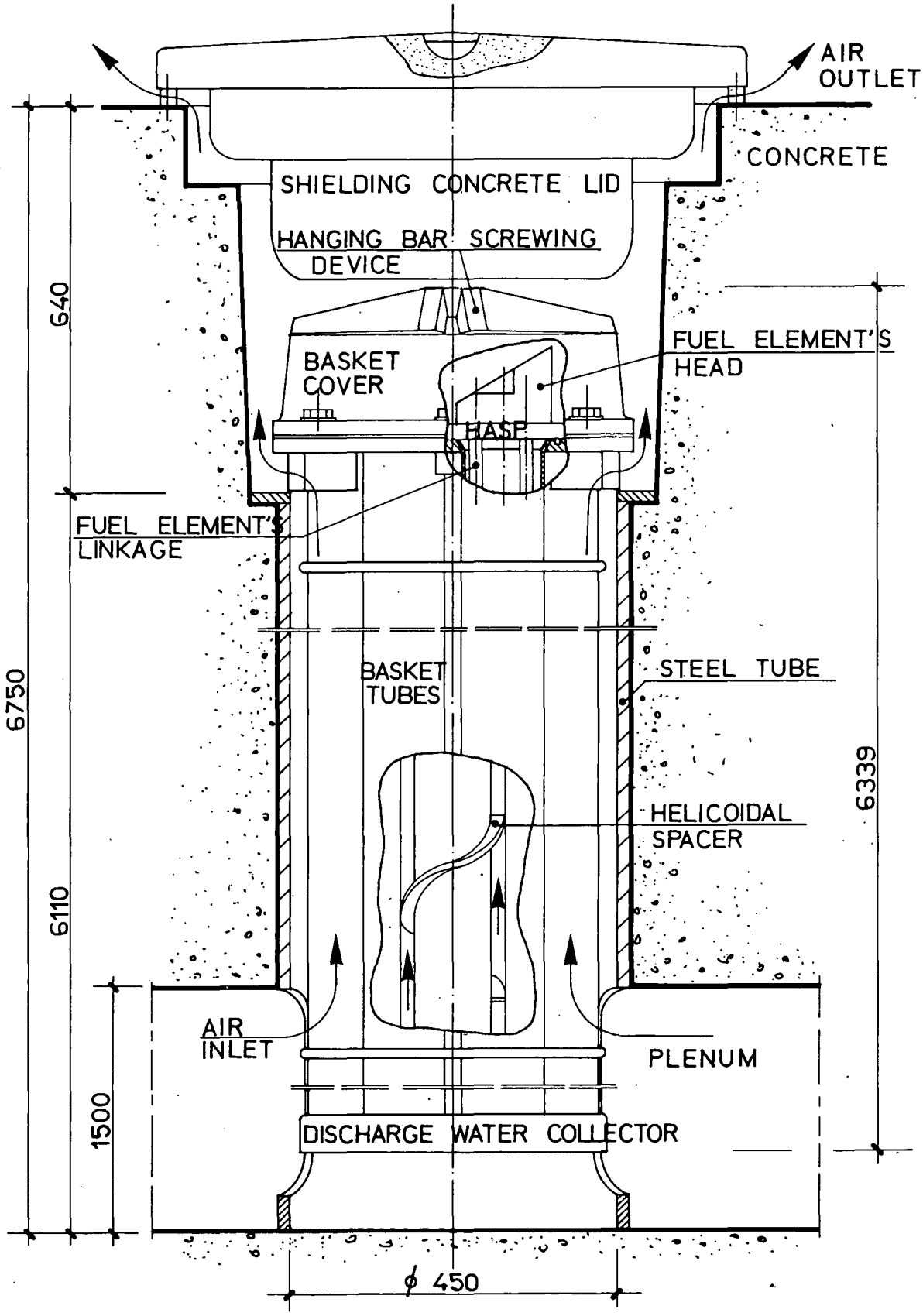
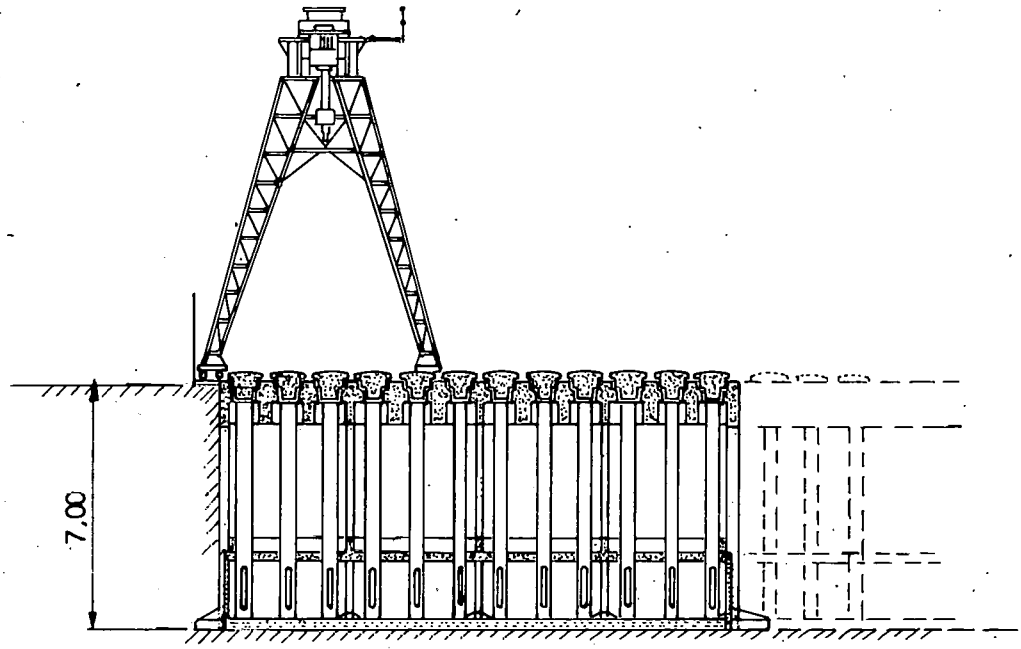


FIG. 1 STORAGE BASKET AND WELL



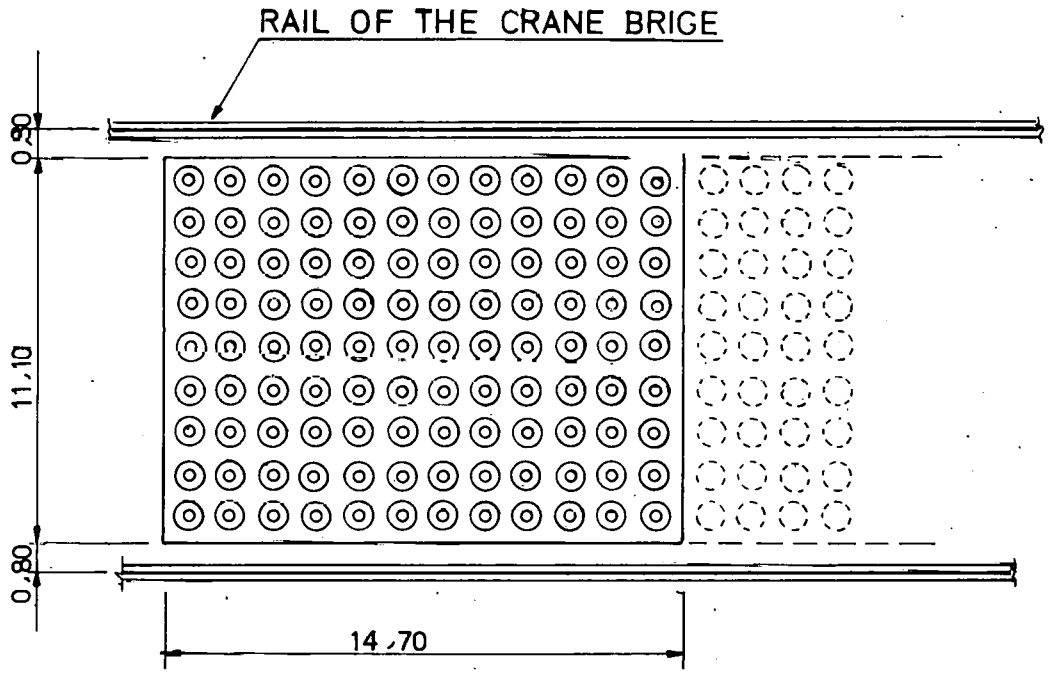


FIG. 2 SECTION AND PLANT VIEWS OF ONE STORAGE MODULE

STRUCTURAL

EVALUATION

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EMPIRICAL EQUATIONS FOR PUNCTURE ANALYSIS OF LEAD-SHIELDED SPENT FUEL SHIPPING CASKS

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SUMMARY

In-depth review and analysis of the existing experimental puncture data of unlaminated steel plates and lead-shielded prismatic and cylindrical steel casks are made. Based on these test data, empirical equations applicable to puncture analysis and designs of steel plate and shell components of spent fuel casks are developed in terms of nondimentional incipient puncture energy, shell thickness and curvature parameters.

In addition, the present study also examines the static puncture force behavior and puncture failure mechanism of unlaminated mild steel plates. This results in shedding some additional light on the puncture failure mechanism and better understanding of the puncture force behavior. Certain possible distortions in model-predicted prototype cask puncture responses (through use of the exact replica scale model testing techniques) are also discussed and a correctional measure presented.

1. INTRODUCTION

Title 10, Part 71 of the Code of Federal Regulations sets forth the requirements governing the shipment and performance of shipping casks under accident conditions. Because of the requirement of subjecting the cask to the accident fire test following the puncture test, it is a good design practice that the outer steel shell layer of a lead shielding cask not be punctured; otherwise, some loss of the lead shielding in the puncture region will result under the subsequent fire condition and its effect on package safety should be evaluated.

Earlier systematic experimental studies of the incipient puncture behavior of lead-backed carbon and stainless steel plate and shell assemblies were made at Oak Ridge National Laboratories (ORNL) (Refs. 1 and 2). In these studies, puncture test data were mainly generated by simplified scale model drop tests supplemented by a few prototype and obsolete cask tests. These studies have produced the widely used Nelms empirical plate puncture equation for lead-backed prismatic cask. Correlation between the equation-predicted results and obsolete cylindrical cask test results (Ref. 5) shows that the Nelms empirical puncture equation over-predicted the test results by about 30% and thus, use of this equation in cylindrical cask design may result in nonconservative design.

More recently, Sakamoto, et al. (Ref. 3) conducted a series of scale model tests aiming at studying the effects of cask shell diameter and rounded punch edge radius on incipient puncture energy behavior. The model casks used in the tests

were scaled-down versions of a realistic 80 ton prototype cylindrical cask of steel/lead composite shell construction. Static, impact machine and drop puncture tests were performed. An empirical puncture energy equation which, in general, predicts stronger cask puncture resistance than that of Nelms has been developed.

A closely related puncture study of unlaminated square mild steel plates under both static and projectile impact loadings can be found in Ref. 4. The plates were clamped around edges and penetrated statically and dynamically by unrounded edge punches of various punch materials (tool steel, mild steel, etc.). The projectile puncture impact velocities were mostly in the range of 100 ~200 ft/sec, which are substantially higher than the regulatory cask drop puncture impact velocity of 14.67 ft/sec.

The present study is aimed at developing appropriate puncture equations for cask component analysis. In particular, the study has been motivated by the necessity of developing a sound basis for regulatory review of cask puncture designs. A general energy functional form suitable for studying the incipient puncture energy behavior of lead-shielded cask shells is given in Section 2. The static puncture test data of unbacked mild steel plates given in Ref. 4 are analyzed and empirical puncture equations developed in Section 3. The prismatic and cylindrical shell puncture equations are developed in Section 4. Applications of the developed empirical equations to cask puncture analysis and certain distortions inherent in scale model-predicted cask puncture results are discussed in Section 5.

2. GENERAL CONSIDERATIONS

In order to develop a set of design curves for shipping cask shells subjected to the regulatory puncture drop impact loading, both analytical and scale model testing techniques can be used to generate the required data. In the use of scale models, the tests should be guided by similitude principles (e.g., see Refs. 6 and 7) to insure technically sound and valid test results. A detailed development of similitude principles and scaling laws for similar materials applicable to the present puncture problems can be found in Refs. 6 and 7, and thus, will not be given here.

A measure of puncture resistance for a lead-backed cask shell assembly subjected to a drop puncture loading is the puncture energy absorbing capacity of the assembly up to the initiation of shell puncture. To study the incipient puncture energy behavior, consider a typical three-layered steel-lead-steel composite cask shell model of outer diameter D, length ℓ, and outer-, lead-, and inner-layer thickness of t, t_{pb}, t_i, respectively. The cask shell assembly with the total cask weight of W, is assumed subjected to a symmetric dynamic drop puncture loading with the drop height h of the regulatory drop height of 40 inches. The punch is a cylindrical mild steel bar of diameter d, edge radius r and length ℓ_1. Let σ_u denote the static ultimate tensile strength of the outer layer steel, and $\eta = \lambda \sqrt{(h/40)}$ the relative strain rate parameter with respect to that of the prototype cask/punch system under 40 inch drop, where $\lambda = .6"/d$ is the punch scaling factor.

The main concern here is to find the incipient puncture energy absorbing capacity E of the cask shell as a function of certain important parameters which, varying in degrees, are identified as d, t, D, r, ℓ, η, and σ_u; the other parameters such as punch strength, ℓ_1/d, t_{pb}/d, t_i/d, ℓ/d, etc. may also affect shell puncture energy absorbing capacity. However, such a parametric study is beyond the scope of the present study. Assuming that E is directly (linearly) proportional to σ_u[1], the incipient puncture energy $E \simeq W(h + \delta)$ can be expressed in its nondimensional form as

[1] This was shown in Ref. 4 to be true for the unbacked carbon steel plate puncture case and in Ref. 2 to give a good puncture energy approximation to both lead-backed, carbon and stainless steel plate puncture cases. More accurately, E is proportional to σ_u^m with m perhaps somewhat smaller than 1 for the lead-backed case.

$$\bar{E} = E/\sigma_u t^3 = \bar{f}\ (x, y, z, \eta, \bar{\ell}) \qquad (1a)$$

where δ = the maximum cask C.G. displacement after cask/punch initial contact, $x = d/t$, $y = d/D$, $z = r/d$ and $\bar{\ell} = \ell/D$.

For a rectangular plate of width b and length ℓ

$$\bar{E} = \bar{g}\ (x, z, \eta, k, \bar{\ell}) = \tilde{g}\ (x, z, \eta, c, \bar{\ell}) \qquad (1b)$$

where $k = b/d$ and $\bar{\ell} = \ell/b$

3. STATIC PUNCTURE OF MILD STEEL PLATE

Static and dynamic puncture tests of unlaminated clamped-edge mild steel square plate penetrated by moving unrounded edge steel cylindrical bar punches (projectiles) were performed at Stanford Research Institute and reported in Ref. 4.

The static test data obtained by using tool steel punches are plotted in Fig. 1. From these test data there is obtained

$$\bar{E}_s = 0.5\, c^{0.54}\, x^{1.11} \quad (z = 0,\ 2 \le x \le 20,\ 15 \le c \le 150,\ \bar{\ell} = 1) \qquad (2a)^2$$

$$= 0.5\, k^{0.54}\, x^{1.65} \quad (8 \le k \le 16) \qquad (2b)^2$$

where $c = b/t$, $\bar{\ell} = \ell/b$, $k = b/d$, $x = d/t$ and $\bar{E}_s = E_s/\sigma_u t^3$ with subscripts "s" standing for "static."

It is seen that $\bar{E}_s$ is a rather strong function of plate width parameter c (or k). For example, by doubling the k-value from 8 to 16, $\bar{E}_s$ increases by 45%!

The static test results of ultimate punch force F_u and the corresponding approximate (upper bound) average punch force $F_{av} = (E/u)_u$[3] are shown in Fig. 2 as functions of x, where u is the plate indentation at the incipient puncture state. From these test data, the following empirical equations are obtained:

$$F_u = 110\, d^{0.89}\, t^{1.11} \text{ (kips)}$$

$$1 \le x \le 20 \qquad (3a,b)$$

$$F_{av} = 35\, d^{1.086}\, t^{0.914}\, e^{1/x} \text{ (kips)}$$

Hence

$$F_u/F_{av} = 3.14/\ (x^{0.196}\, e^{1/x}) \simeq 1.8 \qquad (3c)$$

As x decreases from x = 4 down, the F_{av}*, x - curve in Figure 2 is seen to curve-up, which is an indication of appreciable energy being dissipated in the punch due to inelastic punch deformation.

Figure 3 shows the variation of ultimate shear-ultimate tensile strength ratio μ (calculated from the test data in Ref. 4) as a function of x = d/t for various values

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[2] For use of these equations in cask component analysis, see discussions given in Section 6.

[3] The force is an upper bound to the exact average force because contribution of the punch shortening on total deflection and thus, total energy has been neglected.

of c = b/t. It is seen that for a fixed c-value, μ decrease linearly with increase in x while μ increases with increase in c for a fixed value of x. However, within the range of x and c shown, the μ-values lie in the vicinity of 0.6 within ± 10% errors. This indicates that unbacked-plate puncture is of the shear type failure (fracture).

4. PUNCTURE RESISTANCE OF LEAD-SHIELDED STEEL CASKS

The pertinent drop puncture test data can be found in Refs. 1-3 for the prismatic cask case and in Refs. 2, 3 and 5 for the cylindrical cask case. The punches used in these tests were all sharp-edged mild steel cylindrical ones except for those in Ref. 3, which were rounded edge punches with various edge radii. Reference 3 also contains static and impact machine test results. Because the test impact velosities and boundary conditions were not reported for the machine impact test case in Ref. 3, and also the solidly lead-backed cylindrical cask models used in the tests of Ref. 2 deviate too much from realistic casks,[4] the corresponding test results will not be used in the following development of puncture energy equations.

In these tests, drop height in some cases deviated considerably from the regulatory drop height of 40 inches (at which the initial impact velocity $v = v_o = 14.67$ ft/sec). For a "light" structure (i.e., a structure with fundamental period of elastic oscillation being relatively small compared with the punch pulse characteristic or rise time), the effect of drop height (or initial velocity) on the puncture behavior, according to Lae 3 of Ref. 7 is insignificant. It is somewhat doubtful that this conclusion can be applied to the production casks or cask models used in the puncutre tests of Refs. 3 and 5, which appear to be more appropriately classified as "heavy" structures. Therefore, the dynamic empirical puncture equations developed below, which do not contain velocity (or drop height) parameter (due to insufficient data to incorporate this parameter in the equation) should be used only by taking an appropriate measure to account for the uncertainties.

Plate (Prismatic cask) Puncture Case

Figure 4 shows the static plate puncture tests results of Ref. 3 and correlated curves given by the equation

$$\bar{E}_s = .70\ (1 + 0.87z - 7.60z^2)e^{2/x}\ x^{1.82}\ (1 + 0.856z + 1.513z^2) \qquad (4a)$$

$$(0.0246 \leq z \leq 0.148;\ 2.5 \leq x \leq 7,\ k = \text{const.})$$

where z = r/d is the punch edge radius parameter. Equation (4a) was obtained by using the least-square curve fitting technique on the test data. From Eq. (4a), the effect of rounded regulatory punch edge (with edge radius-punch diameter ratio of 1/24) on incipient puncture energy can be approximately estimate by

$$E_{s(z = 1/24)}/E_{s(z = 0)} = 1.024\ x^{0.07} \qquad (4b)$$

The drop puncture test data of Refs. 1 and 2 for carbon and stainless steel plate cases in which the measured ductility limit ε_u greater than 0.4, are plotted in Fig. 5. It is seen that the plate puncture energy absorbing capacity decreases with decrease in scaling factor λ. The major cause for this phenomenon is believed to be due to material strain rate sensitivity effects although assumption of puncture

[4] A preliminary data analysis shows that the puncture resistance capacity of a steel shell backed up by a solid lead core is appreciably stronger than that backed up by a hollow lead shell layer.

energy E being linearly proportional to σ_u may also be partly responsible for this distortion. By factoring Eq. (4b) into the empirical equation for the prototype cask case ($\lambda = 1$), i.e., $E \simeq 2.3\ x^{1.53}$, there is obtained

$$\bar{E} \simeq 2.35\ x^{1.6} \quad (\lambda = 1;\ z = 1/24) \qquad (5)^2$$

$$(2 \leq x \leq 20,\ 8 \leq k \leq 13 \text{ and } \bar{\ell} = 1 \sim 2)$$

which turns out to be almost identical to Nelms' equation (Ref. 2) established for the unrounded punch edge case.

Shell (Cylindrical Cask) Puncture

It is somewhat questionable that the prismatic cask test results are representative of the limiting (zero curvature) case of cylindrical cask test results. This is because the limiting case is equivalent to the puncture case of infinitely wide plate. Hence, in the following development of empirical puncture equations, Eq. (4a) or (5) will not be used as the limiting shell puncture case.

The static shell puncture test results of Ref. 3 are plotted in Fig. 6. These test data can be closely fitted into a set of curves represented by the equation

$$\bar{E}_s = 1.38\ (1 + y + 46y^2)e^{2/x}\ x^{1.3} \quad (z = r/d = 1/24;\ y = d/D) \qquad (6)$$

as shown in Fig. 6. Because of limited test data points, the range of applicability of Eq. (6) is necessarily confined to the vicinity where data points exist. The energy results predicted by Eq. (6) are seen to correlate very well (within 10% error) with the experimental results except for the test data point corresponding to $x = 1.514$, $y = 0.0522$.

To obtain the corresponding dynamic puncture equation, the dynamic puncture factor is first determined from Eq. (6) and those three drop puncture test data of Ref. 3 at $y = 0.1$ (corresponding to points 1-3 in Fig. 7). Before this is made, the $\bar{E}$-values corresponding to the drop puncture test cases in Ref. 3 are first extrapolated back to the baseline case of $\eta = 1$ and $h = 40$ inches by using the correction factor $[1.3/(1 + 0.3\ \eta^{0.24})]\ [(1 + u/h)/(1 + u/40)]$; here the first bracketed quantity approximately corrects material strain rate effects while the second one corrects less accurate energy calculation results by the authors of Ref. 3 (i.e., use of $E = Wh$ instead of $E = w(h + u)$). This results in reduction of E - values corresponding to data points 1 - 3 in Fig. 7 less than 10% and point 4 by 14%! From Eq. (6) and these corrected test results, the dynamic puncture factor of 1.3 and thus, the following dynamic puncture equation is obtained:

$$\bar{E} = 1.3\ \bar{E}_s = 1.8\ (1 + y + 46y^2)e^{2/x}\ x^{1.3} \qquad (7)^2$$

$$(30y \leq x \leq 12;\ 0.067 \leq y \leq 0.2;\ z = 1/24;\ \bar{\ell} = 2 \sim 3)$$

It should be noted that the dynamic puncture factor of 1.3 applies only to the static and dynamic test results of Ref. 3. That is, if the conditions from which static test results are obtained deviate considerably from those of Ref. 3, this factor may no longer be 1.3.

Figure 7 shows the adjusted drop puncture test results of cask models given in Ref. 3 and of obsolete casks reported in Refs. 2 and 5 and corresponding results calculated from Eq. (7). The original obsolete cask test results (points A-D) with the corrected energy values (calculated from the expression $E = W(h + u)$ and adjusted to the incipient puncture energy values through use of Tabor's lead penetration equation given in Ref. 8) have been extrapolated through use of Eq. (4b) from $z = 0$

to the regulatory punch case of z = 1/24. Excellent correlation between the experimental and empirical (equation-predicted) results is seen within the ranges of parameters given in Eq. (7). However, additional full-scale cask or larger reduced-scale model cask tests of drop height of 40 inches are desirable to supplement the rather limited existing data so as to improve and extend the range of applicability of Eq. (7). This is because the test data points 1-3 and A-D in Fig. 7 were obtained at drop heights of 97.81, 67, 6, 24 and 18 inches, respectively, which deviate considerably from 40 inches required by the regulation and scaling laws. Although adjustment on the incipient puncture energy values have been made here to account for strain rate and potential energy distortion effects, this may not be enough because the extent and direction of drop height distortion effects on puncture behavior are not well understood.[5]

Also shown in Fig. 7 is the $\bar{E}$, x-curve given by the Nelms empirical plate puncture design equation:

$$t = (W/\sigma_u)^{0.7} \text{ or } \bar{E} = 2.3\, x^{1.6} \quad (d = 6")$$

It is seen that the Nelms puncture equation may over- or under-predict the shell puncture resistance capacity depending on the values of x and y.

6 PUNCTURE DESIGN EQUATIONS AND SOME REMARKS ON-SCALE MODEL TESTS

As discussed previously, because of existence of uncertainties in the puncture test data from which puncture equations (5) and (7) have been developed, use of these equations, in addition to customary design practice of applying appropriate safety factor, reduction of cask puncture energy absorbing capacity by at least 20% should be made in cask puncture design. Furthermore, in using these equation one should bear in mind that these equations were developed on the basis of the test data obtained at room temperature for initially undamaged and unstressed, single-layered steel plates or shells with the measured ductility limit of $\varepsilon \geq 0.4$. Consequently, without some additional justifications or modifications, direct use of these equations in cask shell design should not be made if the cask initial conditions deviate appreciably from those mentioned above. The appropriate empirical equations for various cask component puncture cases are summarized below.

(a) Lead-Backed Cylindrical Cask Shell Puncture Case - The appropriate incipient puncture energy equation is obtained by multiplying the right hand side of Eq. (7) with 0.8. By setting d = 6", E = W(40 + δ) where δ is the shell indentation plus punch shortening at incipient shell puncture, the resulting equation can be solved for W/σ_u to yield

$$W/\sigma_u = 0.37\, J(D)\, e^{t/3}\, t^{1.68}/(1 + 0.1t^{0.87}); \quad J(D) = 1 + 6/D + (41/D)^2 \qquad (8a)$$

$$(0.5" \leq t \leq D/30;\ 45" \leq D \leq 90;\ \ell = 2 \sim 3D)$$

where the following approximate deflection expression (deduced from the test data for the lead-backed, stainless steel plate case in Refs. 1 and 2) have been used:

$$\delta \simeq 4t^{0.87} \text{ (inches)} \qquad (8b)$$

The $W/\sigma_u J$, t-relationships are shown in Fig. 8.

(b) Lead-Backed Prismatic Cask Shell Puncture Case - In a similar manner as in the preceding case by using Eqs. (5), (8b) and E = W(40" + δ) there is obtained

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[5] The limited experimental study of velocity effects on plate puncture behavior reported in Ref. 1 is believed to be inadequate to be used to conclude that such effects are not important.

$$W/\sigma_u \simeq 0.8\, t^{1.4}/(1 + 0.1\, t^{0.87}) \quad (0.3" \leq t \leq 3",\ 48" \leq b \leq 80",\ \ell = 1 \sim 2b) \quad (9)$$

which is also plotted in Fig. 8.

(c) Unbacked Square Steel Plate Puncture Case - In view of lacking of an appropriate dynamic puncture energy equation for this case, the static puncture energy equation, Eq. (2) can be used. Thus,

$$E/\sigma_u = 3.65\, b^{0.54} t^{1.35} \simeq (W/\sigma_u)(40" + \delta) \quad (10a)$$

for $0.3" \leq t \leq 3"$, $45" \leq b \leq 90"$ and $\ell = b$. The plate identation δ at the incipient plate puncture state can be estimated from the following expression deduced from the static puncture test results of Ref. 4:

$$\delta = 0.016\, \sigma_u b^{0.54} t^{0.24} \quad (10b)$$

Scale Model Puncture Tests

It should be noted that scale model testing techniques are often used to demonstrate the adequacy of a cask design under the regulatory puncture loading. The most widely used scale models are the exact "replica models," which are the exact geometrically scaled versions of the corresponding prototype casks (without any material and initial impact velocity distortions). From theoretical and practical considerations, it is usually impossible to satisfy all scaling laws associated with such models (cf. Ref. 6 and Law 2 of Ref. 7) in cask puncture test dynamics. As a result, the model-predicted full-scale cask puncture responses will be distorted.

Some sources of such distortions are (i) material strain rate sensitivity properties, (ii) difference in cask and model center-of-gravity displacements, δ_p and $\delta_M = \delta_p/\lambda$ (λ = scaling factor), after initial impacts, which affect total potential energy results, and (iii) practical difference in cask and model material strength properties. Some corrections are needed to insure that the material strengths of production casks are properly predicted. Without monitoring material properties of both model and production casks, a proper way is to assume that the cask shell material has the minimum strength properties listed in standard handbooks and take a correctional measure to affect the scale model test parameters. A similar remedial measure can be taken to correct the strain rate and potential energy distortion effects. The simplest (approximate) way to take into account these distortion effects appears to be to correct the initial model impact energy and apply it in the test. The energy correction factors r_e, r_m, and r_n associated with the potential energy, material and strain rate distortions, respectively, are then

$$r_e = (40" + \delta_p)/(40" + \delta_p/\lambda) \quad (\lambda = 6"/d = \text{scaling factor})$$

$$r_m \simeq [(\sigma_u)_M/(\sigma_u)_P]^m \quad (m \leq 1)^*$$

$$r_n = r_n(\eta)^*$$

where subscripts M and P stand for "model" and "prototype," respectively. The total energy correction factor is $r_t = r_e r_m r_n$. Hence, the correct total energy to be applied in the model test to predict the correct prototype cask puncture behavior is

$$E_M' = W_M'(h_M' + \delta_p/\lambda) = r_t E_M = r_t(40" + \delta_p/\lambda)W_M \quad (W_M = W_p/\lambda^3)$$

Therefore, if the model drop height h_M' is kept at 40 inches, the correct model weight should be

$$W_M' = r_e r_m r_n W_M$$

*In the absence of a more accurate value for m and expression for $r_n(\eta)$, m = 1 and $r_n(\eta) = 1.3/(1 + 0.3\ \eta^{0.24})$ (cf Fig. 5 in which λ is set to η) can be used.

7. CONCLUDING REMARKS

Existing model and prototype puncture test data of both lead-shielded prismatic and cylindrical type steel casks and unbacked steel plate assemblies subjected to static or dynamic puncture loading were reviewed, analyzed and used in developing empirical puncture equations governing the incipient cask plate or shell puncture energy and punch force. Study of the static plate puncture test data confirmed that the puncture mechanism is a shear failure type with fracture shear value in the vicinity of 60% of ultimate tensile strength of the plate. Similarly, study of the incipient plate and shell puncture energy data indicated that, in addition to plate or shell thickness and punch diameter, the other parameters such as punch edge radius, strain rate and cask shell curvature or plate width, varying in degree, have appreciable effects on cask puncture energy absorbing capacity.

The numerical results of the present study are presented in the figures. The empirical equations developed herein for cask shell puncture design under certain restricted conditions are summarized in Section 6. Certain scale modeling distortions arising from use of the exact replica scale model testing techniques to predict cask puncture behavior were also discussed and a corrcctional measure to remedy such distortions formulated.

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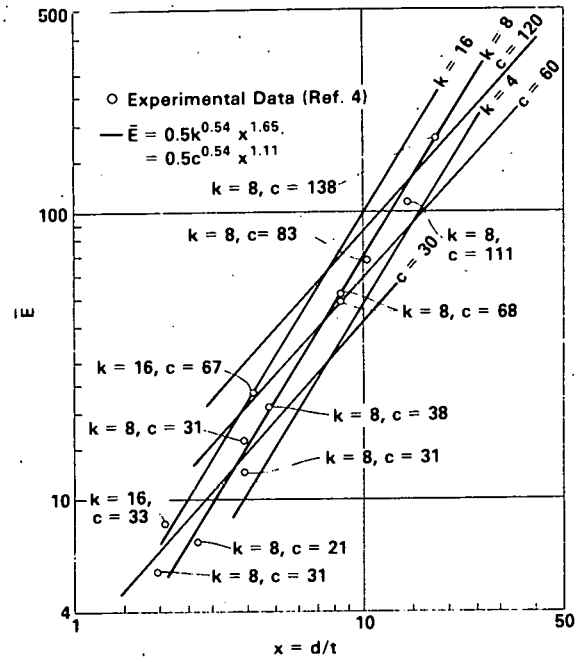


FIGURE 1. Incipient Puncture Energy for Single (Unbacked) Mild Steel Plates with Clamped Edges

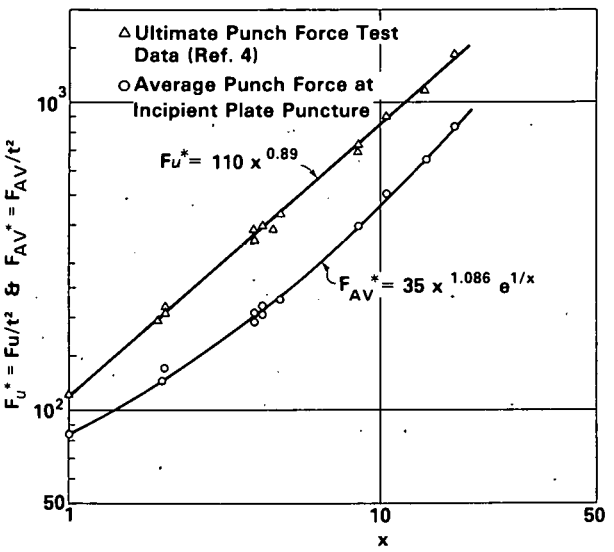


FIGURE 2. Punch Forces F_μ and F_{AV} for the Single Carbon Steel Plate Case (z = 0)

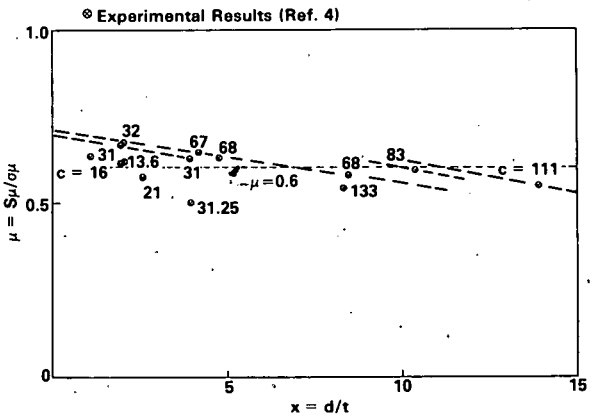


FIGURE 3. Static Fracture Shear Stress—Ultimate Strength Ratio μ as a Function of x for Various Values of c = b/t—Unbacked Carbon Steel Plate Puncture Case (z = 0)

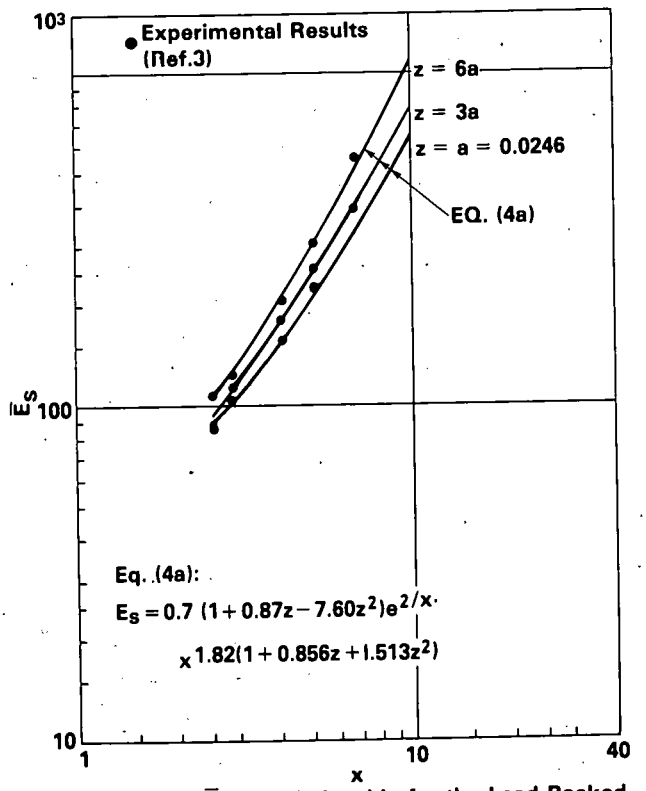


FIGURE 4. $\bar{E}$, x—Relationship for the Lead-Backed, Low Carbon Steel Plate Assemblies-Static Case

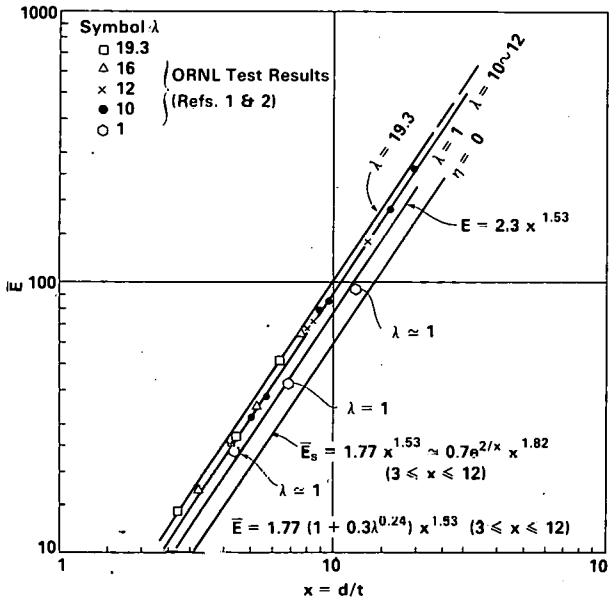


FIGURE 5. $\bar{E}$, x—Relationship for Lead-Backed Carbon Steel Plate Assemblies (z = 0)

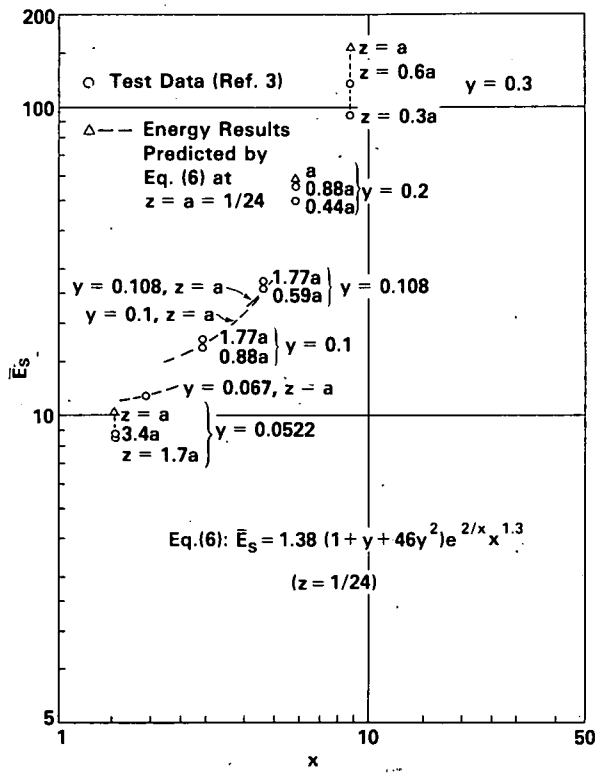


FIGURE 6. Static Incipient Puncture Energy Results for Lead-Shielded Carbon Steel Model Cask Shells

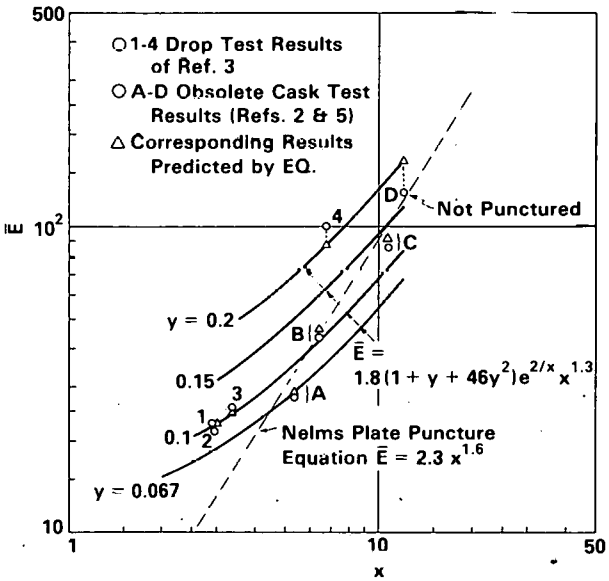


FIGURE 7. Drop Test and Equation-Predicted Incipient Puncture Energy Results for Lead-Shielded, Hollow Cylindrical Cask Models

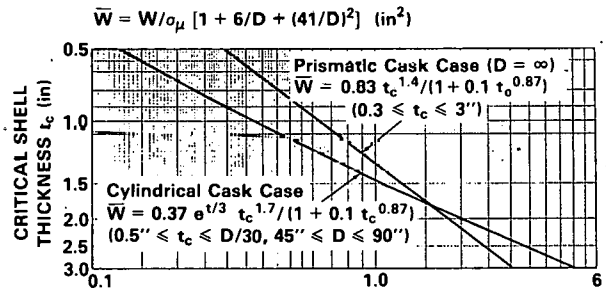


FIGURE 8. Puncture Design Curves for Lead-Backed Prismatic and Cylindrical Steel Cask Shells

PUNCTURE PHENOMENA IN LOW VELOCITY IMPACT OF RADIOACTIVE WASTE SHIPPING CONTAINERS

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Design of radioactive waste shipping containers requires consideration of impact phenomena at low contact velocity. Use of analytical procedures based upon high velocity impact may lead to serious errors as the structural vibrations play a major role in the energy transfer. A traditional "stereomechanical" analysis treats the impacting bodies essentially as mass points rigidly connected. The local response of the material is then examined and response times are in the order of microseconds due to accoustic wave effects. When vibrational effects enter, the response times are in milliseconds for low frequencies and failure may not occur at initial impact but later when vibrational beating phenomena enter into the problem.

To study the impact problem of thin plate or shell structures backed with high density material, Lawrence Livermore Laboratory initiated an analytical and experimental research program using circular plates. The circular plates consisted of three layers and a ballast ring and were impacted axisymmetrically by dropping them on a circular steel punch. The outer plate thickness, the high density backing material and thickness were varied as was the impact velocity.

Initial analytical studies using a detailed finite element analysis, including the effects of plastic-work hardening in the material were made on two geometries. Although the finite element program is accurate, it is expensive and time consuming. Therefore, an alternate analytical approach was developed based upon a system frequency spectra analysis. A Rayleigh-Ritz variational analysis was used to determine the natural frequencies and their corresponding modes.

RAYLEIGH-RITZ ANALYSIS

A half cross-section of the circular plate and punch is shown in Figure 1. All materials were treated as elastic with distributed mass and the natural frequencies and mode shapes were calculated using a Rayleigh-Ritz variational approach. This analysis assumes a series form for the harmonic deformation of the plate with arbitrary coefficients. The maximum potential and kinetic energies of the system are calculated and the difference between them is minimized by selection of the arbitrary coefficients. A system of homogeneous, linear equations are thus formed involving the circular frequency.

Vibrations of axisymmetric circular plates have deflection modes in the form of

zero order Bessel functions of the first kind which may be expressed in terms of a

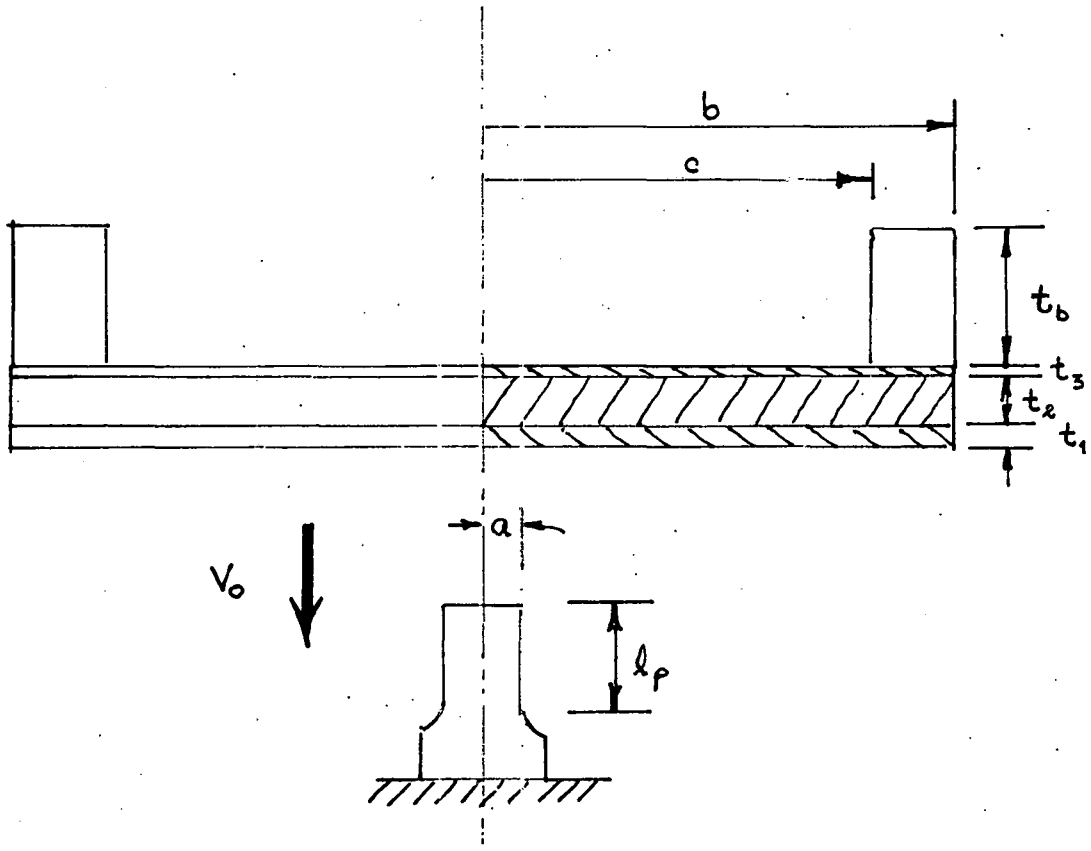


FIGURE 1

power series in the radial variable involving even powers. The harmonic plate deflection was chosen in the form:

$$u_z = \sum_{i=1}^{I} A_i U_i(r) \sin \omega_i t \tag{1}$$

where

$$U_i(r) = \sum_{n=0,2,4,}^{N} C^i_n (r/b)^n \tag{2}$$

ω_i is circular frequency; u_z is plate deflection; t is time; b is radius of the plate; and r is the radial variable.

The harmonic form was chosen such that the initial plate deformation was zero and the initial velocity distribution could be chosen later. Individual natural frequencies and mode shapes are represented by ω_i and U_i respectively. For a single mode response the deflection is:

$$u_z = AU(r) \sin \omega t \tag{3}$$

Potential energy is stored in the form of strain energy in each laminate of the

plate, in the ballast and in the punch. The plate laminates were treated as thin plates each having the same deformation shapes. They are free to slide but transmit pressure between each other. The ballast was treated as a plate ring and the punch as an elastic uniaxial stress member with shear stress.

The potential energy in the plates is at any time:

$$V_{plates} = \sum_{i=1}^{3} \pi D_i \int_o^c \left\{ \left[\frac{\partial^2 u_z}{\partial r^2} \right]^2 + \left[\frac{1}{r} \frac{\partial u_z}{\partial r} \right]^2 + 2\nu_i \left[\frac{1}{r} \frac{\partial u_z}{\partial r} \frac{\partial^2 u_z}{\partial r^2} \right] \right\} rdr$$

where D_i is theflexural rigidity of the plate:

$D_i = \frac{E_i t_i^3}{12(1-\nu_i^2)}$; E_i is Young's modulus; t_i is plate laminate thickness; and ν_i is Poisson's ratio.

The potential energy in the punch becomes:

$$V_{punch} = \frac{1}{2} \int_{Vol} \sigma \varepsilon dv = \frac{\pi E_p}{\ell_p} \int_o^a u_z^2 \, rdr + \frac{E_p \pi}{8(1+\nu)} \int_o^a \int_o^{\ell_p} \left(\frac{\partial u_z}{\partial r} \right)^2 \frac{z}{\ell} \, dzdr \qquad (5)$$

where E_p is Young's modules of the punch; ℓ_p is length of the punch; a is the punch radius.

Substituting equations (1) and (2) into (4) and (5) for any single mode and assuming maximum valve yields:

$$V_{max} = \sum_{i=1}^{3} \frac{\pi D_i}{b^2} \sum_m \sum_n C_m C_n \frac{mn(m-1)(n-1) + mn + 2\nu_i mn(m+n-2)}{(m+n-2)}$$

$$+ \frac{\pi D_b}{b^2} \sum_m \sum_n C_m C_n \left[1 - \frac{c}{b} \right]^{(m+n-2)} \frac{mn(m-1)(n-1) + mn + 2\nu_b mn(m+n-2)}{(m+n-2)}$$

$$+ \frac{\pi E_p b^2}{\ell_p} \sum_m \sum_n \frac{C_m C_n (a/b)^{m+n+2}}{(m+n+2)} + \frac{\pi E_p \ell_p}{16(1+\nu_p)} \sum_m \sum_n \frac{mn}{m+n} C_m C_n \left(\frac{a}{b} \right)^{m+n} \qquad (6)$$

where D_b is the equivalent flexural rigidity of the ballast and ν_b is the Poisson's ratio of ballest material.

The kinetic energy is due to the distributed mass in the plate laminates, the ballast and the punch. The expression for the kinetic energy in the plate is:

$$T_{plate} = \sum_{i=1}^{3} \pi \rho_i t_i \int_{radius} (\dot{u}_z)^2 rdr \qquad (7)$$

where ρ_i is the mass density in the plate laminate or ballast and t_i is the thick-

ness. The dot above u_z signifies the derivative with respect to time.

The kinetic energy in the punch is:

$$T_{punch} = \pi\rho_p \ell_p \int_0^c (\dot{u}_z)^2 \, r\,dr \tag{8}$$

The total kinetic energy in the system expressed at maximum time is:

$$T_{max} = \sum_{i=1}^{3} \pi\rho_i t_i b^2 \omega^2 \sum_m \sum_n \frac{C_m C_n}{m+n+2}$$

$$+ \pi\rho_b t_b b^2 \omega^2 \sum_m \sum_n \frac{C_m C_n}{m+n+2} \left[1-\left(\frac{c}{b}\right)^{m+n+2}\right]$$

$$+ \pi\rho_p \ell_p b^2 \omega^2 \sum_m \sum_n \frac{C_m C_n}{m+n+2} \left(\frac{a}{b}\right)^{m+n+2} \tag{9}$$

The difference between these two expressions should be a minimum:

$$V_{max} - T_{max} = \text{minimum} \tag{10}$$

Therefore: $$\frac{\partial(V_{max} - T_{max})}{\partial C_j} = 0 \qquad j = 1,2\text{---} \tag{11}$$

Substituting equations (6) and (9) into (11) yields:

$$\sum_n A_{mn} C_n = 0 \tag{12}$$

where:

$$A_{mn} = \frac{mn(m-1)(n-1)+mn}{(m-n-2)} \left\{ \frac{2\pi}{b^2} \sum_{i=1}^{3} D_i + D_b \left[1-\left(\frac{c}{b}\right)^{m+n+2}\right] \right.$$

$$+ mn \frac{4\pi}{p^2} \sum_{i=1}^{3} D_i \nu_i + D_b \nu_b + \frac{2\pi E_p}{\ell_p} \frac{(a/b)^{m+n+2}}{(m+n+2)}$$

$$\left. + \frac{E_p \pi \ell_p}{8(1+\nu_p)} \frac{mn}{m+n} \left(\frac{a}{b}\right)^{m+n} \right\} - \frac{\omega^2 \pi b^2 2}{(m+n+2)} \left\{ \sum_{i=1}^{3} \rho_i t_i + \rho_b t_b \left[1-\left(\frac{c}{b}\right)^{m+n+2}\right] \right.$$

$$\left. + \rho_p \ell_p \left(\frac{a}{b}\right)^{m+n+2} \right\} \tag{13}$$

Equation (12) has nontrival solutions for values of ω^2 which make the determinant of A_{mn} zero. Increasing the number of terms yields the higher natural frequencies and convengence of the low eigenvalues from above.

After the eigenvalves are determined, the coefficient of the highest polynominal term is set equal to one, and the ratios C_i/C_n, i = 1, to (n-1), are determined yielding the eigenfunctions $U_i(r)$.

MODAL ANALYSIS

The plate deflection curve may now be written as:

$$u_z = \sum_{i=1}^{I<N} A_i U_i(r) \sin \omega_i t \tag{14}$$

where

$$U_i(r) = \sum_{n=0,2,4}^{N} \frac{C_n^i}{C_N^i} \left(\frac{r}{b}\right)^n \tag{15}$$

The eigenvalves, ω_i, and eigenfunction coefficients, C_n^i C_N^i, were determined by the Rayleigh-Ritz analysis. The modal analysis coefficients A_i will be determined based upon system frequency spectra examinations using a least squares approximation. The number of frequencies used must be less or equal to the number of polynomial terms in the eigenfunction. The error in eigenvalue and eigenfunction are higher for the higher frequencies and will contribute unnecessary noise in the response. Therefore, for most case "I" was selected less than "N".

The initial conditions are:

$$u_z = 0 \quad t = 0, \quad o \leq r \leq b \qquad \dot{u}_z = F(r) \quad t = 0, \quad o < r < b \tag{16}$$

The first condition is satisfied by the form of u_z and the second is satisfied by least-squares.

$$\int_o^b \left\{ \sum_i \omega_i A_i U_i(r) - f(r) \right\}^2 rdr = \text{minimum} \tag{17}$$

The integral is minimized with regard to each coefficient yielding:

$$\sum_j A_i \omega_i \int_o^b U_i(r) U_j(r) rdr = \int_o^b f(r) U_i(r) rdr \tag{18}$$

This may be written:

$$\sum S_{ij} D_j = R_i \tag{19}$$

where: D_j is $A_j \omega_j$; R_i is $\int_o^b f(r) U_i(r) rdr$; $S_{ij} = \int_o^b U_i(r) U_j(r) rdr$.

S_{ij} is a symmetric array and if the eigenfunctions were orthogonal, it would also be a diagonal array. For the determined digenfunctions and for a uniform initial velocity, v_o, this becomes:

$$S_{ij} = b^2 \sum_m \sum_n \frac{C^i_m}{C^i_M} \frac{C^i_n}{C^i_N} \Big/ (m+N+2) \tag{20}$$

$$R_i = v_o b^2 \sum_m \frac{C^i_m}{C^i_M} \Big/ (m+2) \tag{21}$$

This linear system is easily solved yielding the modal coefficients A_i.

DISPLACEMENT AND STRESS ANALYSIS

Displacements, velocities and accelerations may be easily obtained by differentiation of equation (14) and evaluation at anytime or position.

Since the punch is treated as a uniaxial member, the stress may be expressed as:

$$\sigma_{zz}^{\text{punch}} = \frac{E_p u_z}{\ell_p} \qquad 0 \leq r \leq a \tag{22}$$

If plastic work hardening is assumed equation (22) is modified as follows:

$$\sigma_{zz}^{\text{punch}} = \frac{E_p u_z}{\ell_p}(r,t) \quad ; \sigma_{zz} > -\sigma_y \tag{23}$$

$$\sigma_{zz}^{\text{punch}} = -\sigma_y + \frac{\tilde{E}_p}{\ell_p}\left[u_z + \frac{\sigma_y \ell_p}{E_p}\right]; \quad \sigma_{zz} < -\sigma_y$$

where σ_y is tensile yield stress; E_p is plastic hardening modulus; E_p is elastic Young's modulus.

The moment and shear force distribution in each plate laminate may be written as follows:

$$M_r = -D_i \left(\frac{\partial^2 u_z}{\partial r^2} + \nu_i \frac{1}{r} \frac{\partial u_z}{\partial r} \right) \tag{24}$$

$$M_\theta = -D_i \left(\frac{1}{r} \frac{\partial u_z}{\partial r} + \nu_i \frac{\partial^2 u_z}{\partial r^2} \right) \tag{25}$$

$$Q_r = -D_i \frac{\partial}{\partial r} \left(\nabla^2 u_z \right) \tag{26}$$

where D_i and ν_i are the plate flexternal figity and Poisson's ratio of the plate laminate and

$$\nabla^2 = \frac{\partial^2}{\partial r^2} + \frac{1}{r} \frac{\partial}{\partial r}$$

The net pressure acting on any plate laminate may be written as:

$$p(r) = D_i \nabla^2 \nabla^2 u_z + \rho_i t_i \ddot{u}_z \tag{27}$$

where ρ_i is mass density; t_i is thickness; $\ddot{u}_z$ is plate acceleration.

The plate stresses may now be determined

$$\sigma_{rr} = \frac{12 M_r z}{t_i^3} \; ; \; \sigma_{rr}{}^{max} = \frac{6 M_r}{t_i^2} \tag{28}$$

$$\sigma_{\theta\theta} = \frac{12 M_\theta z}{t_i^3} \; ; \; \sigma_{\theta\theta}{}^{max} = \frac{6 M_\theta}{t_i^2} \tag{29}$$

$$\sigma_{rz} = \frac{3 Q_r}{2 t_i} \left[1 - \frac{2z}{t_i} \right]^2 \; ; \quad \sigma_{rz}{}^{max} = \frac{3 Q_r}{2 t_i} \tag{30}$$

EXPERIMENTAL PROGRAM

Twelve inch diameter laminated plates composed of a 316 stainless steel outer plate, a lead or uranium middle plate and 0.050" thick 316 stainless steel plate were dropped upon a mild-steel punch from different heights. The experimental drop structure is ballasted with 400# of weight to increase impact energy. Strain gages have been placed on the lower plate and punch and accelerometers were located on the plate and the drop structure. The test program varies lower plate thickness, backing plate material and thickness and punch diameter. Preliminary experimental data shows structural vibration response dictates failure as first two resonant modes are clearly visible. General agreement exists between the analytical and experimental program and criteria for failure are being studied.

DISCUSSION

The Rayleigh-Ritz and system frequency spectra analyses were programmed for computer calculations and represent an efficient means to consider the puncture phenomena. This approach permits the inclusion of other linear effects such as spring constraints between the ballast and test plates as well as edge constraints. Some of the parameters must be adjusted based on initial experimental results but the referred mathematical model then predicts accurately future tests.

This approach clearly indicates that the structural response of the entire radioactive waste shipping container must be considered when examining the design relative to puncture. The lowest eigenfunction corresponds approximately to the static deformation configuration and governs the dynamic response. The static deformations are 1.3 to 3 times higher. The higher vibration modes tend to radiate energy away from the puncture process. Analyses based upon high velocity puncture will contribute large errors and fail to focus on correct design parameters.

ABSTRACT

Punch penetration phenomena of a radioactive waste shipping container when subjected to low velocity impact are studied both analytically and experimentally. Model cask materials in the form of circular test samples of stainless steel, backed with high density material (lead or uranium) are dropped on a steel punch with impact velocities of 220 in/sec or less. The experimental program involves 25 static tests and 67 dynamic tests in which plate thicknesses, backing material and punch diameters are varied.

A detailed finite element analysis including the effects of plastic-work hardening in the materials and plate laminate separation was made on two geometries; ballasted with an impact velocity of 66.7 in/sec and unballasted with an impact velocity of 200 in/sec. The ballast mass was chosen such that the initial kinetic energy of both systems were the same. This analysis indicates that structural vibrations play a major role in impact due to the low natural frequencies of the system and low impact velocities. In these cases, stereomechanical treatment of impact is seriously in error. (See Impact, W. Goldsmith, Edward Arnold Ltd., London, 1960.) An impact analysis including system vibration effects were indicated and although the finite element program is accurate, it is expensive and time consuming.

Therefore, an alternate analytical program was developed based upon a system frequency spectra analysis. A Rayleigh-Ritz approach was selected, including the laminated plates, deformable punch and elastic ballast in the potential energy, and distributed mass in the kinetic energy. A modal analysis using the first five frequencies was performed and stress calculations included the effects of plastic deformation. Excellent agreement was obtained between the Rayleigh-Ritz and the finite element analyses. In most cases, it was observed that initial impact was not sufficient to cause onset of punch yield or plate puncture but the plate vibration effects would initiate failure at a later time.

The Rayleigh-Ritz program was used to analyze the different experimental configurations and then compared to experimental data. This research indicates that low velocity impact studies of radioactive waste shipping containers must consider the phenomena of structural vibrations. The analytical approach developed offers an accurate, quick and inexpensive method of examing low velocity impact. Work is progressing to study the vibration modes of actual shipping casks by similar methods.

INTERNAL PRESSURE CHANGES OF LIQUID FILLED SHIPPING CASKS DUE TO THERMAL ENVIRONMENT

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The amount of moisture present in a shipping container can have a significant effect on the internal pressure of the cask for certain transportation conditions. Specifically those conditions which lead to high internal coolant temperatures. This can be as a result of the fire accident environment, the shipment of casks which contain high internal heat loads or casks with moderate internal heat loading and poor thermal dissipation properties. Water exhibits a large vapor pressure at elevated temperatures, as do most liquid coolants, and therefore the effects of the resulting phase change on the pressure of the system must be carefully considered.

A liquid coolant can be converted into a gas, or a gas into a liquid with the proper changes in pressure and temperature. Steam is a gaseous substance which is usually referred to as a vapor instead of a gas, since the word vapor is normally used for a gas near the condensation point. However a gas is commonly defined as a vapor if it exists at a temperature lower than its critical temperature and therefore can be liquefied by a change in presure alone.

Thermodynamically, a liquid coolant can be considered a pure substance if it is completely homogeneous and its chemical composition does not change. However, it may exist in more than one phase as with water. For example, steam and liquid water or ice and liquid water would be considered a two phase mixture of a pure substance. Some substances such as air can be treated as a pure substance for thermodynamic purposes, only if no phase change occurs. This is because the chemical composition of liquid air is different from the gaseous phase.

For an equilibrium mixture of pure substances the thermodynamic properties can be established experimentally and related by a thermodynamic equation, just as for a single pure substance. However gases can be mixed in any proportions to form an infinite number of different mixtures, depending on the constituents and their relative proportions in the mixture. The experimental work necessary to chart the behavior of all possible mixtures would be prohibitive. Therefore, it becomes necessary to establish a method to account for the behavior of a mixture of gases. It is desirable to be able to express the thermodynamic properties of the mixture in terms of the properties of the constituents and their relative proportions in the mixture.

Consider a mixture of gases made up of constituents a, b, c, ..., existing in equilibrium at a pressure p, and occupying a volume V. Each constituent occupies

the same volume V that the entire mixture occupies and has the same temperature T as that of the mixture. The mass of the mixture must equal the sum of the masses of the constituents,

$$M = M_a + M_b + M_c + \ldots = \Sigma M_i.$$

Similarly, the number of moles of the mixture is the sum of the moles of the individual constituents,

$$N = N_a + N_b + N_c + \ldots = \Sigma N_i.$$

The mole fraction X of the constituents is defined as, $X_i = N_i/N$ and therefore, $\Sigma X_i - \Sigma N_i/N = 1.0$

In an equilibrium mixture of gases, each constituent occupies the entire volume at the common mixture temperature while exerting only a fraction of the entire pressure. This fact leads to Dalton's law of partial pressures which states that the total pressure of a mixture of gases is equal to the sum of the partial pressures which the constituent gases would exert if each existed alone in the mixture volume at the mixture temperature. In a mixture of ideal gases each constituent is assumed to be unaffected by the presence of other constituents. Substitution of the equation of state for ideal gases into,

$$N = N_a + N_b + N_c + \ldots = \Sigma N_i$$

yields, $$\frac{pV}{R_o T} = \frac{p_a V_a}{R_o T_a} + \frac{p_b V_b}{R_o T_b} + \frac{p_c V_c}{R_o T_c} + \ldots = \Sigma \frac{p_i V_i}{R_o T_i}.$$

But, $V = V_a = V_b = V_c$ and $T = T_a = T_b = T_c$

therefore, $$p = p_a + p_b + p_c = \ldots = \Sigma p_i$$

which is a mathematical statement of Dalton's law.

If the mixture of gases a, b, c, ..., were separated such that each constituent was at the pressure and temperature of the mixture, we would find that the sum of the partial volumes of the constituents would be equal to the total volume, or

$$V = V_a + V_b + V_c + \ldots = \Sigma V_i$$

which is a mathematical statement of Amagat's law of partial volumes.

Both Dalton's law and Amagat's law are consistent with the kinetic theory of gases. These laws assume that no intermolecular forces exist in a mixture of gases and that each constituent acts as if no other constituents were present. Therefore, the perfect gas equation can be applied individually to each constituent.

When a liquid occupies a space greater than its volume, it evaporates as a vapor into the space above the liquid. This is because of the differing kinetic energy levels of the liquid; the higher velocity molecules have sufficient energy to overcome the surface restraint and leave the liquid. With the liquid in a confined space, vapor molecules strike the liquid surface where they condense, being held by the potential attractive forces of the liquid. Thus the maximum pressure exerted by the vapor will occur when the rate of vaporization of liquid is equal to the rate of condensation of the vapor. For this equilibrium between saturated liquid and saturated vapor, the pressure is referred to as vapor pressure. Vapor pressure is a function of temperature, Figure 1, therefore the boiling temperature of a liquid is controlled by pressure. The vapor pressure is independent of the relative amounts of liquid and vapor present because both the rate of vaporization and the rate of

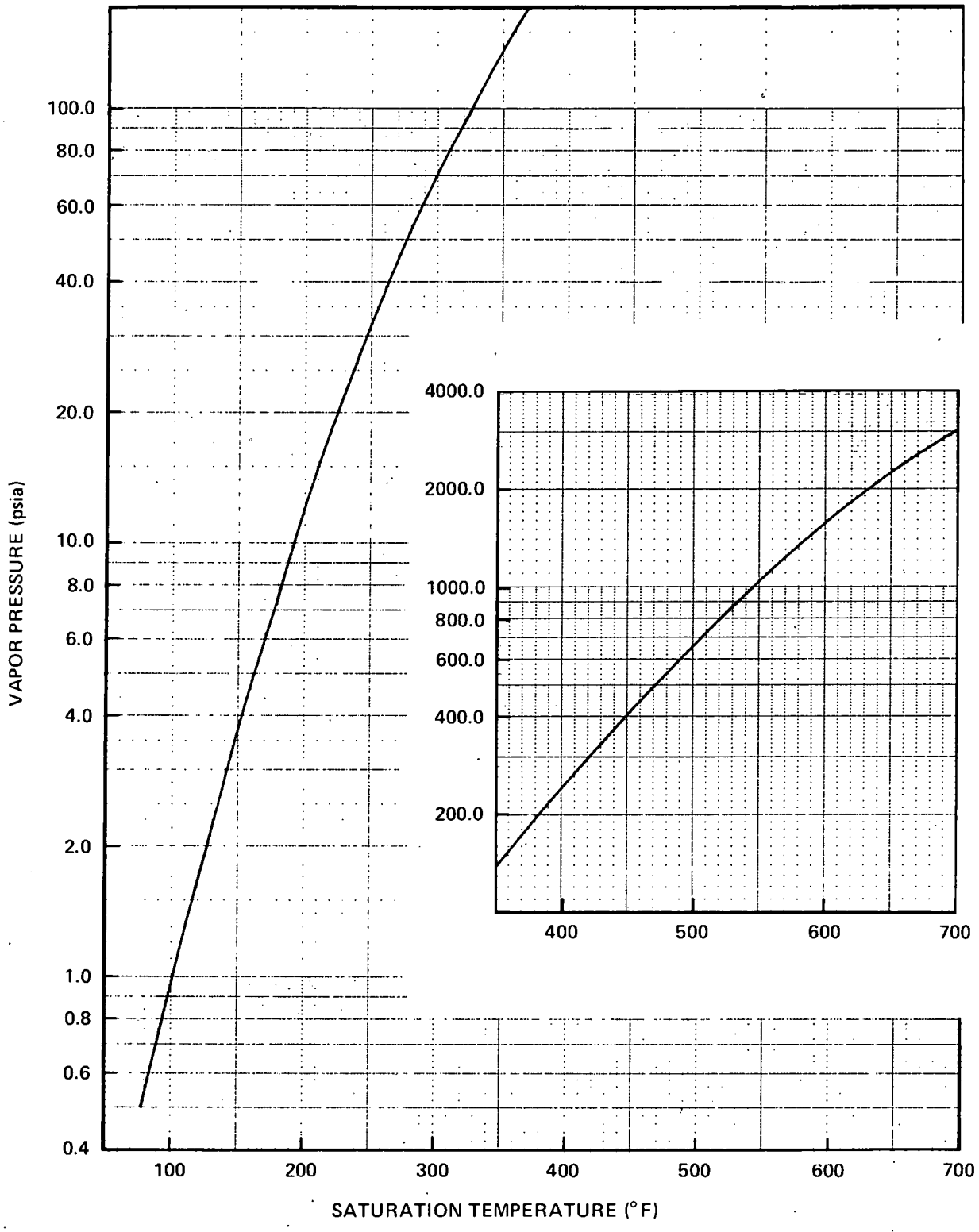


Figure 1. Vapor Pressure as a Function of Saturation Temperature for Pure Water

condensation are a function of liquid surface area. Changing the magnitude of the surface area of the liquid will change the time for reaching equilibrium, but it will not affect the equilibrium pressure.

At low pressures, gases other than the vapor from the liquid can be present without seriously affecting the rates of condensation and vaporization. Due to this fact, the vapor pressure-temperature relationship for the pure substance can be applied to a low-pressure system consisting of one liquid component and several gaseous constituents to predict the maximum partial pressure of the vapor. Since the partial pressure of the vapor is dependent on temperature, the sum of the partial pressures of the other gaseous constituents in the mixture is equal to the total pressure of the entire mixture less the partial pressure of the vapor. If the temperature of the mixture were raised while keeping the total pressure constant, the result would be that more liquid would evaporate since the vapor constituent would remain in equilibrium with its liquid. Finally, the liquid would disappear completely and for any additional increase in temperature, the vapor would be superheated. With no liquid component present the mixture of gases and superheated vapor would exhibit partial pressures equal to their mole fractions.

Using some of the thermodynamic relationships previously discussed, it is possible to develop a model of the liquid filled shipping cask based on an idealized two-phase mixture. A mathematical relationship based on this model can be derived which predicts internal cask pressures as a function of initial thermodynamic loading conditions, initial fluid volume ratio and final mixture temperature. The governing assumptions of the liquid filled cask model are as follows:

1. Constant volume, rigidly unyielding tank.

2. Closed system containing a two phased mixture of a pure substance combined with other pure substances in the gaseous phase.

3. The contents of the cask (liquids, gases and vapors) are at one uniform mixture temperature.

4. The mass of liquid which undergoes phase change to a vapor is insignificantly small with respect to the mass of the remaining liquid.

5. Enough liquid always remains in the tank to supply the phase change mass required for the saturated vapor condition.

6. The liquid acts as an incompressible fluid, but is thermally expandable.

7. The gas acts as both a compressible and thermally expandable fluid.

8. Both the vapor and the gas can be represented by the ideal gas equation of state.

Some of the parameters and symbols used in the equations developed are,

p = total pressure, psia
T = temperature, °R
V = tank volume, other volumes subscripted, ft^3
1 = initial condition
2 = final or mixture condition
sv = saturated vapor
a = gas (air)
f = liquid (water)

The internal tank pressure is made up of the pressure contribution of the saturated vapor plus the partial pressure of the cover gas, in most cases air.

$$p = p_{sv} + p_a \quad .$$

The final mixture pressure after the sealed cask has reached equilibrium at some elevated temperature is given by,

$$p_2 = p_{sv_2} + \frac{N_a R_o T_2}{V_{a_2}} \quad ,$$

where the pressure contribution of the air is predicted from the ideal gas equation of state evaluated at the final mixture temperature and the pressure contribution of the vapor is simply the corresponding vapor pressure of the liquid, usually water, at the final mixture temperature. The number of moles of air contained in the cask can be calculated from the ideal gas equation of state evaluated at the initial condition,

$$N_a = \frac{p_1 V_{a_1}}{R_o T_1} \quad .$$

Substituting the number of moles of air into the cask pressure equation yields,

$$p_2 = p_{sv_2} + \frac{p_1 T_2}{T_i} \frac{V_{a_1}}{V_{a_2}} \quad .$$

The volume ratio of air can be obtained from the tank total volume, using the initial and final volumes of water contained in the tank,

$$\frac{V_{a_1}}{V_{a_2}} = \frac{V - V_{f_1}}{V - V_{f_2}} \quad .$$

The initial volume of liquid is assumed to be known and the final volume of liquid can be found from the initial and final liquid densities,

$$V_{f_2} = V_{f_1} \frac{\rho_{f_1}}{\rho_{f_2}} \quad .$$

If we define the initial fluid volume ratio and the fluid density ratio (initial to final) as follows,

$$R_f = \frac{V_{f_1}}{V} \quad \text{and} \quad \gamma = \frac{\rho_{f_1}}{\rho_{f_2}} \quad ,$$

and make the proper substitutions into the equation for the volume ratio of air, we are left with,

$$\frac{V_{a_1}}{V_{a_2}} = \frac{1 - R_f}{1 - \gamma R_f} \quad .$$

Now that the change in volume of air is known as a function of initial and final conditions we can substitute into the pressure equation as follows,

$$p_2 = p_{sv_2} + \frac{1 - R_f}{1 - \gamma R_f} \frac{p_1 T_2}{T_1} \quad .$$

The above equation is a relationship which can be used to determine internal cask pressure as a function of initial pressure, temperature, fluid volume ratio and the final mixture temperature, along with the use of a vapor pressure curve and density table for the liquid.

Two figures are presented for use in reviewing internal presssure changes of liquid filled shipping casks due to their thermal environment. The resulting curves are for water/air filled casks with an initial fill temperature of 80°F and internal pressure of 14.696 psia at time of sealing. Figure 2 relates cask internal pressure as a function of mixture temperature for various values of initial fluid volume ratio. This figure illustrates that large increases in pressure do not occur until approximately eighty percent liquid fill is reached for any given temperature. The results also show the large difference in pressure between $R_f = 0.0$ (just enough water to satisfy the saturation conditions) and a completely dry air internal cask environment. Figure 3 relates initial fluid volume ratio to mixture temperature for lines of constant internal cask pressure. This figure is useful in preliminary pressure determination, for estimated mixture temperature and liquid fill ratio.

It can be seen from the curves presented that some cask designs are more sensitive than others to pressure change effects. For a given mixture temperature, errors in R_f in the lower range are relatively unimportant, however at large R_f values small errors in R_f can result in large pressure changes. Therefore, designs which have small gas free volumes are more sensitive to the effect of mixture temperature on the cask internal pressure. Measurement errors in the amount of gas free volume can become quite important when the initial fluid volume ratio is large. The error in predicted design temperature can also become very important for large values of R_f.

The following are some common pressure calculation problems frequently seen by NRC for liquid filled cask designs.

1. Failure to consider change in gas free volume due to liquid expansion.

2. Failure to consider affect of Dalton's law on the behavior of a two-phase mixture with cover gas. Applicants quite often only considers vapor pressure after the liquid boiling point has been reached.

3. Incorrect temperature used for pressure calculations. Coldest surface temperature quite often controls the vapor pressure contribution, while the fluid bulk temperature is controlling for the gas pressure.

4. Failure to consider water in liquid or vapor form which is present in so called dry cask designs.

A relationship has been derived which can be used to determine internal cask pressures as a function of initial thermodynamic loading conditions, initial fluid volume ratio and final mixture temperature. The results have been presented graphically in a parametric form for the case of water/air filled casks. These results are particularly useful for preliminary design verification purposes. They provide the means for determining if the space provided for contained expansion of liquid coolant or a liquid shielding medium is adequate for the expected design conditions.

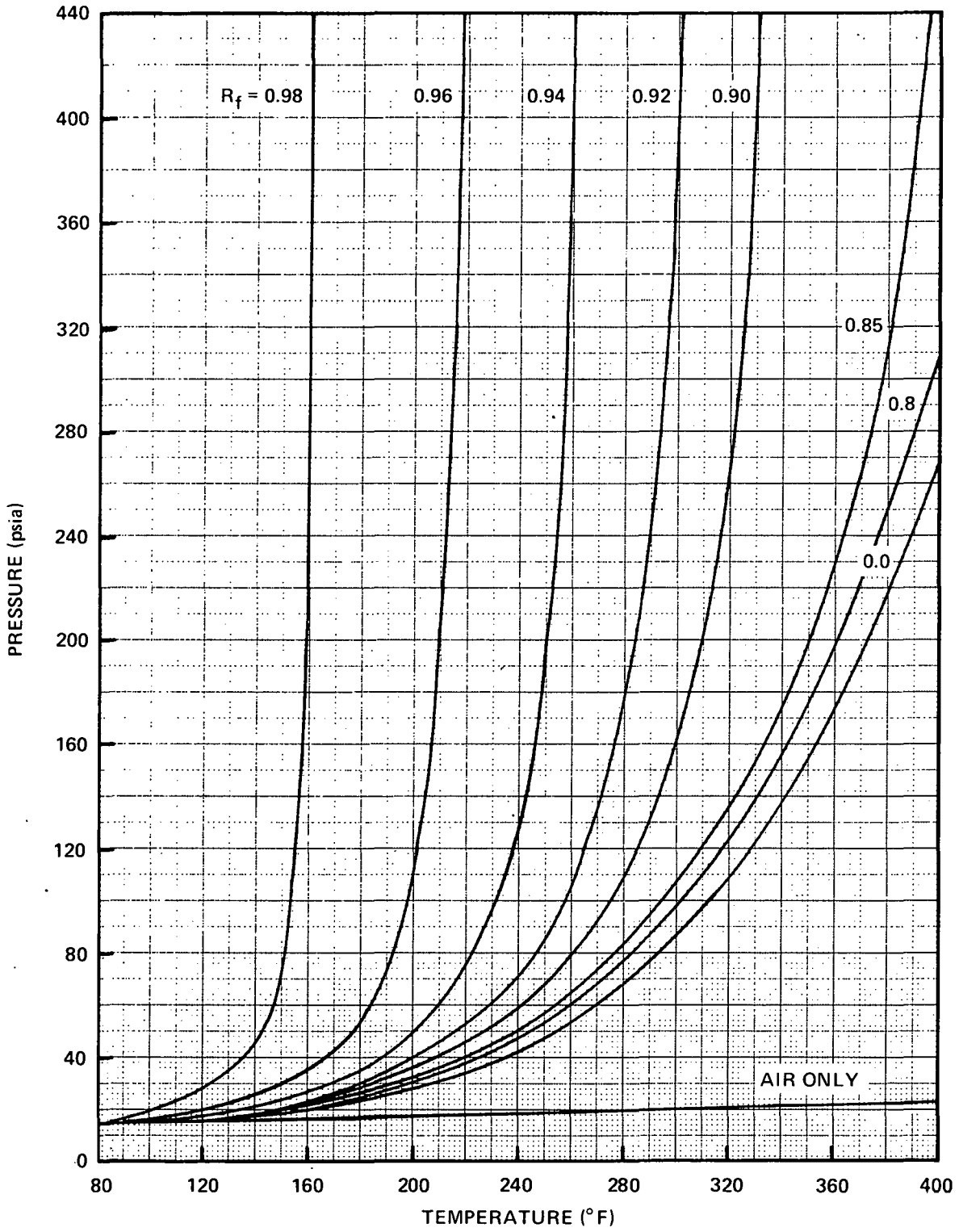


Figure 2. Cask Internal Pressure as a Function of Mixture Temperature

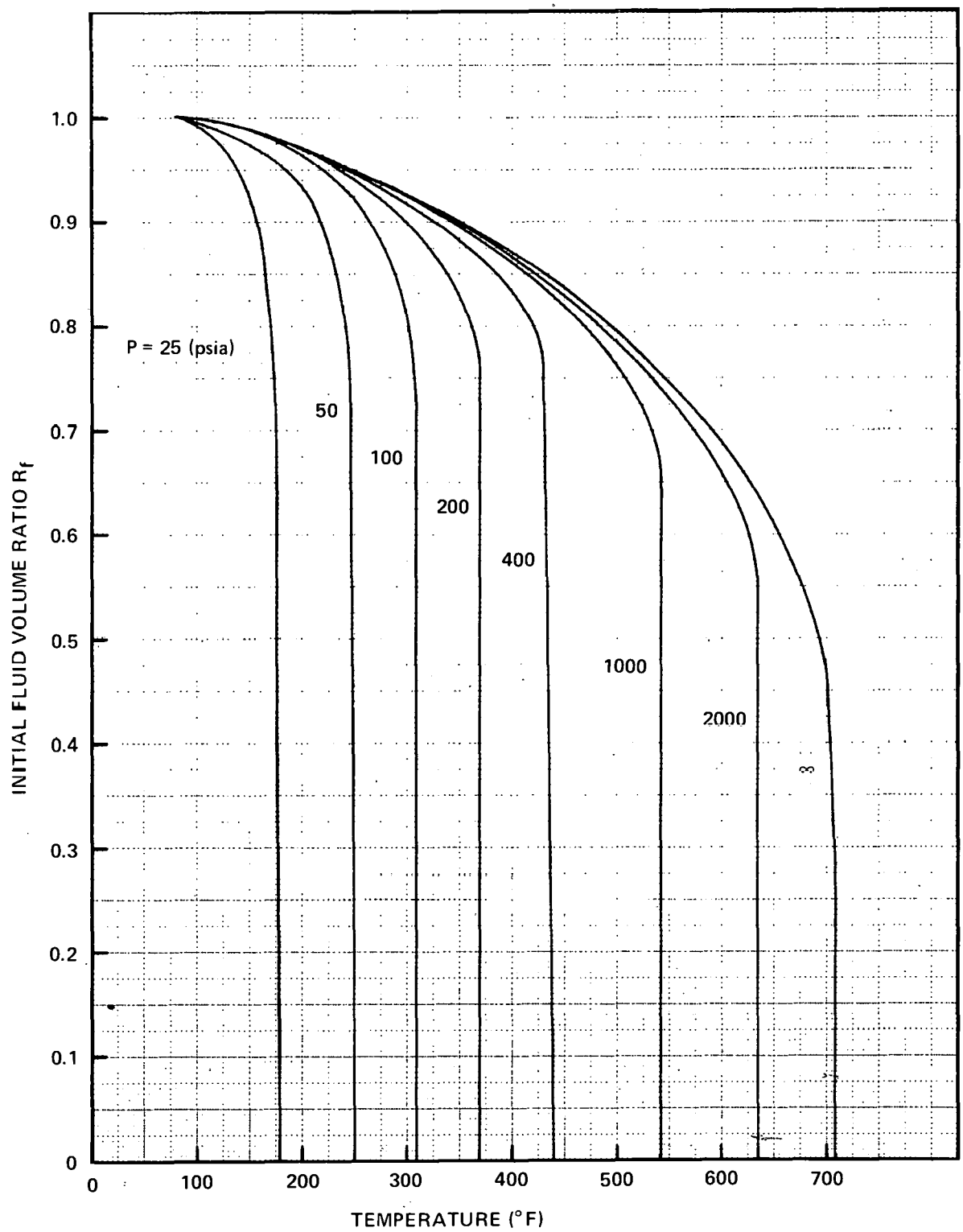


Figure 3. Initial Fluid Volume Ratio vs Mixture Temperature

DESIGN OF HEMISPHERICAL SHELL IMPACT LIMITERS

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I. INTRODUCTION

Hemispherical impact limiters were designed to protect the contents of a large (100 inch (2.54M) diameter, 400 inch (10.16M) long, 200 ton (181 Tons Metric)) cylindrical new fuel shipping container. The design specification stipulated that the container meet the requirements of AECM 0529[1]* and further, that the maximum deceleration of the container not exceed 60 g's. It was further stipulated that the limiters be fabricated from formed carbon steel plate and impact deformation was to be restricted to preclude contact with the contents.

In order to develop a reliable limiter design, a series of analytical and experimental investigations were conducted. Preliminary finite element analysis of the limiter was conducted but it was found that the expense of running the complex computer models necessary to model the structure precluded their use as a design tool. Limit analysis methods were used to determine the static load-displacement characteristics of the hemispherical limiter. A series of model tests were then conducted on small scale specimens to verify the analytical expressions. The analytical methods were then modified to account for strain rate sensitivity. Dynamic tests were conducted on the small scale specimens with good agreement between test data and theory.

The first step in designing the limiters was to develop an understanding of the physical phenomena involved in the deformation process and to determine the basic load-deflection characteristics of a statically loaded hemispherical shell. It was assumed that for the relatively low impact velocity, the local deformations are sufficiently large and the impact velocities are sufficiently small so that the structural elements could be treated as rigid-plastic and the distributed inertia of these elements could be neglected. The calculational model consists of a rigid mass attached to a massless uniform rigid-perfectly plastic spherical shell. (Further tests and analysis confirmed that the elastic behavior of the shell was indeed negligible.) During impact the kinetic energy of the mass is converted into plastic work in deforming the shell. The absorbed energy is equal to the area under the load-displacement curve. The maximum acceleration of the rigid mass is equal to the maximum load divided by the mass. Thus, the load-displacement behavior of a shell pressed against a

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*Numbers in parenthesis refer to references at the end of the paper.

flat rigid surface plays an important part in the design of such limiters. In the static case we may consider the base of the hemisphere to be fixed and the apex loaded by means of a rigid plate moving parallel to the shell axis of revolution.

In order to determine the feasibility of using a hemispherical shell as impact limiter, a preliminary test was performed at the University of Pittsburgh using an existing aluminum hemisphere, 24 inches (61 cm) in diameter and .05 inches (.127 cm) thick. The results, discussed in Section II, indicated the following.

The shell first deformed into a contact region against the rigid surface, then buckled into an axisymmetric dimple, and finally, due to the large circumferential membrane stresses developed, the dimple buckled into a number of lobes (five lobes were developed in the specimen tested). Since the radius to thickness (r/t) ratio for this specimen was much larger than anticipated in the actual design, the results could not be considered totally representative of relatively thicker shells. This observation proved to be correct as later tests on shells with lower (r/t) ratios produced axisymmetric deformations.

However, based upon this test, several important conclusions could be drawn. First, the gross behavior of the hemisphere could be characterized by assuming that the hemisphere deformed as shown in Figure 1. The portion AB of the shell is essentially undeformed. At B, the meridian bends to a sharper radius. At C there is contact between the shell and the rigid plate, and at D the curvature of the meridian is reversed. The portion DE is unloaded and does not undergo further plastic deformation as the shell is loaded. The characteristic circles B, C, and D move radially outward along the shell meridian as the loading on the shell is increased. During unloading, the shell remains essentially in the shape depicted in Figure 1, i.e., there is negligible elastic springback.

To develop an analytical design procedure, the load deflection characteristics of the shell must be modeled through a deformation range of tens to hundreds of times the shell thickness. The point of application of the load is constantly changing. A finite element analysis of this structure is very difficult. However, because of the symmetry of the problem, the relatively simple deformed shape and the fact that the plastically worked regions are relatively well defined, a limit analysis procedure was found to be a viable alternative.

II. STATIC EVALUATION

A. Finite Element Analysis

When confronted with a design/analysis problem of this level of complexity, the usual reaction is to reach for one of the large structural analysis programs currently available and begin to model the problem using finite elements. The present authors were not an exception. The ANSYS[2] computer program was chosen for use because of our familiarity with the program and because of the wide range of features available (plasticity, large deflection capability, gap elements, etc.) that would be necessary to model the problem.

The first step in modeling such a complex problem is to qualify the program to handle the expected range of loadings/deflections. Smaller, simpler, and less expensive problems should be run to determine that the program will produce the results desired in the larger problem.

A "trial" problem was formulated which consisted of modeling the hemisphere with axisymmetric isoparametric elements using two elements through the shell thickness. The loading consisted of a point load acting at the apex of the hemisphere. The loading was applied in small increments and the large

deflection option in ANSYS was utilized. A bi-linear stress-strain curve was used to represent the material.

The results of this analysis were n encouraging. When the load was at or near the plastic collapse load of the structure, a small change in load in- ment produced large geometry changes. The plasticity convergence ratio (a sure of how closely the program is following the stress-strain curve of the material) was unacceptably large. Because of the relatively large geometry changes between load increments, the element geometry became distorted locally. The deformation patterns which resulted were not felt to be representative of actual behavior. It was evident that the actual design and analysis costs would be prohibitive.

This conclusion was based upon the running time per load step and the number of load increments required to approximate the loading, deformed geometry, and stress-strain curve and the fact that the actual problem would require a more extensive model, as discussed later.

In an attempt to reduce the computer time required, a second "trial" problem was formulated using the triangular plastic shell element in ANSYS. The symmetry of the problem was again utilized to reduce the number of elements necessary to model the hemisphere. A 15 degree sector of the sphere was modeled. As before, the loading was applied at the apex only. The plasticity and large deflection options were utilized.

The results of this second problem were also unacceptable. Strain discontinuities between elements and excessively large element rotations were calculated. It was evident that the shell element was not adequate to model the hinge circles which form in the shell and move radially along the meridian. The computing time was also estimated to be too large to consider this type of analysis as a design tool.

The analysis of the hemisphere, while very expensive, is however, not beyond the capability of current finite element analysis. It was determined that the physical problem could be represented quite adequately. The model envisioned for final analysis would have consisted of axisymmetric isoparametric elements, probably using four elements through the thickness. The loading would be applied through a rigid plate which would be connected to the outer surface elements on the sphere by interface elements. The interface elements have the capability of carrying only compressive (or tensile) loading and sensing when an initial closed gap has opened and vice-versa. Using these interface elements on the outer surface of the sphere, the moving contact point between the rigid platen and the shell can be accurately represented. This, of course, introduces yet another nonlinear effect into an already complex problem. The solution time

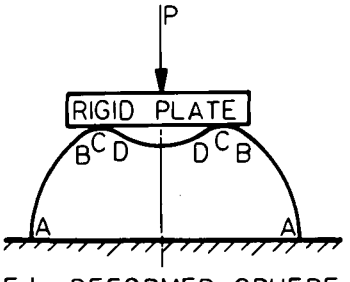


FIGURE I -DEFORMED SPHERE

was estimated as one to several hours on a large, high-speed computer (such as the CDC 7600). Thus, at commercial rates, a cost of well over $1,000 was estimated for each shell geometry investigated. Clearly, this is not an effective design tool.

B. Limit Analysis Methods

The determination of the static load-displacement behavior of the sphere requires the determination of a sequence of limit loads for the sphere in successive deformed configurations.

Updike[3] determined the load-deflection behavior for statically loaded thin-walled hemispheres. He found that the load versus deflection behavior is given by

$$P = 10.9\,\sigma_o t^{3/2}\,\Delta^{1/2}\,, \tag{1}$$

and the energy absorbed is

$$W = 7.26\,\sigma_o t^{3/2}\,\Delta^{3/2}\,. \tag{2}$$

where σ_o = yield stress of the material, t = shell thickness, and Δ = the displacement of the rigid loading plate relative to the base of the hemisphere (Figure 1).

Jones and Wagner[4] performed a similar analysis of a thin-walled cone (Figure 2). The load deflection behavior is given by

$$P = 2K\,\sigma_o t\left\{\left[\frac{tr_c\,[2+(2\theta-\pi)\,\tan\theta]}{2\cos}\right]^{1/2} + t/4\right\}, \tag{3}$$

where

$$r_c = r_c^o + \frac{\tan\theta}{2}\Delta\,. \tag{4}$$

K = 2.5; r_c^o is the initial value of r_c and the other parameters as previously defined.

Using Equations (3) and (4), the load deflection curve for a hemisphere can be approximated as follows. Erect a set of apex tangents to the hemisphere. The limit load is then calculated from Equation (3) at the point of tangency of a cone with the hemisphere. The deflection is a straightforward geometrical relation, $\Delta = R(1-\cos\theta)$. Figures 2 and 3 illustrate this procedure.

The preliminary test conducted at the University of Pittsburgh was then compared to the predictions of Equations (1) and (3). The aluminum sphere had been sectioned after testing and tensile specimens prepared and pulled. The resulting stress-strain curve indicated a yield of stress of approximately 23,000 psi. The stress-strain curve was approximately bi-linear and could be approximated well as rigidly-perfectly plastic. The results are shown in Figure 4. The undulating nature of the measured curve is attributed to the buckling of the hemisphere.

C. Static Tests

Having gained some confidence in the accuracy of these methods, design calculations were then performed for the actual container. It was determined that a radius to thickness ratio of approximately 39 was required to meet the

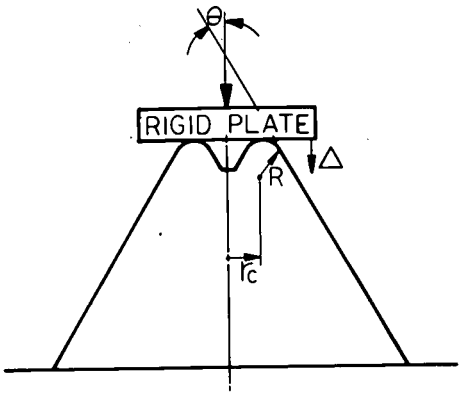


FIGURE 2—CONICAL SHELL GEOMETRY

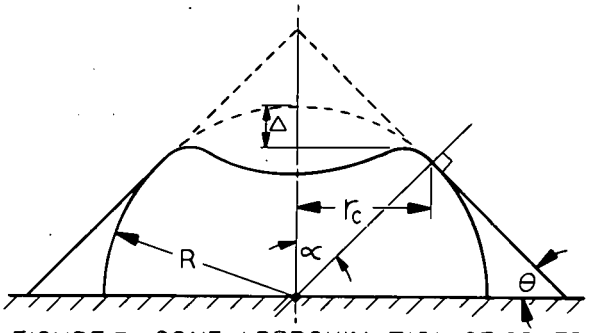


FIGURE 3—CONE APPROXIMATION OF SPHERE

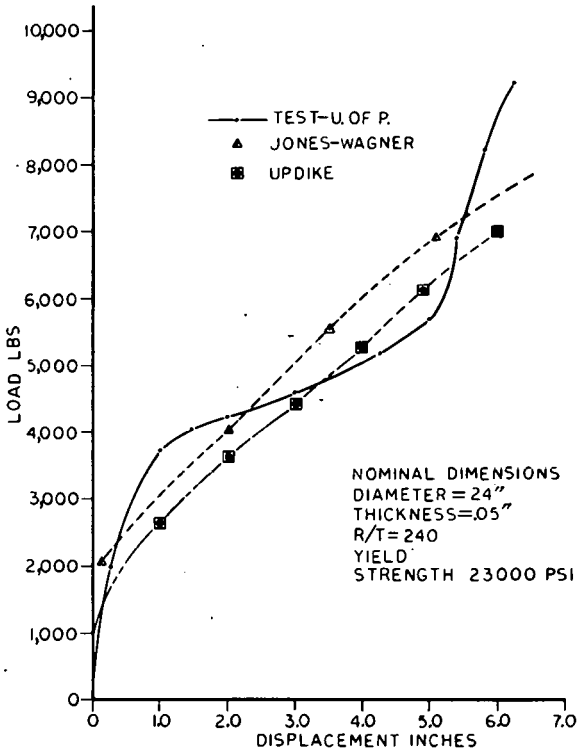


FIGURE 4-STATIC LOAD DISPLACEMENT CURVES FOR UNIVERSITY OF PITTSBURGH TEST.

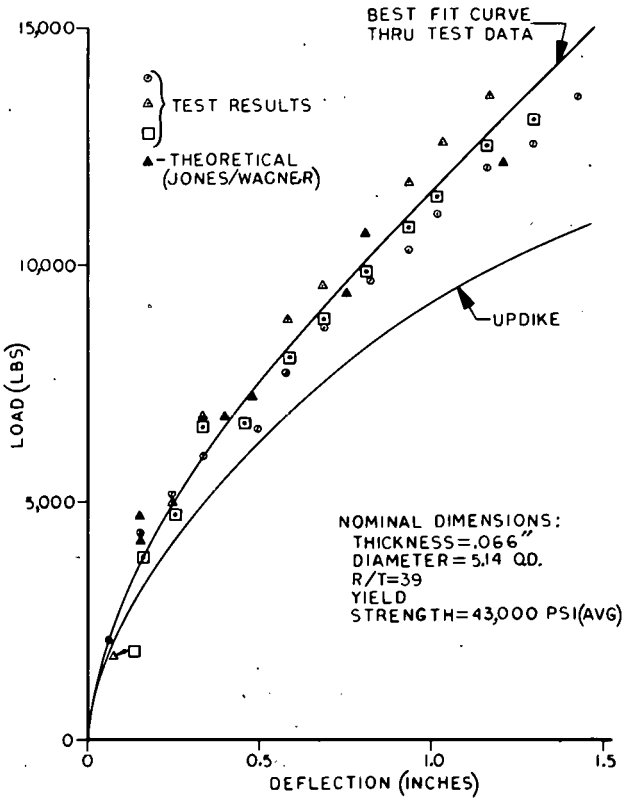


FIGURE 5-STATIC LOAD DISPLACEMENT CURVES FOR 5" TEST SPECIMENS

energy absorbing requirement and not exceed the acceleration limit. In order to verify these calculations, a series of scale model tests were run. The test specimens used were approximately five inches (12.7 cm) in diameter with a thickness of .066 inches (.167 cm). Coupons taken from the specimens were pulled in a tensile testing machine to determine the yield point of the material. Brinelle hardness tests were conducted at points along the meridian of shell to determine the change in yield strength due to the spin-forming operation. These measurements were calibrated using the results of the tensile specimen tests.

The results of these tests are shown in Figure 5. As can be seen, the test data tended to fall into a narrow band with a relatively small amount of scatter. The theoretical predictions of Updike (Eq. 1) tended to underpredict the measured values while the Jones and Wagner prediction was closer to the experimental results. However, it is noted that the shape of the force-deflection curve is well represented by Eq. (1) and the form of the equation is convenient for calculations. Hence, it was decided that the force and energy equations from Updike (Eqs. 1 and 2) would be used for design purposes but the constant terms in both equations should be increased by approximately 60% to account for the higher measured values. The following equations resulted.

$$P = 17.53\sigma_o t^{3/2} \Delta^{1/2} \quad , \tag{5}$$

$$W = 11.68\sigma_o t^{3/2} \Delta^{3/2} \quad . \tag{6}$$

III. DYNAMIC EVALUATION

A. Dynamic Analysis

In the previous analyses the effects of elastic strain, strain hardening and strain rate sensitivity are neglected. The static tests indicated that the first two of these approximations do not introduce significant error into the analysis. The effect of strain rate, however, appears to be quite important. This effect may be included in an approximate manner by suitably adjusting the yield stress.

The dynamic yield strength for mild steel was obtained from Ref. 5. While it appears that the ratios of dynamic to static yield strengths are higher than those found from other sources, the experiment used to develop these data closely approximates an actual 30-foot drop. Therefore, it was concluded that the dynamic yield strength obtained from Figure 25 of Ref. (5) would be used for the dynamic analysis.

The procedure used to calculate the dynamic load-deflection curve is as follows. The impact velocity of the cask is calculated by equating the initial potential energy before release to the kinetic energy at impact and solving for the impact velocity. For a 30-foot (9.14M) drop this is 44 feet/sec (13.40 M/sec). The dynamic yield strength of the hemisphere is calculated by multiplying the ratio of dynamic to static yield strength by the static yield strength determined for the material. Using Equations (5) and (6) the load required to deform the sphere and the energy absorbed in a small increment of deformation (d) is calculated. The remaining kinetic energy of the cask is determined by subtracting the energy absorbed in this increment of deformation from the initial kinetic energy. The reduced velocity of the cask may then be calculated based upon the remaining kinetic energy, and a revised dynamic yield strength can be estimated. This procedure is repeated until the kinetic energy has been totally dissipated. This procedure was programmed on a time-sharing computer for convenience. Figure 6 shows the load deflection curve obtained using this procedure for the five-inch diameter test specimen used in the dynamic tests. Integration steps of .05 inch (.127 cm) were used in the analysi

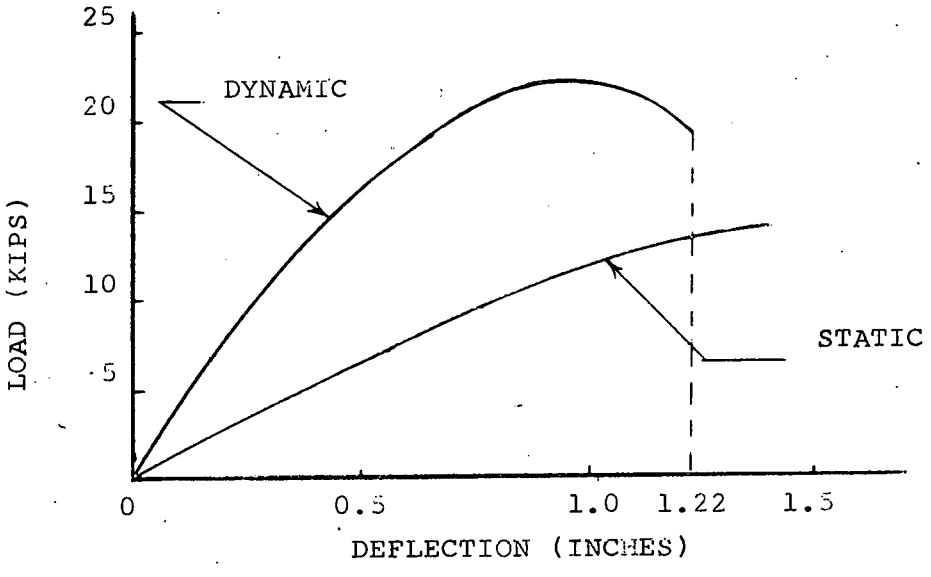


FIGURE 6. DYNAMIC ANALYSIS COMPARED TO STATIC TEST RESULTS FOR 5" SPECIMEN

The reactive force is much larger in the dynamic case than the static case, as expected. It is also noted that the largest force does not occur at the maximum deformation, as would be expected from the static tests.

B. Dynamic Testing

In order to verify the dynamic calculations, scale model tests were performed. The test consisted of dropping a 50 lb. (22.7 Kg) weight through a distance of 30 feet (7.62 M) onto hemispheres of the same dimensions used in the static tests. The test set-up is shown schematically in Figure 7. Two drop tests were conducted. The deformations were measured to be 1.19 inches (3.02 cm) and 1.27 inces (3.22 cm) for an average of 1.23 in. (3.12 cm). Using the dynamic analysis method described in the previous section, a deformation of 1.22 inches was calculated. The ratio of the energy absorbed in the dynamic analysis to that in the static analysis was found to be approximately 2.0, and the ratio between the maximum dynamic and static force was found to be 1.56.

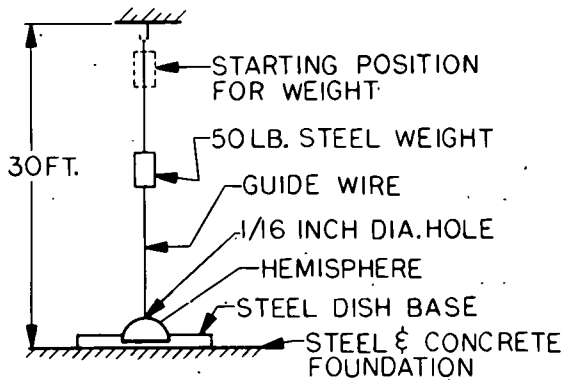


FIGURE 7 -SCHEMATIC DIAGRAM OF DYNAMIC TEST SET-UP

These parameters were used in the design of the actual limiter.

IV. RESULTS AND CONCLUSIONS

Based upon the excellent agreement between the analysis and the scale mo testing, it was concluded that the impact limiter could be designed to perform as required. The actual design of the limiter indicated that a material thickness range of 1.5 to 2.2 inches (3.81 to 5.59 cm) could be used without exceeding the 60 g impact requirement or the deformation limit of approximately 18 inches (45.72 cm). It is noted, however, that care should be taken in designing limiters of this type to use the actual yield strength of the material. The actual material properties are invariably higher than the specified minimum properties and the variation is usually large enough to have a significant effect upon design.

The design procedure outlined herein is recommended for use in the design of hemispherical shell impact limiters. Equations (5) and (6) should be used, along with the dynamic yield strength of the material, in the dynamic design analysis of the limiter. Equations (3) and (4) are useful in that they provide an additional check on the accuracy of the calculated static forces, which is important in gaining confidence in the design especially if no model testing is done. The computer program described in Section III can be made available for a nominal fee to handle reproduction and mailing by contacting the authors. Running time is extremely fast and core requirements are minimal.

The advantages of using hemispherical shell limiters include ease of fabrication, predictable performance, and a well-shaped dynamic load-displacement curve. The limiter is easy to inspect for damage and will not deteriorate during long (normal) usage or weathering. The hemispherical shell limiter can withstand off-normal impact up to relatively large drop angles. Many ductile metals can be used for construction. Very thin limiters should be avoided, however, because of the unpredictability of buckling. The relative weight of a hemispherical limiter compared to other limiter designs has not been investigated.

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CORRELATION OF EXPERIMENTAL SHIPPING CASK MODEL STRUCTURAL BEHAVIOR BY COMPUTER ANALYSIS

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SUMMARY

As an aid in the planning and investigation of an experimental research program on the effects of environmental accidents upon spent fuel shipping containers, a study is being made to correlate the results of tests of scale model shipping casks with those of complex structural computer models. After surveying the available codes, a general purpose, dynamic, nonlinear computer program (ADINA) was chosen to model several experiments. Among these are: stainless steel cylindrical containers dropped on end from a height of 12 inches; lead and uranium filled cask models dropped 12 inches and 30 feet, in both end and side orientations.

The results will be used for selection of the test parameters by the prediction and planning of failure test conditions, as well as confirmation of previous test results.

Together, the experimental and theoretical analyses will provide a realistic assessment of the effects of accident loadings on spent fuel shipping containers.

INTRODUCTION

As part of ongoing research sponsored by the Department of Energy, Division of Environmental Control Technology, into the structural response of spent fuel shipping containers subjected to severe environmental accidents, Battelle-Columbus Laboratories is conducting a program of experiments and mathematical analyses to study the dynamic effects of such conditions on shipping containers. To date, studies have been completed in the areas of material impact properties, scale modeling analysis methods, and model design. In addition, several experimental drop tests have been completed using scale models of typical shipping containers. A detailed discussion of some of the results of the test program can be found in the paper presented by R. A. Robinson and J. A. Hadden.

An additional task has been undertaken to use an available dynamic nonlinear structural analysis computer code to assist in planning the experimental test conditions. Only a limited number of cask model specimens are available and the general program test plans are to obtain data up to loss of cask integrity. Thus computer analyses are being used to make the most efficient use of the specimens in data collecting by predicting the specimen response to several testing conditions.

Since the purpose of this study does not include development or modification of existing computer codes, several candidate programs were screened to determine the most suitable, i.e., one that employs "state-of-the-art" computer modeling techniques to compare with experiments. This paper will cover our initial research, which was directed toward the decision of which computer program to use, the initial modeling of simple experiments, and our continuing investigation into the more complex model behavior.

In a related research program, computer modeling techniques designed specifically for shipping containers are being developed at Los Alamos Scientific Laboratory. One of that program's objectives is to develop computer codes which will facilitate the efficient, repetitive analysis and assessment of shipping casks undergoing specific impact conditions. These codes will incorporate experimental data from the Battelle cask model testing program.

COMPUTER CODE SELECTION

Our research into computer program capabilities was based on the following constraints:

(a) The programs must be "standard", that is, commercially available in the same form at different locations, allowing for periodic updates and revisions. However, the results obtained at different installations should not vary significantly. The main point of this limitation is that local, special purpose programs would not be considered.

(b) Programs must be capable of handling all types of experimental models. This implies three-dimensional, dynamic, and nonlinear material and geometry capabilities, precluding use of simpler codes (e.g., two-dimensional, shell elements, etc.) to handle the elementary cases.

(c) Programs should be current, based on the latest structural technology. The purpose of this task does not include computer code development and modification to suit shipping cask accident models.

The following computer programs were selected for further study based on the above restrictions:

- ANSYS[1]
- MARC[2]
- NASTRAN[3]
- ADINA[4]
- PETROS[5].

Of these programs, only MARC and ADINA could handle dynamic, three-dimensional, geometric and material nonlinearities, and large strain problems. Although ADINA has a more limited capability than MARC (e.g., no nonlinear shell elements, pre- and post-processor, or sliding interface connections), due to the fact that ADINA is operational on BCL's computer, and since our experience has shown that it is more convenient to use and faster running than MARC for simple problems, ADINA was selected for correlation of experimental results.

MODEL SELECTION

Containment Model, End Impact

The analytical effort started with the computer predictions of simple geometry experimental results, in order to compare output with a minimum of assumptions. The first model analyzed represents a linear scaled version of an inner containment shell

of typical fuel element shipping casks. This scale model consisted of a Type 304 stainless steel cylinder with flat heads, no lead or uranium shielding, and no contents (Figure 1). It was dropped in a vertical position from a height of 12 inches, onto a thick steel pad which was effectively a rigid surface, the condition assumed in the computer model. Also, the experimental model used a solid steel cone on the bottom to produce a uniform impact loading.

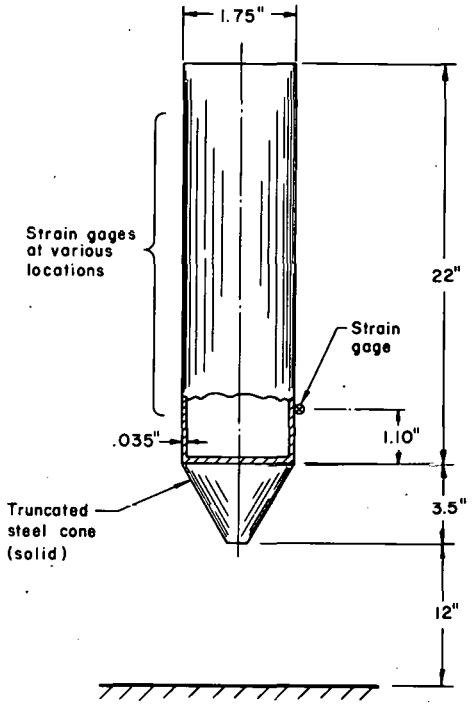


FIGURE 1. SCHEMATIC OF 1/8-SCALE CONTAINMENT MODEL

Although the loading was dynamic, it was not of sufficient magnitude that any geometric or material nonlinearities were needed in the model.

No temperature effects were included in the computer analysis; room temperature properties were used.

Since the geometry and loading are axisymmetric, the computer model used was two dimensional.

Cask Models, End Impact

The studies of the stainless steel containment model were intended to provide an indication of the agreement between theoretical and experimental data. In the second series of models which were studied, the models approximated actual shipping cask designs, in which the containment is surrounded by a shielding material (either lead or uranium), which in turn is enclosed by an outer stainless steel shell. Figure 2 is a sketch of a typical scale model (1/8-scale, uranium shield).

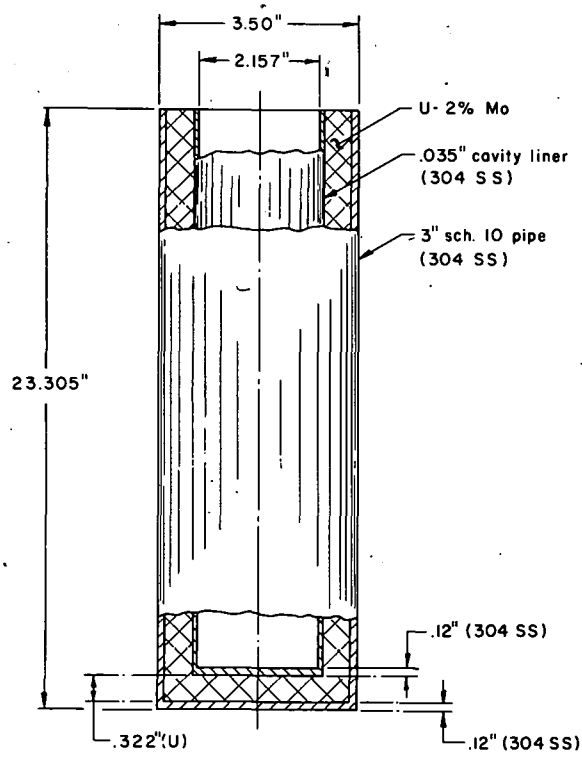


FIGURE 2. SCHEMATIC OF 1/8-SCALE URANIUM-SHIELDED CASK MODEL

The testing matrix for these cask models includes end drops from low heights (up to 12 inches) and 30 feet, in addition to non-axisymmetric drops discussed below. The 1- and 30-foot end drop tests have been completed for the lead shielded models, while the uranium models' testing is planned to begin this summer.

For the 1-foot drops, elastic material properties will be used in the computer analysis. Subsequent analysis will involve nonlinear behavior to compare with experimental data obtained by dropping the models from 30 feet onto a one-half-inch steel plate backed by over seven feet of reinforced concrete. Although large strain and displacement capability may not be necessary for the steel and uranium models, the lead behavior will be nonlinear. The dynamic behavior of lead will be modeled by a visco-elastic stress-strain relation, while uranium will be characterized as a brittle material.

Cask Models, Side Impact

The last complexity added to the computer analysis is the extension to three dimensional behavior, in order to model the effects of side and off-angle drops from heights of 30 feet, and greater.

ANALYSIS

Before a discussion of the analytical methods can begin, it is necessary to review the specific type of dynamic analysis performed by the ADINA code. Basically, the ADINA program uses an implicit integration scheme to solve directly for the displacements at each time step. Since this method is unconditionally stable, the time step is limited only by convergence of the solution.

This is an important point, not so much for finding an accurate solution, but because the purpose of this task is to correlate with experimental behavior. It is therefore necessary to model the containers so that the computer results are no more accurate than experimental data. The recorded experimental data excluded all frequencies greater than about 3000 HZ. However, it is quite easy to produce computer output displaying frequencies up to the MHZ range, thus including behavior unrecorded in the experiments.

Consequently, the bulk of this initial modeling effort involved "fine-tuning" a model that filtered essentially all frequencies above the 3000 HZ limit. Bathe and Wilson[6] present an excellent discussion of the relationship between computed output frequency and solution time step. For example, if the time step is equal to the period of a stress wave, it will be completely damped from the output. It was found that, if the chosen time step was $\geq 4 \times 10^{-5}$ sec, frequencies greater than 3000 HZ were damped sufficiently to allow comparison to experimental data after only a few time steps. Figure 3 shows this computer "damping" effect for a model of the stainless steel cask liner (54 nodes, 17 elements).

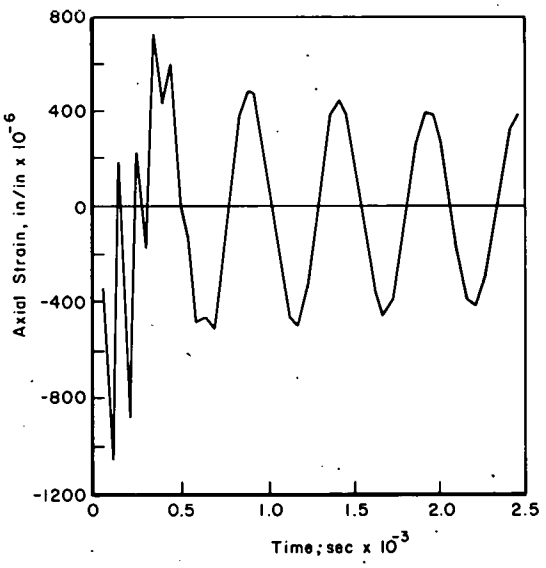
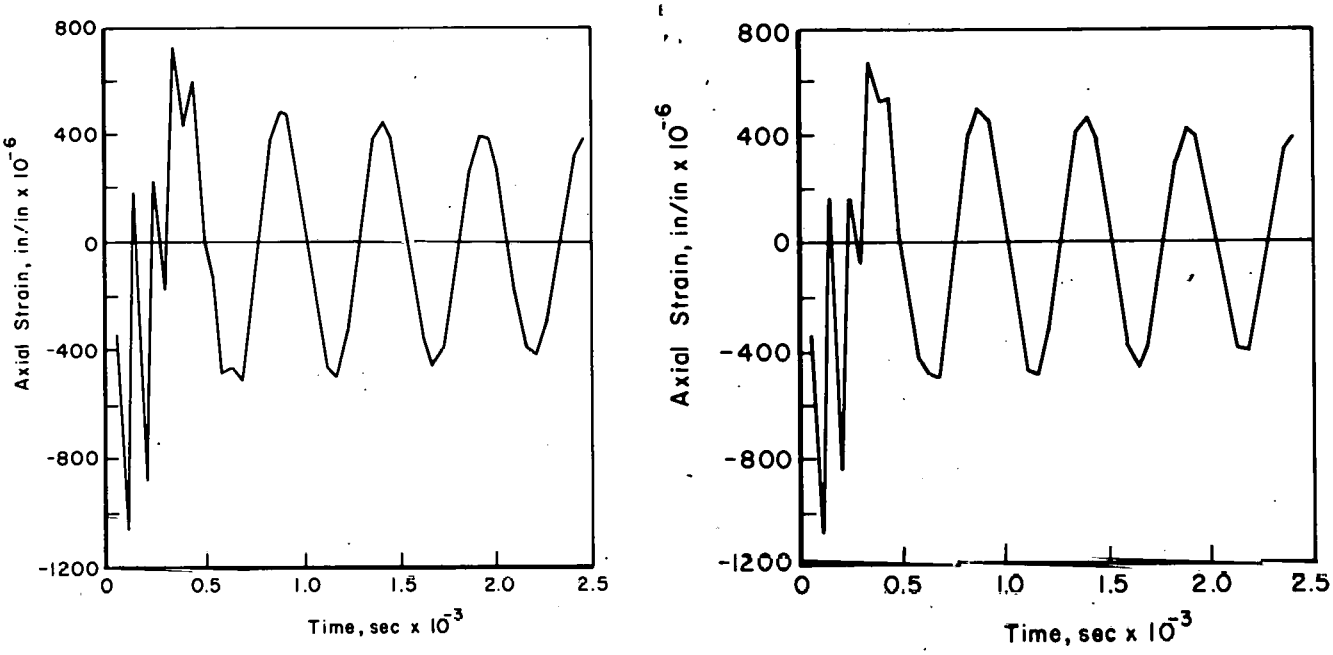


FIGURE 3. EFFECTS OF COMPUTER INTEGRATION TECHNIQUES ON OUTPUT STRAIN FREQUENCY AND AMPLITUDE: 54 NODES, 14 ELEMENTS, STAINLESS STEEL CONTAINER, 12-INCH DROP, LOCATION = 1.10 INCHES FROM IMPACT SURFACE

Containment Model, End Impact

Since this was the initial analysis of a test model, it was decided to perform the analysis using three models of varying detail. These ranged from a 308 node, 252 element model, to one with only 54 nodes, 14 elements. As mentioned above, and confirmed by actual computer analysis, the output results (model natural frequency, stress, strain waves, etc.) were negligibly affected by model geometry; rather, the time step was the dominant parameter. Figure 4 shows a comparison between two of the model descriptions with reference to the strain history based on identical time steps. Running time was roughly proportional to the number of nodes. From this comparison, it appears that a coarse model description is adequate for experimental correlation.



(a) Coarse Mesh (54 nodes, 14 elements) (b) Fine Mesh (286 nodes, 76 elements)

FIGURE 4. COMPARISON BETWEEN COARSE (a) AND FINE (b) MESH GEOMETRIES: TIMESTEP = 4.8×10^{-5} SEC, 50 TIMESTEP INCREMENTS, EMPTY STAINLESS STEEL CAN, 12-INCH DROP

Cask Models, End Impact

At this time, computer analyses have not yet begun for the lead-shielded models. Since this task was initiated after most of these experiments were completed, it was more in keeping with the project goals to forecast uranium-shielded model behavior. Two types of computer models were designed for this analysis: one which described each shell of the shipping cask model, including uranium; another which treated the model as a single, composite material. The former consists of 400 nodes, 151 elements, the latter has only 60 nodes, 15 elements.

In these uranium shield models, the interface nodes are rigidly connected because: (1) the uranium/steel shells are interference fit, hence no sliding is expected to occur during end impacts; (2) the ADINA code does not presently have the capability to model sliding joints. In the composite model, material properties were averaged by weight, hence the values are close to those of uranium.

Initial study of the uranium shielded models has described only linear, elastic behavior, such as will be tested experimentally using the uranium models dropped from heights of about 12 inches. The computer results will be used to estimate the drop height limit at which inelastic behavior begins to occur. The material model for uranium does not as yet include brittle phenomena, since this is not critical for the low magnitude impact loadings in the initial testing phase. Before the accident drop tests begin, computer models of the uranium casks will contain appropriate plasticity relations to describe uranium behavior. Following these analyses, similar models of the lead-shielded casks will be created to correlate nonlinear behavior obtained from those experiments.

One additional distinction between the computer models and the test models should be mentioned. The actual impact of a container onto a target involves multiple impacts (of decreasing intensity). To accurately model this phenomena requires a nonlinearity to be introduced at the impact surface boundary condition. That is, as the model compresses, then "rebounds" from the impact surface, the model will break contact with this surface. For simplicity, the computer model does not allow the impact surface to displace in the axial direction. These secondary impacts, however, occur beyond the time frame of even a few cycles. The boundary condition does cause minor differences in results: it does not appear to cause any difference in either the wave speed, or compression magnitude, but does affect the frequency of cycles beyond the initial one (see Results below).

Damping is not explicitly included in the computer model. Any damping that appears is actually due to integration error effects (discussed earlier). In the experimental model, of course, real damping is included. The net effect is that the apparent computer damping partially offsets the lack of explicit damping in the model.

In a series of tests designed to simulate more severe accident conditions, the uranium-filled containers will be dropped from heights greater than 30 feet, which necessitates the use of both material and geometric (large strain/displacement) nonlinear capabilities, as well as greater material strain-rates insofar as dynamic properties are concerned.

Cask Models, Three Dimensional Geometry

The effects of three dimensional analysis will be incorporated by modeling the lead-filled containers under a side and off-angle impact orientation.

The final series of computer models will involve the same types of three dimensional analyses as before, but uranium shielding is used instead of lead. The material model for uranium will include brittle material behavior.

RESULTS

Stainless Steel Container Model

The computer results for the single stainless steel container model were obtained assuming impact onto a rigid surface. Consequently, the theoretical analyses can only be compared to those experiments which used a thick steel slab as an impact surface. Figure 5 shows a comparison between computed and theoretical results, for a 12-inch drop test. It should be remembered that the oscillations occurring in the first cycle are the high frequency components which decay due to integration techniques. The time step is such that "absolute" damping occurs for frequencies greater than 21,000 HZ, moderate damping for frequencies between 3,000 and 21,000 HZ, and only slight attenuation for frequencies below 3,000 HZ. (In addition to decay of wave amplitude, there is a corresponding increase in the wave period.) The chosen time step was selected to minimize attenuation of those frequencies recorded experimentally, while "filtering" out those higher frequencies as quickly as possible.

Good agreement between computed and experimental strain amplitude was achieved. However, the computed strain wave frequency was about 30 percent lower than the experimental value, beyond the initial cycle. Using the ADINA code to predict the natural frequency of the computer models, a value of 2100 HZ was found. As Figure 5 indicates, there is approximate agreement between the computer and experimental results, for the first cycle, insofar as strain amplitude and frequency are concerned. Beyond the first cycle, the experimental model rebounds from the impact surface, while

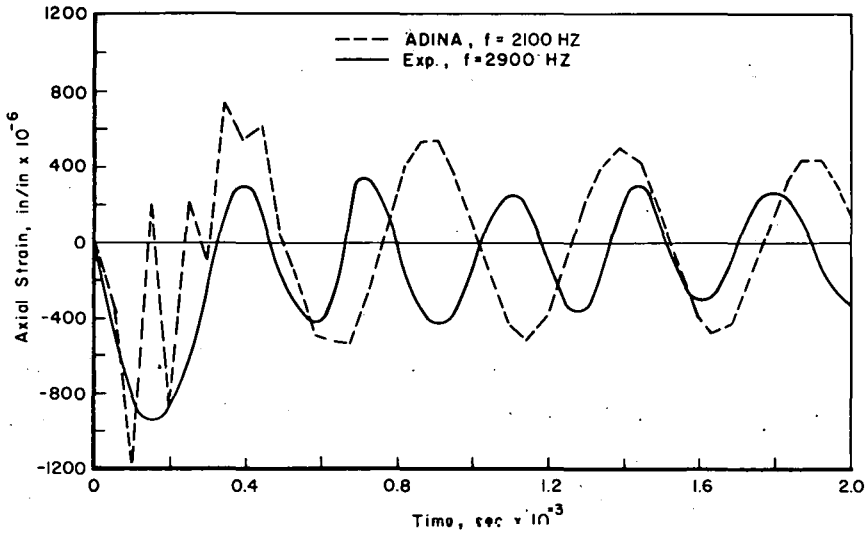


FIGURE 5. EXPERIMENTAL VERSUS COMPUTED RESULTS FOR 1/8-SCALE MODEL DROPPED ON END FROM HEIGHT OF 12 INCHES ONTO HARD, UNYIELDING SURFACE

the computer model remains fixed. Hence, the resonant frequency of the model is expected to change. In addition, damping causes experimental amplitudes to decay below computer results.

Since the experimental wave frequency (2900 HZ) is very close to the upper limit of the instrumentation, the computer model was used to investigate the high frequency components that were not recorded. Figure 6 is a graph of the strain wave showing the high frequency attenuation effects of varying the time step. For the smaller time step, wave frequencies as high as 20,000 HZ are displayed. This high frequency component adds peaks of less than 30 percent to the amplitude of the carrier wave (2100 HZ). (Note, the strain waves shown in Figure 6 were obtained after the initial, or converging, cycle had passed, as shown in Figure 3.) The relative effects of higher frequency stress waves on model behavior during impact has not been undertaken in this task.

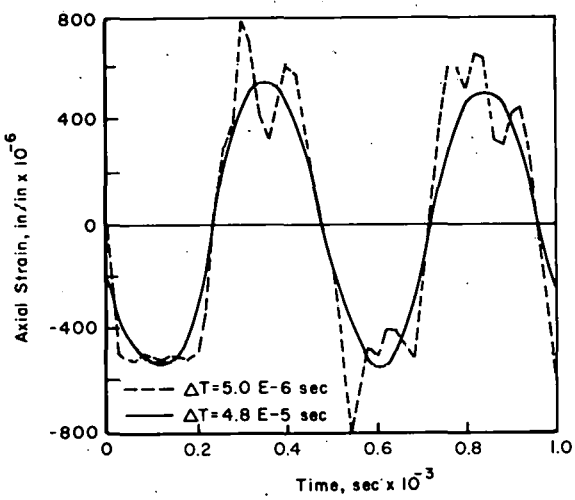


FIGURE 6. EFFECTS OF HIGH FREQUENCY ATTENUATION FOR DIFFERENT TIMESTEPS

Results are expected during the next few months to predict experimental behavior.

CONCLUSIONS

Overall, the magnitude of the impact loading is comparable for both experimental and computer output. Consequently, the computer models can be used to study the impact stresses and strains throughout the cask model, rather than be limited by strain gage locations. In addition, the computer models offer a wide range of impact loadings at low costs.

The ability of present-day general purpose nonlinear, dynamic computer programs to assist in the planning and evaluation of experimental shipping container modeling studies is the objective of this research task. Additional experimental results will be predicted, and variations between computer and experimental output will be used to further investigate the effects of impact loads on shipping container design. These analyses will expand to include uranium and lead inelastic behavior, three dimensional geometry, and greater impact effects.

In conclusion, the combination of theoretical and experimental analyses will provide a more realistic assessment of the feasibility of using present-day modeling techniques and an improved understanding of the actual structural response of shipping containers under adverse environments.

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THE USE OF ENDOCHRONIC PLASTICITY THEORY IN MODELING THE DYNAMIC INELASTIC RESPONSE OF SHIPPING CONTAINERS: A PRELIMINARY ASSESSMENT*

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Comprehensive structural design and risk assessment of shipping containers used in energy material transportation systems, particularly in the shipment of radioactive materials, requires accurate computer modeling of dynamic inelastic material response. In particular, predicting total plastic deformation of a complex shipping container subjected to the rapid transient loads associated with a variety of severe accident conditions requires extensive computer time and places extreme demands on both software and the basic physical models underlying the analyses. The representation of dynamic inelastic materials response behavior in a form that is compatible with state-of-the-art computational methods and also accurately incorporates all of the features revealed in materials tests is thus of vital importance to the development of rational risk assessment. In an attempt to make maximum use of recent developments in continuum mechanics describing dynamic plastic material behavior, a program has been initiated at Argonne National Laboratory by the Transportation Branch of the DOE Division of Environmental Control Technology to assess the merits of endochronic plasticity in predicting the inelastic response of shipping containers designed for the transportation of hazardous materials. This paper reviews relevant details of the theory and discusses a recently developed finite-element program which has been modified to use an endochronic constitutive formulation. A series of sample problems for a variety of dynamic loadings are the basis for a preliminary evaluation of the theory in terms of its potential for saving computer time and its ability to accurately predict observed experimental behavior.

Classical plasticity, which forms the basis for virtually all theoretical models now used in general purpose finite element codes, requires that one trace the evolution of the material yield surface in stress space and the location of the stress state of each point in a structure with respect to this surface. Calculations relating to yield surface behavior and logical checks of position on the yield surface are costly in computer time and storage requirements. Endochronic plasticity, through its formulation in terms of a time-like measure which is itself a material property, dispenses with the need for a yield surface and, hence, there is a great potential savings in computer time. This potential saving has been explored by modifying a recently developed convected coordinate finite element program to accept an endochronic formulation of plastic response. The resulting program, which appears to be the first finite element program based on endochronic plasticity, has been used to compute the stress and deformation history of a series of rotationally symmetric structures subjected to rapid transient loads. Comparison of these results with equivalent classical plasticity solutions and with corresponding experimental data have been made.

*Work performed under the auspices of the U.S. Department of Energy, Division of Environmental Control Technology, Transportation Branch.

In the following sections, we shall first outline the endochronic plasticity theory, then based on this theory, the constitutive equations for plane stress and plane strain problems will be presented. The next section gives a brief description of the corotational finite element method. The final section of the paper describes a series of sample calculations currently being completed to assess the potential of endochronic plasticity in evaluating the dynamic response of shipping containers.

I. AN OUTLINE OF ENDOCHRONIC PLASTICITY THEORY

A major field that has attracted intense research during the past decades is material science. While this interest has expanded our knowledge of material behavior, the overall picture is far from complete. For example, for engineering purposes, materials have been classified as elastic, viscoelastic, plastic, viscoplastic, etc., and both linear and nonlinear mathematical models have been developed. One can use a very simple linear elastic theory to describe many observed phenomena but clearly such a model fails to describe most material behavior under extreme load conditions. It is reasonable to ask whether we can determine one single constitutive law which describes all observed elastic and inelastic phenomena. An attempt at this goal, initiated by Valanis [1], unifies the mechanical response of materials in terms of a general formulation called "endochronic" theory. It is unified in the sense that conventional elastic and plastic behavior can be deduced from the general theory by correct definition of specific material parameters.

The endochronic theory of plasticity is based on thermodynamics consideration together with general continuum mechanics principles. Deformation is considered as an irreversible process and, hence, the maximum available energy contained in the material will be decreased whenever deformation occurs. In other words, the entropy of the material system increases during the deformation process and the amount of entropy production can be obtained from laws of thermodynamics. Thus, deformation from the virgin state passes through "elastic" and "plastic" states as a continuous process even for unloading, and the current stress and strain states can be related by thermodynamic states of the whole material system.

In conventional elastoplastic theory, a distinction between elastic and plastic regimes must be made by introducing a yield function coupled with an assumed hardening rule. Since the definition of a yield surface and a hardening law are somewhat arbitrary, the resultant stress and strain history would contain major uncertainties. These uncertainties are greater for more complex loading history than the simple tension or compression. The endochronic theory by not requiring a yield function or hardening rule eliminates the aforementioned uncertainty.

The essence of endochronic theory is the introduction of the endochronic time scale in terms of which the material memory is defined and what determines the current stress state. This time scale is not the absolute time scale as measured by a clock, but is itself considered as a material property. The time scale is composed of a monotonically increasing function of deformation. For example, it can be arc length in a two-dimensional strain space. Because of this time scale, each component of strain (or stress) is coupled with all others. Thus the cross-hardening effect (i.e., straining in one component causing hardening in another component) can be taken care of easily.

As far as computational scheme is concerned, the conventional elastoplasticity problems are initiated using the elastic approximation. One then checks whether the stresses are inside the yield surface or not. If the stress state is outside of the yield surface, then the flow rule is applied to obtain the plastic strain. The differential form of the endochronic theory is similar to the incremental form of conventional flow theory; however, for a given strain (or stress) increment, the corresponding stress (or strain) increment can be obtained immediately and does not involve an iteration procedure. Also, there is a one-to-one correspondence between stress and strain for a prescribed deformation history. Because no "yield" is required, the endochronic theory is better suited to materials which gradually exhibit strain hardening (e.g., lead, mild steel, aluminum, copper, concrete, etc.).

In summary, the feature of endochronic theory that most differentiates it from classical stoplasticity theory is that only one constitutive law describes the whole spectrum of

material behavior without regard to whether a structure is being loaded or unloaded or whether it is in the elastic or plastic range.

II. THE CONSTITUTIVE EQUATIONS

The constitutive equations for isotropic materials under isothermal and small strain conditions are given by Valanis [1], with the assumption of elastic hydrostatic response, as follows:

$$s_{ij} = 2\int_0^z \mu(z - z')\,\frac{\partial e_{ij}}{\partial z'}\,dz' \,, \tag{1a}$$

$$\sigma_{kk} = 3K\varepsilon_{kk} \,, \tag{1b}$$

where s_{ij} and e_{ij} are deviatoric stress and strain tensors; $\underset{\sim}{\sigma}$ and $\underset{\sim}{\varepsilon}$ are stress and strain tensors; $\mu(z)$ and K are the heredity function and bulk modulus, respectively. The symbol z denotes a positive monotonically increasing time scale with respect to a time measure such that

$$d\zeta^2 = d\underset{\sim}{\varepsilon} \cdot \underset{\sim}{\underset{\sim}{P}} \cdot d\underset{\sim}{\varepsilon} \,, \tag{2}$$

where $\underset{\sim}{\underset{\sim}{P}}$ is a material tensor. Now if we let $\mu(z) = \mu_o e^{-\alpha z}$, and $dz = d\zeta/(1 + \beta\zeta)$, μ_o, α, and β being material constants, and substitute the above relations into equation (1a) we get

$$2\mu_o \frac{d\underset{\sim}{\varepsilon}}{d\zeta} = \frac{\alpha\underset{\sim}{\varepsilon}}{1 + \beta\zeta} + \frac{d\underset{\sim}{s}}{d\zeta} \,. \tag{3}$$

Under the assumption of plane stress, the above constitutive equations can be reduced to the following:

$$\begin{Bmatrix} d\sigma_1 \\ d\sigma_2 \end{Bmatrix} = \left(\frac{E_o}{1 - \nu^2}\right)\begin{bmatrix} 1 & \nu \\ \nu & 1 \end{bmatrix}\begin{Bmatrix} d\varepsilon_1 \\ d\varepsilon_2 \end{Bmatrix} - \frac{\alpha_1(\beta d\zeta)}{3(1 + \nu)(1 + \beta\zeta)}\begin{bmatrix} 2-\nu & 2\nu-1 \\ 2\nu-1 & 2-\nu \end{bmatrix}\begin{Bmatrix} \sigma_1 \\ \sigma_2 \end{Bmatrix} \,, \tag{4a}$$

$$(\beta d\zeta)^2 = \beta_1^2(d\varepsilon_1 + d\varepsilon_2)^2 \,; \tag{4b}$$

here, E_o and ν are the elastic modulus and Poisson's ratio, respectively. Similarly, for the plane strain case, we have

$$\begin{Bmatrix} d\sigma_1 \\ d\sigma_2 \end{Bmatrix} = \frac{E_o}{(1 + \nu)(1 - 2\nu)}\begin{bmatrix} 1-\nu & \nu \\ \nu & 1-\nu \end{bmatrix}\begin{Bmatrix} d\varepsilon_1 \\ d\varepsilon_2 \end{Bmatrix} - \frac{\alpha_1(\beta d\zeta)}{1 + \beta\zeta}\left[\begin{Bmatrix} \sigma_1 \\ \sigma_2 \end{Bmatrix} - K(\varepsilon_1 + \varepsilon_2)\right] \,, \tag{5a}$$

$$(\beta d\zeta)^2 = \beta_1^2(d\varepsilon_1 + d\varepsilon_2)^2 \,. \tag{5b}$$

Note that the material parameter α_1 and β_1 appearing in equations (4)-(5) can be determined from a simple tension test. It is also worthwhile to note that the stress increments are linear functions of the strain increments. The coefficients of these linear functions depend only on the strain and deformation histories. This makes the endochronic theory particularly attractive when used in the displacement approach of the finite element method, since the strain and deformation histories are known in this approach.

III. THE COROTATIONAL FINITE ELEMENT METHOD

One of the most common and powerful numerical methods is the finite element method in which the structure is divided into a finite number of substructures. Each substructure is

called an element, and has a simple, regular geometry. The solutions at some specific locations of the elements are chosen as unknown parameters of the element and are called nodal parameters. The solutions elsewhere in the element are then assumed to be known functions of the nodal parameters. These functions are called shape functions. The shape functions are arbitrary; they must be compatible among elements and converge to the actual solutions as number of elements increased. The external and internal work associated with each element can easily be computed in terms of the nodal parameters once the shape functions are defined. The governing equations for the structure in terms of the nodal parameters are then obtained and solved by assuming that the work of the structure is the sum of the work of each element.

It is obvious that the accuracy of a finite element model depends on the shape functions used. For linear problems, the shape functions for an axisymmetric shell model are easily constructed in terms of the displacements and rotations at the nodes of an element. However, the construction of shape functions becomes more involved when the displacements of the shell are no longer small. This complexity arises from the demand to satisfy Kirchhoff's shell assumptions; i.e., if we require the plane normal to the midsurface to remain normal to it during deformation of the shell, then the shape functions become so complicated that high order polynomials are needed. These high order shape functions increase the computation time greatly. Furthermore, classical constitutive equations, well developed for small displacement problems, do not apply since rigid-body motion of an element causes a pseudo-stress rate in conventional constitutive equations.

These difficulties are greatly reduced when the corotational coordinate finite element method is used [2,3]. In this method, a coordinate system that rotates but does not deform with the element is defined for each element. Simple deformation shape functions that satisfy Kirchhoff's shell assumptions are defined in each element in terms of its initial corotational coordinates. This deformation displacement can be shown to be obtainable by subtracting the rigid body displacement from the total displacement. Corotational strain and rate of deformation are then defined in terms of the deformation displacement. Since the rigid body motion is excluded from the deformation displacement, corotational strain, etc., the constitutive relationship becomes simple; hence, any constitutive laws derived for small displacement problems can be used in the corotational coordinate system for large displacement.

IV. NUMERICAL EXAMPLES

An existing finite-element code based on the corotational coordinate finite element method was modified to accept a constitutive formulation in terms of endochronic plasticity theory as derived in the previous section. While not in a form suitable for the full range of problems to be investigated during the evaluation period of this activity, the resulting initial modification is a suitable tool for comparing classical plasticity and endochronic plasticity for certain small-strain, large displacement problems.

The first example is that of a clamped spherical cap subjected to suddenly applied uniform pressure on its convex surface. The geometry, dimensions, and material properties of the cap are: radius of curvature, $R = 22.27$ in.; thickness, $h = 0.41$ in.; half angle of cap, $\alpha = 26.67°$; elastic modulus, $E_o = 10.5 \times 10^6$ psi; Poisson's ratio, $\nu = 0.3$; yield stress, $\sigma_y = 24{,}000$ psi; plastic modulus, $E_p = 2.1 \times 10^5$ psi; mass density, $\rho = 2.45 \times 10^{-4}$ lb sec^2/in.^{4}. The equivalent material parameters in the endochronic formulations are $\alpha_1 = 49$, $\beta_1 = 8.75$. The magnitude of the step pressure is 600 psi.

The apex displacements of the cap obtained by using the classical elastic, classical elastoplastic, and the endochronic constitutive laws for the plane strain case are shown in Fig. 1. The results obtained for the plane stress case are shown in Fig. 2. For comparison, the results of Nagarajan and Popov [4] were also presented in both Figs. 1 and 2. From these results, we see that noticeable differences are introduced when the endochronic theory is used. The results using an endochronic constitutive law compare favorably with the conventional elastoplastic isotropic work-hardening assumption as far as dynamic response at the apex of the cap is concerned. In the endochronic approach, due to the earlier deviation from the linear elastic line (as can be seen from the uniaxial stress-strain curve), the results show e "plastic" deformation than the classical theory. The mean value of displacement is thus

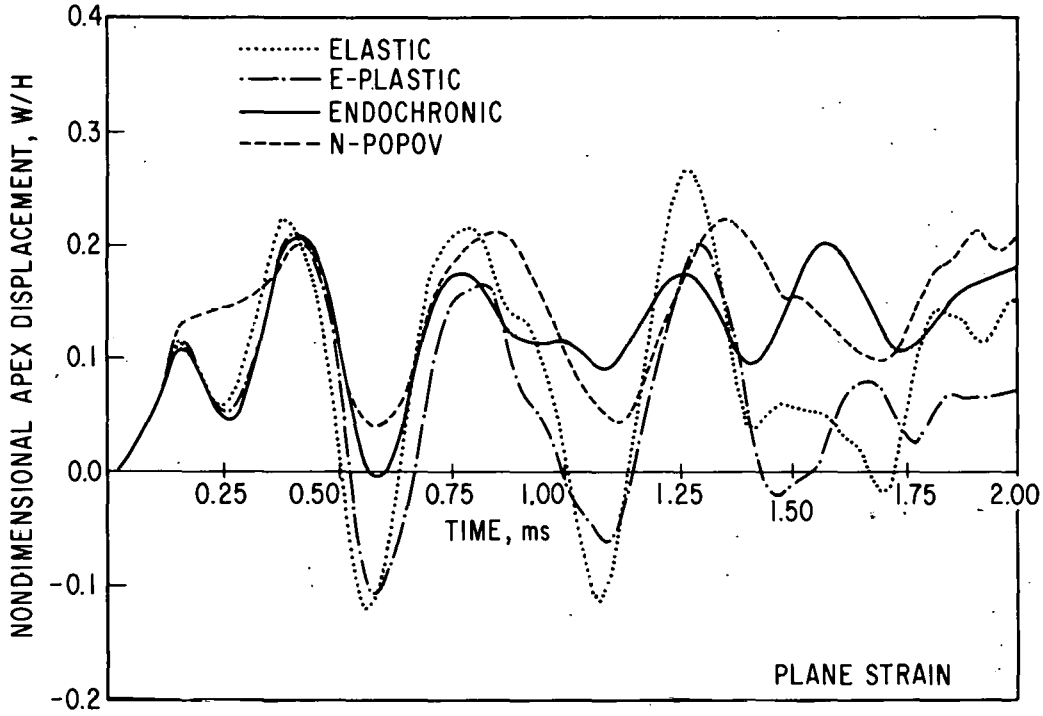


Fig. 1 Dynamic response of spherical cap under uniform external step pressure of 600 psi (plane strain).

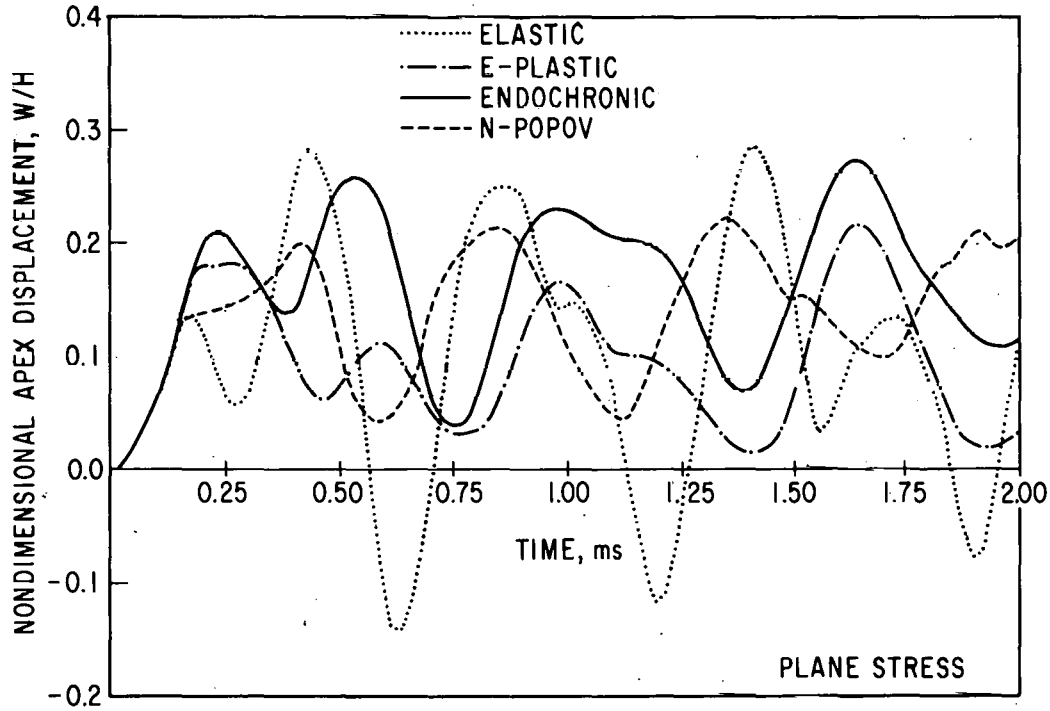


Fig. 2 Dynamic response of spherical cap under uniform external step pressure of 600 psi (plane stress).

higher in the endochronic theory than in the classical theory. It can also be seen that the amplitude of vibration is somewhat smaller in the endochronic result than in the classical result. This is due to the greater dissipation of energy and the resultant greater damping effect when compared to the elastoplastic and linear elastic cases. It seems that the use of endochronic theory produces a "softer" model compared to the models generated by the classical theories. This, in a sense, would make the endochronic theory more conservative than the classical approach in many design situations. For example, a shipping container designed on the basis of an endochronic model of dynamic plastic response would be slightly thicker than using the classical plasticity theory if a limit on total deformation under given loads was the basic design criterion.

Next, a hardening system was examined on the clamped circular plate laterally loaded over a circular central region with sheet explosive. It is assumed that the plane stress condition will be valid for a thin plate. Figure 3 shows the resulting nondimensional apex displacement versus time for 6061-T6 aluminum. The plate geometry is: thickness, h = 0.0625 in., diameter = 6 in., applied impulse I = 1.67 N·s over a central circular region of diameter 2 in. The experimental results, together with another finite difference numerical result DEPROSS [5], were taken from reference [6]. It is seen that all numerical results presented here are in good agreement with the experimental results. Figure 4 shows similar results for 1022 steel with impulse I = 2.20 N·s. Here, all numerical results compare favorably with each other. Although some discrepancy exists between experimental and numerical results due to the slippage on its clamped boundary as indicated in reference [6], the overall dynamic responses are still in good agreement.

V. CONCLUSIONS

The calculations that were performed comparing classical and endochronic theory revealed that the endochronic approach can result in a 25 to 30 percent reduction in computer time for equivalent solution accuracy. This result, combined with the apparent accuracy of the material representation, indicates that the use of endochronic plasticity has great potential in evaluating the dynamic response of shipping containers. It is anticipated that the saving in computer time which was noted in these initial examples will be even more pronounced in more complex cases. This is true because the numerical complexity associated with the method does not increase as one moves to more general problems. That is, one does not have the complexity associated with multidimensional yield surfaces as exist in classical plasticity.

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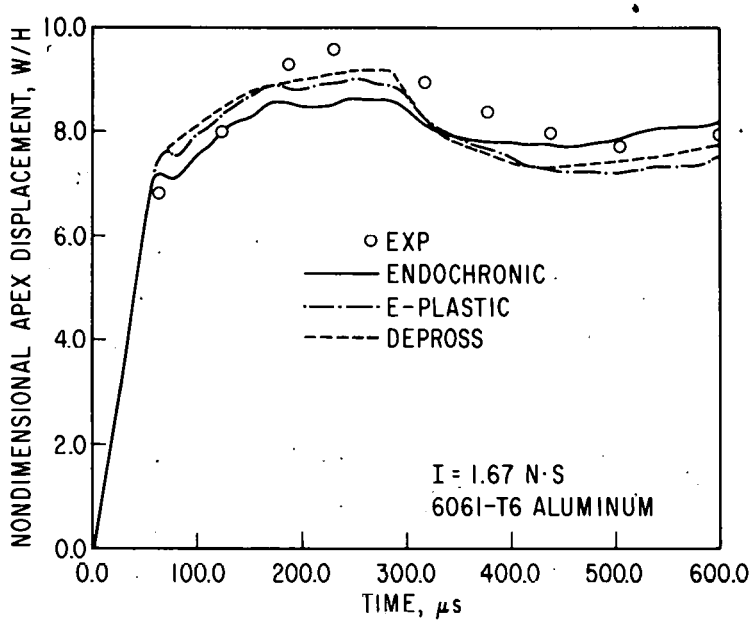


Fig. 3 Dynamic response of impulsively loaded clamped circular plate (6061-T6 aluminum, Imp = 1.67 N·s).

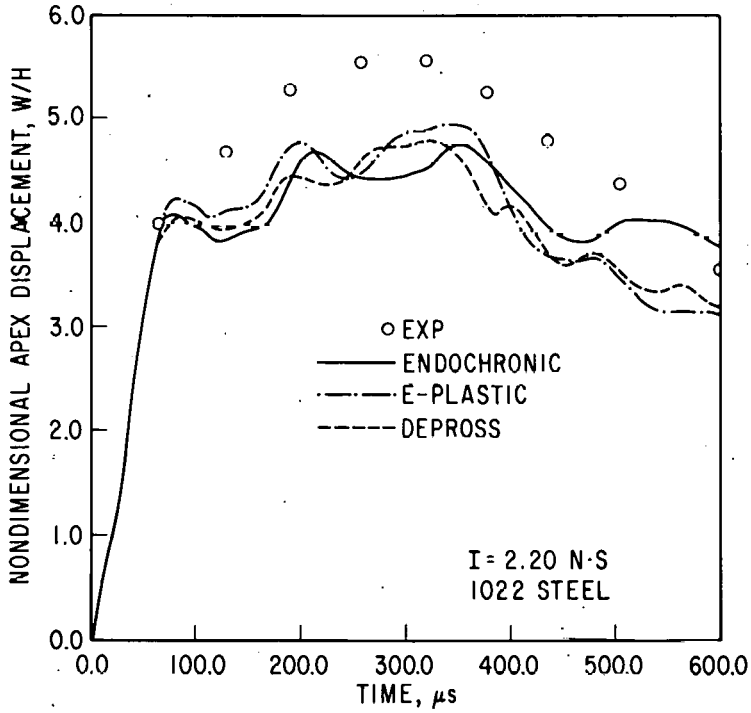


Fig. 4 Dynamic response of impulsively loaded clamped circular plate (1022 steel, Imp = 2.20 N·s).

CARGO RESPONSE TO RAILCAR IMPACT AND TIEDOWN LOAD ANALYSIS*

by

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ABSTRACT

An analytical study that investigated the loads produced during coupling of railcars carrying heavy shipping containers is described. The structural model of the impact event is represented by a lumped parameter technique. Each discrete mass lump possesses longitudinal, vertical, and rotational degrees of freedom. The resulting computer simulation provides for nonlinear railcar coupler stiffness and linear damping forces in the coupler and container tiedowns. Results include response to parametric variations in container weight, impact speed, and tiedown stiffness. Container dynamic response and tiedown loads are found to depend heavily on these parameters. Also, railcar bending and subsequent vertical motion are shown to be important contributors to these responses. When experimentally substantiated, the model can serve as a useful tool in the design and evaluation of shipping container tiedown structure.

INTRODUCTION

When massive energy materials containers are moved by rail, considerations must be given to forces produced by railcar coupler impacts. These forces may be large and can affect the container, its tiedown structure, and the railcar. Few rational dynamic analyses exist for the problem of dynamic interaction between these three systems. Current design practice employs static analysis procedures, using empirical load factors, to design tiedown systems, but no dynamic analysis has confirmed the applicability of these static analysis methods. There has been a need to develop a computer model that will handle the vertical, rotational, and longitudinal motions of both the container and the railcar. This is especially important for the heavier spent nuclear fuel containers that are transported by railcar.

In this paper an analytical model of railcar impact during coupling is developed using a lumped structural parameter technique.[1] The model is appropriate for horizontally laded

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flexible containers and most flatbed railcars. A vertically laded rigid container can also be simulated. A digital computer simulation of this model, RICTL, (Railcar Impact Container Tiedown Loads), has been developed by the Los Alamos Scientific Laboratory (LASL). This simulation presently has provision for a nonlinear coupler as the only nonlinearity. However, the program is written so that nonlinear subroutines can be easily added for the tiedown configuration and for other longitudinal members of the railcar. With RICTL in its present form, we have studied the effects of variations in certain analytical model parameters. These parameters include container weight, impact speed, and tiedown stiffness (and corresponding damping). The effects of varying these parameters include the response accelerations of the railcar and the container and tiedown loads.

This paper contains the general method of analysis and the application to a specific railcar carrying three different cargo weights with variations in coupling speed and tiedown parameters, a discussion of the results of these analytical studies, and the general conclusions of the study.

A. Method of Analysis

The structural dynamic method of analysis used to study the problem of coupling impact for a railcar carrying an energy material shipping container is based on the lumped parameter technique of structural modeling. Figure 1 shows a side view schematic of a specific railcar[2] and coupler, a shipping container with its tiedown configuration, and the mass lumping schematic with degrees of freedom assigned to each lumped mass. Mass 11 represents the initially stationary train that the railcar strikes. The container geometry is based on the NFS-M100 cask system, but with three weights--13.6, 27.2, and 63.5 metric tonnes. Seven mass lumps represent the initially moving railcar, and three mass lumps represent the shipping container. The mass lumps are interconnected by massless springs and/or dampers as indicated by the members designated ℓ_{ij} between mass lumps i and j where i,j = 1,2,...N. The nonlinear coupler is designated by stiffness $k_{7,11}$ and damping $c_{7,11}$ as shown in Fig. 1. The impact velocity (V_I) is given as the initial velocity in the direction shown for each of the ten masses that constitute the railcar-container structural dynamic system.

The lower schematic of Fig. 1 indicates the motion degrees of freedom for the system. The designation u_i refers to the positive horizontal displacement of the i^{th} mass; v_i refers to the positive vertical displacement of the i^{th} mass; and θ_i refers to the positive rotational motion of the i^{th} mass. There are 29 degrees of freedom. The dynamic equations of motion in matrix form are:

$$[M]\{\ddot{x}\} + [C]\{\dot{x}\} + [K]\{x\} = \{F\} \qquad (1)$$

with initial conditions

for $\dot{x}_j = \dot{u}_i$, (i = 1, 2, 3 ... 10)

$\dot{x}_j = V_I$ and

all other $\dot{x}_j = 0$.

Also, in the eleventh equation of Eqs. (1), $F_{11} = -R_F$, the sliding friction force of the initially stationary train being struck (Fig. 1). All other {F} = 0. The vector {x} refers to the appropriate absolute motion of each of the 29 degrees of freedom; [M] refers to the mass and mass-moment-of-inertia matrix; [K] refers to the stiffness matrix; and [C] refers to the damping matrix. Equations (1) constitute a set of 29 second order nonlinear differential equations, which are solved with the digital computer program RICTL.

The mass matrix [M] is a diagonal matrix of 29 elements derived from railcar and container mass and mass-moment-of inertia data. Table I includes the numerical data used for the specific railcar and cargo considered. Details of lumping methods are included in Ref. 3.

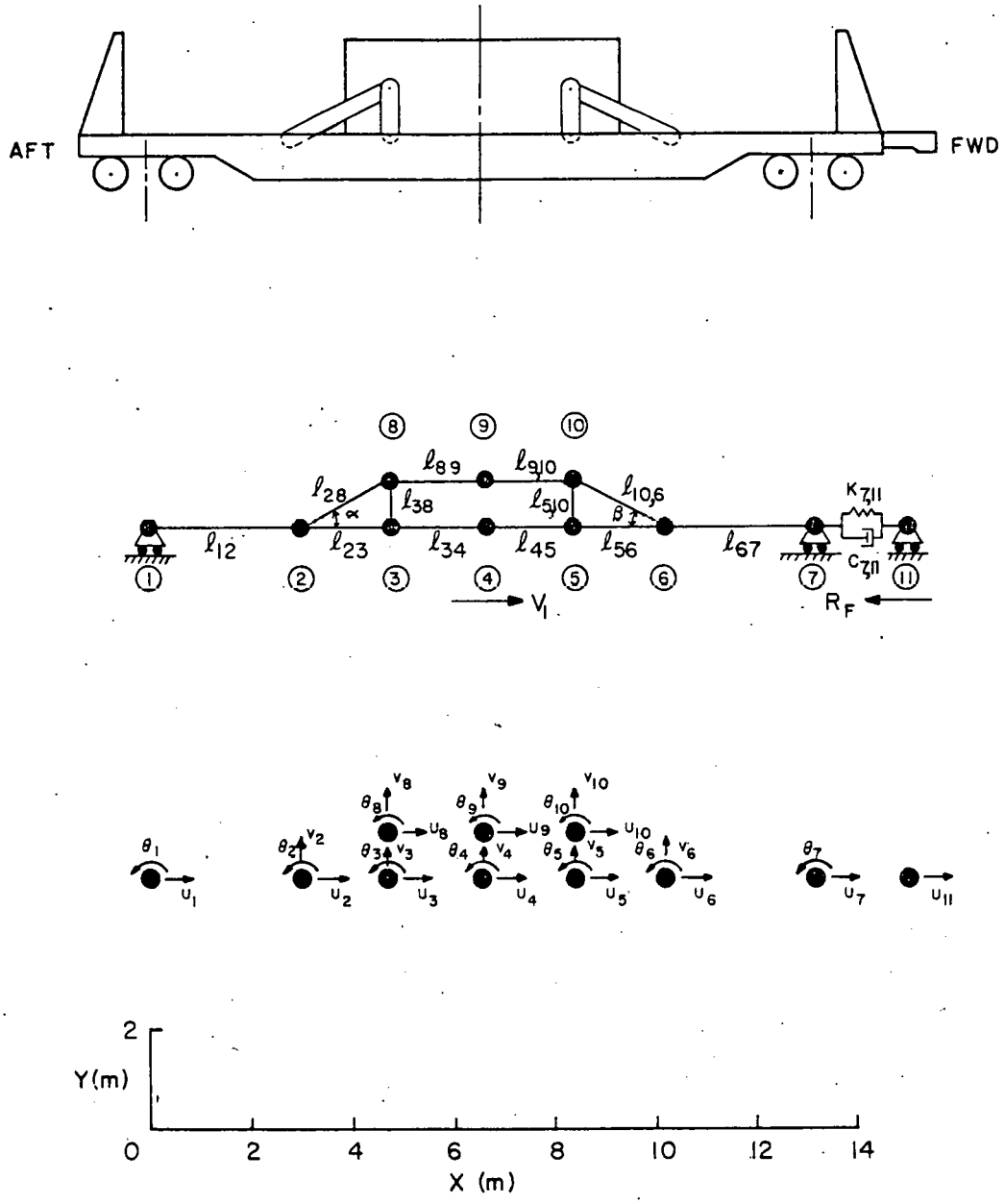


Fig. 1. Structural modeling technique for railcar, container, and coupler.

The stiffness matrix [K] is derived in the standard way[1] by first obtaining a transformation matrix [β] from absolute to relative coordinates. The axial and bending component stiffnesses [κ] are as given for beams,[1] and the resulting stiffness matrix [K] was formulated by the matrix product

$$[K] = [\beta^T]\,[\kappa]\,[\beta]\ . \qquad (2)$$

TABLE I

ANALYTICAL MODEL DATA

Railcar data from Appendix L, "Stress Analysis for 50 Ton 53 Ft. 6 In. Bulkhead Flat Car," Published by Association of American Railroads, January 1966.

Length between trucks = 13.182 m (519 in)
Container weights of 13.6, 27.2, 63.5 tonnes (15, 30, 70 tons)

Weights of railcar

(1) Distributed weight = 15,292 N/m (87.32 lb/in)
(2) Bulkheads (2) = 16,414 N each at x = 0, x = 13.182 m (3,690 lb each at x = 0, x = 519 in)
(3) Weight of trucks (2) = 31,805 N each at x = 0, x = 13.182 m (7,150 lb each at x = 0, x = 519 in)

Moduli (Pa)

Railcar $E_R = 2.0684 \times 10^{11}$ (30,000,000 psi)
Container $E_C = 2.0684 \times 10^{11}$ (30,000,000 psi)

Tiedown damping (est. $\zeta = 0.05$) - $(\frac{N}{m/s})$

Impact velocity (km/h) V_I = 8, 11.3, 14.5, 17.7

Stationary train - Weight = 226.8 tonnes - 2.224×10^6 N (500,000 lb)

- Resistance force (coeff. of friction of 0.2) = 4.448×10^5 N (100,000 lb)

The damping matrix [C] was formulated by first assuming linear viscous damping in the tiedown members and in the coupler. These are the only damping forces included. They were formed in a relative damping matrix [γ], and then [C] was formulated by the matrix product

$$[C] = [\beta^T]\,[\gamma]\,[\beta] \quad . \tag{3}$$

The nonlinear spring ($k_{7,11}$) associated with the draft gear of the coupler was as given by Magnuson and Wilson[4] as a standard Miner RF-333 draft gear. A subroutine was written that updates the $k_{7,11}$ element in [K] based on the current (time dependent) relative displacement in the coupler ($u_{7,11} = u_{11} - u_7$).

A Runge-Kutta numerical integration routine was used to solve Eqs. (1). A constant time step increment was used. The results of computer runs indicate that a small enough time step for convergence does not require excessive computer time to simulate the impact event for sufficient duration to obtain maximum motions and loads.

The constraint forces (including tiedown loads) were computed from the following relationships for every time step:

$$\{X_{rel}\} = [\beta]\{X\} \quad , \tag{4}$$

$$\{\dot{X}_{rel}\} = [\beta]\{\dot{X}\} \quad , \tag{5}$$

$$\{F\} = [\kappa]\,X_{rel} + [\gamma]\,\dot{X}_{rel} \quad , \tag{6}$$

where {F} is the constraint force vector, and

$\{X_{rel}\}, \{\dot{X}_{rel}\}$ = the relative displacement and velocity of the ends of each member carrying a constraint force.

B. Discussion of Results

The analytical results were obtained using the RICTL code. The study was parametric in nature with container weight, impact velocity, and tiedown stiffness and damping as parameters. Complete results of the study are presented in Ref. 3. The results of interest presented here include peak values of the time profiles of

(1) Horizontal deceleration of the railcar at the point of impact,
(2) Vertical response acceleration of the railcar at struck-end container attachment,
(3) Horizontal deceleration of the container c.g.,
(4) Vertical acceleration of the container c.g.,
(5) Struck-end tiedown load horizontal component,
(6) Struck-end tiedown load vertical component, and
(7) Struck-end tiedown load bending moment.

The tiedown loads were expressed in terms of the total load of the two struck-end tiedowns (which produced the maximum tiedown loads) and involved the axial load in each of the tiedown members $\ell_{5,10}$ and $\ell_{10,6}$. These axial loads were converted to equivalent horizontal and vertical loads and a bending moment of a single member attached rigidly between Mass 10 and Mass 5.

Figure 2 shows the variation in peak acceleration with the container weight for the variables of interest. The significant result shown in Fig. 2 is that, for heavy containers, the vertical acceleration of the railcar is nearly as large as the horizontal deceleration. This result emphasizes the relative importance of vertical dynamics compared with horizontal dynamics. This implies that bending effects of the railcar are important in considering tiedown loads. Figure 3 shows the peak tiedown load variation with container weight. This plot confirms the fact that bending effects in the tiedowns need to be considered. Although the net vertical component is small compared with horizontal tiedown loads, the bending moment is certainly significant.

Figures 4 and 5 show variation of the same peak load variables as Figs. 2 and 3 with impact speed as the parameter. The largest container weight, 63.5 tonnes, was used for these studies. The curves all indicate that the peak loads vary linearly, or approximately linearly, with impact speed. This is an entirely predictable result, since the impulse of impact is proportional to the momentum, which varies linearly with impact velocity. Variations from linearity are due to the peak loads being based on vibratory responses that peak at different relative times during the impact event. In the low speed case (8 km/hr), the coupler load peaks at .006 s when the coupler stiffness is high (10.51×10^7 N/m). In the higher speed cases, the coupler load peaks at .038 s when the coupler stiffness is low (2.1×10^7 N/m) during deflection of the draft gear. Thus the nonlinear stiffness characteristic of the coupler accounts for the deviation from linearity of some of the peak response loads.

The peak horizontal deceleration of the railcar is linearly related to impact velocity because it occurs 40 μs after initial impact for all impact speeds. At this time the coupler stiffness is still in the linear range, and the draft gear is undeflected. The container response acceleration peaks are linearly related to impact velocity because, although they occur at times when the draft gear is deflecting, they occur at the same time (.012 s for horizontal response and .017 s for vertical response) for each impact velocity.

Figures 6 and 7 show the peak load variations of the 63.5 tonne container configuration as a function of container tiedown axial stiffness. The tiedown stiffness embodies the same values of stiffness in each of the four members $\ell_{2,3}$, $\ell_{3,8}$, $\ell_{5,10}$, and $\ell_{10,6}$. Also, since the

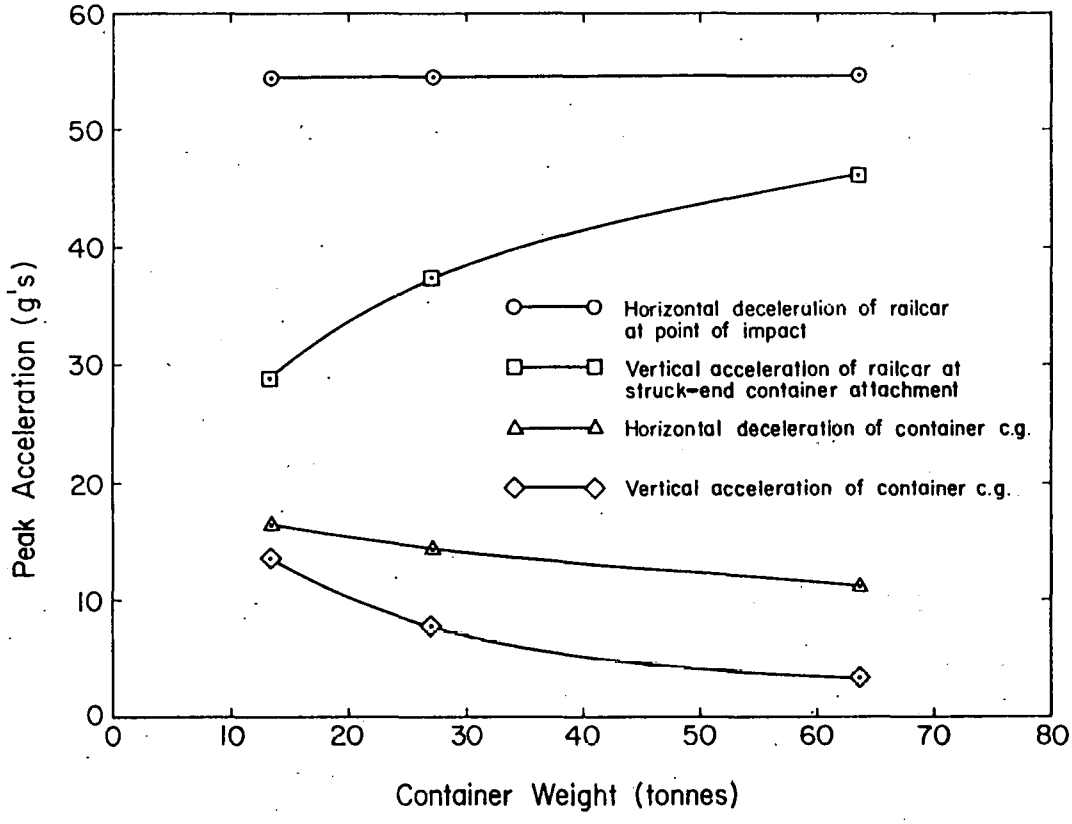


Fig. 2. Peak acceleration during impact at 17.7 km/h (11 mph).

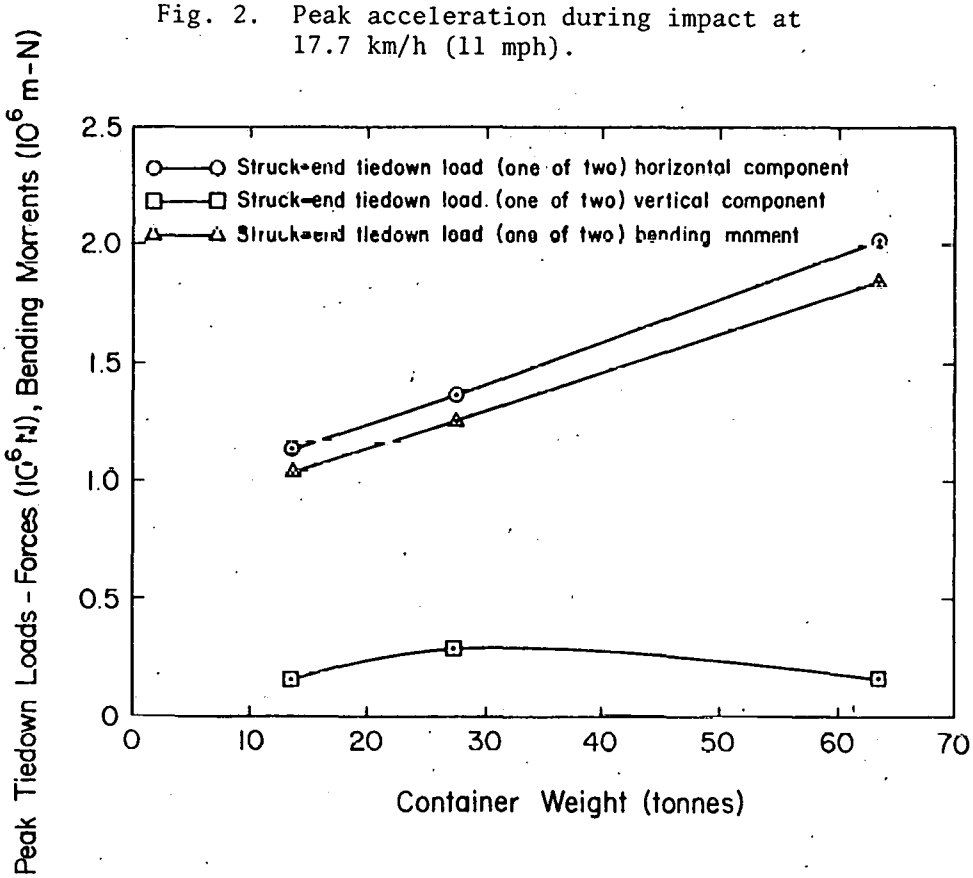


Fig. 3. Peak tiedown loads during impact at 17.7 km/h (11 mph).

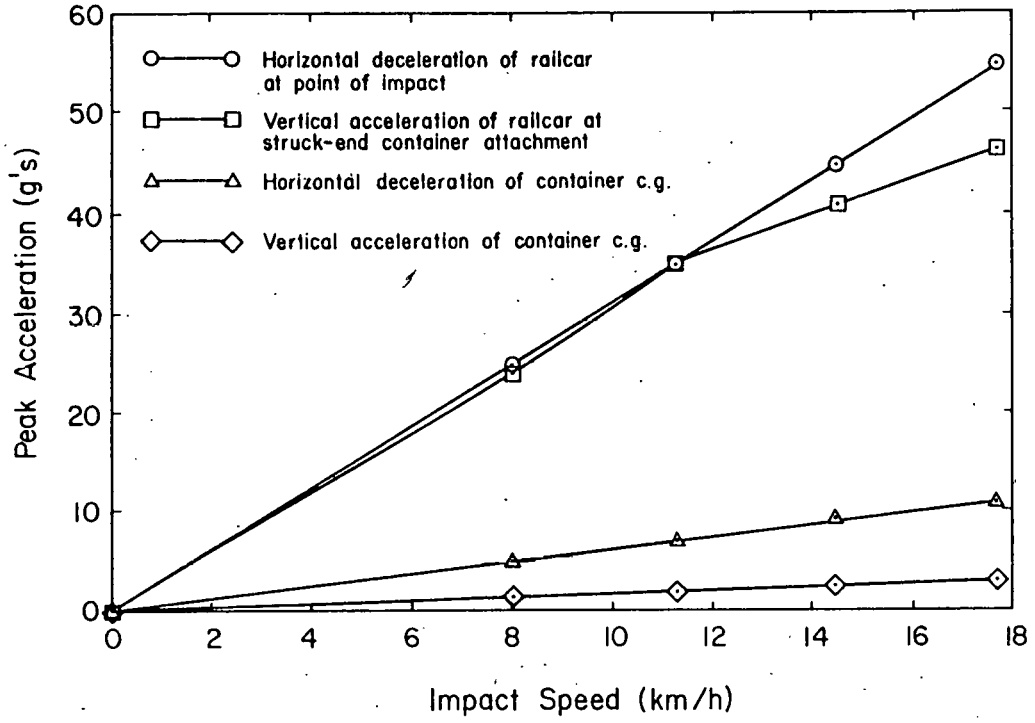


Fig. 4. Peak acceleration during impact of railcar carrying 63.5 tonne container.

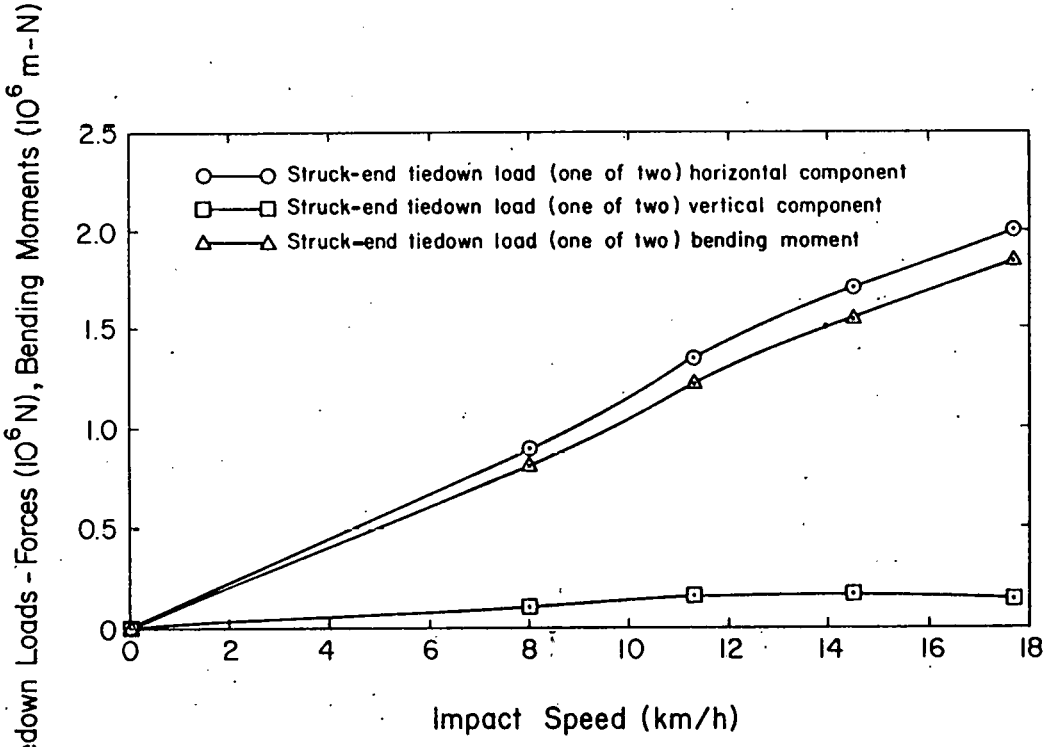


Fig. 5. Peak tiedown loads during impact of railcar carrying 63.5 tonne container.

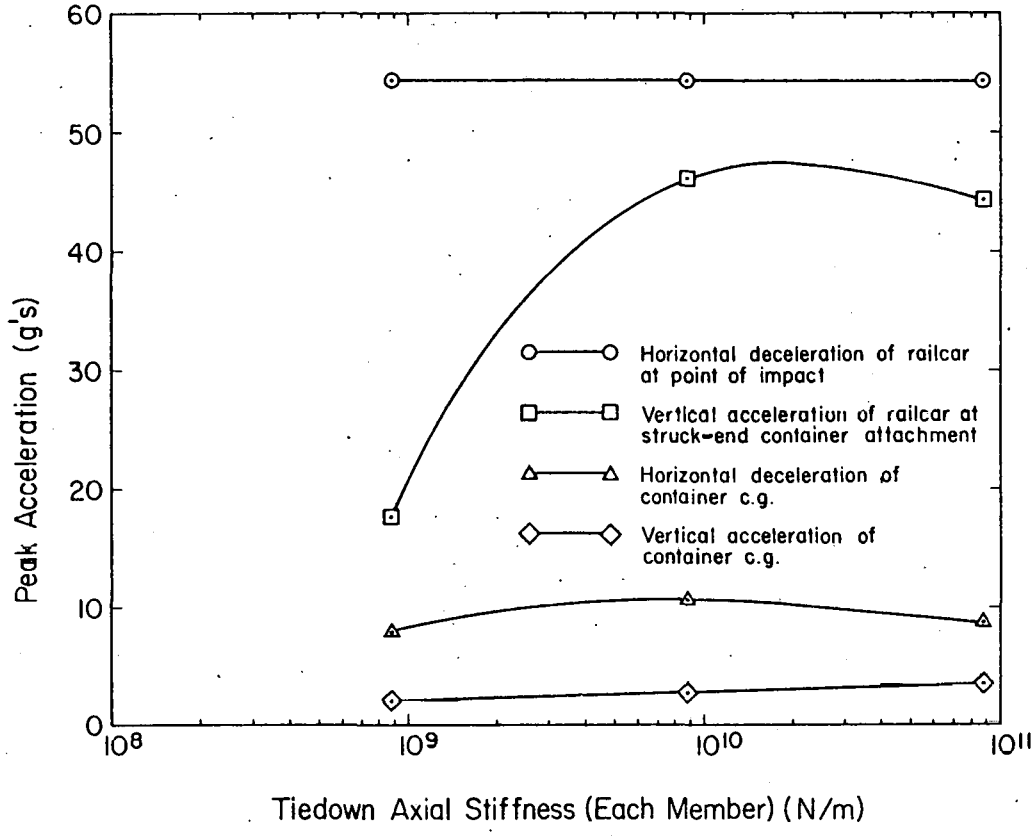


Fig. 6. Peak acceleration during impact at 17.7 km/h (11 mph) of railcar carrying 63.5 tonne container.

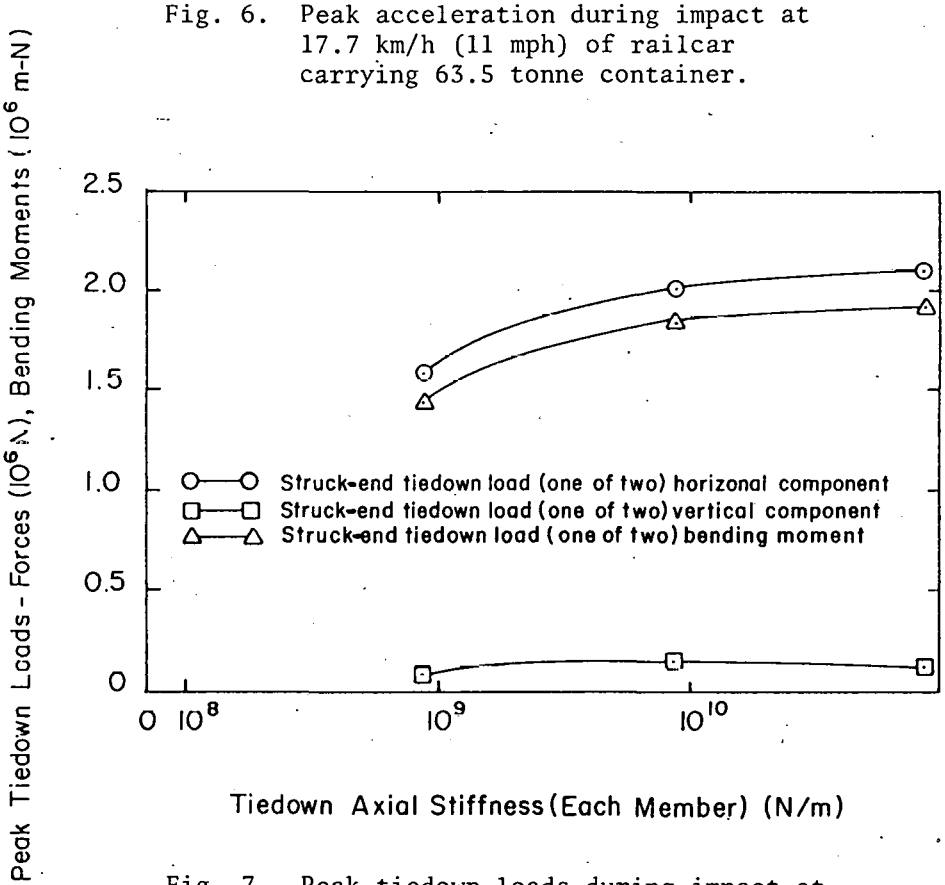


Fig. 7. Peak tiedown loads during impact at 17.7 km/h (11 mph) of railcar carrying 63.5 tonne container.

tiedown damping was assumed to vary with the tiedown stiffness, the damping constants were varied by the same ratio as the stiffness. The most notable result is that the response variables that depend on the tiedown interconnections to transmit loads are markedly reduced when the tiedown stiffness is reduced. The railcar vertical acceleration plot of Fig. 6 and tiedown load plots of Fig. 7 show this effect.

Time profiles were computed for the heaviest weight container-railcar configuration striking an initially stationary train at the highest coupling impact velocity (17.7 km/hr or 11 mph). For brevity, these profiles are not included, but they were computed to show the oscillatory behavior of the impact event. The impact deceleration has an apparent dominant frequency of 125 Hz, which is revealed initially in the tiedown load of member ℓ_{106}, the tiedown having the dominant loading. This result appears to be approximately in accord with the result obtained by Magnuson and Wilson in their model that has only longitudinal degrees of freedom for the ATMX car tiedown structure (dominant frequency of 100 Hz).[4] The magnitude of peak deceleration that we obtained (54.5 g's) also compares with the Magnuson and Wilson results for cargo weights between 178,000 N and 445,000 N, which gave 52 g's and 58 g's respectively for a spent fuel cask system with 3.2 mm free travel space. The configuration of Magnuson and Wilson and our configuration are not exactly the same, so a direct comparison is not possible. However, the closeness of the two results for longitudinal motion variables is of interest.

CONCLUSIONS

The results indicate that the RICTL simulation of railcar coupling impact dynamics produces results that are plausible when compared with results of other impact analyses.[4,5] The analysis method applied to the design of cargo tiedowns should be a significant improvement over current static design methods that employ equivalent empirically derived load factors in the design of tiedowns.[5] Also, vertical motion and bending are found to be important effects to be considered in the design of tiedown structure. More cannot be said about the adequacy of the simulation until a degree of correlation is established between the RICTL model and actual impact tests.

ACKNOWLEDGEMENTS

The efforts of T. A. Butler, E. G. Endebrock, J. B. Payne, W. F. Corcoran, and J. W. Straight in helping to code and debug the RICTL program are gratefully acknowledged.

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2. Association of American Railroads, "Specifications for Design Fabrication and Construction of Freight Cars Adopted Sept. 1, 1964," Appendix L - Stress Analysis for 50 ton 53 ft. 6 in. Bulkhead Flat Car, Chicago, Ill., (January 1966).

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4. Clifford F. Magnuson and Leonidas T. Wilson, "Shock and Vibration Environments for Large Shipping Containers on Railcars and Trucks," Sandia Laboratories Albuquerque Report SAND 76-0427 NUREG 766510 (June 1977).

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Dup

SUMMARY

RADIOACTIVE FUEL CASK RAILCAR HUMPING STUDY*

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The purpose of this study was to determine shock spectra for testing cask tiedowns for railcar shipment. Even though radioactive shipments are not to be humped, it can happen. The response of two radioactive shipping casks due to railroad humping shocks was calculated using a spring-mass model. The two railcars for these casks had different coupling mechanisms and different tiedown arrangements. Humping tests had been performed on one of the railcars (ATMX-600) and the resulting shock spectra was used to adjust the spring-mass model to get matching results. One car (designed for cask shipment) was equipped with Freightmaster E-15 end of car coupler and had about 3.2 mm (1/8 inch) free travel of the cask skid relative to the car. The other car, (ATMX-600) equipped with Miner RF-333 draft gear was designed for nuclear weapon shipment and adapted to nuclear waste shipment by fastening the casks to the floor. Both car frames were built by the same manufacturer and are very similar.

The response of the casks was put in shock spectra format and a parametric study was performed with various cask weights. Additional studies were done on the effects of fastening the loose cask, and using the Freightmaster end of car coupler on the ATMX car. Half-sine response spectra were overlaid to include the natural frequency of the cask tiedown. The resulting shock amplitude was plotted against the cask weight for each car.

The results show a constant acceleration level for all the weights on the car with hydraulic end of car coupler which results from constant force at that impact velocity. The cask acceleration can be reduced by fastening it to the car, rather than allowing it to move freely through some small space. This study also shows that the cask response can be optimized on railcars without hydraulic draft gear by adjusting the tiedown stiffness to keep the tiedown frequency different than car frequencies.

The utility of using shock spectra allows the cask designer the flexibility of choosing dynamic or static test levels that are based on the total system.

* This work was supported by the U. S. Department of Energy.

RADIOACTIVE FUEL CASK RAILCAR HUMPING STUDY

INTRODUCTION

The designer of radioactive fuel casks must provide for shipping loads during transportation. Due to the size and weight of the larger casks, rail transportation is used. It also happens to contain the most severe shock environment if humping loads are considered. The DOT Code of Federal Regulations, Part 49 (49CFR - para. 174.84 dtd. 1976) states that "placarded flatcars may not be coupled into with more force than is necessary to complete the coupling." In spite of this regulation, there is a consensus among shippers that violations may occur and that it would be prudent to design for humping loads. Records from the Sandia Environmental Data Bank show that 99.8% of humping velocities are below 17.78 km/hr (11 mph). This velocity is just below the level where damage becomes apparent on most railcars. Consequently, the 17.78 km/hr (11 mph) impact is used as the maximum test velocity.

This analytical study was done to determine shock spectra for testing cask tiedowns for railcar shipment. Mathematical models of two railcars were prepared. One car was an ATMX-600 car (DOE weapon shipping railcar) which had been used to ship radioactive wastes. Humping shock spectra (obtained from tests) were availablea for this car. Detailed knowledge of the car helped in adjusting the coupler spring stiffnesses to match the test shock spectra. Load-deflection information on couplers is both non-linear and scarce. The other railcar that was modeled was designed to transport a nuclear-fuel cask. Its frame is very similar to the ATMX car frame and was constructed and built by the same company. This car, however, had a hydraulic end-of-car cushion behind the coupler designed by Freightmaster. Force-velocity curves were available and were used for the coupling spring.

This study was performed using a code called GENRE, which solves a single axis spring-mass model with nonlinear spring capability. The cars and casks were impacted into another car of similar weight at four velocities. The highest was 17.78 km/hr (11 mph) which is reported here.

RAILCARS

The ATMX-600 railcar is a hopper car built on an as-cast steel frame of a flatcar and equipped with Miner RF-333 draft gear. The other car was a flatcar with cast steel frame and equipped with a Freightmaster E-15 hydraulic end of car coupler. This coupler provides a constant force during coupler deflection. The coupling force varies with impact velocity. This car is hereafter called the cushioned car os spent fuel cask system.

CASKS AND TIEDOWNS

The radioactive shipping casks shipped in the ATMX-600 cars were designed by Dow Chemical Company and reported in Reference 1. The casks were held to the floor by large bolted brackets. The cask on the cushioned car was mounted to a skid which fit around a shear structure on the car, with no additional restraint. There was about 3.2 mm (1/8 inch) gap between the skid and shear structures which was modeled as a nonlinear spring.

MODELING

The railcars were divided into 7 masses each about 2.43 m (8 feet) in length. The spring-mass model of the ATMX-600 and casks is shown in Figure 2, and for the cushioned car and cask in Figure 1. The weights of the casks were varied in four steps to determine the effect of weight on response. The acceleration-time response of the car mass that supported the cask was input to the code PSPECMO to obtain a shock spectrum of the driving point for the cask(s).

RESULTS

A half-sine spectrum was overlaid to envelop the natural-frequency of the tiedowns and the peak acceleration of the half-sine was graphed versus the four cask weights. The curves showed an increasing acceleration level with weight for the cask with the 3.2 mm (1/8 inch) free travel so the cask was tied down and analyzed. This produced a constant acceleration vs. weight curve as shown in Figure 3. The curve for the ATMX-600 showed that the draft gear produces the lowest acceleration for 712,000 N (160,000 lbs.) cargo weight and began to bottom out with greater weight. A comparison was made by replacing the Miner draft gear with the Freightmaster E-15 gear. This produced a constant acceleration vs. weight curve as shown in Figure 4. From the foregoing, it is obvious that if a railcar is dedicated to moving casks that a cushioned car is the better drafting method. It also shows the need for compatibility between the end-of-car coupler and the tiedown stiffnesses. Figures 5 - 8 are the response spectra for the 890,000 N (200,000 lb.) cargo weight of the conditions mentioned above.

Tieing down the loose cask resulted in lowering the responses in the 50 to 60 Hz region and increasing it in the 30 to 40 Hz region. Cushioning the ATMX car lowered the 6 and 400 Hz responses and increased the 10 to 20 Hz response.

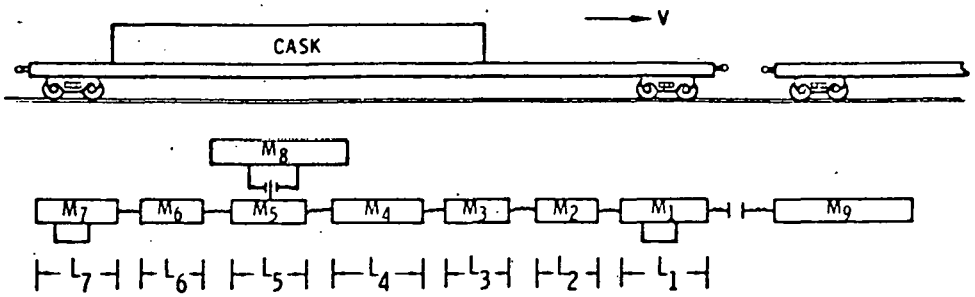
The value is using spring mass modeling and using test shock spectra is that it permits parametric studies to be performed to optimize the overall car, cask, tiedown design. The utility in using shock spectra for cask tiedowns alone is that it permits a judgment of whether to test the cask statically or dynamically if the cask tiedown natural frequencies and cask-cargo natural frequencies are known.

For a full treatment of this analysis see Reference 2.

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1. ATMX-600 Railcar, A New Concept in Radioactive Waste Shipments, Frank E. Adcock, Proceedings from Third International Symposium on Packaging and Transportation of Radioactive Materials, August 16 - 20, 1971, 71501 (Vol. 1) Richland, Washington Conference.

2. Shock and Vibration Environments for Large Shipping Containers on Rail Cars and Trucks, Clifford F. Magnuson and Leonidas T. Wilson, Ref: SAND76-0427, Sandia Laboratories, Albuquerque, New Mexico.



MODEL CHARACTERISTICS

	WEIGHT* (N)	WEIGHT* (lbs)		LENGTH (m)	LENGTH (in)
M_1	78100	(17553)	L_1	2.67	(105)
M_2	22600	(5079)	L_2	2.44	(96)
M_3	23600	(5301)	L_3	1.83	(72)
M_4	41100	(9237)	L_4	2.44	(96)
M_5	23600	(5301)	L_5	1.83	(72)
M_6	22600	(5079)	L_6	2.44	(96)
M_7	78100	(17553)	L_7	2.67	(105)
M_8	= (1) 178000	(40000)			
	= (2) 445000	(100000)			
	= (3) 712000	(160000)			
	= (4) 890000	(200000)			
M_9	801000	(180000)			

SPRING RATES

K_{1-2} = 8.98×10^9 N/M (.5130 x 10^8 lb/in)

K_{2-3} = 9.95×10^9 N/M (.5684 x 10^8 lb/in)

K_{3-4} = 16.21×10^9 N/M (.9256 x 10^8 lb/in)

K_{4-5} = 16.21×10^9 N/M (.9256 x 10^8 lb/in)

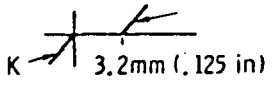
K_{5-6} = 9.95×10^9 N/M (.5684 x 10^8 lb/in)

K_{6-7} = 8.98×10^9 N/M (.5130 x 10^8 lb/in)

K_{1-9} @ 17.78 km/hr (11.05 mph) F = 1712570N (385000 lb)

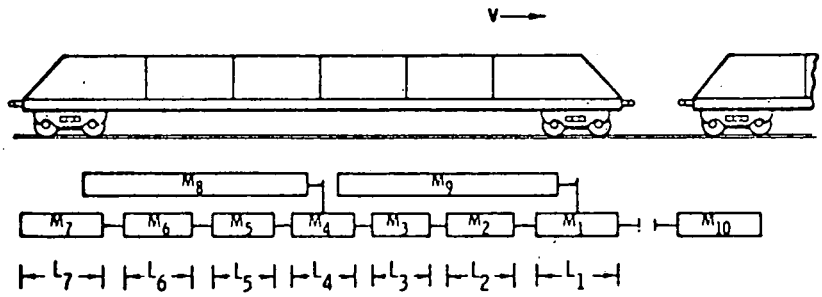


K_{5-8} = 1.75×10^9 N/M (.1000 x 10^8 lb/in)



*SEE PAGE 4, PAR. 3.4.1.2 OF ASTM METRIC PRACTICE GUIDE E380-72

Figure 1. Spring-Mass Model; Spent Fuel Cask System

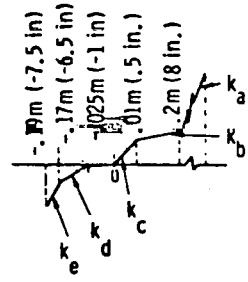


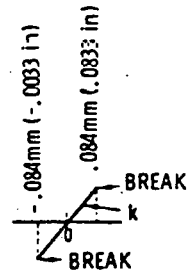
MODEL CHARACTERISTICS

	WEIGHT* (N)	(lbs)		LENGTH (m)	(in)
M_1 =	91200	(20500)	L_1 =	2.90	(114)
M_2 =	68900	(15500)	L_2 =	2.44	(96)
M_3 =	68900	(15500)	L_3 =	2.77	(109)
M_4 =	68900	(15500)	L_4 =	2.13	(84)
M_5 =	68900	(15500)	L_5 =	2.77	(109)
M_6 =	68900	(15500)	L_6 =	2.44	(96)
M_7 =	91200	(20500)	L_7 =	2.90	(114)
$M_8 = M_9$ = (1)	89000	(20000)			
(2)	225300	(50650)			
(3)	356000	(80000)			
(4)	445000	(100000)			
M_{10} =	890000	(200000)			

SPRING RATES

K_{1-2} = 1.75×10^{10} N/M ($.1000 \times 10^{9}$ lb/in)
K_{2-3} = 2.10×10^{10} N/M ($.1200 \times 10^{9}$ lb/in)
K_{3-4} = 2.98×10^{10} N/M ($.1700 \times 10^{9}$ lb/in)
K_{4-5} = 2.98×10^{10} N/M ($.1700 \times 10^{9}$ lb/in)
K_{5-6} = 2.10×10^{10} N/M ($.1200 \times 10^{9}$ lb/in)
K_{6-7} = 1.75×10^{10} N/M ($.1000 \times 10^{9}$ lb/in)
K_{1-10} k_a = 7.00×10^{8} N/M ($.4000 \times 10^{7}$ lb/in)
k_b = 2.10×10^{7} N/M ($.1200 \times 10^{6}$ lb/in)
k_c = 10.51×10^{7} N/M ($.6000 \times 10^{6}$ lb/in)
k_d = 8.76×10^{7} N/M ($.5000 \times 10^{6}$ lb/in)
k_e = 12.26×10^{7} N/M ($.7000 \times 10^{6}$ lb/in)
K_{1-9}, K_{4-8} = 4.95×10^{10} N/M ($.2827 \times 10^{9}$ lb/in)





*SEE PAGE 4, PAR. 3.4.1.2 OF ASTM METRIC PRACTICE GUIDE E380-72

Figure 2. Spring-Mass Model; ATMX-600 Rail Car with Two Fuel Casks.

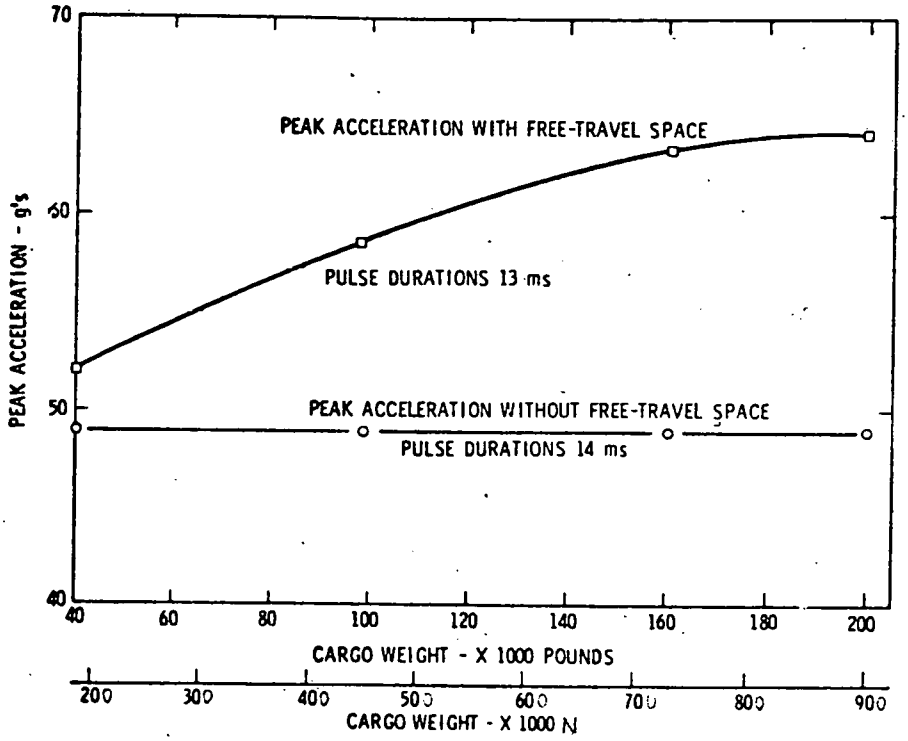


Figure 3. Peak Acceleration and Pulse Duration Half-Sine Pulses - Spent Fuel Cask System Longitudinal Axis.

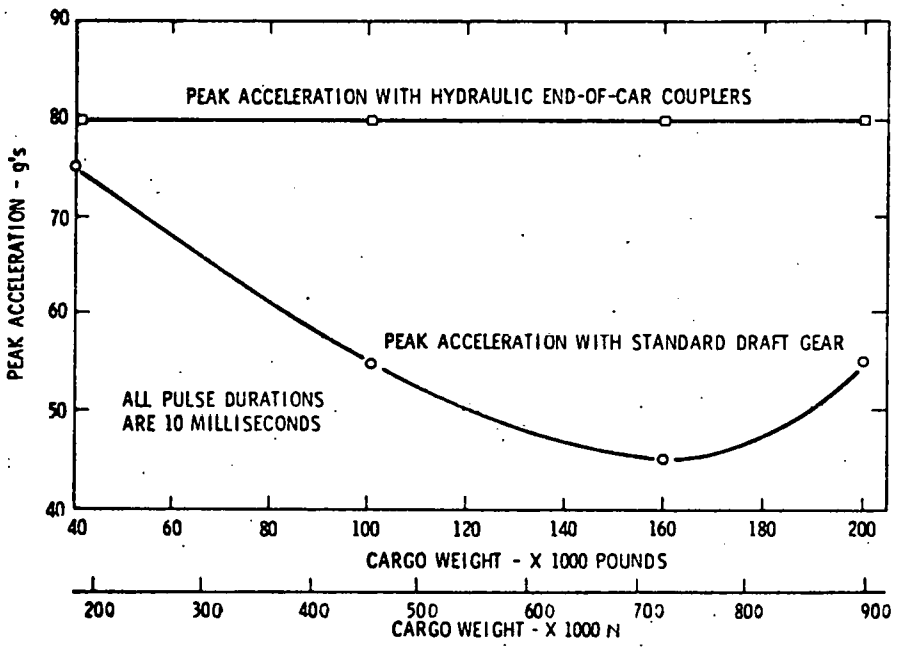


Figure 4. Peak Acceleration and Pulse Duration Half-Sine Pulses - ATMX System Longitudinal Axis.

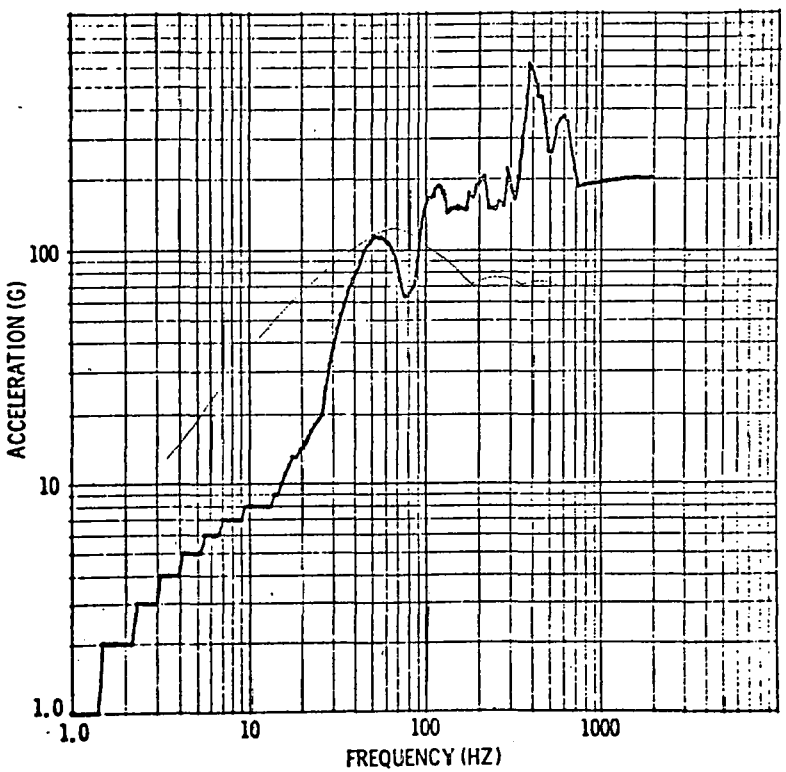


Figure 5. Response Spectrum Analytical Results.

Spent Fuel Cask System with 3.2 mm (1/8 inch) Spacing, 890,000 N (200,000 pound) Cargo, 17.78 km/hr (11.05 mph) Impact Velocity, 3 percent Damping, Longitudinal Axis

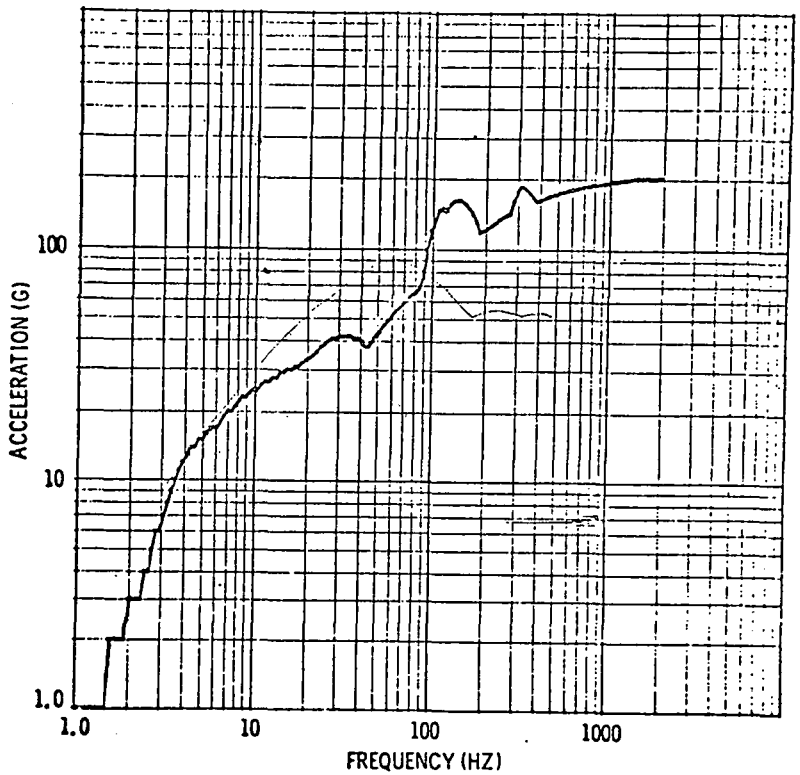


Figure 6. Response Spectrum Analytical Results.

Spent Fuel Cask System, Cargo Tied Down, 890,000 N (200,000 pound) Cargo, 17.78 km/hr (11.05 mph) Impact Velocity, 3 percent Damping, Longitudinal Axis

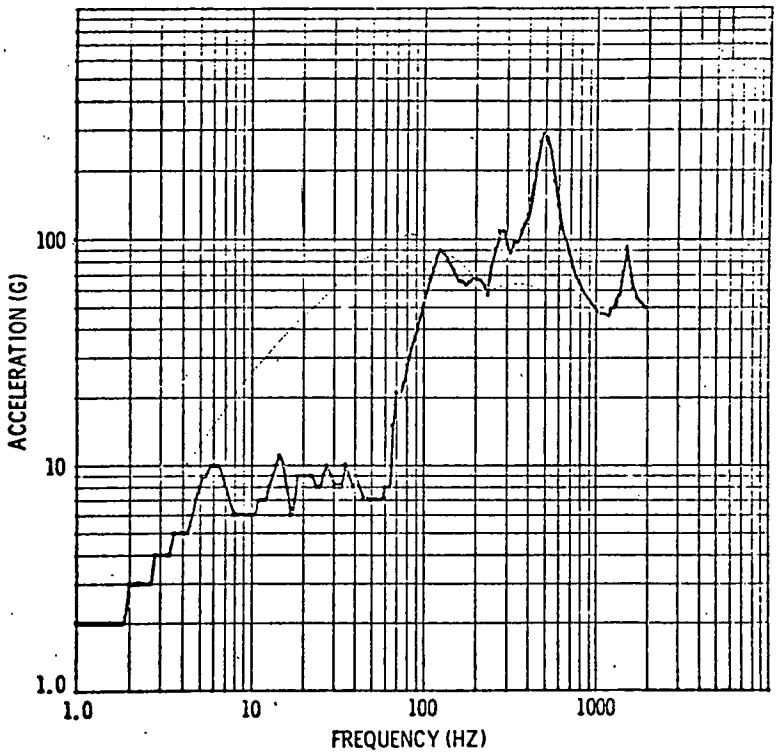


Figure 7. Response Spectrum Analytical Results.

ATMX Car, Standard Draft Gear, 890,000 N (200,000 pound) Cargo, 17.78 km/hr (11.05 mph) Impact Velocity, 3 percent Damping, Longitudinal Axis

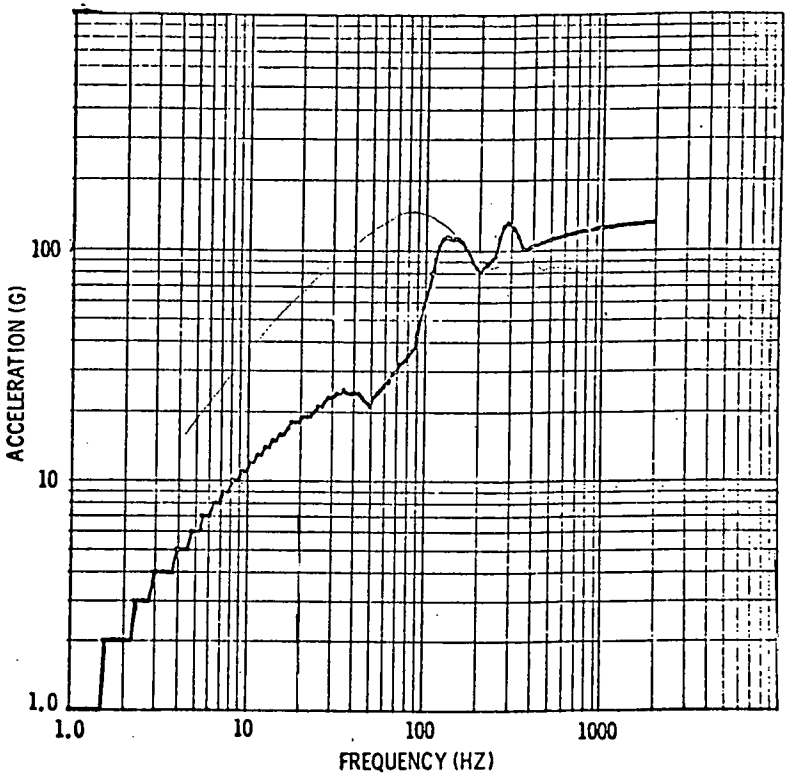


Figure 8. Response Spectrum ANalytical Results.

ATMX Car, Shock Attenuating Couplers, 890,000 N (200,000 pound) Cargo, 17.78 km/hr (11.05 mph) Impact Velocity, 3 percent Damping, Longitudinal Axis

Dup

THE ORNL RADIOACTIVE CARRIER DROP TEST FACILITY: THE DESIGN, CONSTRUCTION, AND OPERATING CAPABILITY*

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W. D. Box, L. B. Shappert, J. H. Evans
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Oak Ridge, Tennessee 37830

ABSTRACT

A large drop test-facility has been constructed at the Tower Shielding Facility (TSF) at the Oak Ridge National Laboratory. A 670-metric ton impact pad was constructed of reinforced concrete and armor plate and located between two 96-m-tall towers of the TSF. The towers, with heavy-duty hoisting equipment attached at their tops, are 39 m apart and are capable of lifting and dropping casks from heights of 80 m. Casks weighing 160 metric tons can be tested at 9 m, whereas lesser weight casks can be handled at greater heights. This paper describes the design, construction, and operating capability of this facility, unique in the free world.

INTRODUCTION

In October 1975, the Oak Ridge National Laboratory approved the design, construction, and installation of a permanent drop pad at the Tower Shielding Facility (TSF). This installation was to replace the small, temporary drop pad at the TSF.

The structure would be a reinforced-concrete, truncated Aztec pyramid, 8 m by 15 m at the base, 4 m by 8 m at the top, and with a total height of 3 m. The impact surface, a 64-metric ton (MT) structure, 5.5 m long, 2.5 m wide, and 0.63 m high, would be constructed of laminated steel (four layers, each 15.88 cm thick). The TSF site was selected because of the existing availability of hoisting equipment that was capable of lifting large loads to great heights, its remote location atop a small mountain in a controlled access area, and its underground control room for remote operation. The TSF consists of four towers, each of which is 96 m high, set in a rectangular array 30 m by 60 m (Fig. 1). Each of the towers is guyed with two pairs of 5-cm-diam cables. The hoisting system is connected to the top of each tower. The impact surface would be located midway between towers III and IV, which are 60 m apart.

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*Research sponsored by the Division of Environmental Control Technology, U.S. Department of Energy under Contract W-7405-eng-26 with the Union Carbide Corporation.

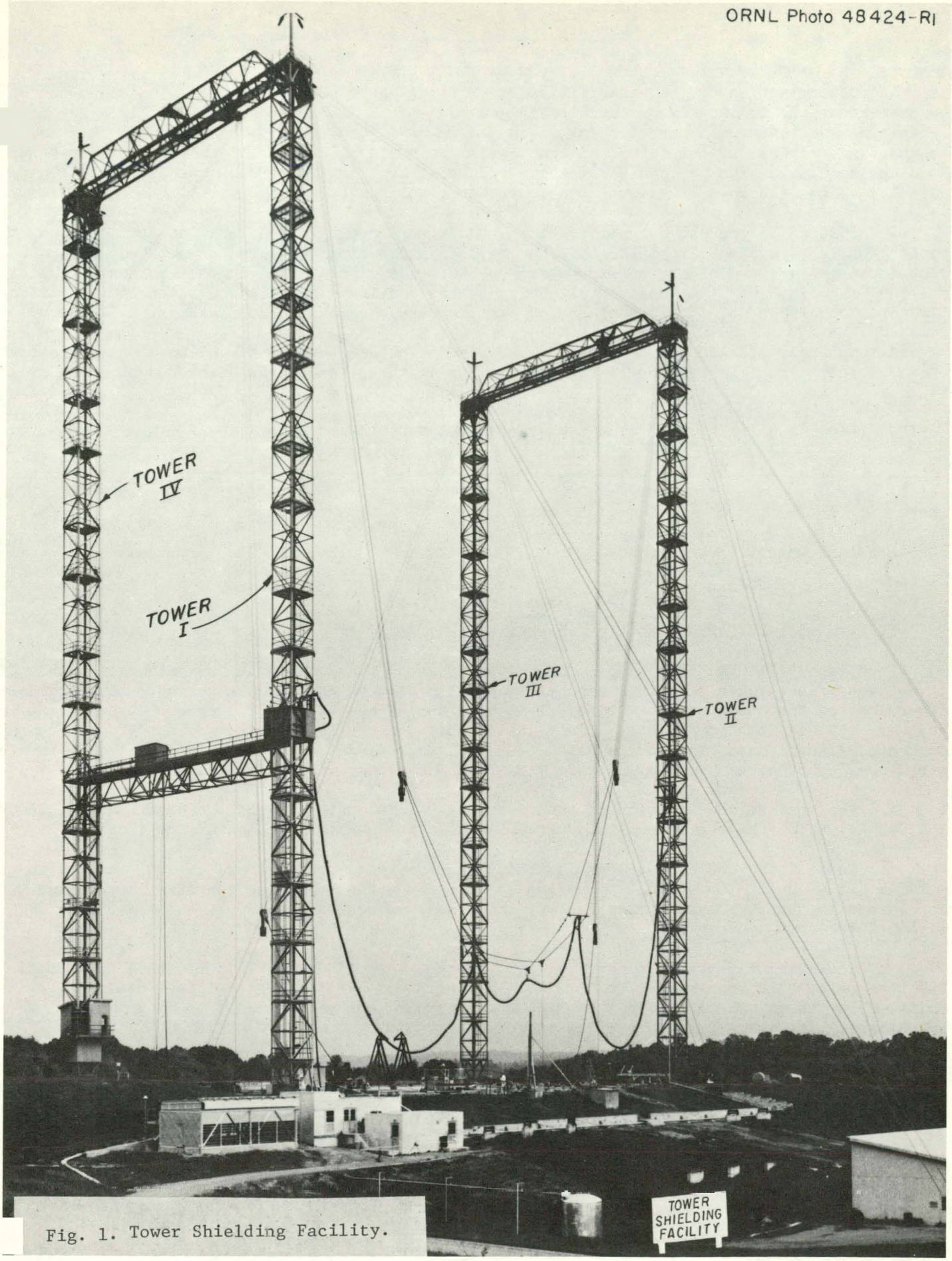


Fig. 1. Tower Shielding Facility.

TOWER SHIELDING FACILITY LIFTING CAPABILITY

The determination of the capabilities for lifting casks using the TSF is based on the assessment of the load-carrying capabilities of each of the TSF components: inclined guy cables (stay-type cables between the tip of each tower and the ground), horizontal tie cables (connecting cables at the top of the towers), structural members of the tower legs, dynamic loading associated with the cable snap-back occurring at the time of cask release, soil-bearing capacity at the footings of each of the tower legs, and loads in the supporting sheaves.

In addition to the above considerations, the tower capabilities are constrained on the basis of the system kinematics. The main contributing factors associated with these considerations are sag of the horizontal tie cables (the displacement in the vertical direction of the midpoint of the horizontal cable as measured from the unloaded tower configuration), tip-tower displacement, and loss of preload in the horizontal cable support system.

A functional relationship based on the force balance at the tip of the tower, that involves cask weight, load in the horizontal tie cables, and drop height was developed as shown by Eq. (1).

$$W = [(90.221 - h)/29.108](21.567 - 0.67T + 1.69/T^2) + 4.103 + (1.834 - 0.0368h + 0.000204h^2)^{1/2} \quad (1)$$

where

W = cask weight (metric tons),

h = drop height (meters), and

T = tensions in horizontal tie cables (newtons).

Equation (1) is the functional relationship that was used in a parameter study to investigate the tower lifting capabilities. The drop height vs cask weight relationship was determined on the basis of Eq. (1) by first considering the events that produce overloading on any of the system components. Only the results of the full parametric study conducted are presented here. These results are based on the following limitations associated with the structure, constrained by a set of safety factors which were imposed in the analysis.

1. The maximum load of each of the tower legs, in the vicinity of the base, was limited to 350,000 kg, which was the upper limit established in the original design of the facility.

2. The lifting cable system is limited by a safety factor of 6, based on the ratio of maximum line pull to nominal breaking load of the cable. Because the steel guy cables were considered to have the same safety factor, their maximum working load was equal to 276,000 N (1,655,000/6).

3. The maximum bearing load of the soil at the base of the smallest of the two footings has been calculated to be equal to 2.7×10^6 N, which was based on the original design allowable soil-bearing value of 2×10^5 Pa. Subsequent field work has yielded soil bearing values which are almost twice those used in the initial design.

4. The stresses in the tower leg structural components due to the dynamics associated with the snap-back of the cables that occurs immediately after release of the cask is limited to 5×10^7 Pa.

5. The loss of preload on the horizontal tie cables is not a primary controlling factor, since loss of preload occurs only after the inclined guy cables have been overloaded with respect to the normal safety factor of 6.

Figure 2 is a graph of the parametric study using Eq. (1), which for this purpose was based on a maximum load in the inclined cables of 2.8×10^5 N. The initial section of the curve, up to h = 15.0 m, is limited by the static soil-bearing pressure of 2.0×10^5 Pa. The curve from points A to B represents cables associated with a safety factor of 6. Point C is the point at which the maximum impulse is imparted to the impact pad.

ANALYSIS AND DESIGN OF THE CASK IMPACT PAD

Based on the analysis of the TSF previously described, the design loading of the cask impact pad was a 151-MT cask dropped from a height of 32 m. This combination produces the largest impulse (9.0×10^5 N-sec). The design approach was to provide the flat horizontal impact surface by embedding steel armor plates in a concrete mass. Since the site was fixed, this mass had to rest on approximately 12 to 15 m of overburden. Static-soil tests[6] were made which indicated that the ultimate bearing capacity ranged from 1.0×10^6 to 1.3×10^6 Pa. Based on the average value of the tests, 1.2×10^6 Pa was chosen as the design load on the rigid-pad base. Assuming this load, the pad was designed using the American Concrete Institute's working stress method for conventional isolated rectangular footings. This conservative approach was considered necessary to ensure that the pad with embedded plates was rigid. This rigid mass resting on the soil was then idealized as a single-degree-of-freedom spring-mass system under impulse loading. A generalized linear impulse theory for undamped systems[1] was used to determine the peak spring deflection. The resulting peak dynamic spring force was compared with the stated design load on the pad base to verify the design.

Refraction seismograph techniques were used to provide an estimate of the dynamic shear modulus of the soil. This technique measures the compressional (P-wave) wave velocity. Relationships for p-wave velocities, shear-wave velocities, and Poisson's ratio were used[2] to arrive at a design value. The P-wave velocity was found to vary from 910 m/sec to 1050 m/sec. For Poisson's ratio = 0.35, the shear wave velocity varied from 455 m/sec to 530 m/sec. The lower value was used, and the dynamic shear modulus of the soil[3] was determined to be 2.0×10^8 pa after incorporating the reductions recommended in WASH 1301[3]. The vertical-soil spring constant was determined to be $K_y = 7.5 \times 10^9$ N/m.

Employing the concept[1] of dynamic load factor (DLF) and impulse and an assumed triangular load-time curve, the following general equation governs:

$$Y = Y_o \cos \omega t + (Y_o/\omega) \sin \omega t + Y_{st} \int_0^t f(\tau) \sin \omega(t - \tau)\, d\tau \qquad (2)$$

DLF is defined as the ratio Y/Y_{st}, where $Y_{st} = F_1/K$, and F_1 = peak value of load on the load-time curve. Curves relating maximum DLF values to (t_d/T) ratios are given by Biggs[1] for various load applications where t_d = load duration and T = fundamental system period. For the assumed triangular load pulse with finite rise time, neither F_1 nor t_d are known, and reasonable estimates of t_d had to be made. Furthermore, the Biggs curves[1] show that the maximum DLF occurs when the load duration is approximately equal to the natural period. Since our concern was with maximum response, the curves were used to search for the maximum dynamic effects. For an initial estimate of t_d = 0.06 sec, which was considered to be probable based on previous tests for smaller casks, Y_{max} = 0.2 cm.

It should be restated that the actual shape of the load-time curve is unknown, necessitating the assumption of the triangular load pulse with finite rise time. Within this assumption, several values of t_d were used with the Biggs curves in an attempt to bracket the

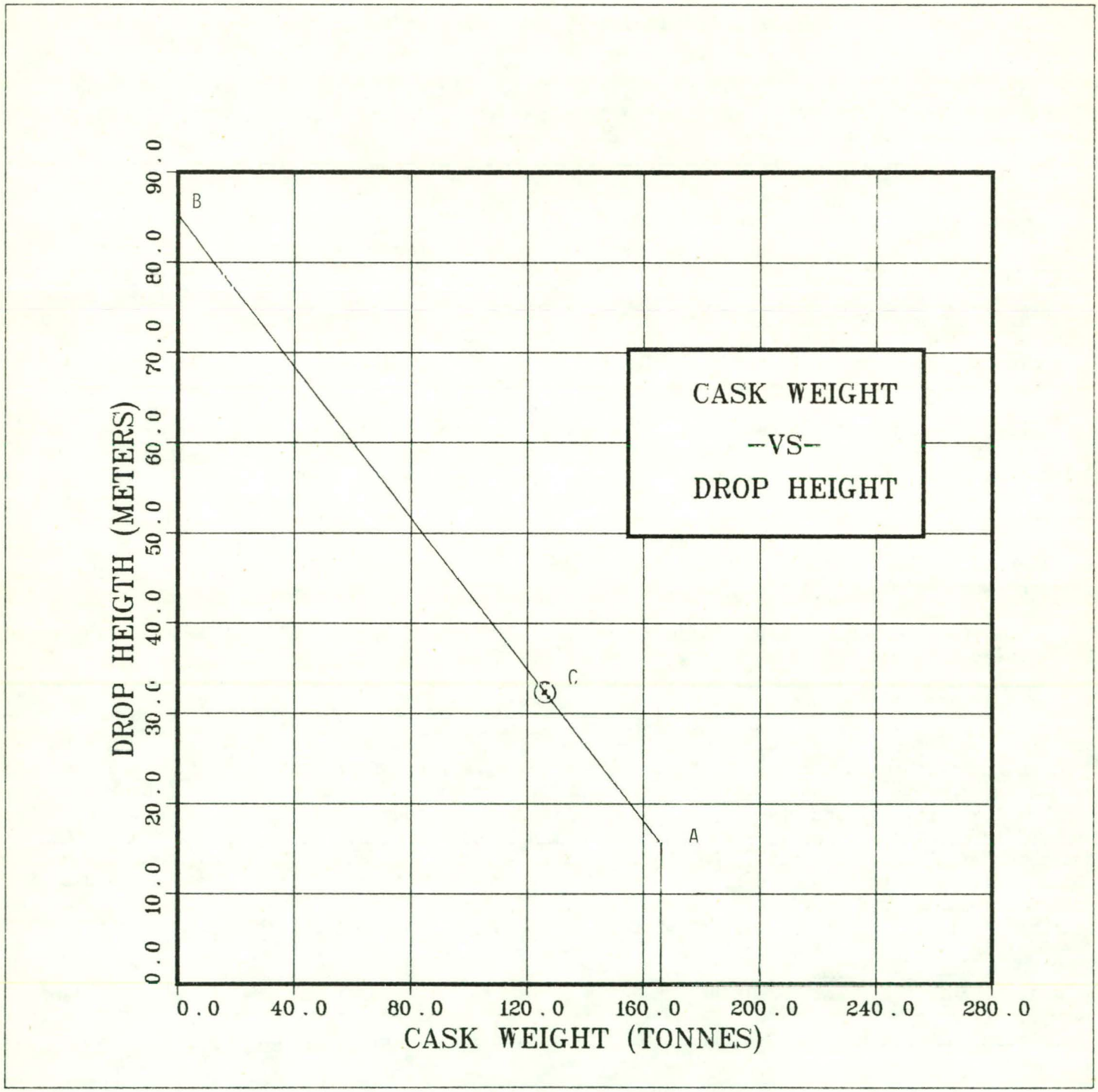


Fig. 2. Cask weight vs drop height.

response. The extreme case of t_d = 0.01 sec produced a maximum dynamic deflection of 1.2 cm. The corresponding peak spring force is 9.0×10^7 N which produces a dynamic subgrade reaction on the pad base of 7.2×10^5 Pa. This value is less than the design value of 1.2×10^6 Pa. The energy absorbed by the soil spring varied from 1.5×10^4 to 5.4×10^5 J for the previously stated assumed values of t_d. The kinetic energy of the cask is $1/2\ mV^2 = 4.9 \times 10^6$ J. Within stated limits, an estimate of the available energy for absorption by the cask is 99.7% for t_d = 0.06 sec and 89% for t_d = 0.01 sec.

The design details of the impact pad are described as follows:
Surface area: 4.88 by 7.92 m,
Bottom area: 9.75 by 12.8 m
Depth: 3.05 m
Pad weight: 670 MT
Transition steps: 4 at 0.81 m horizontal to 0.76 m vertical
Embedded steel armor plate: four 0.15-m layers at 2.44 by 5.5 m
Embedded hold-down bolts: 40 at 3.2 cm diam (ASTM A 307)
Armor plates floated on 0.15-m-thick high-strength grout
f_c concrete: 2.9×10^5 Pa
Reinforcing steel: ASTM A615 Grade 60

CONSTRUCTION

Construction of the drop pad started in April 1977 and was completed in June 1977. The initial site preparation consisted of removing the small, temporary drop pad that weighed 45 MT. The resulting hole in the ground was enlarged to 13 m long, 10 m wide, and 3 m deep.

After the initial layer of reinforcing steel was assembled on the base of the hole, a 15-cm layer of concrete was poured on the base to support the reinforcing steel. The balance of the reinforcing steel was assembled and welded in place. Fourteen metric tons of reinforcing steel was used in the installation. Forty 3.18-cm-diam ASTM A307 anchor bolts for the laminated steel armor plates were placed for embedment in the concrete. Forms were erected for the steps of the pyramid, and each step or layer of concrete was poured and allowed to set (Fig. 3).

Prior to pouring the final step of the pyramid, the steel armor plates were installed to permit any necessary in-place adjustments. Prior to setting the bottom plate in place, a 15 cm layer of quick-setting, high-strength grout was poured on which to float the plate and provide a transition medium for the high compressive stresses. The balance of the laminations were also bonded with grout (Fig. 4).

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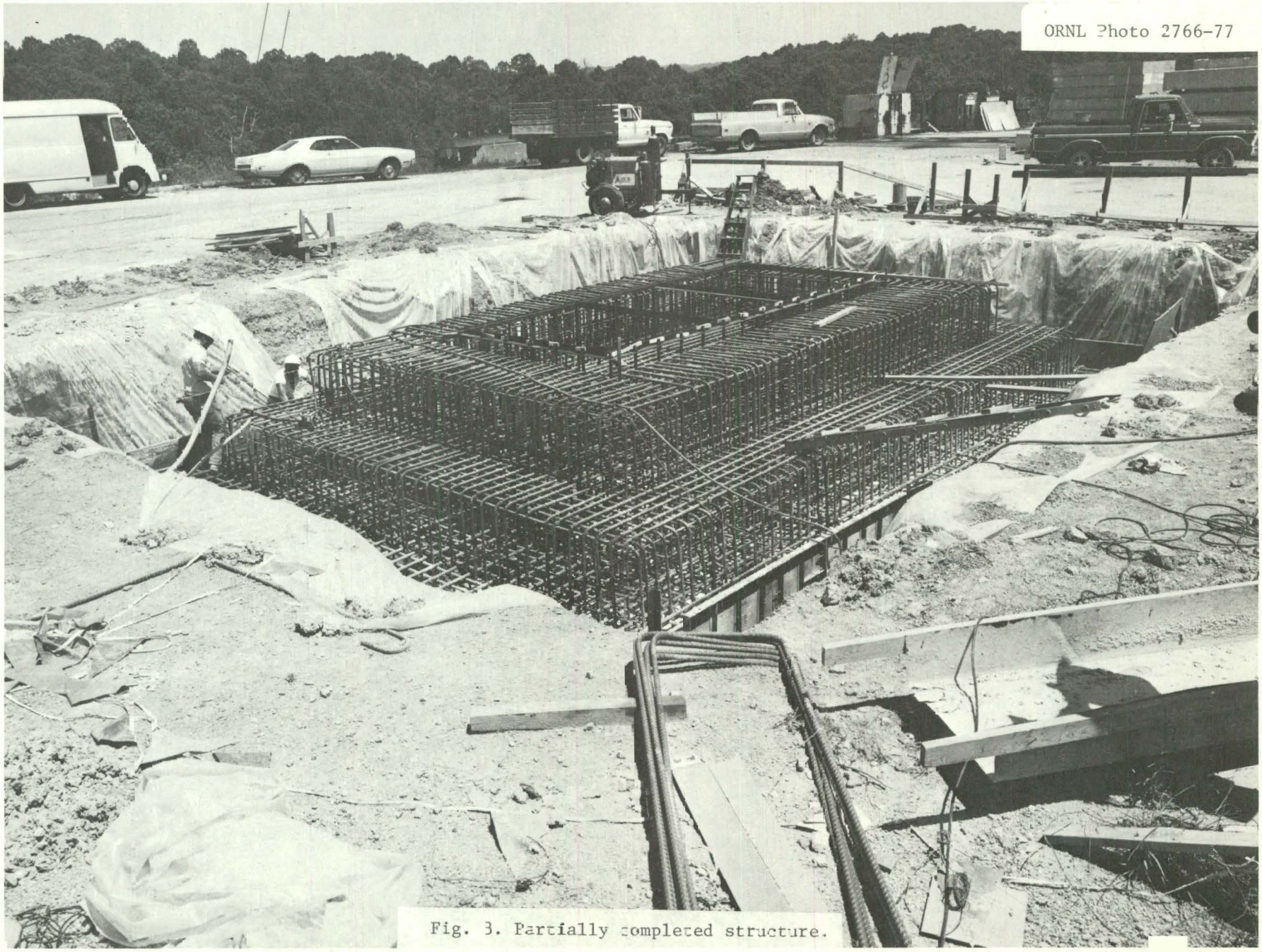


Fig. 3. Partially completed structure.

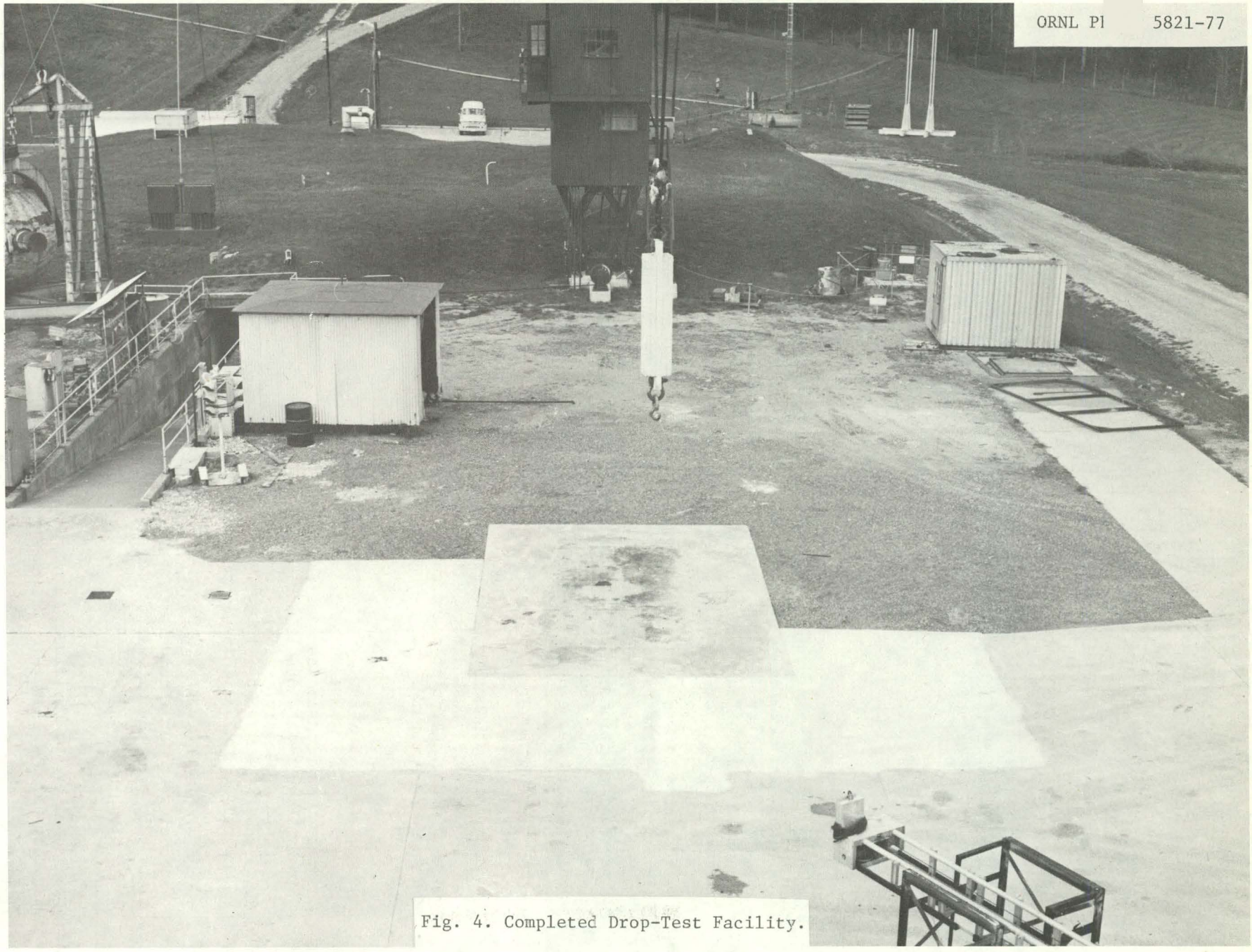


Fig. 4. Completed Drop-Test Facility.

THE OBSOLETE CASK PROGRAM: DROP TESTS OF THE HWCTR CASK*

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ABSTRACT

The Heavy-Water Component Test Reactor (HWCTR) cask was subjected to five impact tests that involved four 9-m drops and one 1-m drop onto a solid essentially unyielding surface. The cask, weighing about 21,000 kg, was first dropped from a height of 9 m at an oblique angle in which the long axis of the cask was oriented at an angle of 30° from the impact surface. This was followed by a 9-m edge impact (corner) with the center of gravity directly above the point of impact. For the third 9-m impact test, a balsa impact limiter was fitted over the closure end, and the cask was dropped in a vertical position so that the impact limiter would strike the impact pad first. In the fourth 9-m impact test, the cask was dropped in a vertical position onto the closure end without the impact limiter. In the final test, the cask was released in a horizontal orientation onto a 15.25-cm-diam steel pin from a height of 1 m. The hoisting equipment at the Tower Shielding Facility (TSF) was used to make the drops. The cask was instrumented with both piezoelectric accelerometers and bonded-resistance strain gages. Cask damage was predicted using analytical techniques prior to each drop. Calculated and actual damage agreed within the limits of engineering accuracy.

INTRODUCTION

The Obsolete Cask Program was initiated to study the behavior of large casks under specified destructive test conditions[†] and to establish the adequacy of the calculational techniques that are presently employed to predict cask damage. The test used to produce severe damage consisted of a free-fall by the cask from an elevation of 9 m with impact on an essentially unyielding surface. The cask was oriented to produce maximum damage in each test.

The 21,000-kg Heavy-Water Component Test Reactor (HWCTR) cask, originally designed to transport special reactor fuel elements at the Savannah River Plant site, was the first large cask tested in this program. It was tested at the Tower Shielding Facility (TSF), which has the capability of lifting heavy casks to great heights.

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TEST CASK DESCRIPTION

The cask, as tested, is approximately 4.5 m long, 75 cm in diameter, and weighs about 21,000 kg (Fig. 1). The outer shell is composed of 1.25-cm-thick laminated steel, which udes an outside layer of stainless steel (0.25-cm-thick) and a layer of carbon steel; the r shell is 0.94-cm-thick carbon steel. Lead shielding (24 cm thick) is located between the inner and outer shells. The inner-cask cavity is 25.4 cm in diameter and 4.06 m long. The end closure of the cask consists of a stepped plug held in place by 20 Acme square-threaded studs and nuts, each 2 cm in diameter, and is sealed with a 0.6-cm-thick, flat neoprene gasket. The cask was designed for pool loading and unloading and is equipped with drain lines and plugs.

The original cask was modified to ensure that it would behave more like those that are currently being designed. The modifications included (1) the removal of exterior valves and valve guards, and (2) the removal of a 22-cm "pivot trunnion" section from the end of the cask, thus exposing an inner flat head, which then served as the end of the cask.

TEST FACILITIES AND EQUIPMENT

The TSF, located at the Oak Ridge National Laboratory (ORNL), was the site of the impact tests. The TSF consists of four towers, each 96 m high, set in a rectangular array measuring 30 m by 60 m (see Fig. 2). Each tower is guyed with two pairs of 5-cm-diam cables.

The tests were conducted between towers III and IV, which are separated by 60 m. The hoisting system, connected to the tops of the towers, is capable of lifting and releasing casks weighing up to 35,000 kg about 55 m off the ground (or casks weighing 100,000 kg can be lifted about 9 m). The temporary drop pad at the test facility (now replaced by a permanent pad) consisted of a block of cast concrete weighing approximately 45,000 kg that had been set into the ground. The impact surface was covered with a 10-cm-thick piece of armor plate, measuring approximately 2.5 m long by 1 m wide.

An existing drop-test release mechanism was strengthened to accommodate loads in excess of 27,000 kg. The pivoting release hook in this mechanism is held in place with a frangible nut that can be fractured by a pyrotechnic charge, permitting the hook to swing free and release its load.

The instrumentation system utilized for the cask drop tests had three functions: (1) to monitor the deceleration of the cask body upon impact, (2) to monitor the shock imparted to the steel and concrete impact pad and components of the facility, (3) to monitor the circumferential and longitudinal strain imparted to the cask's steel shell upon impact.

OBLIQUE DROP TEST

The purpose of the oblique drop test was to determine whether impact at such an angle would cause a large cask with a great length-to-diameter ratio (the L/D is 5.5 for the HWCTR cask) to buckle in the middle, resulting in a banana-shaped cask. The cask was dropped from a height of 9 m onto the impact pad at a 30° angle to the horizontal. It rotated rapidly after initial contact with the pad and impacted for the second time on the other end of the cask.

The physical damage to the HWCTR cask as a result of the oblique impact is illustrated in Fig. 3. The damage received at the point of primary impact is shown by the indicated deformation; the cracks extended completely through the outer shell. The shell deformation at the other end resulted from the secondary impact with the edge of the concrete pad. Some deformation of the inner cavity is noted as a result of this secondary impact.

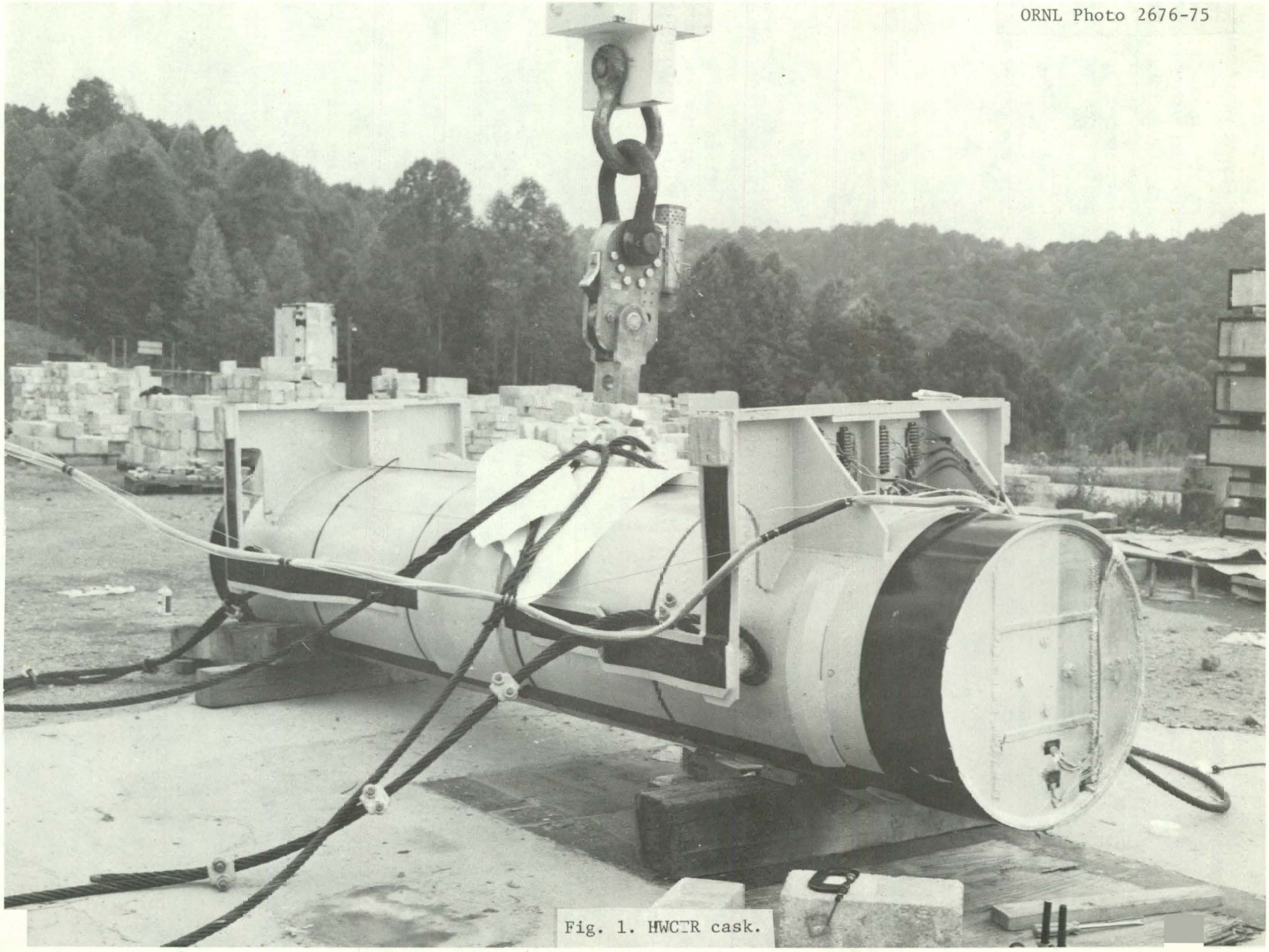


Fig. 1. HWCTR cask.

ORNL Photo 48424-R1

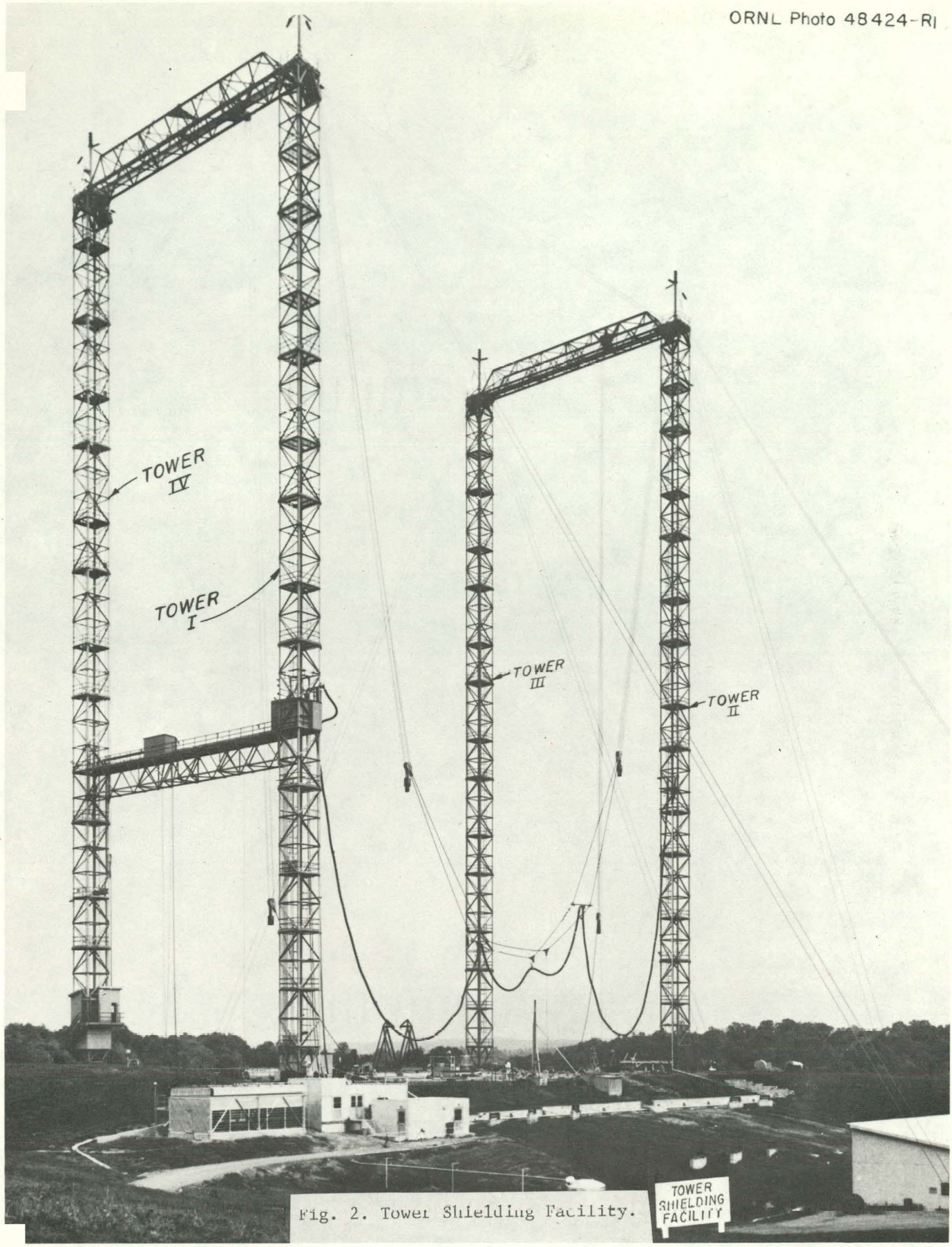


Fig. 2. Tower Shielding Facility.

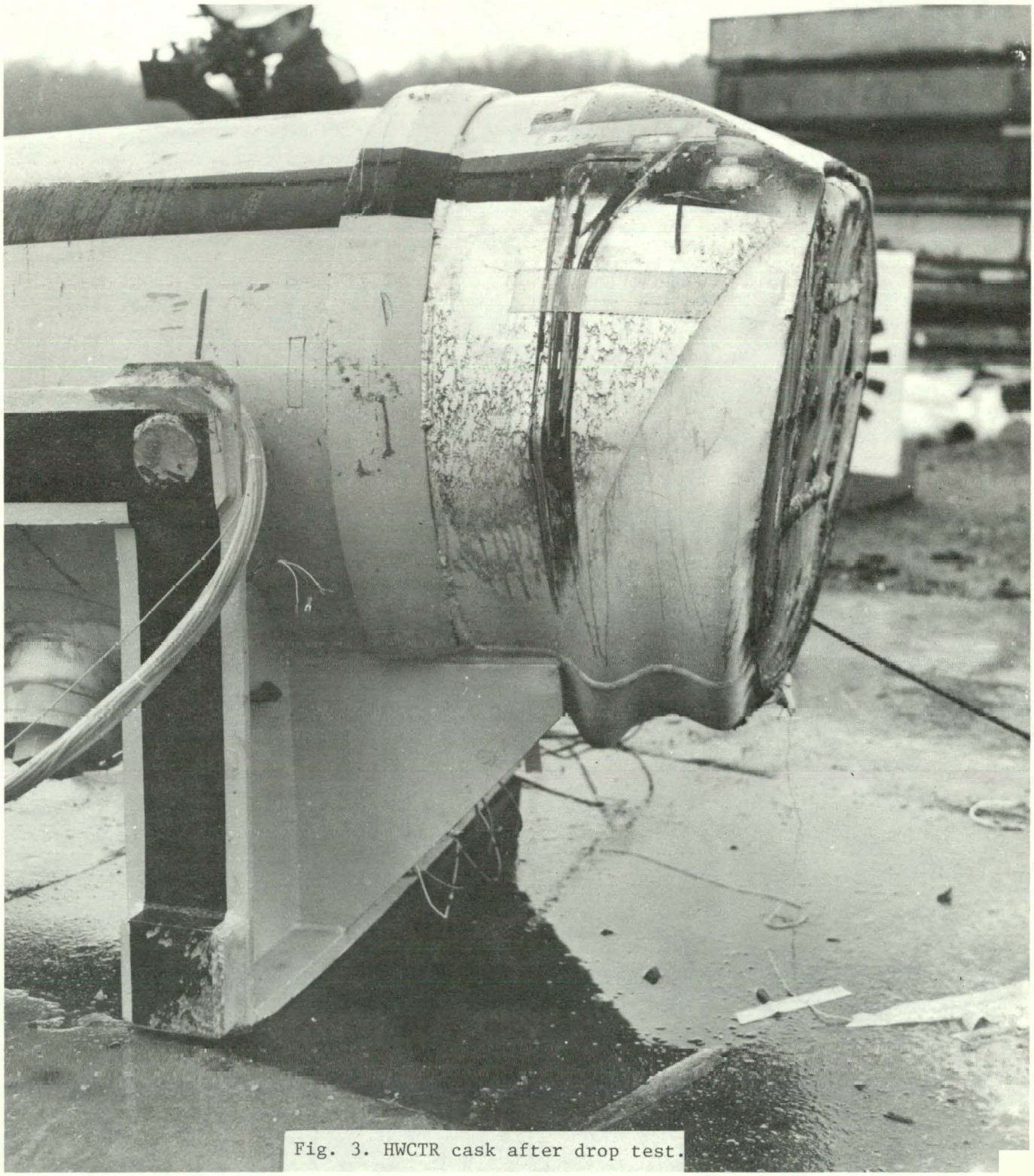


Fig. 3. HWCTR cask after drop test.

A frame-by-frame examination of the high-speed film taken at the time of the impact clearly shows the rotational energy effect due to the configuration in which the cask was dropped. The cask was at a 30° angle with the horizontal and had a velocity of 13.44 m/sec at the time of the initial impact. Between the initial impact and the secondary impact, the closure end of the cask rotated angularly for a distance of 2.39 m in 0.2 sec. The average velocity during this span of time was 21.75 m/sec, which is equivalent to a free-fall from a height of 24.11 m.

CENTER-OF-GRAVITY DROP TEST

The cask, suspended at an 11° angle with the vertical, was dropped from a height of 9 m onto the impact surface. This angle placed the center of gravity of the cask in a vertical plane directly over the point of impact. This particular configuration was selected to utilize the kinetic energy available in order to produce maximum damage to the cask at one specific point.

The instrumentation for the cask consisted of 10 strain gages (equally spaced along the side) and two accelerometers located at the closure end. The gages and accelerometers were connected to recording equipment by means of an umbilical cable. Unfortunately, the Zonics data recorders and storage banks were triggered prematurely; therefore, no data were recorded on these instruments.

Damage from the center-of-gravity drop is illustrated in Fig. 3. The corner deformed, and the shell was damaged; the convolutions are shown in the figure. No cracks were observed as a result of the second drop. Damage from the secondary impact was not considered significant.

After the center-of-gravity drop test, the HWCTR cask was pressurized with air to a gage pressure of 0.21 kg/cm^2.

DROP TEST WITH IMPACT LIMITER

The HWCTR cask, fitted with a balsa impact limiter over the closure end, was dropped in a vertical position from a height of 9 m onto the drop pad at the Tower Shielding Facility.

At the time of impact, the period of deceleration was sufficiently long so that the deformation and compression of the balsa impact limiter could be clearly observed. The balsa was compressed from an initial length of 44 cm to a final length of 15 cm. The steel case of the limiter was folded up into six distinct accordion folds and was split open longitudinally due to the hoop stresses developed during impact. No scratches were observed on the inside painted surface of the limiter, and no discernible damage to the cask closure was detected.

One accelerometer mounted on the lead shielding at the bottom end of the cask opposite the point of impact registered an impact force of 80 g.

DROP TEST WITHOUT IMPACT LIMITER

The HWCTR cask was dropped in a vertical position onto the closure end without the impact limiter from a height of 9 m at the TSF. When the cask impacted onto the drop pad, it remained in a vertical position.

In the area of the closure, all studs and nuts, which normally protrude 3 cm from the lid flange, were destroyed. The neoprene gasket was flattened and extruded about 2 to 3 cm from the seal area, and the outer shell had buckled in several locations. Steel that had been welded to the outer shell to form the support for the cask in its normal horizontal position strengthened it somewhat, and the outer shell damage was relatively minor. Imprints of the studs, 0.16 to 0.32 cm deep, were left on the surface of the armor plate section of the drop pad.

The accelerometers located on the inside surface of the plug closure shielding registered an impact loading of 1240 g. All of the instrumentation functioned properly, and clear traces were obtained from the oscillographs as well as the oscilloscopes.

An attempt was made to pressurize the cask with air. In doing so, a large leak was found on the bottom end of the cask close to where the drain line was welded to the outer shell. It is surmised that one of the joints in the drain line had cracked as a result of the end-on impact, thus permitting air to leak from the joint through a crack in the outer shell. This crack in the outer shell resulted from the stresses produced by the oblique drop test. Since there had been no complete flow path, the cavity was capable of holding pressure prior to this test.

PUNCTURE TEST

The HWCTR cask was dropped in a horizontal position from a height of 1 m onto the upper end of a solid, mild steel bar of circular section, 15 cm in diameter. After the drop, the cask balanced in place on the bar. Figure 4 shows the damage to the cask from the steel bar.

The piezoelectric accelerometer signaled a loading of 3 g at the time of impact. Of the three sets of strain gages mounted on the cask, only two sets survived the test. The reading from these gages indicated an axial strain of 4000 ucm/cm.

COMPARISION OF CALCULATED AND EXPERIMENTAL RESULTS

Existing ORNL computer routines were used to calculate response and predict the damage to the cask as a result of these drops. For the oblique drop, it was predicted that the deformation would be between 15.82 and 22.5 cm. The measured value was 11.4 cm.

It was further calculated that the deformation from the center-of-gravity drop would be between 10.9 and 16.3 cm. This prediction compares favorably with the actual deformation of 14.61 cm.

CONCLUSIONS

The experimentally determined and calculated responses of the test cask were in sufficient agreement to warrant continued use of the basic calculational method employed. The severe deformation that occurred as a result of the center-of-gravity drop test vividly demonstrates the tremendous concentration of force in an impact of this configuration. The oblique drop test effectively showed that considerable rotational energy was imparted to the cask as a result of the initial impact and that this energy was of sufficient force to cause visible damage to the cask.

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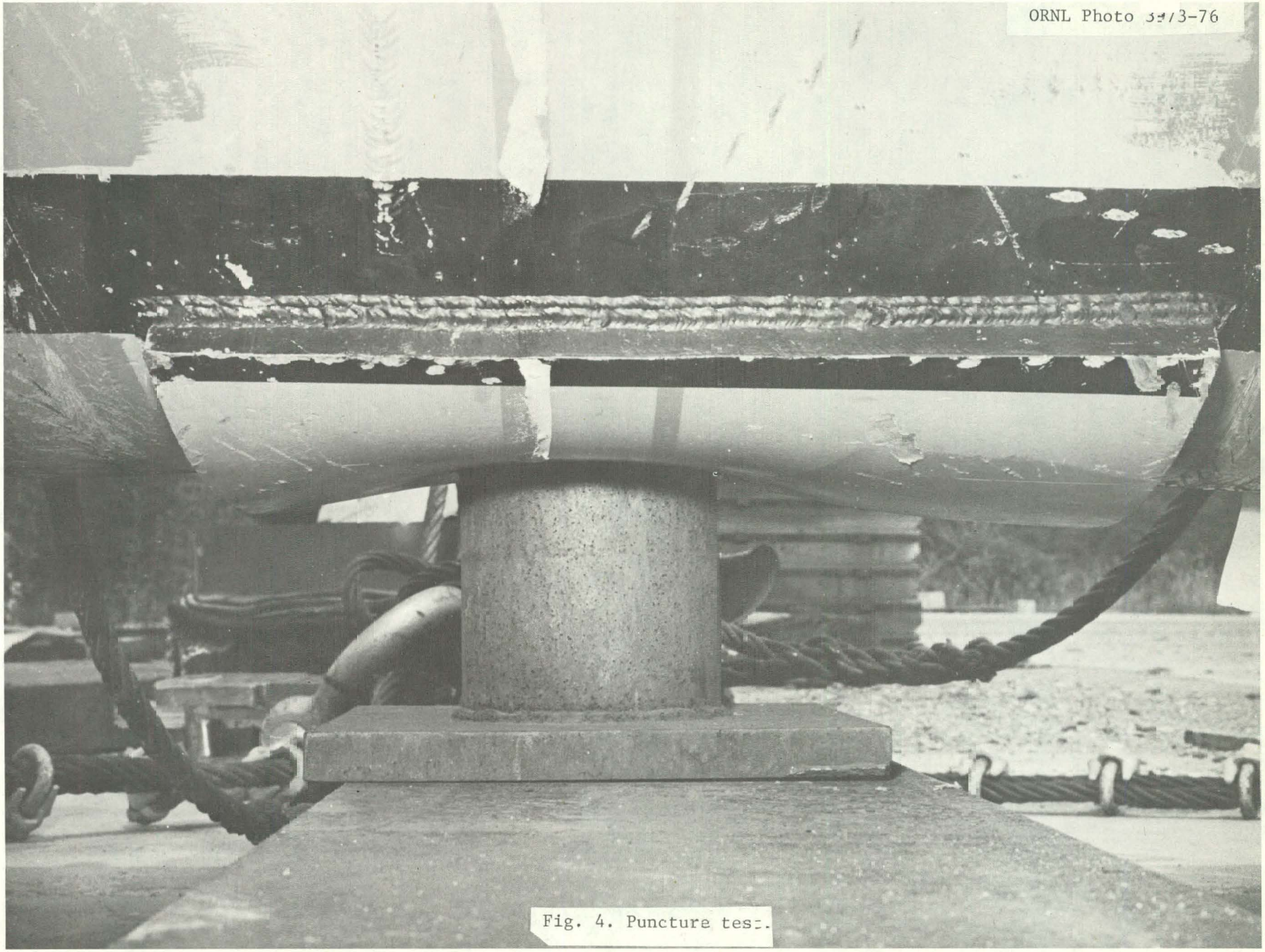


Fig. 4. Puncture test.

MODEL TESTING OF HEAVY SPENT FUEL CASKS AND WASTE PACKAGINGS

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Introduction

The IAEA Regulations, Section VII, No. 701 (c), allow the performance of tests with models of appropriate scale. Therefore model testing has become an often used method in the procedure of approving. Model testing of heavy spent fuel casks and waste packagings is forced by a cost-benefit analysis and a limited capability of test equipment. To verify model laws and to criticize scaling, two test series were performed by BAM. The first test serie carried out with lead cylinders was to testify the right scaling of deformation. Unprotected lead cylinders were dropped from a height of 9 m onto an unyielding surface in an inclined position given by the center of gravity. Deformation was measured. Lead was used due to its nearly ideal plastic behaviour. The second test serie was carried out with rigid steel cylinders as models of a 35 t waste packaging in different scales. These tests were to testify the scaling of forces and of deformations in the elastic range.

Model Testing

The model laws are well known. The deformation corresponds to the scaling factor, the acceleration corresponds to the inversed scaling factor and the stress will be independent from scaling. Modelling in a certain range of scaling is therefore accepted as a reasonable method. But due to the impossibility of scaling the material matrix and the welding procedures e.g. some tests to verify the validity of model results were made.They should be used in the field of public acceptance, too. At first a series of lead cylinders were dropped from 9 m. The length-to-diameter ratio ranged from 1,0 to 3,0 . The weight varied in all cases of L/D-ratios from 10 kg to 40 kg.

All drops were carried out in an inclined position to the center of gravity producing heavy deformations in the plastic range. In this line accelerameters were installed. Beside this measurements pictures by high speed camera gave a survey of the impact behaviour and was used for a rough control procedure in writing the deformation-time-curves. A lot of cylinders were dropped due to difficulties in some test parameters. In some cases the drop position did not correspond to the one in which measurements of acceleration could be taken. In other tests

we had difficulties with the triggering of the high speed camera and with the acceleration measurement devices. All these problems could be solved by strict use of a QA procedure especially developed for testing programs. We could show the reproducibility of test results either by plastic deformation or by measurement of acceleration. Therefore the given data in table 1 are not only single figures but the results of series. The scaling factor from the given dimension is compared with the scaling factor taken from the recorded deformation. Up to the ratio L/D = 2,0 there is only a slight difference, expressed in the last column of table 1. We think, a difference up to about 5% is just the range in which the measurement technique operates. For the very long cylinders the bending effect becomes more effective than the local deformation in the impact area. In addition we have a much greater effect deviation from drop angle, the comparison shows more differences than for shorter cylinders. But as an practical result the scaling of plastic deformation is shown to be in good relation to the scaling itself. That may be helpful in discussing the geometrie of spent fuel cask lids and the effects of deformation to the tightness.

As mentioned before some series of drop tests had to be performed. We got reproducable results of acceleration measurements from the second series. In the last column of table 1 we see e.g. the maximum g-data taken from the records. The good correlation to the scaling is the most important result for this paper.

Later on we tested the model series of a 35t Waste Packaging. The Waste Packaging contains 7 200l drums with waste bound to concrete or bitumen. It is used for shipments from the Karlsruhe Research Center to the Asse Salt Mine, the final disposal in the FRG today. It can be described as a very short and rigid cylinder. It has a welded steel construction with a bolted lid. One phenomenon must be mentioned. Dropping this rigid cylinder flat onto the bottom, we always observed rupture of some lid bolts. As we learned from pictures taken by the high speed camera we always observed a distorsion of the lid. Failure of bolts was reproducable. A significant difference between drop tests at ambient temperatures and at a temperature of 233 k could not be observed due to the special form of the bolts and the material. Safety of packaging was reached by additional rigid shearing pins fixing the lid. The reason for the scaling of bolt rupture is a certain configuration in vibration behaviour. Looking to table 2 we find at first, that the stress is independent from the scaling. The elastic deformation corresponds to the scaling factor, at all events in the range of tolerance of the measurements. The data for the calculated frequence represents data for a plate edges fixed. The comparison with the frequence of the lid measured during the impact is relativly good up to a scaling factor of 2,0 . The gap between calculated and measured frequencies for higher scaling factors demands more and deeper investigations.

Summing up the results of model testing we come to the conclusion that in the range up to a scaling factor of about 2 or 3 model testing could be verified. But further investigations, better control of test parameters and collection of other data helpful to verify model laws are necessary. With a small safety factor of 1.2 results of model tests can be used for the dicission in the approval procedure especially the figures in forces, accelerations and deformations.

An other chapter is the modelling of tightness. We only consider smaller leaks because leak rates greater than 10^{-3} Torr.l/s will indicate severe failure. For the smaller leaks we have to differentiate into laminar and molecular flow. In the latter case permeation is to be seen

as a special sort of molecular flow. Both the laminar and the molecular flow through small leaks correspond to the inversed scaling factor, the same gasket material is presumed. Permeation will get more intensive in the prototype than in the model case. Using in most cases elastomere gaskets we will get mixed flows. Actual measurements on a 1/2 scale model of a spent fuel cask showed the dominant values coming from permeation. Having in mind a maximum scaling factor of 3 we don't see greater problems in the scaling of tightness due to the magnitudes of common leak tightness of about 10^{-8} Torr·l/s . Besides this we can see the tightness proportional to the scaling factor as conservative.

Spent Fuel Casks

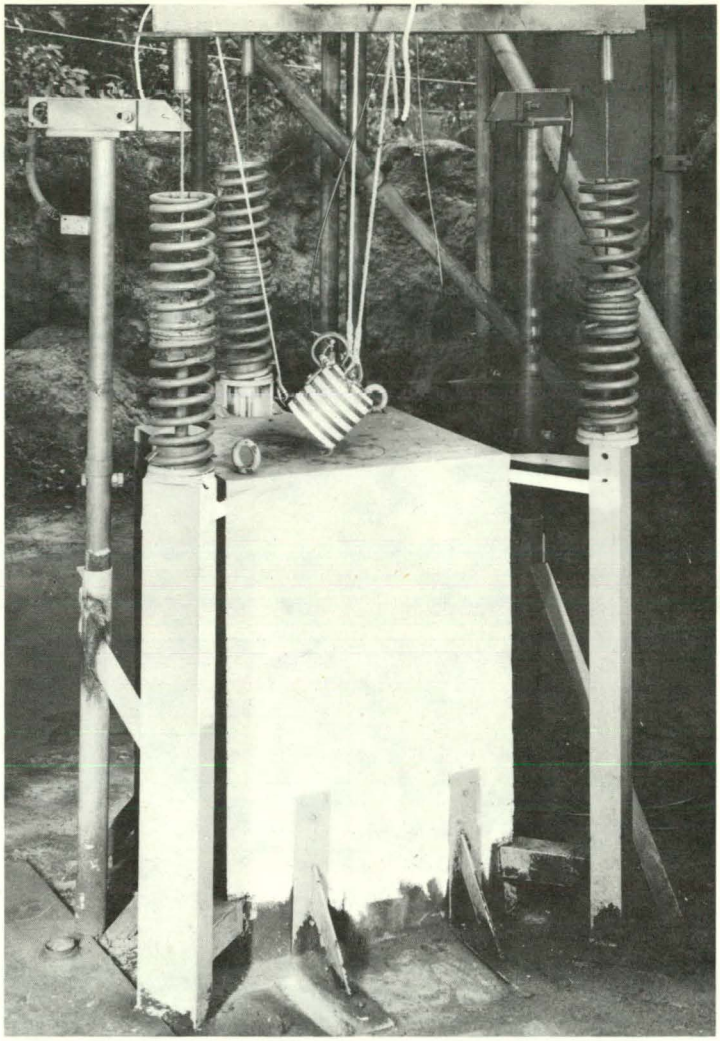
In the last few month we performed drop tests on the TN 12 cask, able to carry 12 Biblis Elements. We got a first survey by testing a 1/12 scale model. Due to the above given data we don't believe that measuring in this range of scaling can be extrapolated up to the prototype. But the second test series with a 1/3 scale model with a weight of about 4 t gave sufficient results in the detail for the procedure of approval. Also it was absolutly necessary to perform the drop and leak tests under QA conditions. So we will have a very detailed test report we are just working on. Only some data shall be given.

Accelerations were measured by recording all frequencies which could be taken by the measuring devices, for example up to 10 kHz. Fig. 1 shows the record printed by the computer plotter. The acceleration-time-curve was fixed by 1024 digit numbers on a punch tape. We than did a Fourier-analysis. Recorded signals were filtered mathematically in fig. 2 and 3. It is evident that the very sharp peak (600g) in fig. 1 are not effective. But the maximum from fig. 3, the acceleration of 260 g has taken into account. The fuel elements and their whole structure will be forced by this value of 260 g. This is the datum needed to calculate deformation and at least unclear safety.

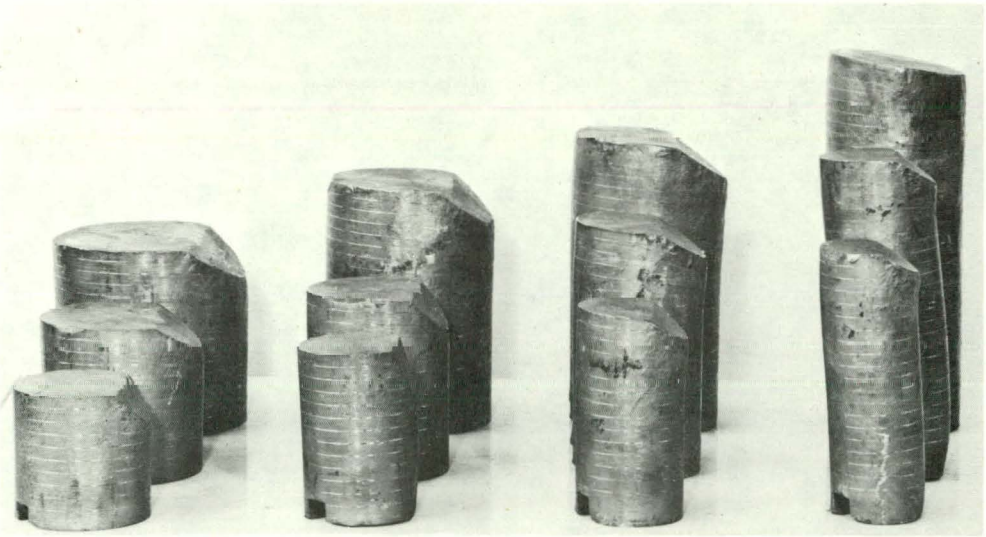
Helium leak testing were done with a sensitivity of 10^{-10} Torr·l/s. Having in mind the permeation rate of elastomere gaskets the tightness after drop test was the same as before tests, namely 10^{-8} Torr·l/s. Even using the conservative procedure in modelling leak tightness, the prototype of TN 12 would be tight after mechanical impact. All this work is to be seen as a basis for extraperlating results given by super type B tests. We refit a 1/2 scale model of the TN 8/9 cask, used in the FRG for shipping spent fuel elements from the reactor station such as Stade to reprocessing facilities.

Spent fuel elements are mostly shipped by truck and train. With a free fall drop from 200 m we cover the possible maximum drop from a bridge and all credible crash velocities in land transport. Three drops were performed. For the first drop we had loamy soil as target. The container was dropped from a CH-53 G-helicopter in a vertical position. The impact position was the same. A complete immersion into the soil was observed. Due to this immersion some calculations were started about the heat transfer after such an accident. The problem is the wet loamy soil which covered the whole surface of the TN 8/9-model. After a short period of time you will find a layer of dry loamy soil around the packaging becoming an additional insulating part of the heat transfer system. We spent a lot of time in the salvage of the TN 8/9 - cask. The second drop target was wet sandy soil. An immersion of about 2/3 was observed. Even after the third drop onto a simmulated "Autobahn" the cask remained at least bubble test tight. Due to the drop position onto the lid under an angle of about 20° from the vertical line some of the lid bolts sheared off. For our knowledge that is the severest damage observed in super type B tests.

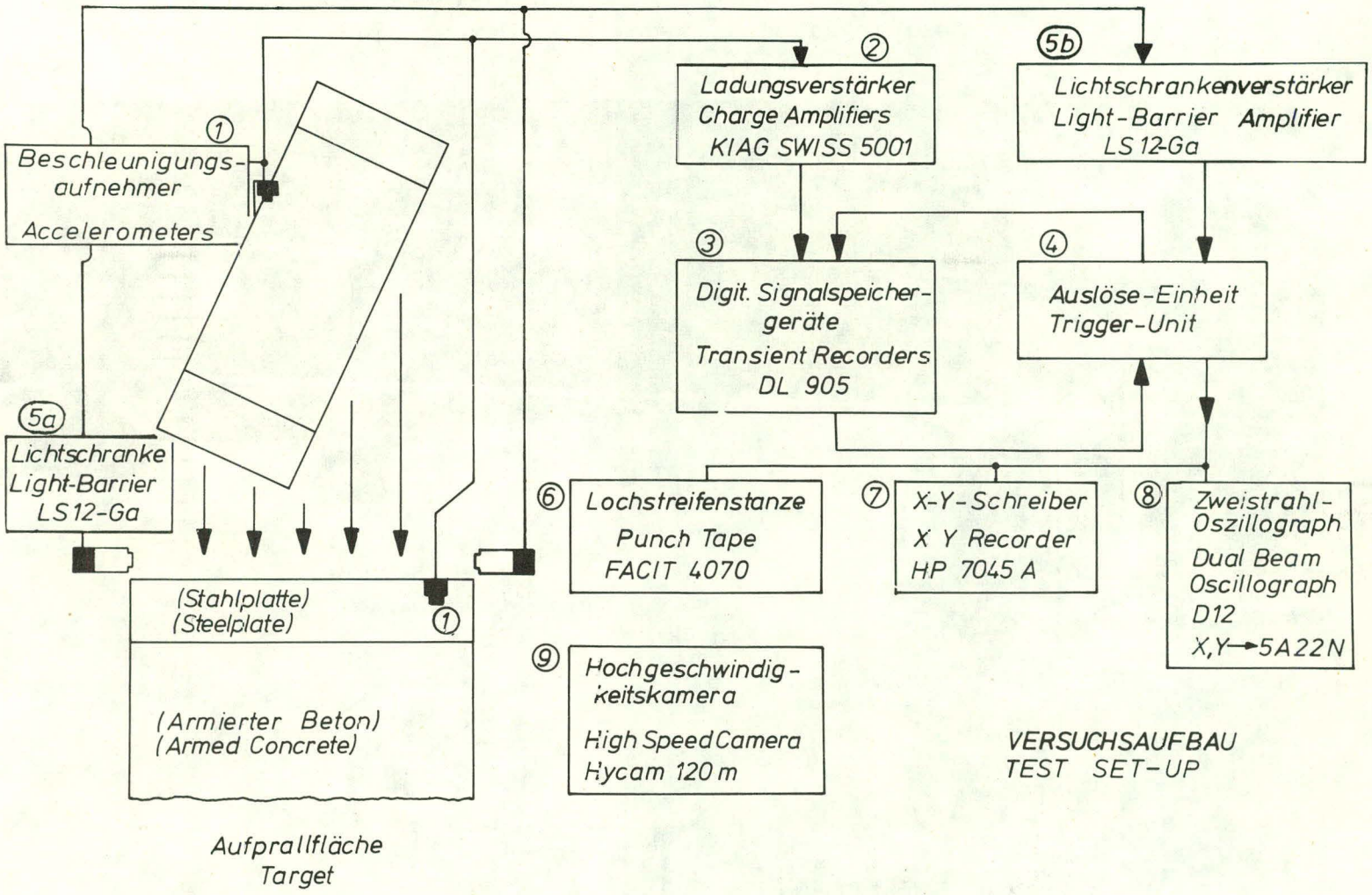
Results given in this paper are from research work just going on. Therefore we and you will miss some further and detailed data, which can only be given later on. But for this symposium reporting without judging might be helpful especially in the field of public acceptance.



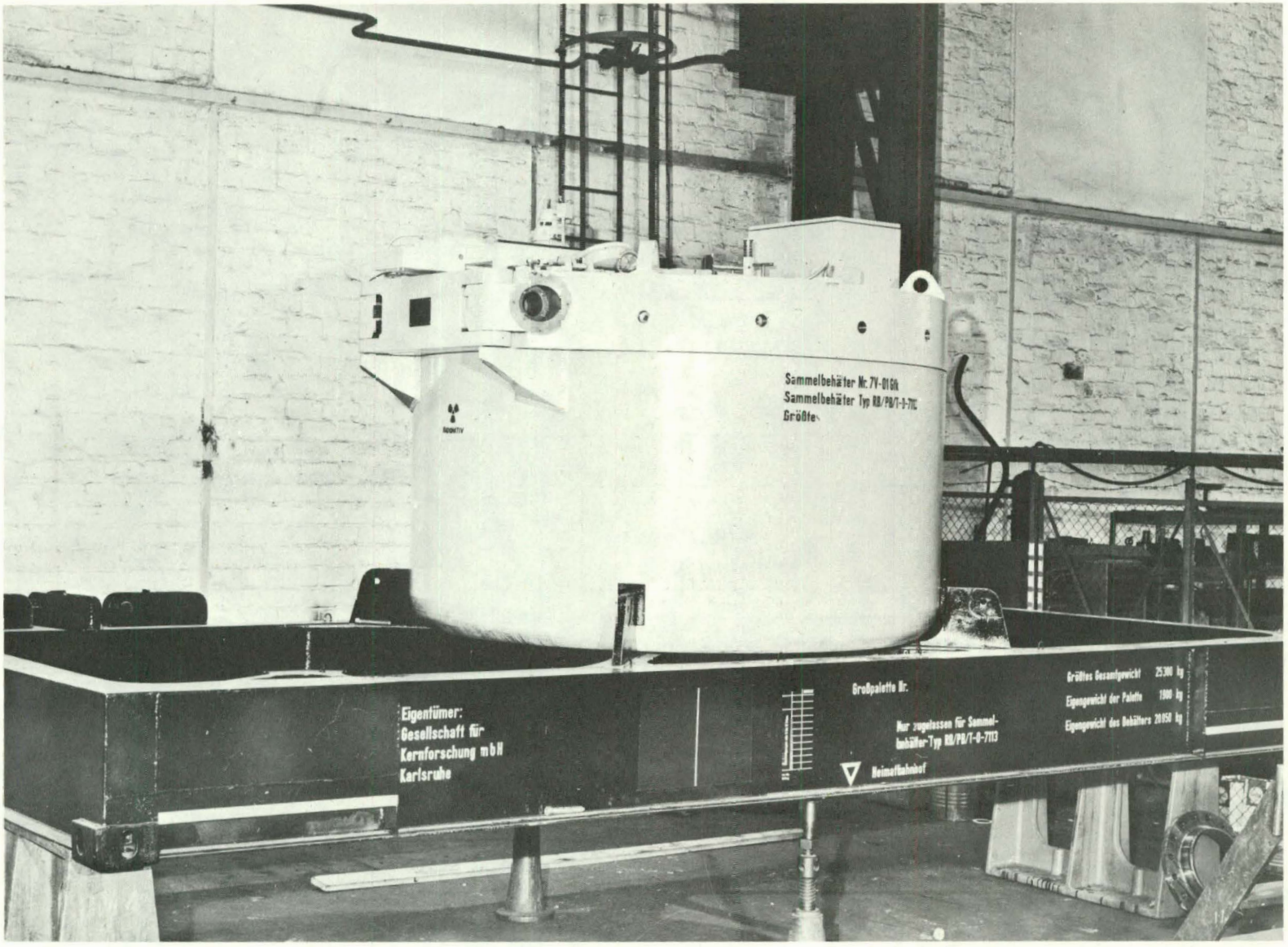
Test Facility for Guided Drops



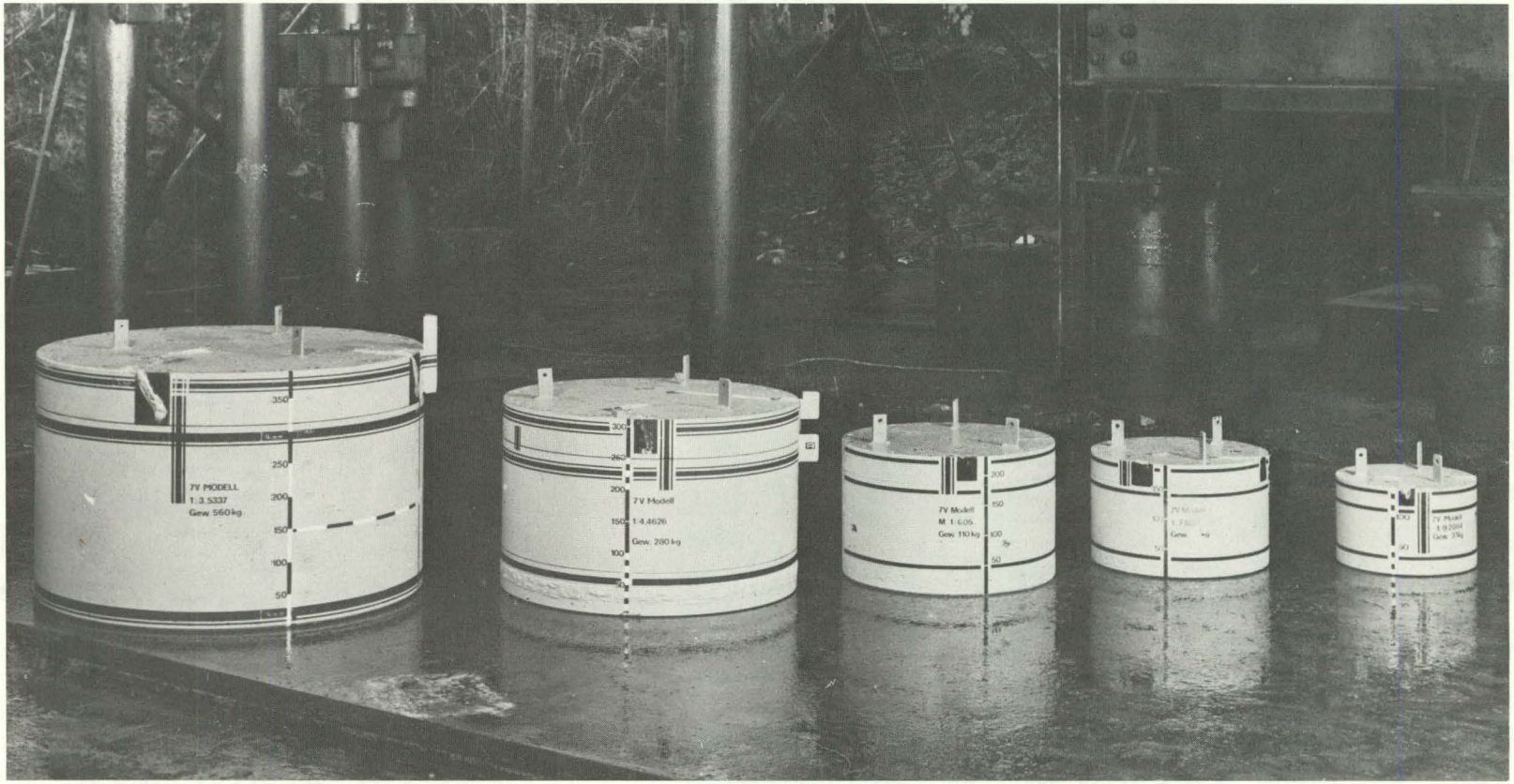
Series of Lead Cylinders after 9m Drop



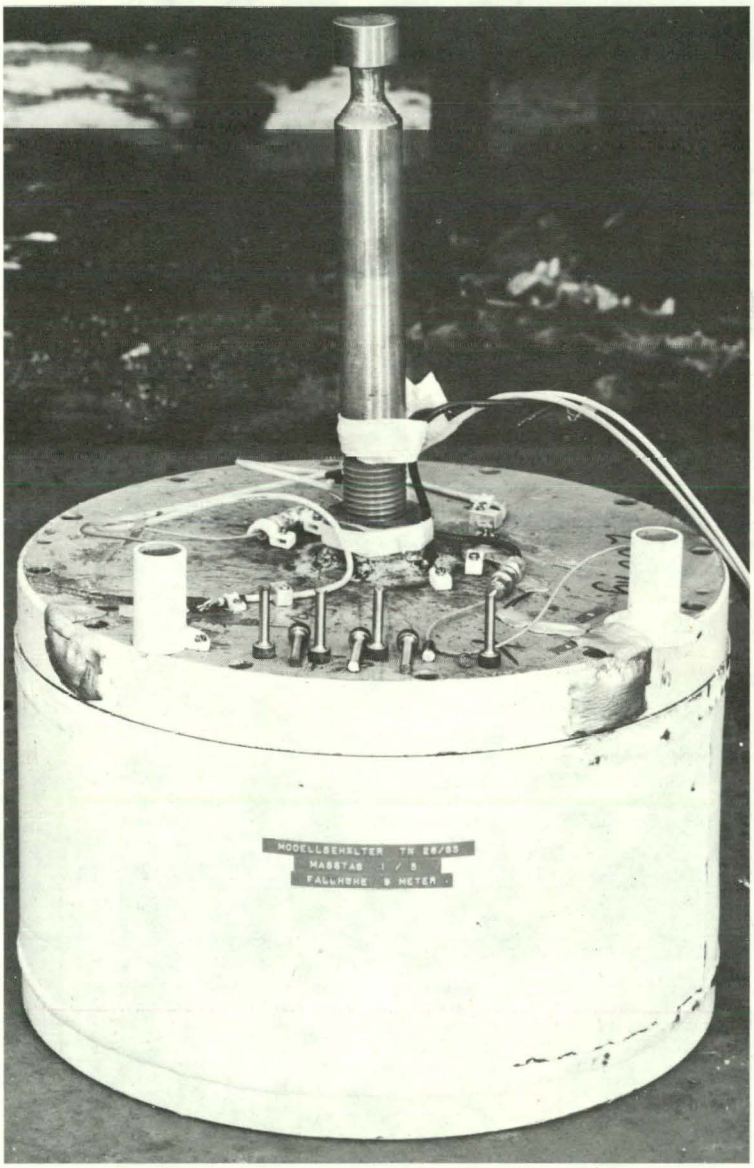
Test Set-up Drop Test Facility at BAM



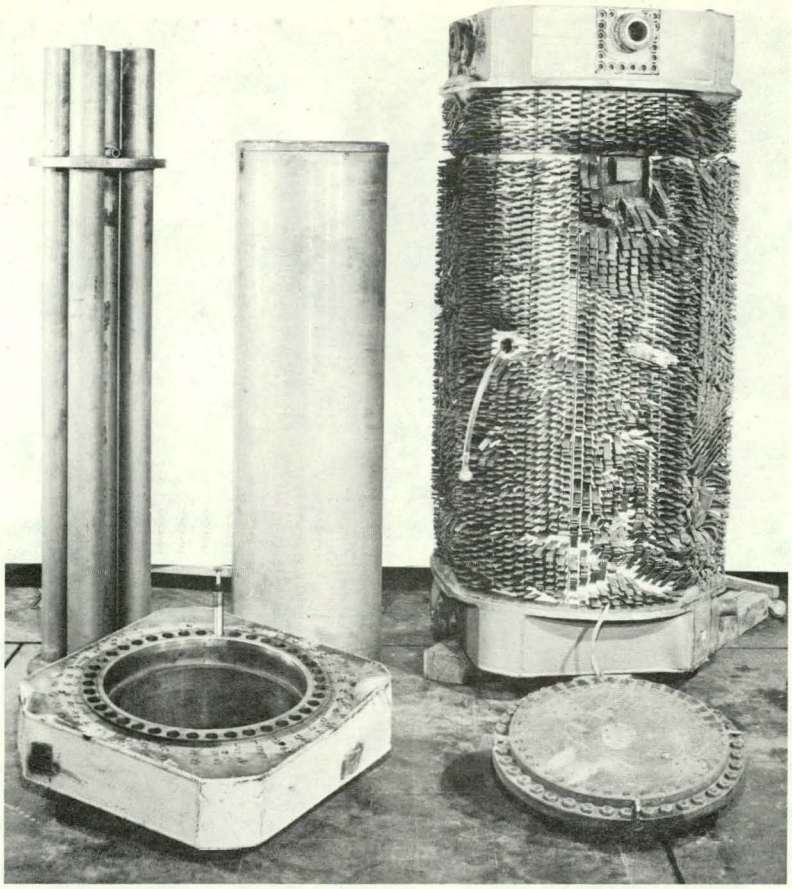
Waste Packaging 7V or Trarsport Palette
Gross Weight 25t



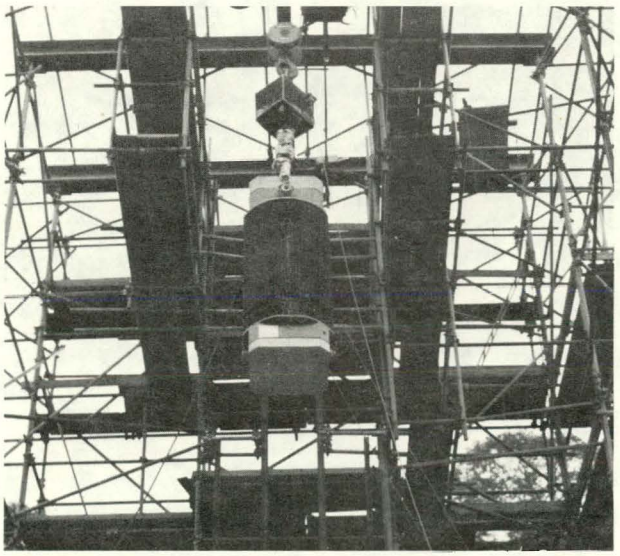
Model Series of Waste Packaging 7V



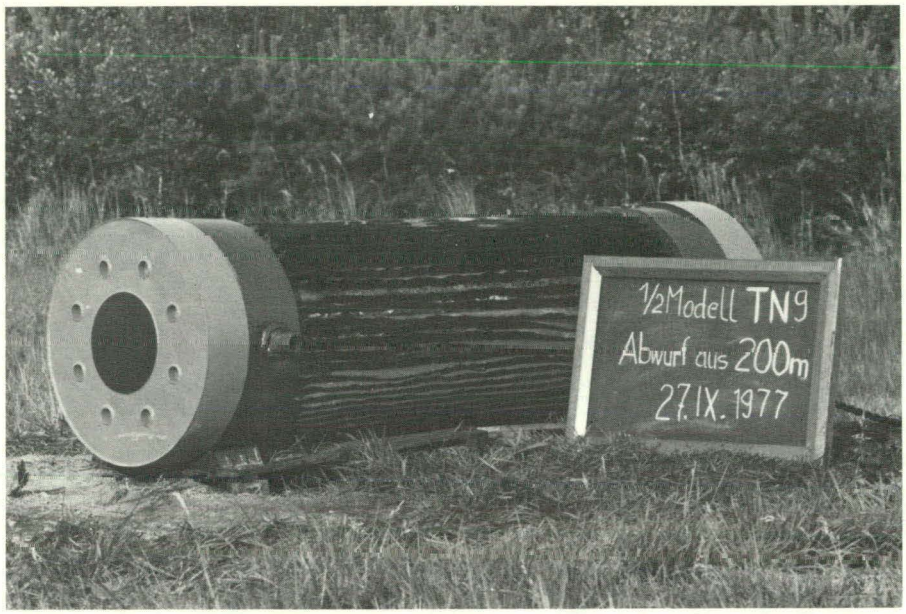
1:5 Scale Model TN 28 Waste Packaging after 9m Drop; Failure of Lid Bolts



1:3 Scale Model TN 12 - dissembled



1:3 Scale Model TN 12 at Drop Test Facility



1:2 Scale Model TN 8/9 Cask (4t) for Spent Fuel Elements



4t Cask TN 8/9 after 200m Drop onto a Concrete Target

Table 1: Deformation of Lead Cylinders

L/D	Weight [kg]	Scaling-Factor	Deformation [mm]	Scaling-Factor from Deformation	Difference in Scaling-Factor [%]	Remarks
1	10	1,57	16,0	1,63	3,5	a_{max}= 1260g } 2nd series
	20	1,25	21,0	1,24	1,0	a_{max}= 980g } 2nd series
	40	1	26,0	1	---	a_{max}= 800g } 2nd series
1,5	10	1,61	16,5	1,54	4,3	
	20	1,26	---	---	---	
	40	1	25,4	1	---	
2,0	10	1,57	15,0	1,65	5,3	
	20	1,24	20,4	1,22	1,9	
	40	1	24,8	1	---	
3,0	10	1,59	(16,5)	(1,45)	(8,6)	Total Bending and Local Deformation
	20	1,28	21,5	1,12	12,8	
	40	1	24,0	1	---	

Table 2: Data of Waste Packaging

Scaling-Factor	Strain Rates Middle of Lid [%]	Screw 1 [%]	Screw 2 [%]	Deformation [mm]	Scaling-Factor (Deformation)	Frequency Calculated [kHz]	Frequency Measured [kHz]
1,00	---	---	---	---	---	1,70	1,67
1,26	0,054	---	0,080	0,153	1,24	2,11	2,00
1,72	0,060	---	---	0,124	1,68	2,89	3,53
2,03	0,065	0,110	---	0,104*)	---	3,33	4,00
2,60	0,052	0,110	0,085	0,073	2,60	4,26	6,67

*) From the average Strain rate 0,058

EXPERIMENTAL STUDIES WITH LEAD-SHIELDED SCALE MODEL SHIPPING CONTAINERS

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ABSTRACT

This paper presents results of preliminary dynamic impact experiments that were conducted with replica scale models of radioactive material shipping containers for assessment of potential safety and environmental consequences. One-eighth, 1/4, and 1/2 linear scale models were fabricated and tested to predict full-scale response of shipping containers employing unbonded lead shielding. Free fall impact experiments are discussed for plain cylindrical stainless steel shells, stainless steel shells filled with lead, and replica scale models of lead-shielded shipping containers. Also, a few standard dynamic and static experiments were conducted to determine the mechanical properties of the lead and other materials used to fabricate the models. The structural response of the models was studied for free fall impact onto an unyielding surface at various drop heights, angles to the impact surface, with and without impact limiters, and with different drop pad surface composition. The test temperature range examined was between -40 and 175 C. Appropriate scale modeling laws are developed in detail in Reference 5. These are substantiated for predicting full-scale response with 1/8, 1/4, and 1/2 linear scale models.

INTRODUCTION

The overall safety evaluation of shipping containers and packaging used to ship radioactive materials is an important part of protecting the public and the environment from unnecessary exposure to radioactivity. Very strict design criteria are used by designers of radioactive material packaging to ensure that the containers are safe for shipping these very hazardous materials and will contain the radioactive contents during both normal conditions and during impact accidents. The complex structural response of the packaging in a dynamic impact accident is a very difficult problem to analyze and frequently one must resort to some type of support testing. The cost of conducting experimental tests on full-scale replicas of the package design can be very high. However, small scale model simulation of prototype designs can be an economical tool for assessing accurately the dynamic impact response of the design.

The scale modeling experiments described in this paper were conducted for the U. S. Department of Energy, Division of Environmental Control Technology as a part

of an overall program[1] on environmental and safety control assessment of energy material shipping container systems.

The objectives of this study were (1) to determine the success with which scale models could be used to predict the dynamic response of radioactive material shipping containers for safety and environmental control assessment, (2) to provide experimenta data for computer code verification and correlation with Los Alamos Scientific Laboratories (LASL) structural dynamics code development programs, and (3) to provide DOE/ECT with a data base for assessment of the adequacy of the safety and environmental control provided by DOE radioactive material shipping container systems under severe accident conditions.

MODEL DESIGN AND FABRICATION

The approach used in the design of the models was to make "distorted replica" scale models*. That is, the models were small-scale replicas in terms of dimensions and materials of construction. However, only the primary containment vessel, lead shield, and lead containment vessel were simulated and the effects of the outer neutron shield and heat transfer fins were ignored in these initial studies.

Based on a survey of existing container designs, the following prototype reference dimensions were selected to allow linear scaling in smaller size.

Length-to-outside-diameter ratio	6.5 (truck) 3.5 (rail)
Lead shield thickness	∿6.6 in.
Shield length	∿180 in.
Containment vessel wall thickness	1/2 to 3/4 in.
Cover plate thickness	∿3 in.

The initial series of scale models was fabricated from stainless steel Type 304 with lead shielding in 1/8, 1/4, and 1/2 linear sizes. Three scales were selected to aid in demonstrating the validity of the scaling laws and to assess the effects of strain rate sensitive materials (e.g., lead) on modeling accuracy.

Since it was thought that the lead slumping process could have a major influence on the dynamic strains induced in the models, the lead pouring techniques used were similar to those used in full-scale prototype systems. The stainless steel weld joints were scaled down linearly from the full size prototype design. All welds were liquid dye penetrant checked, and strain gages that had to be installed during fabrication were monitored continually to make sure that they were functional.

The mechanical properties of the materials used in fabrication of scale models were well characterized. It was considered particularly important to characterize the lead used in the shielding, because lead properties are influenced strongly by small amounts of trace impurities. Thus, to allow complete material characterization, lead samples were obtained from each batch of shielding material used in fabrication of the models. Wherever possible, stainless steel and carbon steel mechanical properties were obtained from the manufacturers material certifications. If this information was not available, standard test samples were taken from the actual construction materials and characterized on standard testing machines.

* The term "distorted replica models" implies that no attempt was made to duplicate a unique cask design in its entirety. Instead, linear scaling of the dimensions of the containment vessel and lead shield portions of a typical truck shipping container was used. Also, the same type of materials that were contained in the prototype design were used in fabricating the scale models.

TEST PROCEDURE

Test Type and Measurements

Free fall drop tests were conducted with several types of container scale models. These consisted of cylindrical stainless steel shells, lead-filled stainless steel shells, and replica scale models. The container model scales were 1/8, 1/4, and 1/2. Free fall drop tests were conducted at heights of 6 inches, 12 inches, 24 inches, and 30 feet. The structural response due to 6-inch and 12-inch drops was in essentially the elastic region. The 24-inch and 30-foot drops caused appreciable plastic strains in the models.

Primary emphasis was placed on the measurement of strain induced during the impact tests. In addition, acceleration measurements were made of the packaging at specified surface locations. The acceleration measurements were normally made within the lead shielding and on the primary stainless steel containment shell. Strain measurements were made at several locations on the stainless steel shell surfaces with "fast response" strain gages. Model temperatures were monitored with standard thermocouples. Both high speed and standard speed movies were made of the packaging as it contacted the drop pad during the impact test.

The severe accident environments considered in the formulation of a test matrix were those described in 10 CFR 71, Appendix B[2]. Other less severe environments were examined to provide useful data for LASL[3] computer code verification and correlation. Other objectives of the simplified tests were (1) to determine the success with which the test results could match scaling law predictions, (2) to examine the repeatability of the results obtained with Battelle's data acquisition system, and (3) to study the effect of varying the impact pad composition on strain.

Test Facilities

The 30-foot drop height experiments were conducted at a special drop test facility which was constructed to meet the IAEA criteria[4] for an unyielding surface for scale models up to 9000 pounds and drop heights up to 38 feet. Heating and cooling facilities were made available to conduct experiments at temperatures between -40 and 260 C. Instrumentation included thermocouples, strain gages, accelerometers, and high speed and standard speed movie cameras. Tests that were performed at heights from 6 inches to 24 inches did not require the special features of the facility that were used for the 30-foot drop height tests. Thus, for convenience, a separate impact pad and drop mechanism were used.

Instrumentation

The data acquisition system used throughout the drop tests consisted of a set of transducers, transducer amplifiers, a multichannel FM tape recorder, an oscilloscope, and an oscillograph recorder. The transducers consisted of several strain gages mounted on the stainless steel surface of the scale models, and one or more accelerometers mounted on the scale models and impact pad. The strain signals were amplified and were then either multiplexed into two signals and recorded directly onto two tape recorder channels, or, alternatively, FM-recorded onto separate channels. Selected signals were also viewed on a storage oscilloscope system during the tests. The recorded signals were played back onto an oscillograph. The measurement system had a bandwidth of 0 to 2000 Hz when used in the multiplex mode, while the system bandwidth without multiplexing was 0 to 3000 Hz. A schematic of the data acquisition system is shown in Figure 1.

RESULTS

Simplified Stainless Steel Shell Models

A limited number of tests were conducted with 1/8 and 1/4 scale models of prototype truck container containment shells dropped from a height of 12 inches. These tests were designed to determine measurement repeatability as well as the amount of data (if any) lost or distorted because of the frequency response characteristics of the data acquisition system. These models were also used to investigate the influence of drop pad composition on the measured strains.

Among the important results of these tests were:

(1) Comparison of the strain data that was processed in the multiplex mode (2000 Hz bandwidth) with unamplified strain data acquired in an experiment using a 100,000 Hz bandwidth system indicated that the multiplexed strain data may have been attenuated slightly. Thus, subsequent tests were conducted with no multiplexing. This resulted in a measurement system bandwidth of 3000 Hz.

(2) Drop pad composition is critical when attempting to quantify experimental results. The drop pad composition was varied from a nearly rigid composition to a "plate-on-elastic foundation" composition. The induced peak strains were about three times lower for drops on the resilient pad than for the nearly rigid pad.

(3) The number of drop tests performed with one model should be sufficiently large to minimize the variance in measured strain. This variance is due primarily to the inability to duplicate exactly the conditions for successive drop tests. Measurements obtained in from six to eight drop tests had substantial variance. Scaling relations calculated from these measurements agreed poorly with scaling law predictions.

Lead-Filled Stainless Steel Shell Models

The test procedure used for the 1/8 and 1/4 scale lead-filled stainless steel containment shell models was modified from that for the simplified stainless steel shell models discussed above. These changes include increasing the frequency response of the measurement system to 3000 Hz, dropping each specimen at least 25 times (an arbitrary choice) so that the impact behavior could be described and compared statistically, and adjusting the drop height to 6 inches to increase the strain magnitude and thus increase the signal-noise ratio.

For this test series, complex lead/steel shell interaction was introduced by filling the center of the stainless steel shells with lead. The same shell specimens that were employed in the 12-inch drop height experiments discussed above were used for this test series. The models were modified by welding a solid steel cone to the bottom (to propagate uniform stress waves), and by filling the central cavity with lead. They were instrumented identically with an array of nine strain gages and one accelerometer. The strain gages measured induced strain in the axial, circumferencial (hoop), and shear (45 degrees) directions. The accelerometer measured axial acceleration on the top surface of the specimen.

To minimize the effect of lead slump on induced strains in the steel shell for a large number of successive drops, the lead in the models was preslumped before recording strain data. Strains induced in the stainless steel shells remained essentially in the elastic region for drops from 6 inches. The magnitudes of strain and axial acceleration as well as the average duration of the initial strain pulse were determined from strip chart time histories. The mean, standard deviation, and rms values of strains and accelerations, as well as the amplitude and time scaling ratios between the two model sizes were calculated.

Ideally, the strains induced in the two different size models, when dropped at identical conditions, would be the same. Since one model was twice the size of the other and had the same impact velocity, the duration of the initial strain pulse for the larger model should theoretically be exactly twice that of the smaller specimen.

The results of the tests performed at a drop height of 6 inches are summarized in Table 1. Strain time histories for this series of experiments are shown in Figure 2. As shown in Table 1, the calculated standard deviations for most of the signals indicated good repeatability of measurements. Since the induced signals could be resolved to about $\pm$ 25 $\mu\epsilon$ for strain, and $\pm$ 200 microsecond for time, the standard deviations indicate that the experimental control was good. The computed values of strain ratios and initial pulse duration for the 6-inch drop height tests agreed well with the scaling relationships predicted by theory.

Results of the 24-inch drop height tests are discussed in detail in Reference 5. At this height, the lead inside the stainless steel containment shell models exerted enough pressure to force the shell stress into the plastic flow regime. Calculations of scaling relations using data from one drop with each of two specimens agreed poorly with scaling law predictions. This can be attributed primarily to the inability to duplicate test conditions exactly, as discussed previously for the simplified stainless steel shell models.

Truck Cask Replica Scale Models

To determine the validity of the scaling factors considered and to assess the effects of strain rate-sensitive materials on modeling of this type, a series of drop tests from 30 feet was performed using three different scales of replica scale models of truck casks. The complete series of experiments that was conducted with these models is described in Reference 5. The influence of scaling, thermal expansion, temperature, impact limiters, lead slump, and impact angle are discussed briefly in subsequent sections. An example of typical strain and acceleration time histories is shown in Figure 3.

Scale Effects

Analysis of the data showed good ability to scale in the various sizes for replica scale models undergoing the severe dynamic impact of a 30-foot free fall onto an unyielding surface in a bottom end-drop orientation at -40 C. Comparisons of the peak induced strains and the observed percentage of lead slump also demonstrated good argreement with the scaling laws. Identical "mushrooming" of the corner at the bottom ends of two specimens was observed for 60-degree off angle drops at 170 C. These results further substantiate the ability to scale model the dynamic structural response of shipping containers.

A summary of the scaling relations calculated from the 30-foot drop test results is shown in Figure 4. Perfect scaling would have occurred if all of the data fell at a measured strain ratio of 1.0 and, depending on the scales, a measured time ratio of either 2.0 or 4.0.

The results of the 30-foot drop tests are considered good, since each test configuration was performed no more than three times. At this height, the models are seldom reusable; severe "mushrooming" and bending often occurs. Thus, the cost constraints for this phase of the research program prohibited fabrication of enough replica scale models from which statistical information such as that for the 6-inch drop tests could be gathered.

Temperature Effects

The magnitudes of thermal strains induced in the models for ambient temperatures of -40 C and for internal containment shell temperatures of 180 C were examined and compared with impact-induced strains from a 30-foot drop height. Thermal-induced

strains at a container temperature of 180 C were about an order of magnitude lower than impact-induced strains at the same container temperature. For the -40 C environment, the thermal-induced strains were of the order of the impact-induced permanent strains. However, comparison of the strains as well as amount of lead slump measured in experiments at -40 and 180 C showed the effect of model container temperature on impact-induced strain was negligible over the range of temperatures considered.

The amount of lead slump measured after impact was about 50 percent smaller for high (170 to 180 C) model temperatures than for low (-40 C) temperatures. This result was expected due to the difference in differential thermal expansion properties between stainless steel and lead. Since lead has a substantially higher thermal expansion coefficient than the outer steel containment shell, the gap between the outer lead surface and the outer steel containment shell would be significantly smaller at the elevated (180 C) temperature than at -40 C, thus allowing less movement of the lead with respect to the shell.

Lead Slump Effect

Experiments were conducted to measure the effect of induced strains during impact for models of containers with lead shielding in the "as-poured" and slumped conditions. These induced strains were slightly larger for the lead slumped conditions. This observation was expected since, during impact, some of the energy is absorbed in the lead slump action. This results in a smaller fraction of the potential energy of the lead being dissipated or transferred to the stainless steel shells.

Impact Angle Effects

A series of experiments were planned to determine the effect on dynamic structural response of the container models for the 30-foot free fall drops at varying impact angles to the impact surface. The replica-scale models were dropped at angles with respect to the drop pad surface of 90 (bottom end drop), 60, 30, and 0 degrees (side drop).

Strains and accelerations induced in the models during secondary impact, or "slap down" varied widely with the use of impact limiters. For 30 degree and 60 degree drops with impact limiters, the strains induced at slap down varied from "barely resolvable" to about 50 percent greater than the strains induced at primary impact. There was no apparent trend in magnitude of secondary impact strains with location for off angle drops using impact limiters. The results of the one off-angle drop made without an impact limiter are shown in Figure 3. The variation of the magnitudes of plastic strain and peak strain at secondary impact are readily apparent. The ratio of peak strains at secondary impact to that at primary impact was about 3.0 near the top, 1.2 near the middle, and 0.3 near the bottom of the model. The ratio of plastic strain induced at secondary impact to that at primary impact was about 100 near the top, 0.8 near the middle, and 0.0 near the bottom.

Comparison of the dynamic strains showed that the off-angle drop condition induced lower strains in the containment shells compared to those induced in bottom end drop conditions. Comparison of side drops under similar temperature and impact limiter conditions to a bottom end drop experiment showed that the maximum induced strains were of comparable magnitude. As expected, the location of the maximum induced strain in the containment shell differed between the side and end drops. The maximum strain was measured at the center of the bottom containment plate for the bottom end impact test and was measured at the top-side midpoint of the containment shell axis for the side drop impact condition.

CONCLUSIONS

From these preliminary scale model experiments, it was determined that the elastic and plastic dynamic structural response of lead-shielded shipping containers probably could be predicted adequately for environmental and safety assessment with replica scale modeling techniques. Strains and accelerations induced during impact vary considerably with time and specific location within the packaging structure. Comparison of induced strain measurements for various impact angles for drops from 30 feet showed that secondary impact can have a large effect on induced strains, particularly if no impact limiters are used. The bottom or side drop conditions resulted in relatively larger strains in the containment vessel and lead shielding. No damage was caused in any of the experiments that would have resulted in loss of containment or shielding of the radioactive contents.

REFERENCES

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(3) Butler, T. A., et al, "Development of Structural and Thermal Analysis Methods", Los Alamos Scientific Laboratories.

(4) IAEA, Advisory Material for the Application of the IAEA Transport Regulations, Safety Series No. 37, 1973.

(5) Robinson, R. A., et al, "Experimental Studies of Dynamic Impact Response with Scale Models of Lead-Shielded Radioactive Material Shipping Containers", Battelle's Columbus Laboratories Topical Report No. BMI-2001, May 1978.

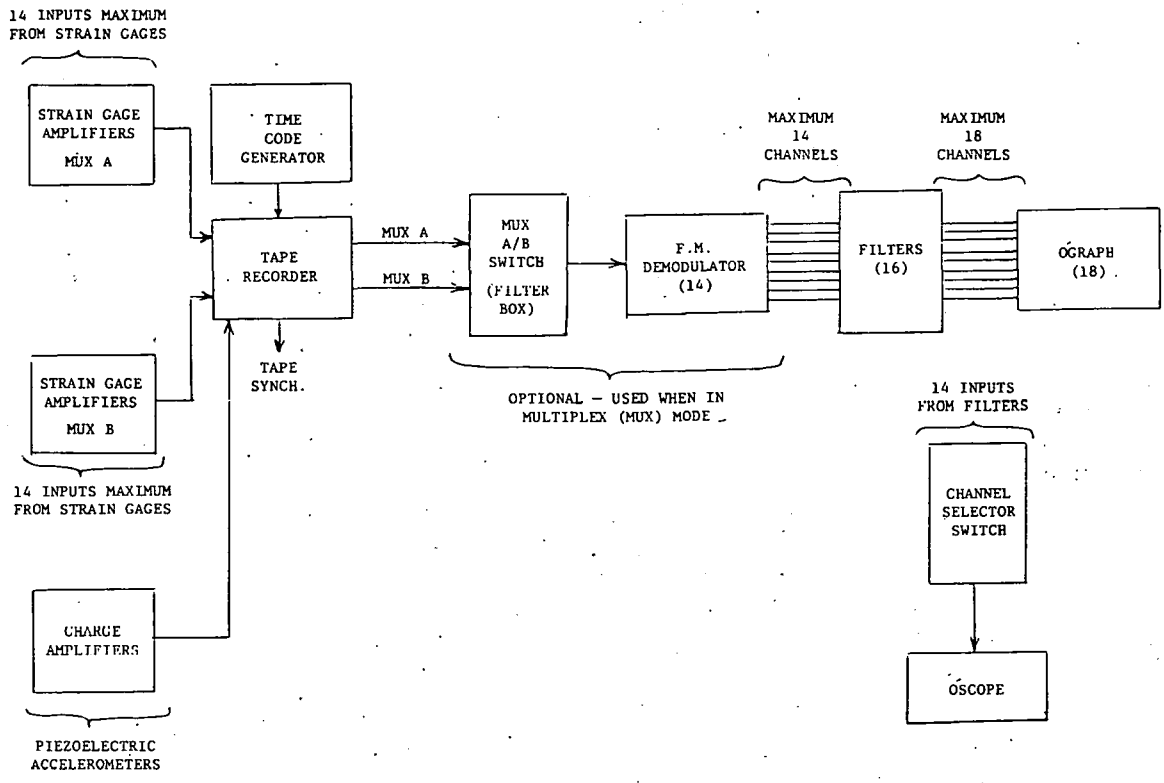


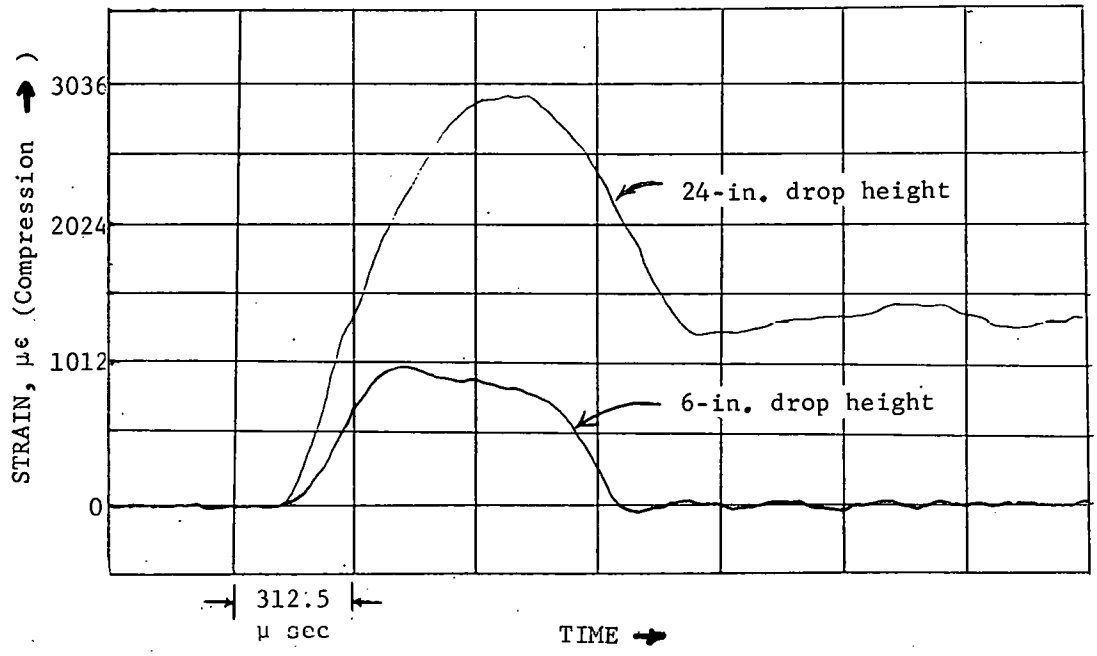
FIGURE 1. BLOCK DIAGRAM OF DATA ACQUISITION SYSTEM

	Location[a]			Strain, με				
				Specimen A[b]		Specimen B[b]		
	A	B	Direction	Mean	Std. Dev.	Mean	Std. Dev.	Mean B/A
S.C. #1	L/18.5	L/22	long.	954	52	1078	27	1.1
2	L/19.6	L/22	shear	316	22	94	16	--
3	L/22	L/22	hoop	556	42	764	42	1.4
4	L/4	L/4	long.	992	85	883	41	0.9
5	L/4	L/4	hoop	365	38	--	--	--
6	L/2	L/2	long.	880	108	781	57	0.9
7	L/2	L/2	hoop	319	27	307	31	1.0
8	L/18.5	L/22	long.	1098	47	881	43	0.8
9	L/22	L/22	hoop	374	25	358	26	1.0
Avg. duration of initial "pulse", msec				1.0	0.0	1.9	0.1	1.9

(a) L is length of cylinder minus cone (L = 22 in. for Specimen A, 44 in. for Specimen B). Location is measured from base of cylinder.

(b) Specimen A is a 1/8-scale model; Specimen B is 1/4-scale.

TABLE 1. RESULTS OF ELASTIC STRAIN DROP TESTS FROM A HEIGHT OF 6 INCHES



1/8 SCALE MODEL: LONGITUDINAL STRAIN AT L/18.5 FROM BOTTOM (S.G. #1)

FIGURE 2. STRAIN TIME HISTORIES FROM Pb/SS SHELL MODEL TESTS

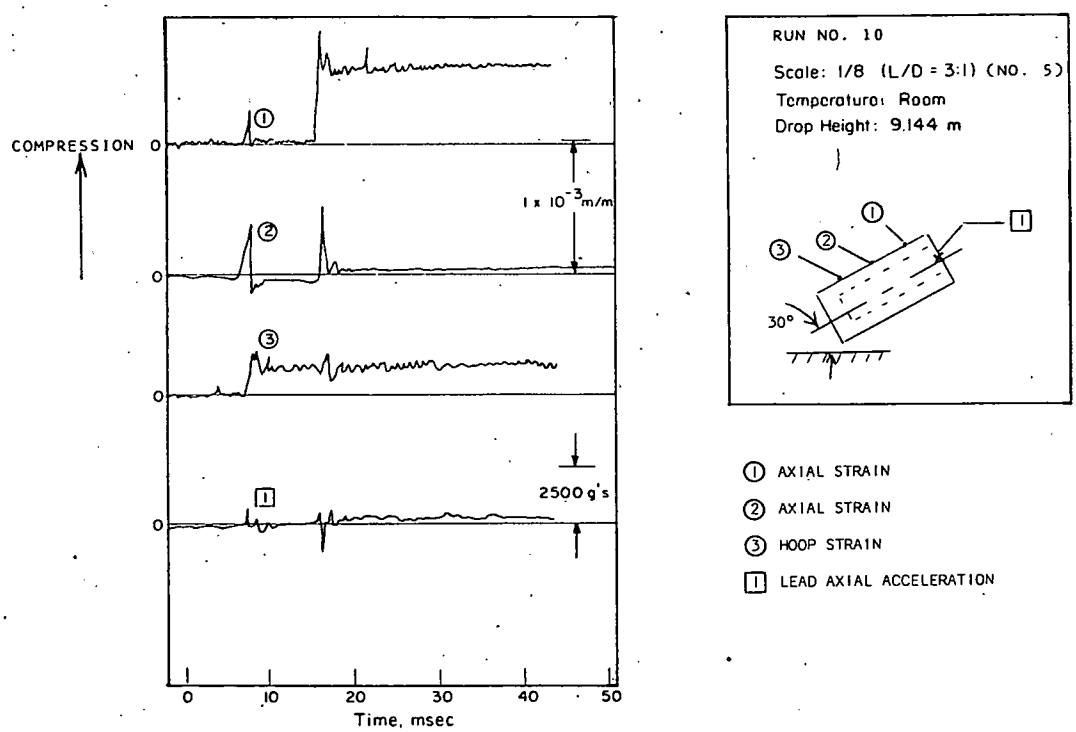


FIGURE 3. SAMPLE TIME HISTORIES FROM 30-FOOT DROP TESTS

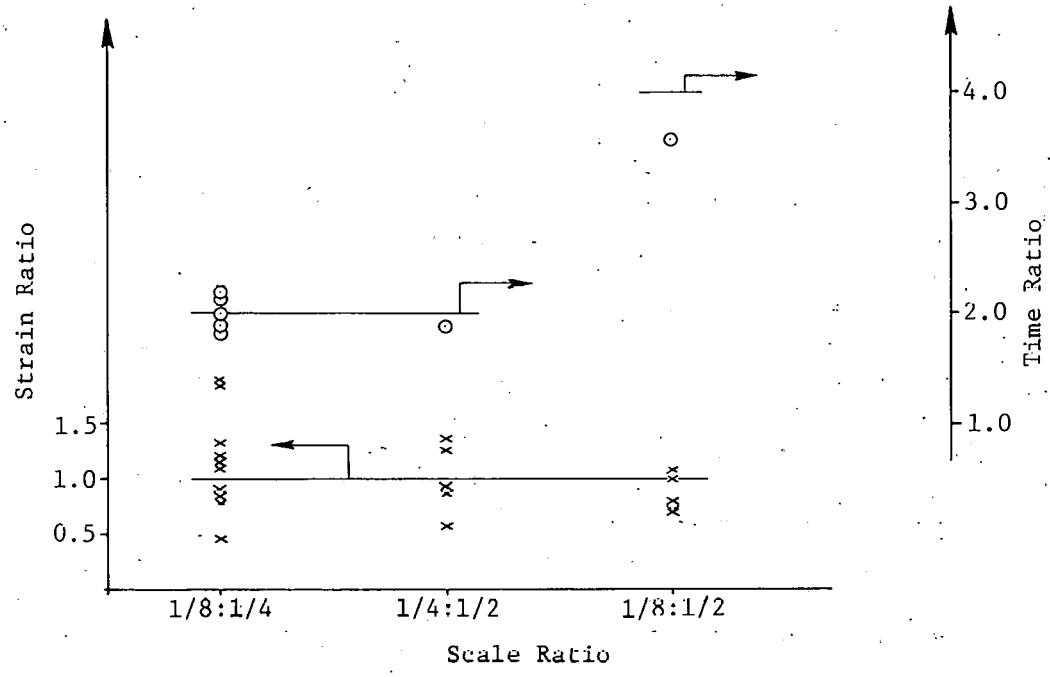


FIGURE 4. SUMMARY OF SCALING EFFECTS CALCULATED FOR 30-FOOT DROP TEST DATA

DETERMINATION OF ALLOWABLE MAXIMUM ACCELERATION LOADS FOR THE TRANSPORT OF IRRADIATED FUEL

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A significant operational consideration in specifying criteria for transporting spent nuclear fuel is the allowable loadings which may be applied to the fuel assembly during transportation and handling. If the applied loads exceed a given maximum value, breach of fuel rod cladding may occur resulting in the release of radioactive fission gases and possibly fuel fragments into the cask cavity. While this failure will not result in a safety hazard during the transport of the fuel assembly, it may result in delays in unloading and preparing the cask for additional shipments.

This paper describes the work performed by Exxon Nuclear Company, Inc. (ENC) to support establishment of criteria which define the maximum allowable loading to which spent fuel may be exposed during transportation without clad failure occuring. The results of these analyses will be utilized to develop design criteria for the Irradiated Fuel Transportation System (IFTS).

PHYSICAL DESCRIPTION OF THE PROBLEM

Typical LWR spent fuel casks provide fuel support in a cavity with a rectangular cross section. The fuel rests in the cavity and depending on its cross section in relation to the cavity, it will have a lateral clearance of as much as 0.8 inch total. In the longitudinal plane, variations in the length of assemblies is accommodated by the incorporation of spacers which take up the unused cavity length. For these analyses an arbitrary maximum clearance between the assembly and the ends of the cavity of 2 inches was used.

The fuel assembly thus rests horizontally on a flat plate with support provided at the spacers and tie plates. Clearance is available above, at the sides and at the ends of the assembly. For these analyses, the cavity walls were assumed to have a relatively high stiffness when compared with the assembly stiffness which was determined from test data.

TRANSPORTATION LOADS AND FUEL ASSEMBLY RESPONSE

The loads[1,2,3] imposed on a spent fuel assembly during handling and transportation operations can be categorized as significant shock loads and rather low amplitude random vibration loads. Shock loads which can be attributed to rough roads or rail humping are the most damaging dynamic loads during the relatively short periods involved in transportation. In addition the analyses must consider the steady-state loads imposed on the cladding by the end-of-life internal fuel rod gas pressure. The random vibration loads imparted to the fuel assembly are due to the input from the carrier suspension system. These loads, derived from road or rail discontinuity, are normally a combination of sinusoidal and random vibration components.

Transportation shock loads[1,2,3] have been shown to be similar to the impact shock response recorded during a fuel assembly vertical drop. Figure 1 illustrates a typical wave form from a PWR assembly vertical free fall drop test. After the fuel rods have bottomed against the tie plate, the classical saw tooth shock impulse is exhibited. The calculated response for free fall impact, using the finite element modelling procedures given below, is in close agreemen with measured results (see Figure 2).

Fuel rod drop tests and related analyses have consistently exhibited linearity between drop height, impact force and fuel response. This linearity is also exhibited in fuel assemblies, even with the complexity of fuel rod to spacer sliding friction and fuel rod to tie plate gaps. The cask end impact can also be simlulated using a linear approximation. This is possible due to the similarities with the free fall drop problem. The only additional condition which must be considered results from the sliding friction between the fuel assembly and the cask cavity. By simulating this sliding friction load with Coulomb damping, a linear function with respect to input acceleration, the total problem has a linear response. Thus the verification of the models below by vertical impact test results and analysis such as shown in Figure 2 provides a firm base for extrapolation to the transportation longitudinal load environment. The input wave form to the models in these analyses is a sawtooth shock impulse which is characteristic of the maximum fuel assembly transport loads.

FUEL ASSEMBLY FINITE ELEMENT MODELS

Longitudinal Loading

The finite element model used to represent the longitudinal impact of the fuel assembly is shown in Figure 3. Identified are the nodal points (which provide coordinate geometry), springs, concentrated (lumped as opposed to distributed) masses, gaps and Coulomb friction elements. A simple one-dimensional lumped-mass spring model with one degree of freedom per node for longitudinal motion was used. The longitudingal shock excitation is applied to the cavity and hence to the point of contact between the cavity base and the fuel assembly. The model takes into account the clearance between the ends of the fuel rods and the tie plates, and between the fuel assembly and cask cavity. The sliding of the fuel assembly on the cavity base, the fuel rod to spacer and fuel column to cladding friction is accounted for by Coulomb friction elements. The fuel rods were assumed to move with respect to the spacers due to the relaxation of the spring properties which occurs following irradiation, thus the potential for impacting on the tie plates is included. The mass and spring element idealization was optimized to yield a stable solution with a reasonable value of numerical integration time step. This was accomplished through mass and stiffness parameter selection which gave a proportioned distribution of frequencies.

The significant data from this model with respect to transportation criteria is the relationship between input acceleration and cladding axial stress and strain.

Transverse Loading

The model shown in Figure 4 was developed to determine the effect of transverse shock loads on the fuel rods and their spacers. An evaluation of mid-span impacting of adjacent fuel rods could also be performed with the model. This model is two-dimensional with rotation and translation coordinates (θ_z and y). The fuel rods are modelled with beam elements; the spacer springs and dimple and midspan contact with adjacent fuel rods are modelled with spring and gap elements. Preload effects on the fuel rods by the spacer springs are included in the gap element representation. Half the length of the fuel rod is modelled using symmetry with appropriate boundary conditions. The transverse shock excitation is applied through the spacer to the eleven gap spring elements shown in Figure 4.

Parameters of interest from this model for transportation criteria are the relationship of input acceleration to: clad bending stress, mid-span fuel rod to fuel rod impact, and spacer dimples and spring to fuel rod impact forces.

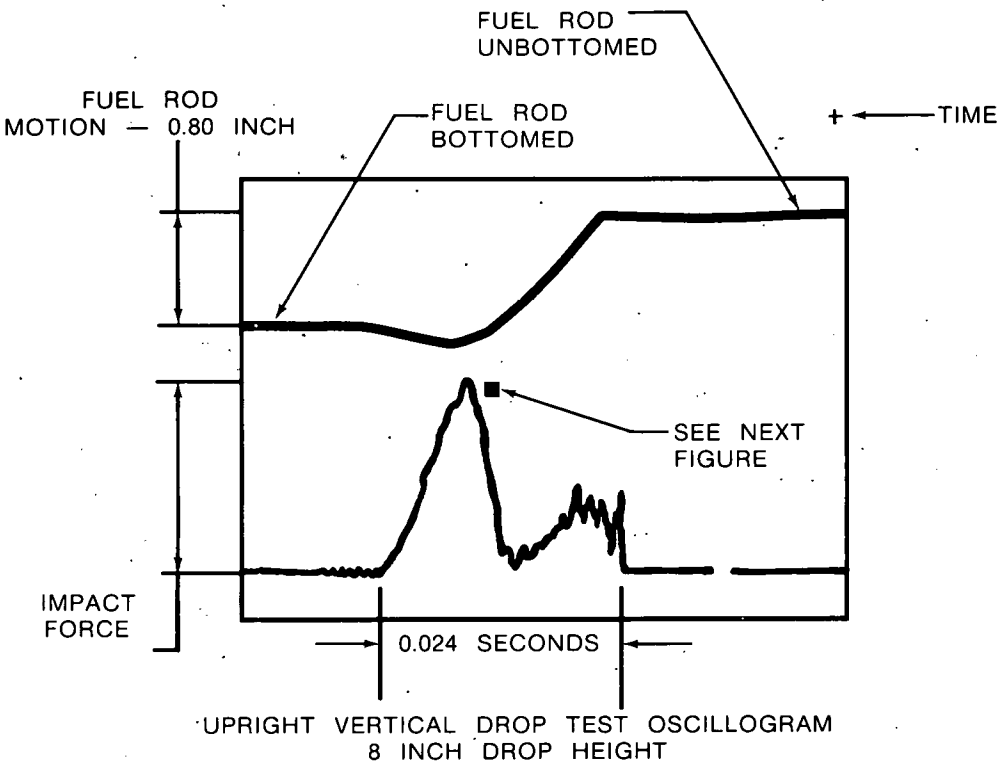
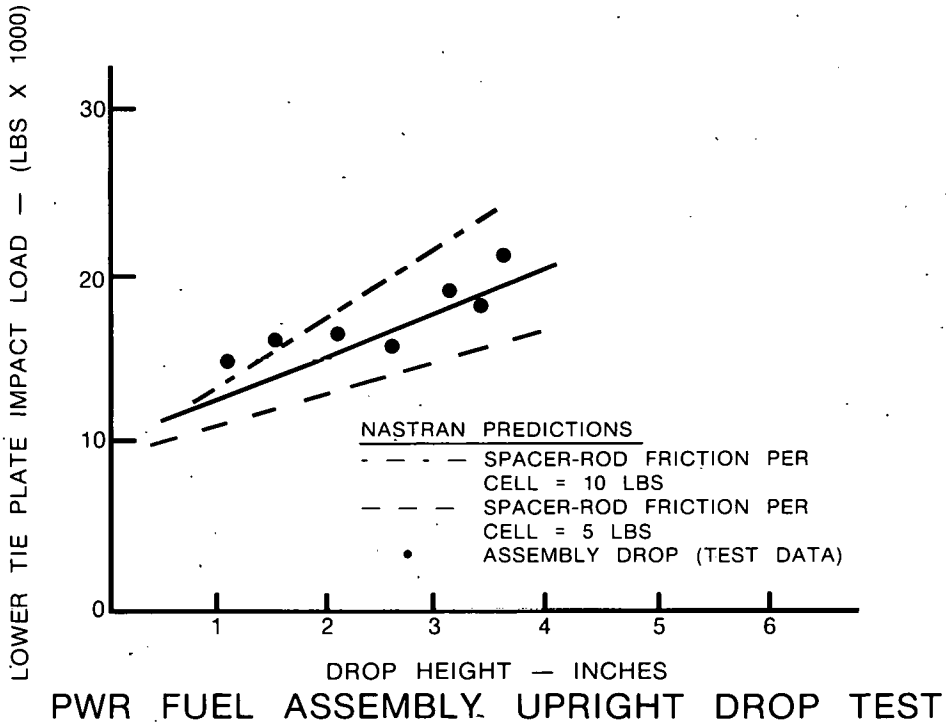
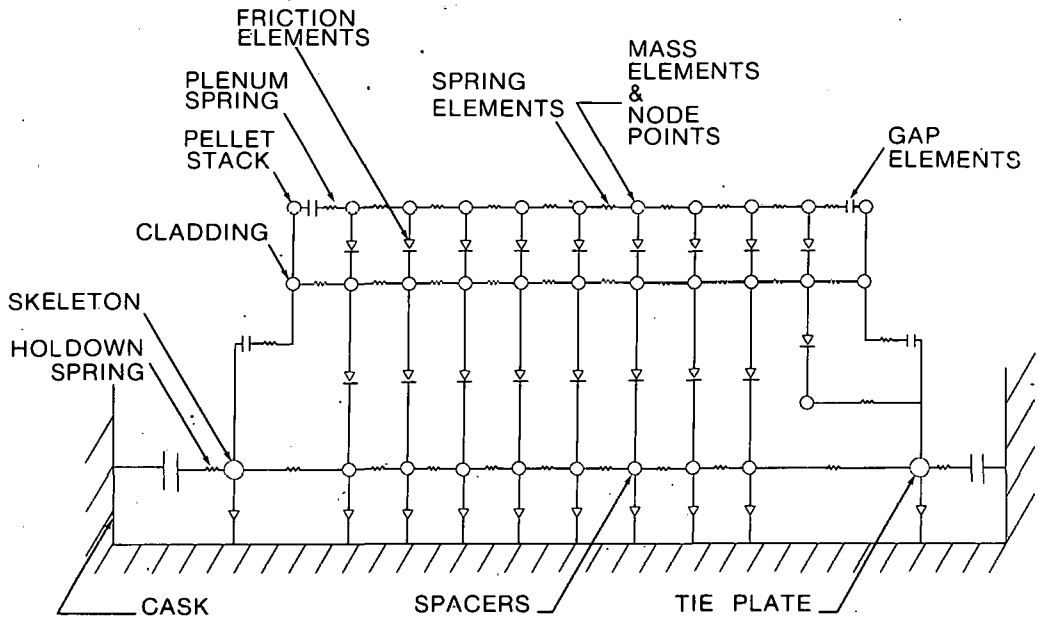


FIGURE 1



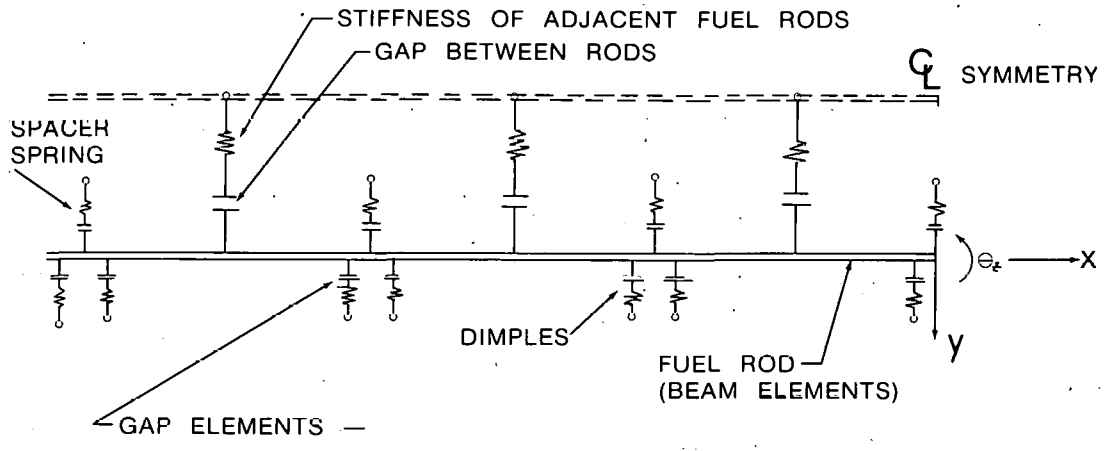
PWR FUEL ASSEMBLY UPRIGHT DROP TEST

FIGURE 2



LONGITUDINAL PWR FUEL ASSEMBLY MODEL

FIGURE 3



PWR VERTICAL FUEL ROD MODEL

FIGURE 4

METHOD OF SOLUTION

The general purpose computer program NASTRAN (NAsa Structural Analysis, MSC version) was used to determine the fuel assembly response to transient loads. In these analyses, the transfer f[illegible]tion capabilities of NASTRAN[4], in conjunction with implicit integration procedures, are u[illegible] to solve the equation of motion. The equations of motion of the fuel assembly lumped mass spring model are:

$$[M]\{\ddot{u}\} + [B]\{\dot{u}\} + [K]\{u\} = P(t) + P_i(t)$$

where;

$[M]$ = mass matrix
$\{u\}$ = a set of coordinates expressed as a column matrix of one term for each node point
$[B]$ = damping
$[K]$ = stiffness matrix
$P(t)$ = linear forcing function
$P_i(t)$ = nonlinear forcing function

The above equations of motion are integrated directly. The introduction of the nonlinearities, $P_i(t)$, in the implicit integration required care in the analysis to prevent instability. These instability problems can be avoided by selection of sufficiently small time steps of integration. No instability problems were encountered with an integration time step of 1.0×10^{-4} second for these transient-impact analyses. As an additional check on the effect of time step, a solution was repeated using an integration time step of 0.5×10^{-4} second. The solutions were approximately identical to the third significant place. A typical run time on a CDC 6600 computer for these analyses was 500 system seconds.

A uniform structural damping, g (or g/2 of critical viscous damping) was used in the system damping matrix [B] as follows:

$$[B] = \frac{g}{W}[K]$$

W is the pivotal frequency with the remaining terms as previously defined. This description of the viscous damping is an approximation, since the damping forces are larger at higher frequencies and smaller at lower frequencies.

CRITERIA DEVELOPMENT

Once a basic understanding of the fuel assembly failure modes are developed, criteria can be established to assure the fuel rod cladding integrity is maintained during shipping and handling. Although the detailed criteria are still to be developed, the bases have been established. As noted above, longitudinal criteria involves cladding axial stress and strain. These criteria are well established for fuel assembly design and are being extended to the transportation loads.

For the transverse model, several potential failure modes and combinations exist. Conservative bending stress criteria are being adopted using ASME Section III criteria. Rod to rod impacts and fuel rod to spacer contact load criteria can best be developed using feature tests with cladding embrittlement simulated by hydriding zirconium tubes to levels which provide an embrittled cladding condition similar to that experienced at end of life. Establishment of these criteria is an ongoing activity and although not safety related is a significant operating consideration.

Evaluation of Results

Using the models described above, it is possible to develop operational criteria to define allowable acceleration loads which may be imparted to the fuel assembly during transportation. The allowable accelerations are based on the derived mechanical loads from the above analyses in combination with creep evaluations of the loads induced in the fuel cladding by internal [illegible]ion gas pressure in the rod. Using these data in combination with cladding strain limits,

the maximum mechanical input to the fuel assembly can be established. Figure 5 shows a typical axial shock response curve.

This value can then be utilized in spent fuel transporter design to assure that the full assembly loads imposed during normal operating conditions do not exceed the criteria. The criteria values will also be utilized in monitoring operation of the IFTS.

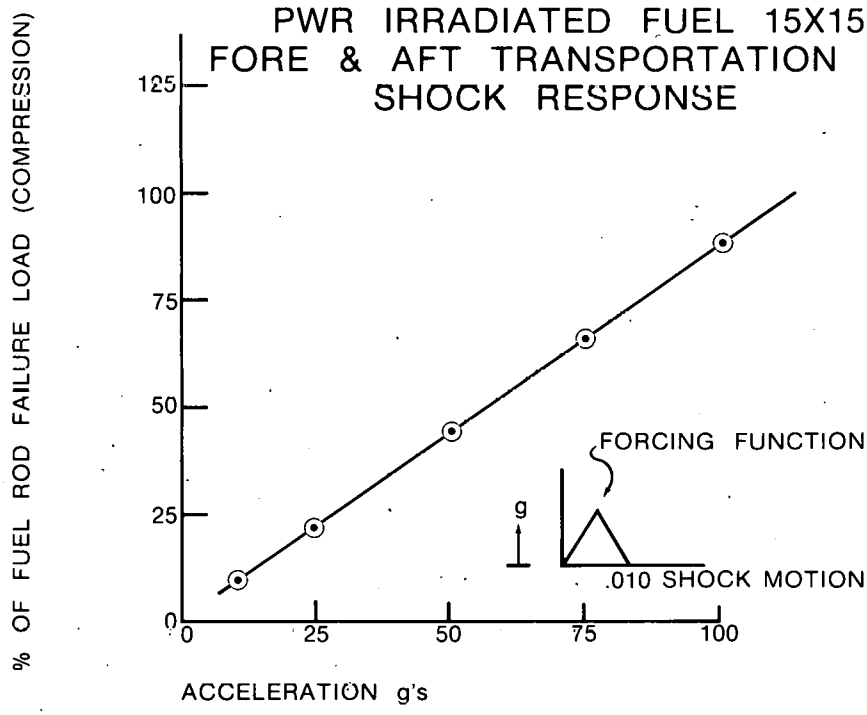


FIGURE 5

CONCLUSIONS

The analyses described provides a portion of the data required to define operational limits of loading for spent fuel assemblies during transportation in shipping casks. Using these data and the associated criteria, suspension and shock mitigation systems may be designed for the transport vehicle.

The data may also be utilized to monitor system operation to assure that the fuel assembly is being transported in a manner which will enhance operability of the total system.

To date these studies have concentrated on PWR fuel payloads. It is expected that similar analyses for BWR fuel assemblies will show a less severe loading limit. This conclusion is based on the structural characteristics of BWR fuel assemblies and their reduced weight.

ACKNOWLEDGEMENT

The authors acknowledge the helpful assistance of Dr. K. R. Merckx in the performance of this work.

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Dup

ANALYTICAL AND SCALE MODELING TECHNIQUES FOR PREDICTING THE RESPONSE OF SPENT-NUCLEAR-FUEL SHIPPING SYSTEMS INVOLVED IN SEVERE TRANSPORTATION ACCIDENTS*

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SUMMARY

This paper describes techniques which can be utilized to predict the response of a spent-nuclear-fuel transportation system in a severe transportation accident involving a high velocity impact into a rigid barrier or a collision with another vehicle. The techniques reviewed include both mathematical analyses and physical (scale) modeling.

The mathematical (computer) modeling can be accomplished in two steps. The first models the overall system including the vehicle and the cask with a lumped-parameter model, which predicts the rigid body motion of the cask as the vehicle crushes. The second step utilizes a detailed dynamic finite element computer model of the cask to calculate deformations which the cask sustains when it encounters a rigid target at a high velocity.

The second part of the paper discusses scale models of casks and vehicle systems used to analyze two hypothetical accident situations. The results of the scale model tests and the mathematical analyses are then related to the results of full-scale tests conducted as the last step in the research program.

It is shown that the techniques described in the paper can give excellent indications of the response of shipping casks and transport systems involved in severe impact situations.

* This work was supported by the United States Department of Energy.

ANALYTICAL AND SCALE MODELING TECHNIQUES FOR PREDICTING THE RESPONSE OF SPENT-NUCLEAR-FUEL SHIPPING SYSTEMS INVOLVED IN SEVERE TRANSPORTATION ACCIDENTS *

Introduction

This paper describes mathematical and scale modeling techniques which may be used to predict the response of spent-nuclear-fuel shipping systems in extremely severe transportation accident environments. More specifically, these techniques may be applied to high velocity impact situations where gross deformation of the vehicular structures and severe shocks to the shipping cask can be expected. A description of both mathematical and scale modeling techniques and examples of how these were used in a research program is included.

The research program conducted by Sandia Laboratories focused on analyzing three hypothetical accident situations. After the analyses had been completed, actual full-scale tests were conducted to verify the results of the individual studies. This paper briefly describes two of the analyses and how the results compared to the actual full-scale tests. More detailed information can be found in references [1] and [2].†

Accident Scenarios Investigated

The three accident situations investigated and for which full-scale tests were staged included:

1. A highway tractor-trailer and shipping cask system impacting a rigid concrete barrier at velocities of 97 and 129 km/hr (60 and 80 mph).

2. A special railcar and shipping cask system impacting a rigid barrier at 129 km/hr (80 mph).

3. A stalled highway tractor-trailer and cask system being impacted broadside by a locomotive traveling at 129 km/hr (80 mph).

These full-scale tests were first described in [3] and all have now been completed. The tests were carefully monitored with high speed photography and onboard instrumentation. The analysis techniques will first be described and the results will later be compared to the response of the full-scale systems. The first two accident scenarios listed above will be used as examples.

Mathematical Analysis

The first step in the analysis was to construct a lumped-parameter model of the complete transport system for each of the tests. In this type of mathematical model, the structure is discretized into mass and coupling (spring) elements and a computer program is used to solve for the response given some initial conditions. As an example, Figure 1 illustrates schematics of the models constructed for the tractor-trailer and railcar impact tests. These models are one-dimensional and were utilized with the SHOCK computer program [4].

* This work was supported by the United States Department of Energy.

† Numbers in brackets indicate references at the end of the paper.

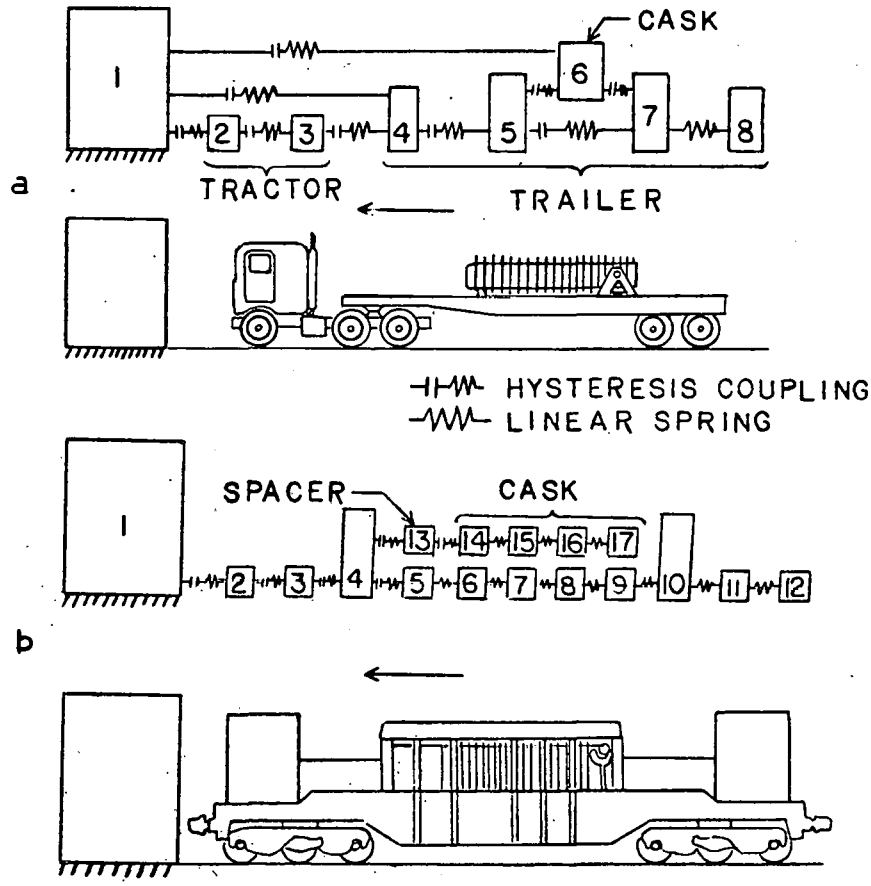


Figure 1. Schematics of the lumped-parameter models. (a) Tractor-trailer model. (b) Railcar and cask model.

The tractor-trailer system was discretized into 7 masses and 10 couplings as illustrated in Figure 1(a). Here mass elements 2 and 3 represent the tractor. Mass elements 4, 5, 7, and 8 represent the trailer and mass 6 represents the cask. In the railcar model of Figure 1(b), the railcar structure is represented by mass elements 2 - 12 and the cask by mass elements 14 - 17. Mass 13 represents a spacer unit which rests directly in front of the cask in the full-scale system.

In each case, the impacts were simulated by giving all the mass elements an initial velocity into a fixed target mass. The response of this system, obtained with the SHOCK computer program, then indicated the dynamic forces and deflections which could be expected in the full-scale test. Results obtained with these models also indicated the dynamics of the cask through the impact and the forces which would be acting on it. This information could then be used to predict the cask response or as an input to a more detailed analysis.

In each case, coupling definitions were based upon structural analysis estimates of their force-displacement behavior. Non-linear couplings with provisions for different loading and unloading paths were used extensively in the model. An example of this type of coupling, usually termed a hysteresis coupling,

is illuatrated in Figure 2. Here it is assumed that after reaching a "yielding" crush load, crushing continues at a constant force until reaching a "bottoming" condition where the force level required for more deflection rapidly rises. Unloading occurs along a different path, as indicated, producing a hysteresis effect and energy absorbtion. The crush (yield) levels have to be estimated either analytically or with the aid of scale models. In the analyses done here, the crush values used were between about 50 and 75% of the yield strength of the material multiplied by the cross-sectional area of the structure.

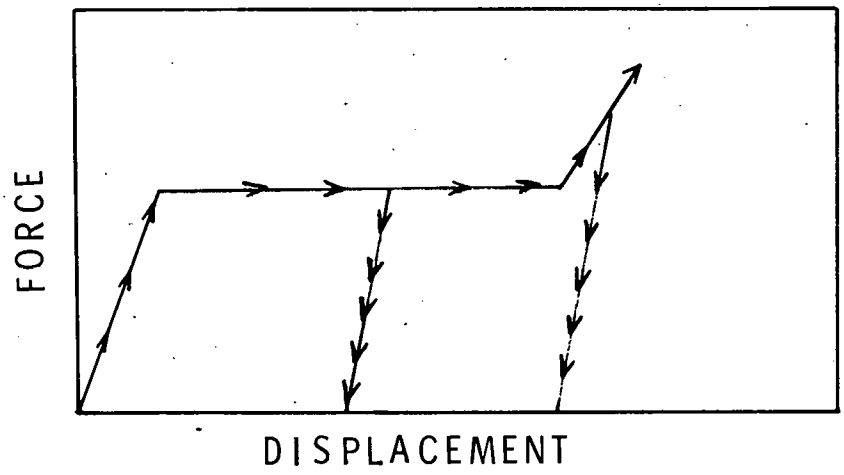


Figure 2. Example of a nonlinear hysterisis type coupling definition.

Calculations with the tractor-trailer model (Figure 1a) indicated that at 97 km/hr the front end of the trailer would plow through the tractor cab and impact the target, crushing the front portion of the trailer. It was calculated that the cask would hit the compressed cab partially crushing the impact limiter. The forces in this case were not sufficiently high to cause any permanent deformations to the basic cask structure.

For the 129 km/hr impact, it was calculated that the cask would hit the target sufficiently hard to cause some deformation to the front end. The extent of the damage which might be expected was then calculated by utilizing a detailed finite-element model of the cask. The HONDO [5] computer program was used for this purpose.

Since the lumped-parameter model indicated that in the worst possible condition, the cask would impact the target at 105 km/hr (65 mph), the finite-element analysis was done for this velocity. Figure 3(a) illustrates the undeformed mesh for the impact end of the cask. This model incorporates a sliding interface between the lead and the steel shells. As with the lumped-parameter model, the discretized body is given an initial velocity equal to the impact velocity. With the finite element model the body moves against a rigid plane which does not permit node penetration and the resulting deformations are

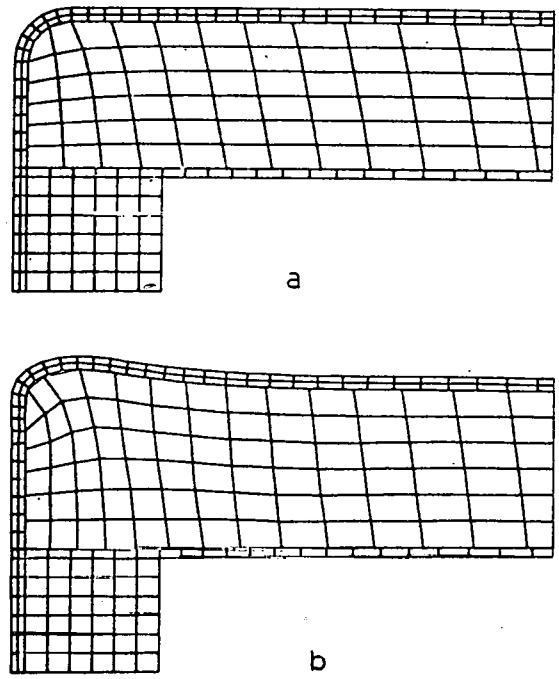


Figure 3. Axisymmetric finite element mesh for the impact end of the shipping cask. (a) Undeformed mesh (b) Calculated deformed mesh for a 105 km/hr impact.

calculated in this manner. Figure 3(b) illustrates the calculated deformed mesh. It was calculated that the cask would mushroom slightly on the impact end with a slight amount of lead slump occurring. It was judged that the cask would maintain its fuel containment capability.

In the case of the railcar impact the lumped-parameter model (Figure 1b) indicated that the front end of the railcar would be severely crushed and that the cask would move forward within the railcar structure partially crushing the spacer unit directly in front of it. The forces calculated to be acting on the cask indicated that the basic cask structure would not be deformed.

The results of these analyses will later be compared to the results of the full scale tests. First, the scale modeling also done as part of this study will be described.

Scale Models

Scale models were constructed for each of the full-scale tests to verify the mathematical analysis and to check the utility of scale modeling. Based on well-established techniques only the pertinent structural features of the full-scale systems were modeled using similar materials. The models were tested at a sled track facility by pushing them up to speed with a small rocket motor and allowing them to coast into scaled targets.

Figure 4 illustrates the model constructed for the tractor-trailer system. This model included the tractor, trailer, tiedowns, cask, and impact limiter structures. Model tests were conducted at nominal velocities of 97 and 129 km/hr. These tests indicated that the cask could be expected to hit the target in very

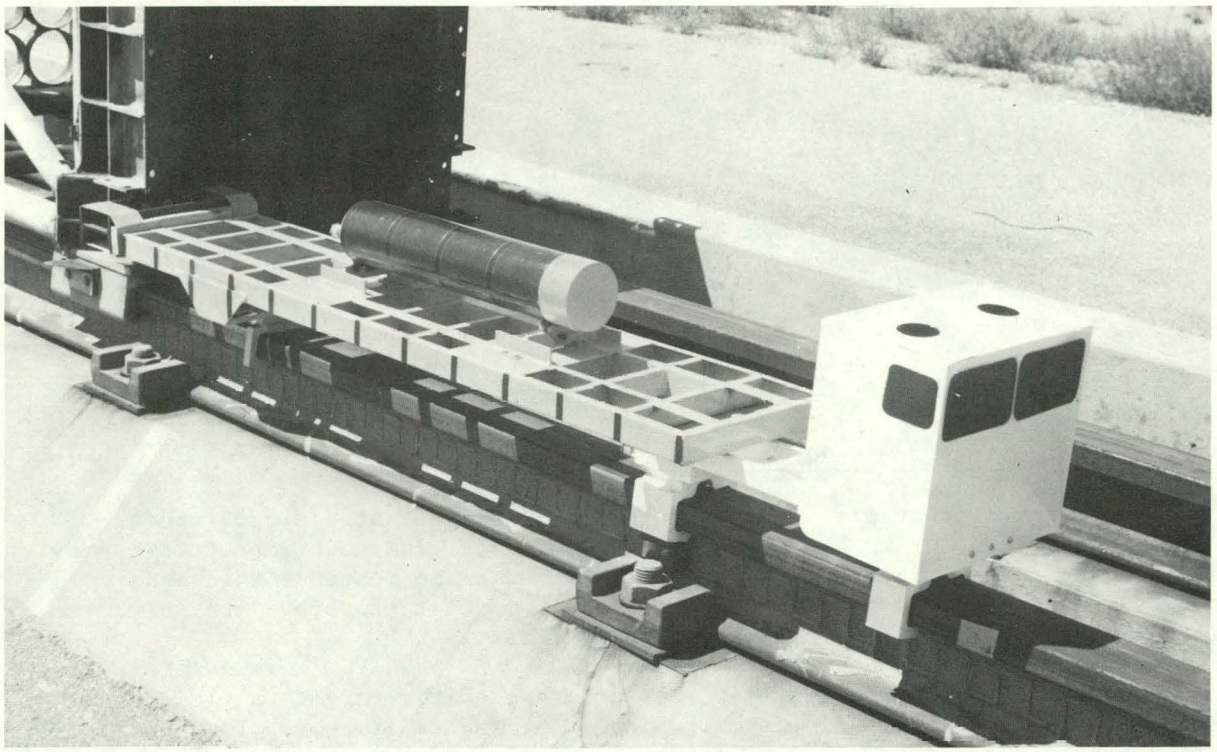


Figure 4. Scale model of the tractor-trailer system. The model was constructed to a one-eighth scale.

close to an end-on condition and that the vehicular system would be completely destroyed. The test at the lower velocity indicated that the cask would only partially crush the impact limiter without sustaining any permanent deformations to its basic structure. At the higher velocity, the scale model indicated that the cask would impact the target hard enough to completely crush the impact limiter and slightly mushroom the impact end of the cask.

Figure 5 illustrates the scale model constructed to analyze the railcar and cask impact. The principal parts of the system including the railcar frame, spacer unit, and cask were modeled. This model was tested at a nominal impact velocity of 129 km/hr. The model tests indicated that the front end of the railcar structure would crush back approximately to the point of attachment of the front railcar truck and that the cask would move forward within the railcar frame partially crushing the spacer and deforming the cage structure which houses the cask. It also indicated that the cask structure would not be deformed in the impact.

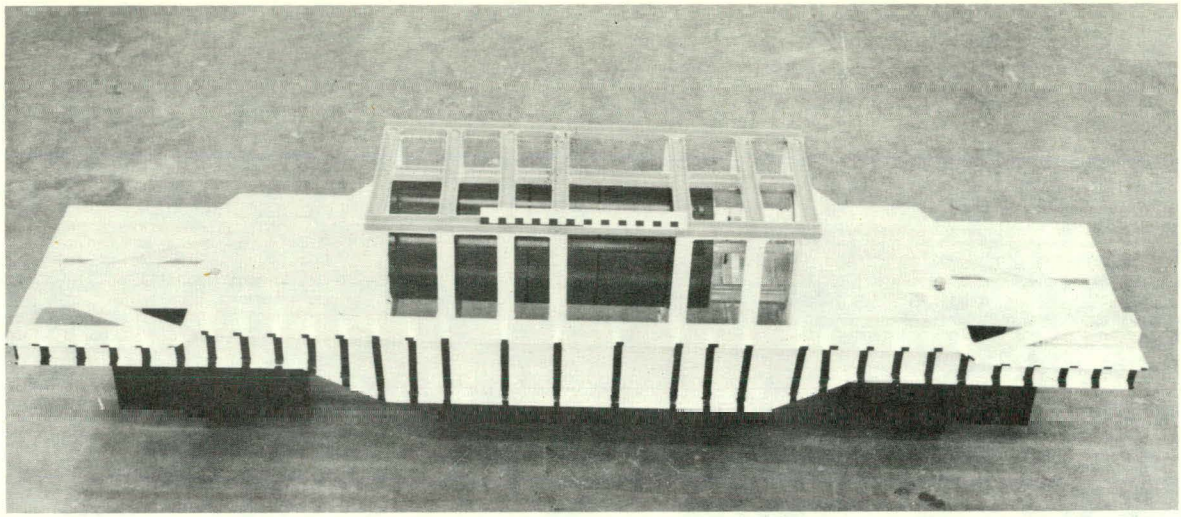


Figure 5. Scale model of the railcar and cask system. The model was one-eighth scale (note the one ft. scale on top).

Comparison of Analysis and Full-Scale Test Results

Full-scale tractor-trailer impact tests were conducted at nominal impact velocities of 97 and 129 km/hr and a full-scale railcar and cask impact test was conducted at a nominal impact velocity of 129 km/hr. The response which had been calculated mathematically and verified with scale model tests proved to be very similar to what was observed in the full-scale tests.

In the first tractor-trailer test the cask reached the target in close to an end-on configuration, partially crushing the front impact limiter. The cask was undeformed except for some external piping and cooling fins. The vehicle responded in a manner very similar to what had been observed in the scale model test.

In the higher velocity test, the impact limiter was partially crushed and ejected upwards. The cask itself impacted the target hard enough to cause some cask deformations on the impact end. A maximum diametral expansion of 3% was observed. This was in good agreement with that predicted by the finite-element model and was almost identical to the cask deformations observed in the corresponding scale model test.

Figure 6 illustrates a velocity-time curve for the cask in the first truck test along with the lumped-parameter model and scale model results. Both the lumped parameter model and the scale model indicated that the cask would not be deformed at this velocity. Excellent agreement was also obtained in the second, higher velocity, test.

The full-scale railcar system also responded much as had been predicted. The front end of the car crushed back to about the midpoint of the front trucks. The cask moved within the cage structure completely crushing the spacer unit but not sustaining any deformations.

Figure 7 illustrates the velocity time history for the railcar cask as calculated with the lumped-parameter model, from a scale model test, and from the full-scale test. Both the lumped-parameter and scale model predicted a slightly greater deceleration than actually occurred. This was principally due to the fact that the spacer unit proved to be much softer than anticipated allowing the cask to decelerate through a greater distance. The agreement, however, is still considered good.

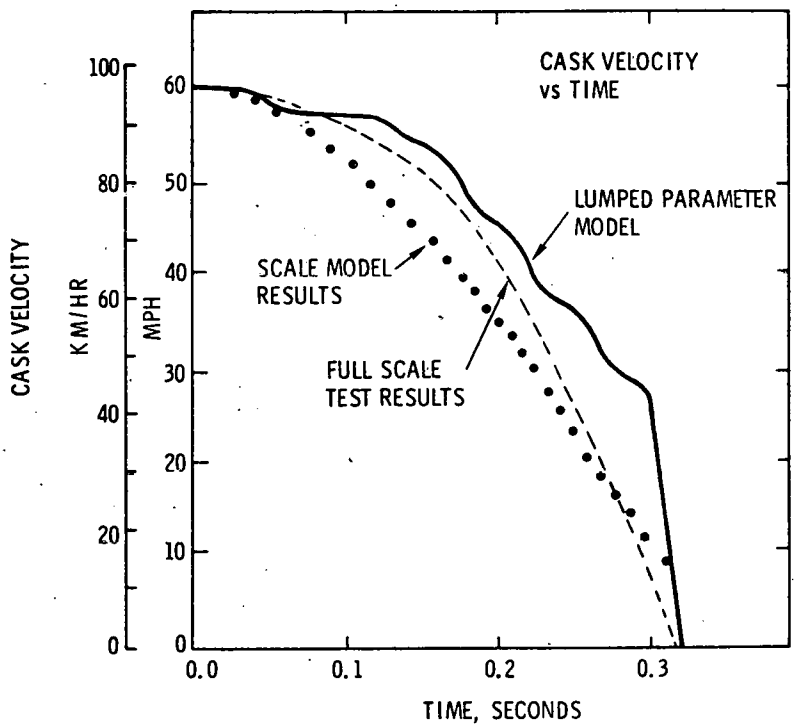


Figure 6. Velocity-time history for the shipping cask in the 97 km/hr tractor-trailer impact test.

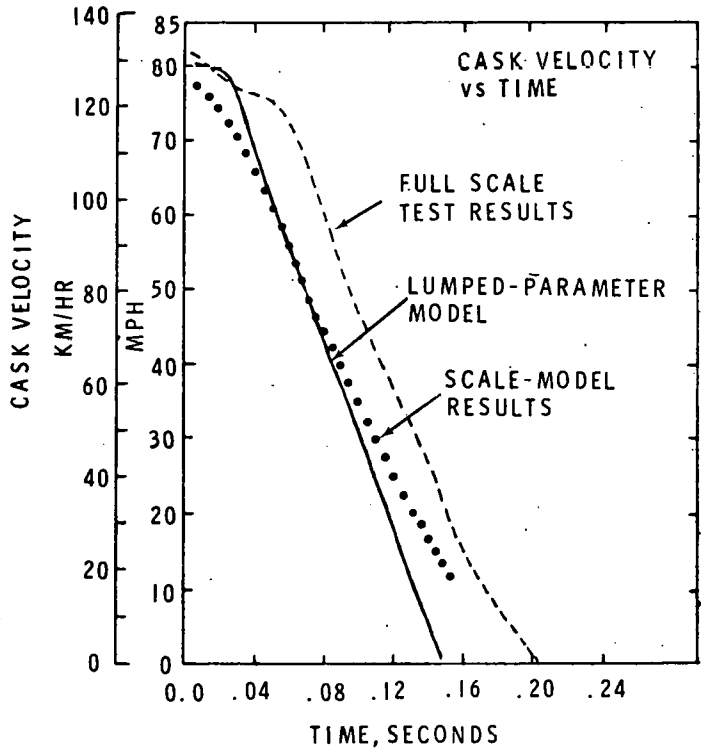


Figure 7. Velocity-time history for the shipping cask in the 129 km/hr railcar and cask impact test.

Conclusions

The analysis techniques reviewed here can given excellent indications of the response of shipping casks and transport systems involved in very severe impact situations. Both the numerical and scale modeling techniques closely predicted the rigid body dynamics of the cask and its condition after the impact. These basic analysis techniques can be extended to other accident situations.

Scale modeling can be a very practical way of predicting damage to shipping casks in extremely severe impacts at skewed angles or onto irregular surfaces. Simplified cask scale models have been shown to accurately predict the response of the full-scale prototype.

At the present time, two-dimensional finite element models, incorporating a sliding interface, if necessary, can also give very reasonable predictions of the impact response of shipping casks. This technique can be applied to end-on or side-on impacts into flat rigid surfaces.

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Dup

FULL SCALE SIMULATIONS OF ACCIDENTS ON SPENT-NUCLEAR-FUEL SHIPPING SYSTEMS

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INTRODUCTION

The Environmental Control Technology Division of the Department of Energy has a coordinated program addressing the problems and perspectives of the transportation of radioactive materials. Initially, to gain a better perspective on transportation and its associated risks, data on the frequency and severity of accidents involving trains, highway vehicles, and aircraft within the United States was collected and analyzed. The basic data collection is contained in the Transportation Environmental Information Center at Sandia Laboratories[1] and correlation of these data are reported elsewhere in this Symposium[2]. This information has been extensively used in a variety of transportation programs at Sandia and other facilities.

While statistical analyses can provide accident frequencies and severities for nuclear fuel transportation, the analyses do not relate the severities to the damage inflicted upon radioactive material shipping containers. Little information exists on the response of spent-nuclear-fuel casks and their transport systems when subjected to severe transport accidents. In fact, very few accidents have occurred involving spent fuel casks, and generally these accidents have not been severe. Additionally, methods to analyze postulated accidents of spent fuel transport systems had not been previously correlated with experimental results. Therefore, ERDA (now the Department of Energy) initiated programs to evaluate the structural response of (1) single spent-nuclear-fuel shipping containers subjected to severe impact environments, and (2) entire transport systems in simulated accidents.

Early full scale tests involved single casks subjected to severe impact environments at both Sandia and Oak Ridge National Laboratory. Casks were dropped at ORNL from a height of 9.1 m at various orientations onto essentially unyielding surfaces to empirically estimate the damage to the containers[3]. Similarly, casks were dropped at Sandia from a helicopter 610 m onto hard prairie soil[4]. Correlation of some of these results showed that, for one cask design tested, the damage received from the 9.1 m drop onto an unyielding surface was much more severe than the 610 m drop onto hard prairie soil. These tests yielded the first full scale empirical data on the spent fuel containers impacted onto different surfaces.

Following the successful completion of the single cask tests, full-scale testing of complete cask transport systems subjected to severe accident environments was initiated at Sandia to gain a better understanding of transport system response[5,6]. Four impact and one fire tests subjecting the cask systems to highway and rail accident scenarios have been conducted to date in the Full Scale Vehicle Testing (FVST) program.

*This work is funded by the Division of Environmental Control Technology, US Department of Energy.

PROGRAM OBJECTIVES

The objectives of the FSVT program are (1) to assess and demonstrate the validity of curr analytical and scale modeling programs for predicting damage in accident conditions by compar... predicted results with actual test results and (2) to gain quantitative knowledge of the extreme accident environments by measuring the response of full scale hardware under actual crash conditions. These tests are not intended to validate present regulatory standards.

To accomplish the first objective, Sandia approached the full scale test program in three distinct phases. Once the accident scenarios were chosen, the first phase used both static and dynamic structural analyses to determine the behavior of the cask and the transport system in the crash environment. Lumped parameter computer codes were used to analyse the transport system response to the environment. If the loads into the cask were sufficiently high, finite element computer codes were used to analyse cask response.

The second phase of the program used scale modeling techniques to assess damage to the container and transport system in 1/8 scale. Initially single cask tests were performed to gain a better understanding of the generic behavior of these containers when subjected to impacts with different surfaces and components of colliding vehicles. In the second part of the model test program, structural models of the cask and transport system were constructed and subjected to scaled crashes. The test results were then correlated with the analytical studies, and predictions were derived for the full scale tests.

The third and final phase of the program involved full scale confirmatory testing. The pre-test mathematical analyses and scale model tests are described elsewhere in the Symposium[7].

To fulfill the requirements of the second objective of the program, high-speed cameras and active and passive instrumentation were used on both the casks and the transport vehicles to record their response under actual crash conditions. Pre- and post test mechanical measurements were used to record any dimensional changes as a result of testing.

TEST SCENARIOS

Spent-nuclear-fuel containers, which contain massive amounts of radiation shielding, are commonly shipped by both highway and rail transport modes. The tests were designed to simulate conceivable yet very severe accidents of cask systems that may be encountered in their normal transport routes. In the selection of test scenarios, the following criteria were applied:

- the casks were to be exposed to very severe accident environments,
- the tests must be amenable to analyses and scale model testing,
- the test costs were considered, and
- the test planners responded as much as possible to the concerns of government, industry, and the public.

A serious effort was made to select test scenarios that could be conceived as realistic, yet based on available accident data, were extremely severe. Accident scenarios which could not be conducted properly (i.e., repeated reliably) were not considered. Scenarios which increase accident severity while providing for simplification in analyses and test procedures were considered and integrated into the test program.

On the basis of these criteria, the selected accident scenarios were

1. Crashes (100 and 130 km/h) of a tractor-trailer rig into a massive concrete barrier,
2. High speed (130 km/h) locomotive/truck transport grade crossing impact, and
3. Impact (130 km/h) of a special cask railcar into a massive barrier, followed by fire.

TEST EQUIPMENT

Financial constraints affected both test definition and equipment procurement. Due to the high costs of modern casks, it was necessary in some cases to use older and obsolete equipment. Out-of-service and older cask shipping systems, used commercial tractor-trailer rigs, and a military surplus locomotive were obtained and modified to make them more representative of current equipment.

The spent fuel casks used in the test program were similar in construction except for weight and payload capacity. Each cask was fabricated with two concentric steel shells, with the annulus between them filled with lead to provide shielding from its radioactive contents. In each case, the head was attached to the cask by means of bolted connections. A summary of the test cask descriptions is shown in Table I.

100 and 135 Km/h TRUCK-CASK IMPACT TESTS

The target used in the two impact tests was designed to be very massive and rigid. A heavily reinforced 626-metric ton concrete wall in the form of a horizontal trapezoid backed with 1580 metric tons of earth provided an essentially unmoving surface with respect to the truck transport (Fig. 1). An object of this size and weight is rarely, if ever, found along normal truck routes.

An obsolete spent fuel cask weighing 20.5 metric tons was obtained complete with its transport trailer and tiedowns. A cutaway of the steel encased lead-shielded shipping container is shown in Fig. 2. The cask was originally designed to carry one PWR or two BWR fuel assemblies. In its original configuration, the cask was mounted with the head facing the rear of the trailer. Since most truck casks are presently shipped head forward, the test cask was reversed on its trailer to simulate current transport environments. The method for attaching the cask to the trailer and the tiedown structure was unchanged. Balsa wood impact limiters designed to the same criteria for modern casks were attached to evaluate the effectiveness of impact-limiting devices in accident conditions (Fig. 1).

Since most casks are pulled with standard cabover, tandem-axle diesel tractors, a similar vehicle was used in the impact tests (Fig. 1). Although worn-out mechanically, the structural members and running gear of the 6.8-metric ton tractor were adequate for testing. To maintain the same test parameters, a nearly identical tractor and trailer rig was obtained for the higher velocity second test. The cask, which was practically undamaged in the first test, was equipped with a replacement impact limiter and was reinstalled on the trailer for the second test.

A fresh (unirradiated) 322 kg fuel assembly from the N. S. Savannah was loaded into the cask for both tests to gain an empirical estimate of damage for a specified fuel configuration and to measure the input response into the assembly. The stainless steel clad fuel assembly was ballasted with lead to simulate the weight of a longer, conventional power reactor fuel assembly.

In both tests, the tractor-trailer rigs were propelled up to speed with rocket motors. The rig was guided with rails laterally and the vertical motion was constrained. At the target, the rig was allowed to displace vertically through the use of unflanged rails. Prior to impact, the pusher rocket sled was stopped by means of a sand braking system.

100 km/h Truck Impact -- In this first test, the truck transport system was propelled into the massive concrete barrier at a velocity of 97 km/h. The response of the cask transport system closely followed that predicted from the pretest analyses and model tests. After the tractor contacted the barrier, the fifth wheel connection broke which resulted in the trailer crushing the cab. As the trailer continued forward, the front end of the frame crushed and buckled upward. The cask and its impact limiter, while still attached to the trailer, moved forward and partially compacted the impact limiter against the crushed trailer frame and tractor cab (Fig 3).

Table I
Test Hardware

	100 km/h Truck Impact	130 km/h Truck Impact	130 km/h Grade Crossing	130 km/h Railcar Impact	Railcar Pool Fire Test
Cask Weight (tonne)	20.5	20.5	22.7	61.8	61.8
External Dimensions (cm)	82.6 dia x 373.4	82.6 dia x 373.4	89.9 dia x 406.4	158.1 dia x 395.0	158.1 dia x 395.0
Interior Cavity (cm)	33.0 dia x 327.5	33.0 dia x 327.5	38.1 dia x 355.6	76.5 dia x 301.8	96.5 dia x 301.8
Lead Shield Thickness (cm)	21.3	21.3	21.3	25.4	25.4
Inner Shell (mm)	12.5	12.5	17.5	9.5	9.5
Outer Shell (cm)	19.1	19.1	28.6	34.9	34.9
Head Seal	Flat elastomeric gasket	Flat elastomeric gasket	Elastomeric O-ring	Metal and elastomeric O-rings	Metal and elastomeric O-rings
Closure Design	16 3/4-in dia studs	16 3/4-in dia studs	8 1-in dia socket head	24 1-1/4-in dia cap screws	24 1-1/4-in dia cap screws
Transport Carrier	Two-axle trailer	Two-axle trailer	Three-axle trailer	Special railcar	Special railcar
Tiedown System	Trunnion and bolted connection	Trunnion and bolted connection	Band type	Enclosed crash frame	Enclosed crash frame

The cask remained horizontal through the initial stages of impact, rose into the air to a 30° angle, and then came to a rest on the trailer in an upright position in front of the target. er the test, inspection of the hardware indicated that the tiedowns remained intact through- the impact. Review of the instrumentation and photometric data analyses indicated that the cask deceleration was approximately 88 m/s^2. The cask deceleration was limited by crushing of the vehicle components and balsa wood impact limiter. Deformations to the cask shell remained elastic; some scraping and crushing of the fins and exterior piping occurred as previously predicted. Post test inspection revealed that the cask head did not leak coolant (colored with dye). The Savannah Core II fuel assembly was easily removed from the cask and was found to be undamaged.

130 km/h Truck Impact -- Since so little damage was sustained to the cask in the first test, the cask was reloaded with another ballasted Savannah Core II fuel assembly and mounted on an identical trailer attached to a similar cab-over tractor. The rig was pushed by rockets up to a 135 km/h impact with the concrete barrier.

The response of the truck transport system was similar to the behavior exhibited in the first test and again closely followed the pretest predictions. The cask tiedown held until late in the impact, slowing the cask down to a velocity of about 100 km/h before striking the buckled trailer frame. As predicted, the balsa wood impact limiter crushed allowing the cask to contact solid structure (Fig. 4). The impacted end of the cask deformed slightly as a result of the higher velocity impact. During disassembly of the cask from the trailer, seepage at a rate of about two drops per minute was observed from the cask head. This seepage later stopped after releasing about 100 cc of dyed cavity fluid. Dents were observed in the cask head which resulted from contact with the trailer fifth wheel king pin (Fig. 5).

Rigid body decelerations measured on the cask were approximately 687 m/s^2. The cask head was removed with great difficulty since the edges of the cask shell had been peened over the head by the impact. Post test inspection of the Core II fuel assembly indicated deformation at the impacted end (Fig. 6). Some fuel pin buckling had occurred, but no clad failure was observed during later dye penetrant and visual inspections.

130 km/h LOCOMOTIVE GRADE CROSSING TEST

The spent fuel cask for the grade crossing test was obtained complete with tiedowns and shipping trailer. Band-type tiedowns held this cask to the transport trailer while trunnion and bolted type tiedowns were used for the other cask. The cask head was retained by eight 1-inch dia socket head screws mounted at an angle with the head. An elastomeric O-ring sealed the head closure. An unirradiated Savannah Core II fuel assembly was placed in the cask.

The test locomotive was a military surplus 109-metric ton six axle locomotive. This locomotive was originally diesel-powered and was driven through six traction motors in two three-axle trucks. Since the diesel engine required extensive repairs and was not needed to accelerate the locomotive in the test, the pinion (driving) gears were removed to allow the locomotive to free wheel.

Prior to the full scale test, analytical investigations were made on several locomotives and grade crossing impact configurations. The first analysis investigated the impact of a large 185-metric ton locomotive into a 22.7-metric ton cask for two configurations[8]. An analysis of the superstructure impact configuration indicated that the cask would substantially crush the locomotive superstructure and would suffer little deformation itself. The analysis of the frame impact configuration, which is not likely since most modern casks are not carried in "low boy" trailers, indicated the cask would deform and would also cause buckling of the locomotive frame. Scale model tests were performed modeling the two configurations; the experimental results closely paralleled that of the analyses.

Further analytical and scale model test investigations for the locomotive and cask used in e test revealed that similar cask responses would result in the superstructure and frame impact

configurations. Review of a rural two track grade crossing indicated that partial frame impact would normally result. Therefore, the glancing or partial frame impact was selected for testing.

In its normal shipping configuration, the cask was carried on a three-axle trailer. An obsolete gasoline powered tandem-axle tractor was used to support the trailer which was placed across a simulated rural two-track grade crossing. In this configuration, the top of the test locomotive's frame would strike the cask approximately 21 cm below the centerline of the cask. The damage to the the shipping container was maximized by positioning the cask in the middle of the test track.

In the full scale grade crossing test, the locomotive was propelled into the test cask at a velocity of 131 km/h. The response of the locomotive and cask paralleled that predicted from the results of the scale model test and structural analysis. The locomotive initially contacted and crushed the trailer frame. Moving through the trailer, the locomotive frame struck the cask approximately 21 cm below the cask centerline (Fig. 7). The locomotive frame buckled as predicted, forming a ramp which pushed the cask upward into the superstructure. Crushing of the superstructure further reduced the velocity difference between the cask and locomotive. After crushing the locomotive cab, the cask rolled off to the right side, tumbled in the dirt, and came to a rest between the track rails. Figure 8 shows the damage to the locomotive after the impact.

Post-test inspection of the cask revealed two 26-mm deep depressions at the points of impact with the locomotive frame (Fig. 9). Leak testing of the cask head indicated a minor air leak in the cask seal. The Savannah Core II fuel assembly was removed from the cask without difficulty. Only slight bowing of the fuel pins was evident in the impact direction. No clad failure was observed.

130 km/h SPECIAL RAILCAR IMPACT TEST

The cask/railcar system used in the test was designed in the early 1960's to carry ten spent fuel assemblies in a composite metal basket. Cooling was provided by circumferential fins on the outer surface and by primary and secondary water cooling systems. The cask was transported by a specially built six-axle railcar, containing a cradle and a crash frame. A cylindrical spacer unit provided axial support to the cask and ensured a snug fit inside the railcar frame. The cask/railcar system was designed to withstand a 4.6 m drop onto an unyielding surface and a thermal environment of 800°C for one hour.

The results of the scale model tests and structural analysis indicated that the cask would be slowed down by the crushup of the spacer unit and the front end of the car frame. It was predicted that the forces on the cask would not cause stresses in excess of the material elastic limit, indicating that the cask body would not be plastically deformed during the impact. It was concluded that deformations of the cask would be limited to fin and external appurtenance damage.

To simulate normal shipping conditions, water coolant, dyed for leak detection, was poured into into the cask cavity prior to testing. The cask was loaded with one prototype depleted uranium and and nine mock (lead and steel) fuel assemblies, and the cask cavity was pressurized to 69×10^3 Pa. The rail cask system was propelled up to speed by rockets mounted on a pusher sled (Fig. 10). Prior to impact, the pusher sled was detached from the railcar and slowed down by a water braking system. In the 131 km/h impact, the front part of the railcar structure and the spacer crushed in a manner very similar to that expected based on the analyses (Fig. 11). The cask's structural shell deformations remained elastic during impact and only minor fin damage was observed (Fig. 12). In the post-test inspection, it was observed that the cask retained its pressure and did not leak. Examination of the prototype depleted uranium fuel assembly indicated no structural damage except for a slight misalignment of the guide at the impact end. Strain gages on the shell of the cask and stainless steel cladding of the fuel indicated that both materials remained elastic during the impact.

SPECIAL RAILCAR FIRE TEST

After the impact test, the cask and railcar were moved to the fire site and placed over a specially constructed fuel pool which measured 0.9-m deep, 9.1-m wide, and 18.3-m long. The railcar was supported by concrete pedestals located centrally in the pool. The post-impact orientation was maintained by placing wheel trucks under one end of the railcar (Fig. 13). This positioning subjected the cask to maximum flame temperatures. JP-4 fuel was piped in continuously during the test to maintain an optimum 15-cm layer of fuel floating above 60 cm of water.

The fire response of the cask was predicted with a radial finite-difference conduction code. Because of the uncertainty in the fire environment, two extremes were considered. A "mild" response was predicted using a fire temperature of 800°C which is similar to current cask design standards, and a "severe" response was calculated using a fire temperature of 1010°C which is typical of test experience at Sandia. The analyses included evaluation of the influence of several factors: a) finned surface of the cask, b) boiling and venting of the internal coolant, c) convective heat transfer from the flames, and d) conduction heat transfer in secondary directions (axial, circumferential). Variations in cask response due to these factors were of second order when compared with the uncertainty in fire characterization.

In the full scale test (Fig. 14), the average flame temperature ranged from 980 to 1150°C. The cask followed the mild response prediction for a period of approximately 90 minutes, after which cask temperatures increased into the severe response range. A white cloud of lead oxide smoke was visually detected at 100 minutes into the test. The fuel supply was then stopped, and the fire burned out at 125 minutes. Due to the weakened condition of the concrete and metal supports and the railcar, the railcar toppled on its side at this point severing all instrumentation connections. Internal thermocouples did indicate that complete lead melt had occurred shortly before the fire terminated. Figure 15 shows a post-test view of the special railcar and cask. Post-test examination revealed that a crack had developed at the top of the cask, allowing additional lead loss as the cask rotated. Fuel assembly containment was not threatened, and an assessment of lead loss and shielding degradation is now in progress.

ACCIDENT SEVERITIES AND PROBABILITIES

Accident statistics were studied to guide the selection of the environmental parameters of this analytical and test program. While it can be concluded that the impact and fire severities comparable to the test levels are possible in real life, the chances of such extreme environments are very remote.

The test scenarios selected generally fall within the "extra severe" or "extreme" categories described in the 1972 Atomic Energy Commission Report (WASH 1238)[8]. These conditions have been estimated in risk assessment studies of transportation accidents to be events of very low probabilities.

With the data from studies of transportation accident statistics[1], one can conservatively calculate the probabilities of occurence for the various accident scenarios considered. These probabilities are presented in tabular form in Table II under the assumption of 11×10^6 km transport distance per year.

Table II
Accident Probabilities

Accident Scenarios	Approximate Interval* (average number of years between accidents)
100 km/h Truck Impact	70
130 km/h Truck Impact	1000
130 km/h Grade Crossing	4500
115 km/h Special Railcar Impact	5900
130 km/h Special Railcar Impact	18000
Combined 130 km/h Special Railcar Impact and 120 minute Fire	10^6
30 minute Railcask Fire, No Impact	120
60 minute Railcask Fire, No Impact	350
90 minute Railcask Fire, No Impact	450
120 minute Railcask Fire, No Impact	700

* Assuming 11 x 10^6 km transport distance per year.

CONCLUSION

In conclusion, the tests demonstrated that (1) scale modeling (impact only) and analytical techniques can reliably predict the response of spent fuel cask systems in severe impact and fire environments, and (2) spent fuel casks can be expected to retain their radioactive contents even after being involved in extremely severe transportation accidents. In the fire test, the cask was subjected to twice the radiant heat flux and three times the duration of the current regulatory thermal requirement before degradation to the cask structure was experienced. In addition, much information has been gained on the behavior of the cask and transport system in extreme environments.

ACKNOWLEDGEMENTS

The structural analyses described in this paper were performed by M. Huerta and A. W. Dennis of the Engineering Analysis Department, Sandia Laboratories, The thermal analysis was performed by R. H. Nilson of the Fluid and Thermal Sciences Department. D. C. Bickel and his staff of the Systems Environmental Testing Department were responsible for the implementation of the scale model and full scale tests. In addition, the author wishes to acknowledge the assistance of D. R. Stenberg, Missouri Research Laboratories, and numerous Sandia personnel not mentioned here who have contributed greatly to the success of this program.

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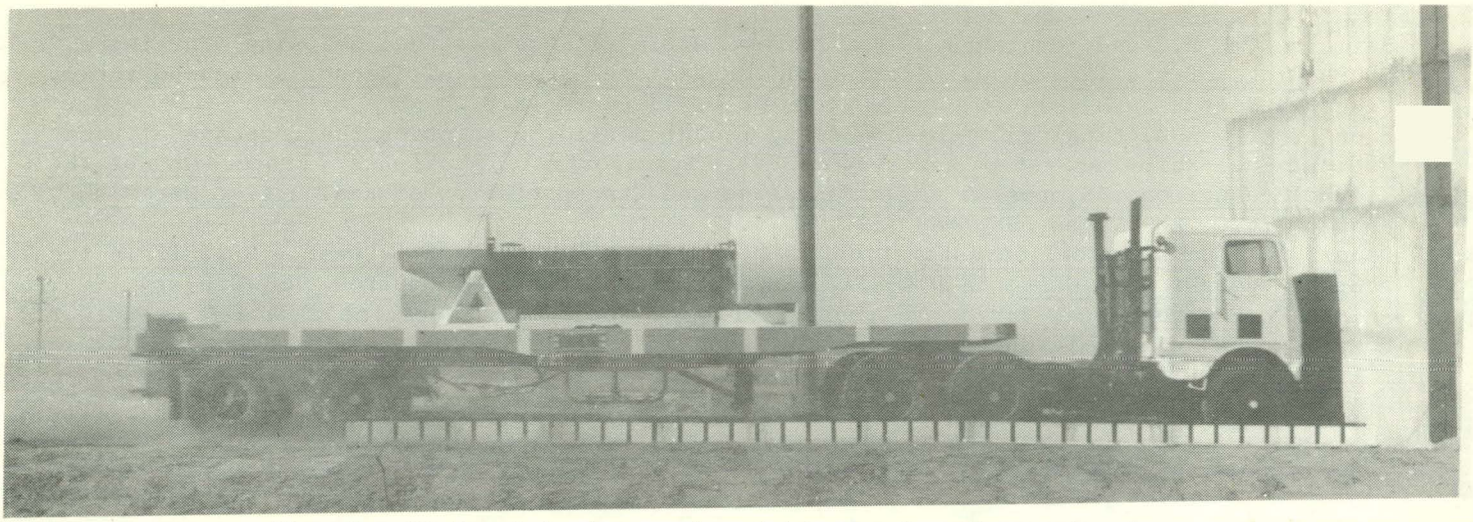


Fig. 1. 20.5-metric ton cask and transport system prior to impacting a 626-metric ton concrete barrier.

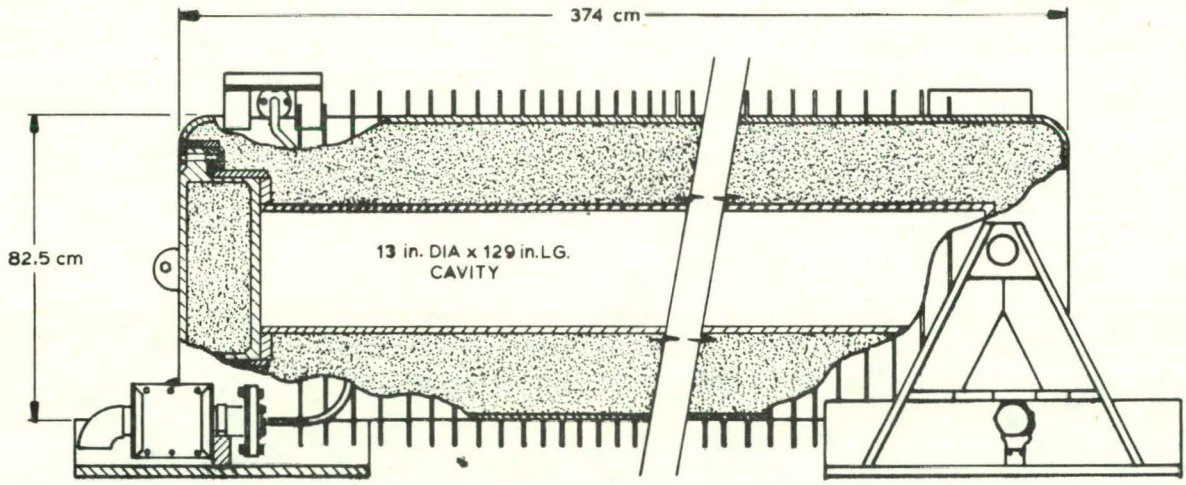


Fig. 2. 20.5-metric ton test cask.

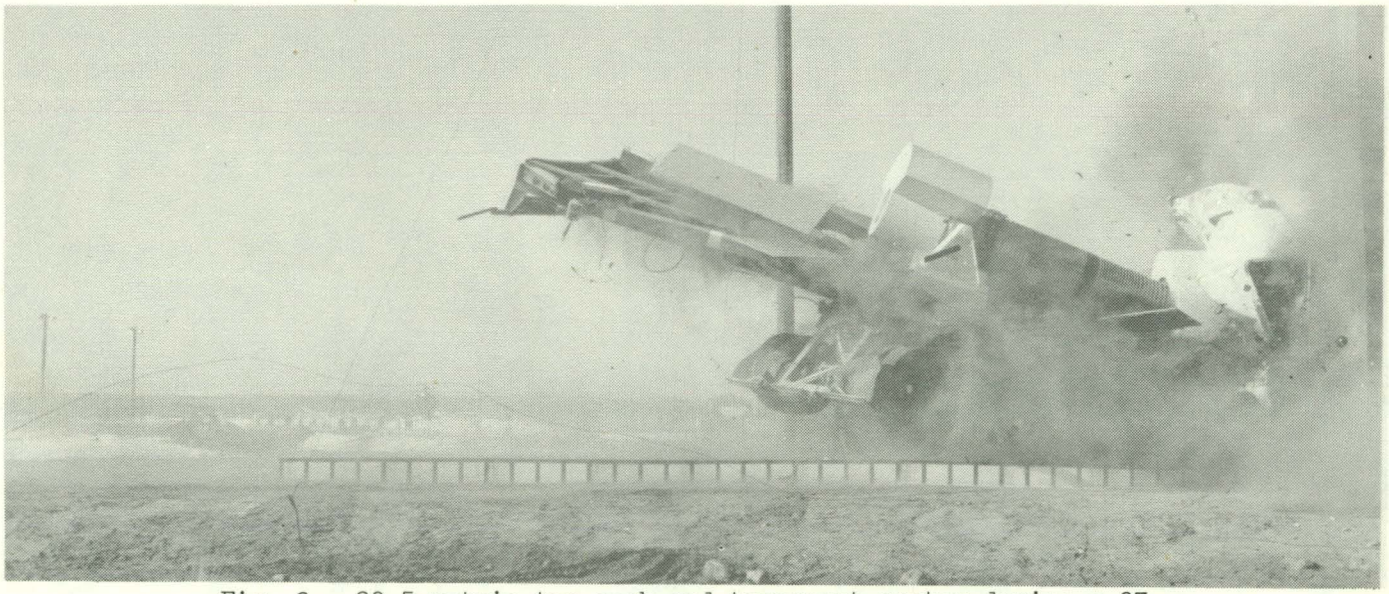


Fig. 3. 20.5-metric ton cask and transport system during a 97 km/h impact with a 626-metric ton concrete barrier.



Fig. 4. 20.5-metric ton cask and transport system in a 135 km/h impact with a 626-metric ton concrete barrier.

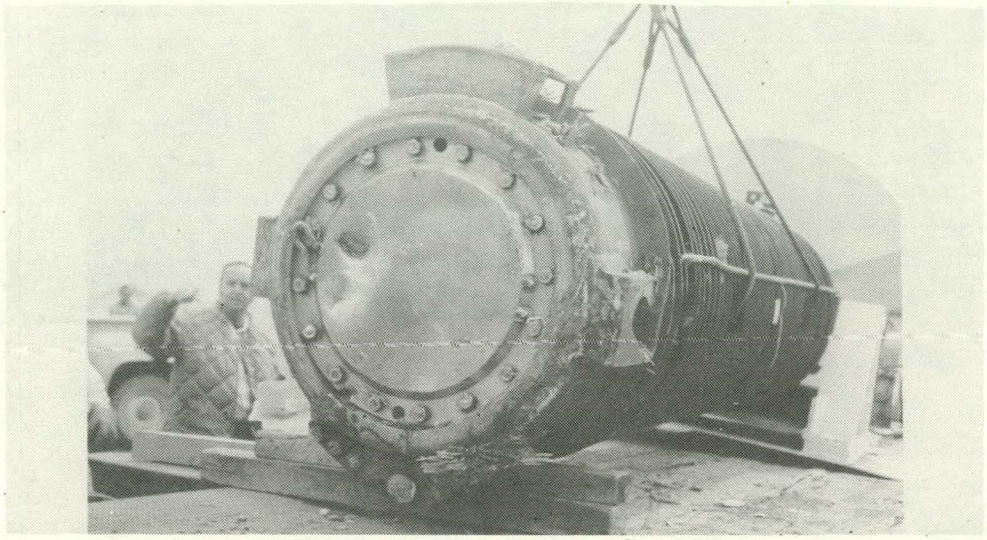


Fig. 5. Cask head after the 135 km/h impact test.

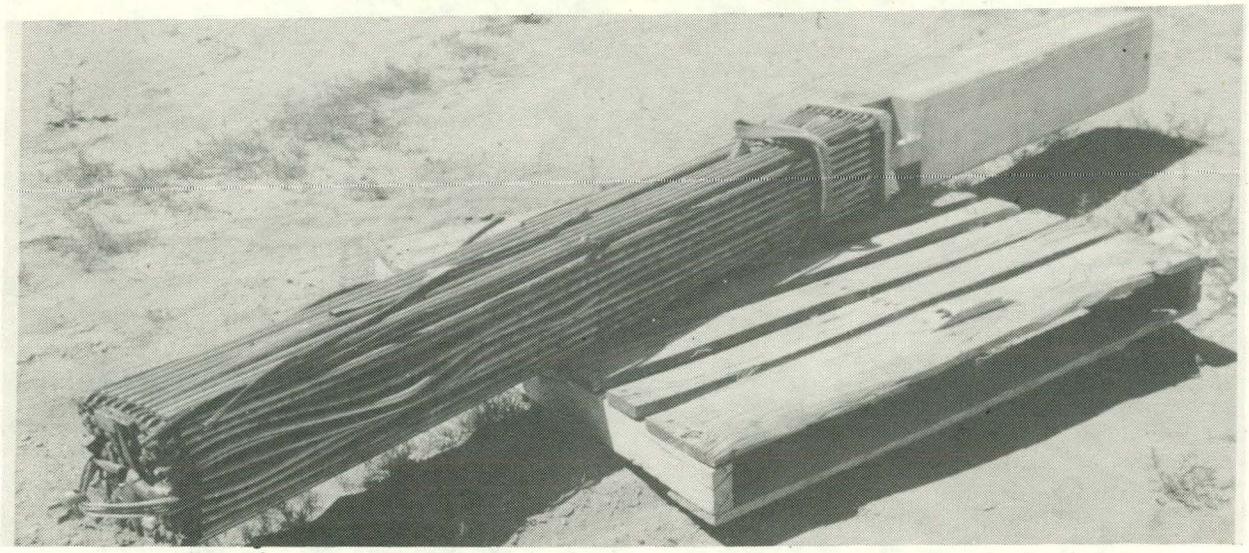


Fig. 6. Savannah Core II fuel assembly after the 135 km/h impact test.

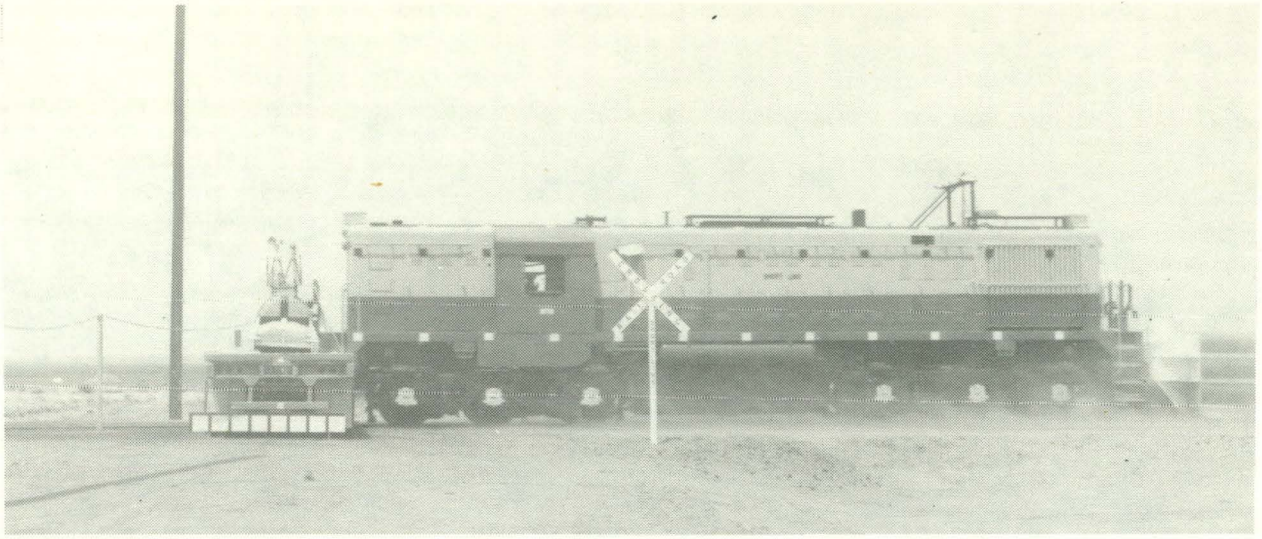


Fig. 7. 131 km/h locomotive grade crossing test.

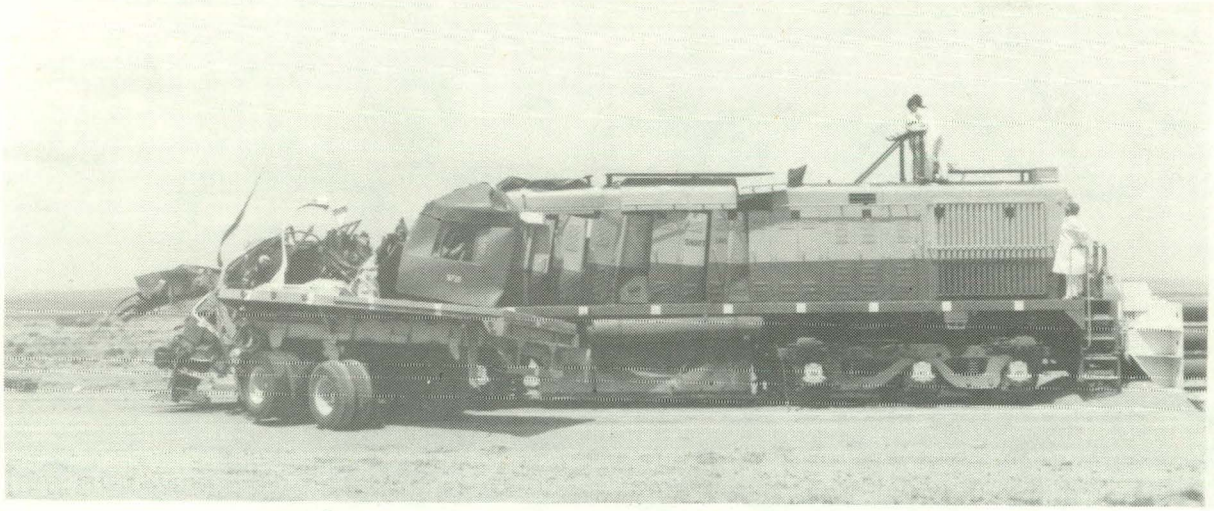


Fig. 8. Post test view of the crashed locomotive.

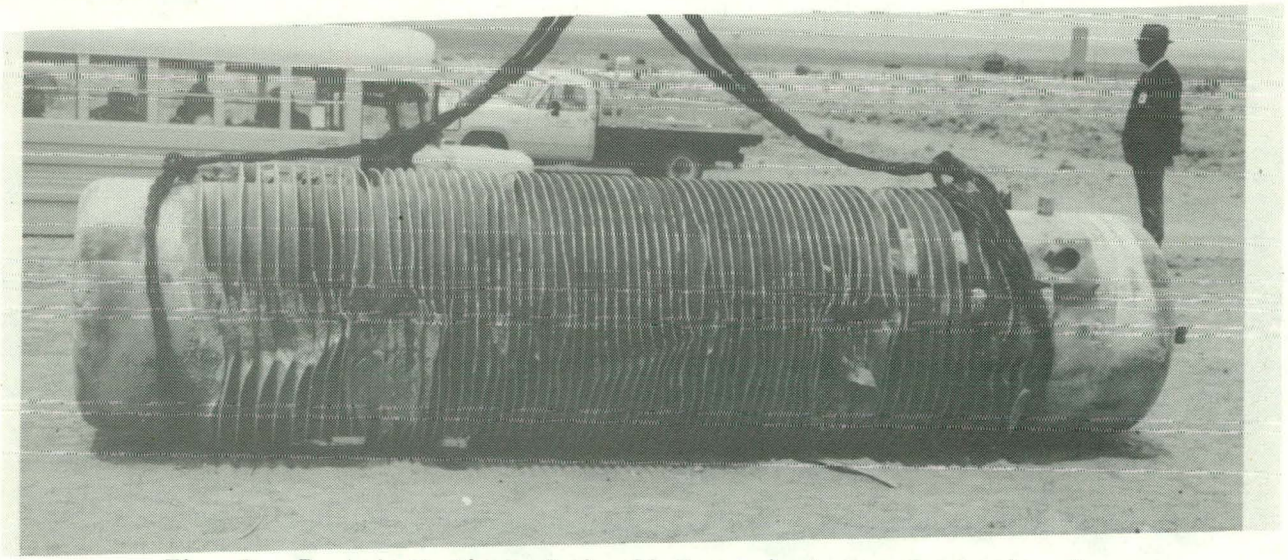


Fig. 9. Post test view of the 22.7-metric ton cask showing the points of impact with the locomotive frame.

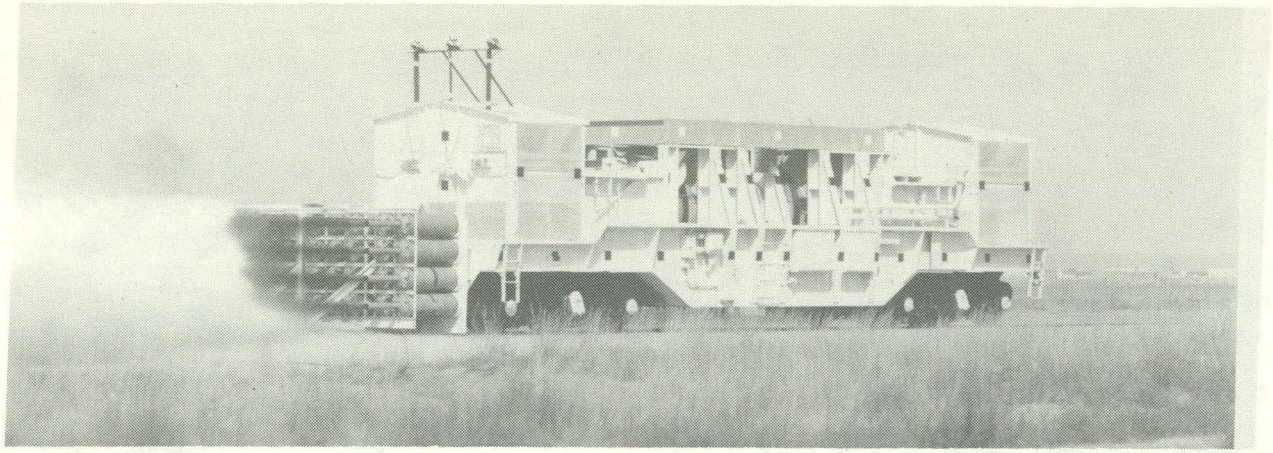


Fig. 10. Rocket acceleration of special railcar shipping cask system.

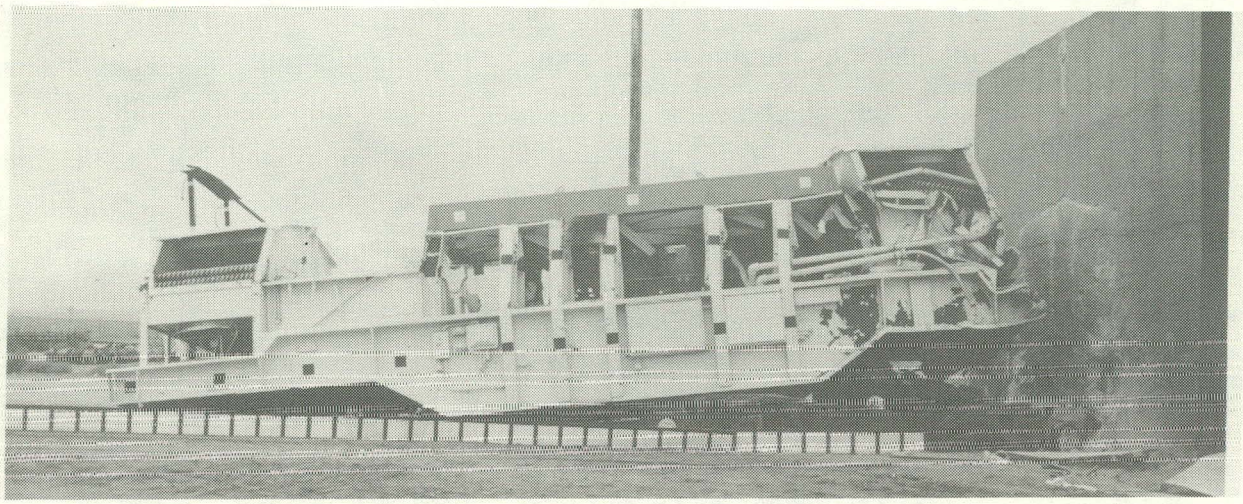


Fig. 11. Special railcar and cask system after a 131 km/h impact against a 626-metric ton concrete barrier.

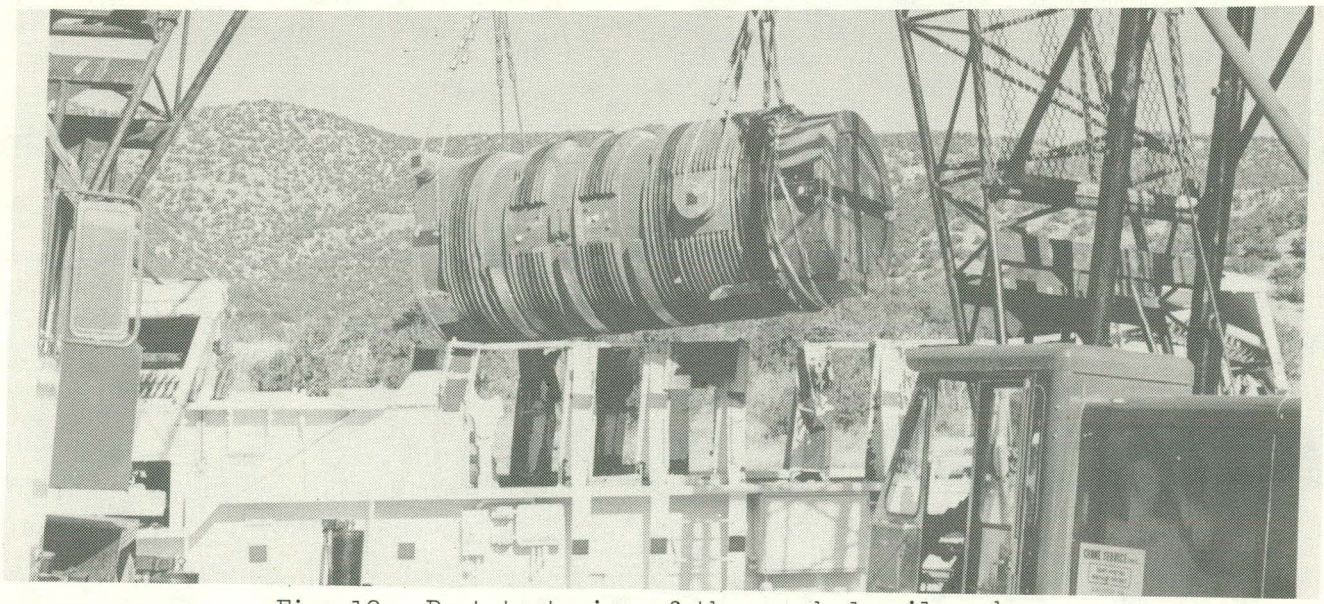


Fig. 12. Post test view of the crashed rail cask.

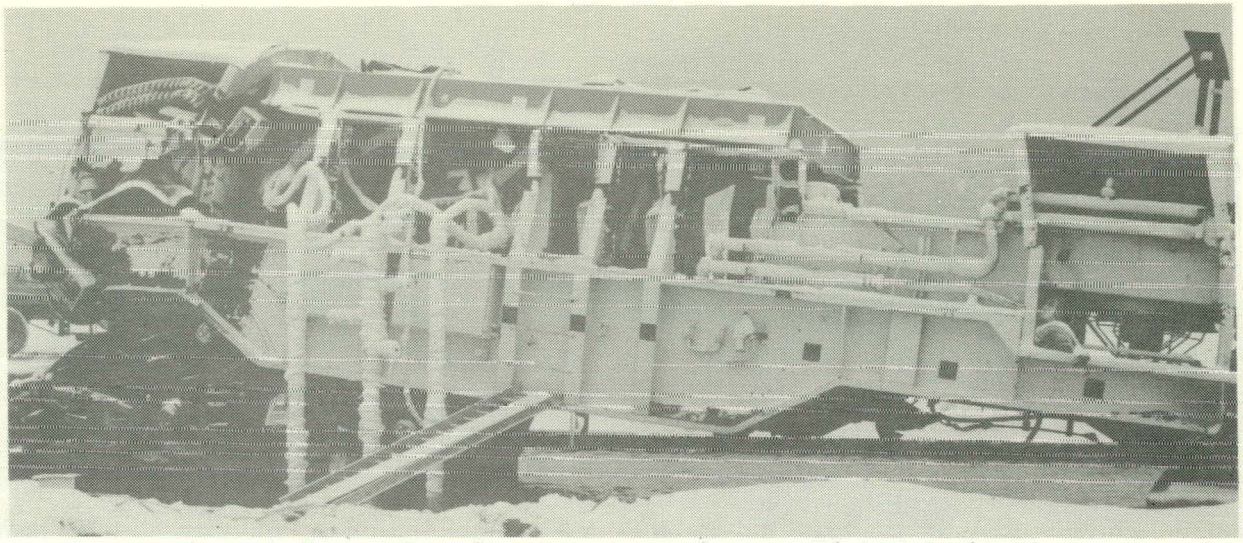


Fig. 13. Crashed railcar and cask in a specially designed pool fire pit.

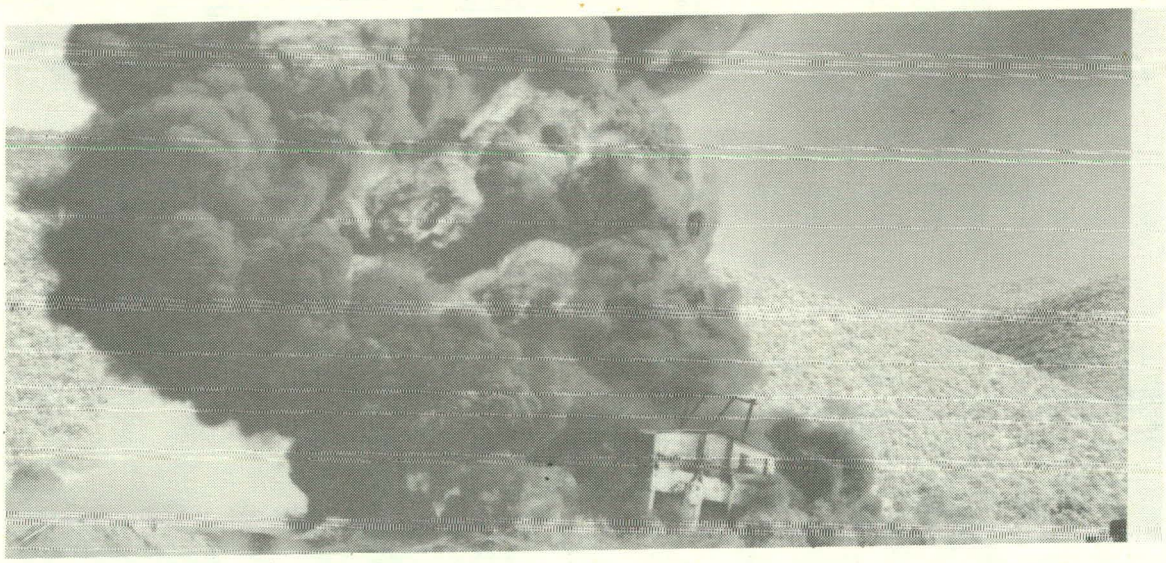


Fig. 14. Pool fire test of the special railcar and cask.

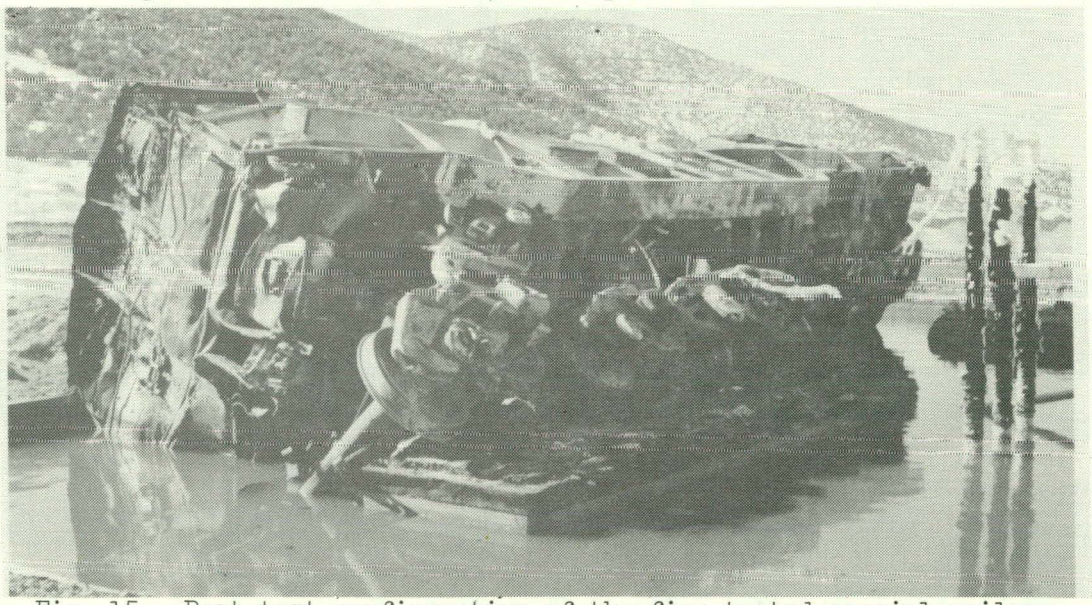


Fig. 15. Post test configuration of the fire tested special railcar and cask.

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ASSESSMENT OF THE HEAT TRANSFER PROPERTIES OF SPENT FUEL SHIPPING CASKS

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ABSTRACT

The purpose of this study is to assess the fire safety of spent fuel shipping cask through the analysis of fire test results. On the basis of results of open fire tests conducted in 1973-1974, of 1/2, 1/2.5 and 1/4 scale models of 80 ton cylindrical casks, heat transfer properties of the cask models in open fire were evaluated by computer analysis and the method for evaluation was confirmed to be feasible. A comparison was made between the temperature-time curves from the tests and those from the computer analyses.

Things obtained are summarized as follows. 1) The temperatures in the upper part of the cask models were generally higher than the computed temperatures, while in the lower part, the both agreed with each other. 2) The average flame temperature can be used as the outer boundary temperature for the analysis, unless the flame temperature fluctuates very much. 3) When the heat transfer by convection to casks in an open fire is computed, equations of forced convection by wind, not those of natural convection, should be used. 4) It is believed that when the flame temperature distribution is not even the heat flux by radiation should be computed with different flame temperature being take into consideration.

INTRODUCTION

Open fire tests of spent fuel shipping cask models were conducted in the years of 1973 and 1974 for the first time in Japan by the Research Sub-Committee of the Japan Society of Mechanical Engineers (JSME) under the Sponsorship of the Science and Technology Agency (STA) of the Japanese Government.

The data obtained in the open fire tests were subjected to computer analyses in 1976, when a research fund was awarded to Sub-Committee of JSME by the Japanese Government.

This paper reports the results of the analyses.

ANALYSIS AND DISCUSSION

The results of open fire tests were described elsewhere[1].

In the computer analysis, a heat conduction computer program "TRUMP"[2] was used. The transient temperature distribution was calculated using the heat input conditions by convection and radiation in order to estimate the mechanism of the heat transfer from fire to the scale models.

In a comparison of the calculated temperature curves with the experimental ones, the feasibility of various factors and analytical methods is described below.

Boundary Conditions

The heat transfer coefficient from the external boundary to scale models is given by equation (1).

$$H_{sb} = h_{sb}(T_b - T_s)^{P_{sb}} + R_{sb}\,\sigma(T_b^2 + T_s^2)(T_b + T_s) \qquad (1)$$

where

H_{sb}: total heat transfer coefficient

h_{sb}: convection coefficient

P_{sb}: convection exponent

R_{sb}: radiation form factor

σ: Stefan-Boltzmann constant

T_b: external temp (flame temp)

T_s: surface temperature

$$R_{sb} = \frac{F_{sb}}{(1/\varepsilon_s)+(1/\varepsilon_b)-1} \qquad (2)$$

$\varepsilon_b = 0.9$: emissivity of flame

$\varepsilon_s = 0.8$: absorptivity of cask surface

$F_{sb} = 1.0$: form factor

$R_{sh} = 0.7347$

As expressed in equation (1), and (2), the heat transfer coefficient from the external boundary contain the terms of radiation and convection.

Establishment of the Dimensional Modol

The cross-sectional temperature distributions in the central part of the 1/2.5 scale model were obtained in both two and three dimentional model analyses by computer.

An example of the temperatures obtained are given in Table 1. The difference between the results from the two and three dimensional model analyses is so small that the simpler two dimensional model analysis was decided to be used in this study.

Procedure of Flame Temperature in the Fire

The average flame temperature and actual flame temperature in the open fire test of 1/2.5 scale model are presented in Table 2.

Fig. 1, and 2 shows the temperature rises in the 1/2.5 scale model which were calculated using either the average flame temperature in Table 2 or the actual flame temperature as the ambient temperature. It is clearly understood from Fig. 1, and 2

that the value is almost the same where the average flame temperature or the actual flame temperature is used.

Table 1 Comparison between 2 and 3 dimensional models

Time (min)	Outer Surface		Lead Shield		Inner Surface	
	dimension					
	3	2	3	2	3	2
5	172.9°C	173.0°C	100.7°C	100.66°C	72.2°C	72.1°C
10	274.2	274.2	197.5	197.6	166.1	166.1
15	367.7	367.8	279.6	279.7	253.3	253.3
20	460.7	460.9	348.8	349.0	313.4	313.6

Table 2 Flame temperatures of 1/2.5 scale model

Time (sec)	Flame Temperature			
	(0°)	(90°)	(180°)	(270°)
0	300°C	400°C	700°C	500°C
50	310	500	900	700
100	500	650	1000	900
150	530	700	1030	950
200	530	700	1000	950
250	450	300	950	800
300	170	100	500	600
Average	460	650	920	810

The surface temperature of the cask rose in correspondence to the fluctuation of the flame temperature, but the inside temperature rose smoothly without being affected by the fluctuation.

So, the theoretical temperature with respect to time is qualitatively in agreement with the experimental one.

For the 1/2 scale model, the heat flux from the surface node was determined by calculation using the average flame temperature (A), or the actual flame temperature (B), as presented in Table 3.

In the thermal analysis of the cask, when a calculation was conducted using either the actual flame temperature or the average flame temperature, as the ambient temperature, no appreciable difference in the results was found.

It suggests that the average flame temperature can be used as the ambient temperature for the thermal analysis unless the difference between the actual temperatures is too large.

Table 3 Comparison of Heat Inputs calculated on the Basis of Average Temperature (A) and Actual Temperature (B)

($\times 10^2$kcal)

Time	0°		90°		180°		270°	
(sec)	A	B	A	B	A	B	A	B
50	0.44	0.70	1.10	1.10	3.05	1.36	2.02	1.61
100	0.88	1.39	2.19	2.25	6.08	4.40	4.04	4.26
150	1.32	1.76	3.27	3.42	9.09	7.57	6.04	6.45
200	1.75	2.03	4.35	4.19	12.1	10.3	8.03	7.95
250	2.18	2.36	5.42	5.02	15.1	12.5	10.0	9.37
300	2.62	2.85	6.49	6.08	18.0	15.5	12.0	11.2

Table 4 Heat Transfer Coefficient

(kcal/m^2 h °C)

			Natural	Forced Convection		
T_b	T_f*	Radiation	Convection	5m/s	10m/s	15m/s
400	200	18.5	7.05 (27.6%)	11.4 (38.1)	20.0 (51.9)	29.7 (60.0)
500	250	25.8	7.13 (21.7)	10.9 (29.7)	19.0 (42.4)	26.3 (50.5)
600	300	34.9	7.09 (16.9)	10.3 (22.8)	17.9 (33.9)	24.8 (41.5)
700	350	46.2	7.09 (13.3)	9.83 (17.5)	17.2 (27.1)	23.8 (34.0)
800	400	59.8	7.02 (10.5)	9.86 (13.5)	16.4 (21.5)	22.7 (27.5)
900	450	75.9	6.98 (8.4)	9.00 (10.6)	15.7 (17.1)	21.8 (22.3)
1000	500	94.8	6.89 (6.8)	8.63 (8.3)	15.0 (13.7)	20.9 (18.0)

* $T_f = \dfrac{(T_b - T_s)}{2}$

Surface temperature T_s = 0°C,

( ): Rate (%) of convection in the total heat transfer coefficient.

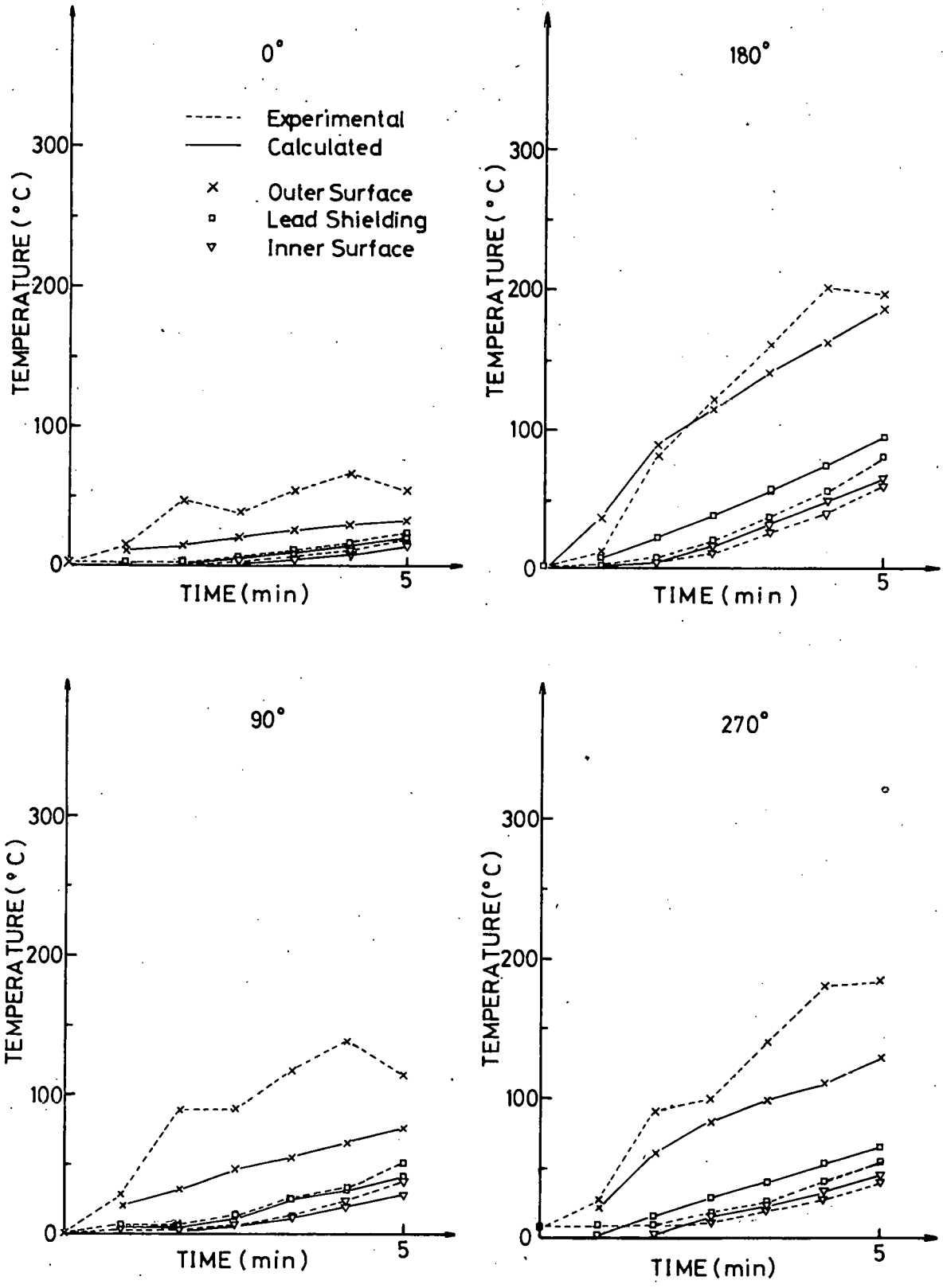


Fig. 1 Temperature vs. Time Curves of the 1/2.5 Scale Model.

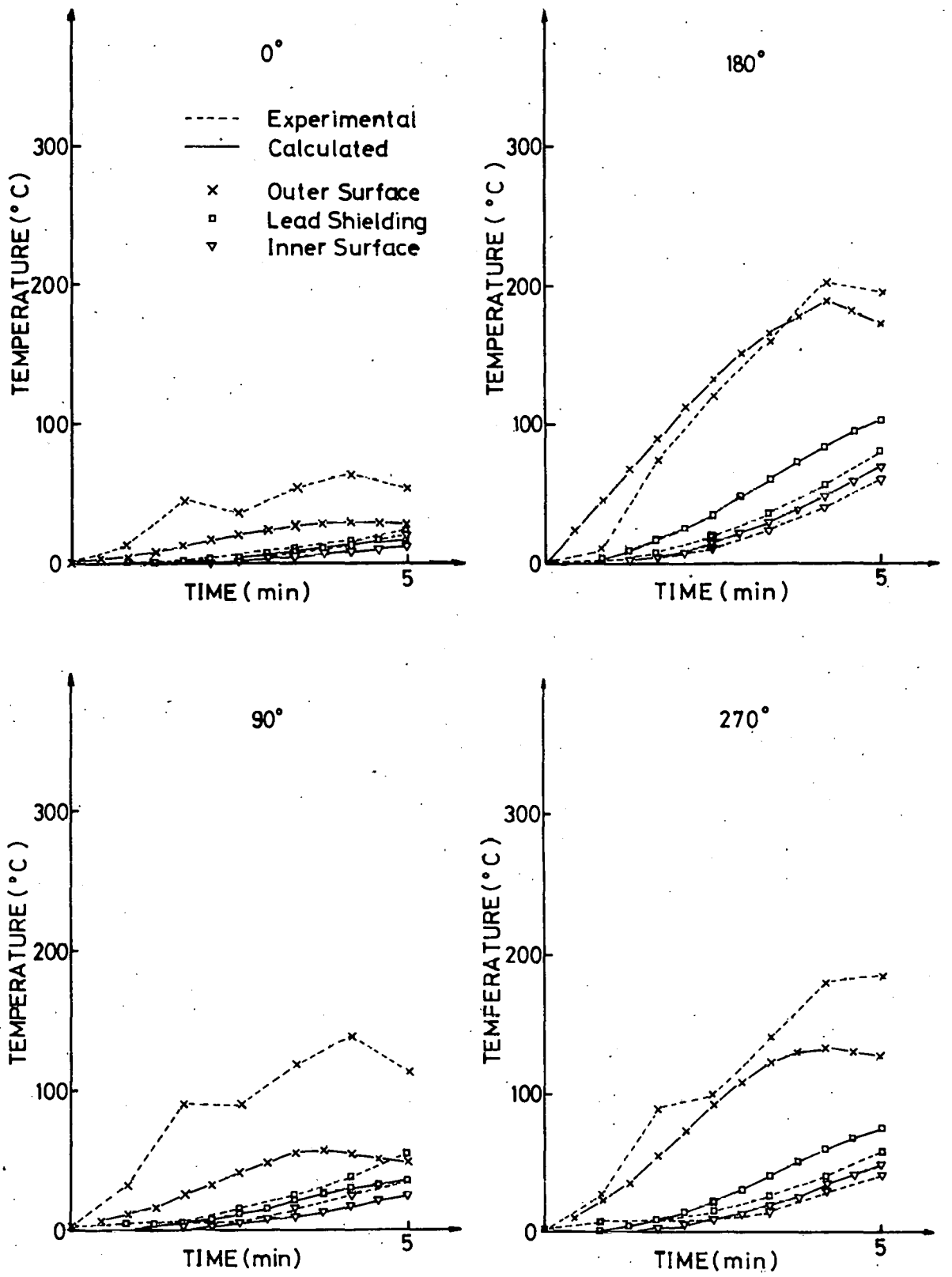


Fig. 2 Temperature vs. Time Curves of the 1/2·5 Scale Model by Actual Flame Temperatures.

Heat Input from Flames

Calculations were conducted on the heat input to the cask from flames under different conditions of the radiation, convection and ambient temperature. In the calculations, the temperature distributions in casks were obtained using the heat transfer coefficient equations (1), and (2).

An example of the calculation results for a 1/2.5 scale model is presented in Table 4, where the initial surface temperature T_s, 0°C was used. It was found from the results in Table 4 that the heat flux by the forced convection was larger than that by the natural convection.

Especially when the flame temperature was lower this trend was more noticeable. When the natural convection or the forced convection of 5, 10, or 15m/s was used in the analysis, the calculated temperature rises in the scale model under the natural convection were lower than those under any given forced convection. However, all of these calculated temperature rises were much lower than the experimental ones.

It is probably because the radiation has larger effect on the temperature rises than the convection when the temperature is so high. The flame temperature differed in the circumference direction in the open fire. The heat transfer coefficient is largely dependent on the radiation, as can be found in Table 4.

So, the calculated heat input varied considerably, when radiation inputs from flames of different temperatures in different directions were taken in to consideration in analyses.

The heat input from 0° direction of the scale model was calculated on the basis of $T_b(0°)=400°C$, $T_b(90°)=700°C$ and $T_b(270°)=800°C$, and Table 4, as follows:

$$Q_1 = 7.05 \times 400 + 18.5 \times 400 = 1.02 \times 10^4 \quad (3)$$

$$Q_2 = 7.05 \times 400 + 0.5 \times 18.5 \times 400 + 0.25 \times 46.2 \times 700 + 0.25 \times 59.8 \times 800 = 2.66 \times 10^4 \quad (4)$$

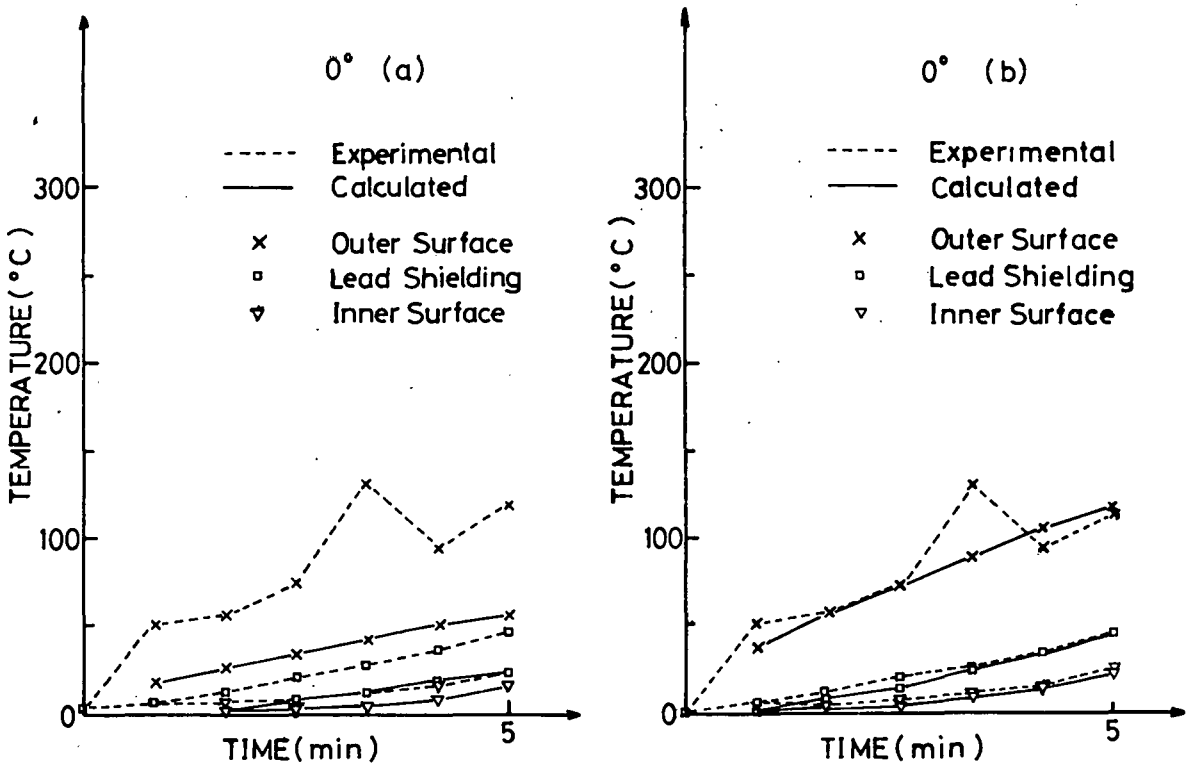


Fig. 3 (a, b) Temperature vs. Time Curves of the 1/2 Scale Model.

where, the effects of radiation from 0°, 90° and 270° directions were assumed to be 0.5, 0.25 and 0.25 respectively.

The temperature rises in 0° direction were obtained by calculation using the heat input from equation (4).

As a result, the calculated temperature rise curve became almost in agreement with the experimental temperature rise curve as shown in Fig. 3.

CONCLUSIONS

Heat transfer properties of the cask models in open fire were evaluated by computer analysis and the method for evaluation was found to be reasonable.

A comparison was made between the temperature-time curves from the tests and those from the computer analyses.

Things obtained are summarized as follows.

1) The temperatures in the upper part of the cask models were generally higher than the computed temperatures, while in the lower part, the both agreed with each other.

2) It was found that when the temperature gradient of flame was small in the model axial direction, the temperature distribution in the cross section of the contral part of the scale model obtained from the two dimensional model analysis by computation was very close to that obtained from the three dimensional analysis.

3) The average flame temperature can be used as the outer boundary temperature for the analysis, unless the flame temperature fluctuates very much.

4) When the heat transfer by convection to casks in an open fire is computed, equations of forced convection by wind, not these of natural convection should be used.

5) The heat flux by radiation should be computed with the unevenness of flame temperature distribution being taken into consideration.

6) The heat input by radiation to the cask was found to be predominantly large in the open fire tests, compared with that by convection from the computer analysis of the results.

In the case of 800°C of flame temperature, the heat input by radiation is estimated to be approximately 80 - 90 %.

ACKNOWLEDGEMENTS

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Dup

THERMAL TESTING A DRY SPENT FUEL CASK

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I. INTRODUCTION

A nuclear spent fuel cask that employs air or an inert gas such as helium as the cooling medium in the fuel cavity is commonly known as a "dry" cask. The gas serves as a passive coolant. There is no forced circulation of the gas by mechanical means. The heat from the nuclear fuel is removed solely by natural convection and thermal radiation from the outer surface of the cask to the atmosphere. In recent years, several casks have been designed for dry shipment as opposed to being water filled. The dry cask has an important safety advantage. The fuel cavity pressure remains less than a few atmospheres under accident conditions, when the cask surface is exposed to intense heat from a fire. This low pressure feature permits the cask to be designed for "zero-release" of radioactive material during an accident. An operational advantage is the eliminating of the need to sample the coolant for radioactivity at any time.

The thermal characteristics of a dry cask differ considerably from a wet cask. A gas is a poorer heat transfer medium than water. As a result, the fuel and cask basket temperatures will be higher for a given fuel heat level. The response characteristics during operating transients will also differ. Detailed thermal tests on a dry cask were performed to verify design parameters and to obtain operational data for shipping and unloading. This paper presents the results of a number of thermal tests performed at the Barnwell Nuclear Fuel Plant (BNFP). The uniqueness of these tests are that they were performed using a full-scale geometric and thermal mockup of a nuclear fuel assembly. This results in more realistic fuel and basket temperatures being measured than those obtained using a simpler, single heater assembly.

II. TEST EQUIPMENT

The cask tested was the NLI 1/2 legal weight truck cask. This cask is shown in Figure 1. It uses water as a neutron shield and a lead-depleted uranium gamma shield sandwiched radially between layers of stainless steel. The cavity coolant is helium filled at atmospheric pressure. Helium was selected because it has better heat transfer properties than air. The cask is capable of carrying either one pressurized water (PWR) fuel assembly or two boiling water (BWR) assemblies at a maximum heat load of 10.6 kwt (36,100 Btu/hr). Two different basket types can be used depending on the integrity of the fuel prior to shipping. So called "leaker" assemblies would be placed within a separate cylindrical "leaker-can" basket consisting of a

leak-tight pressure vessel within the fuel cavity. Nonleaker fuel would be placed in an extruded aluminum basket which also fits within the cask cavity (see Figure 1). The latter is denoted as the normal shipping basket.

The mockup fuel assembly was designed and fabricated by Wachter Associates of Pittsburgh, Pennsylvania (see Figure 2). It consisted of 205 electrically heated rods and 20 dummy rods in a 15 x 15 matrix. The intent was to simulate a typical PWR fuel assembly. A PWR assembly was selected since it represents the thermally limiting situation. The heater rods were directly bonded to a cylindrical test head which was bolted in the place of the cask's normal inner closure head. The 12-foot heated length of the assembly corresponded to the axial location of nuclear fuel when being shipped. A total of 21 thermocouples were positioned in three axial locations on each of seven different heated rods. These temperatures were automatically recorded during the course of the test. Means for measuring a number of local basket and cask surface temperatures were also provided. The cask test head provided for normal fill and vent valves which permit evacuation of the cask cavity, and subsequent filling with helium gas. The electrical current and voltage to the mockup assembly was routinely monitored to assure constant thermal output from the heaters.

III. TEST RESULTS - NORMAL CONDITIONS

The primary purpose of the testing was to obtain the temperatures of the cask and fuel under a variety of arrival conditions at the BNFP. This information was to be used in the development testing of the BNFP cask cooldown system.[1] Data on cask surface temperatures had been obtained previously using a single heater assembly. Computer analyses of the fuel and basket temperatures had also been developed for the package Safety Analysis Report (SAR). However, these analyses did not consider the wide range of arrival conditions which can be expected. For example, they do not consider fuel heat-up rate, and operating conditions at less than the design heat level. An additional objective was to obtain comparative data on the fuel and basket temperatures for the two different basket types.

The tests were performed by placing the heater assembly into the cask and horizontally mounting the cask on a truck trailer (the normal shipping condition). The power input in successive tests ranged from 2 kW to 10.6 kW. The cask was located in shaded conditions with an ambient temperature range between 68° and 72°F.

Table 1 presents the average equilibrium temperatures for the fuel and the basket. The average fuel temperature was calculated by analytically weighing the measured rod temperature with the expected temperature derived from computer analyses. This technique was estimated to have a maximum inaccuracy of about ±30°F. The average fuel and basket temperatures were in reasonably close agreement with previous analyses. Other observations were:

(1) The fuel temperatures, in the leaker can basket, were a maximum of 60°F higher than the normal basket. This was expected since the can represents an additional thermal barrier. (Note: The leaker can basket will be used infrequently and only with "well-aged" fuel.)

(2) The maximum hot-spot temperature was at the axial centerline of an interior rod. It is about 175°F hotter than the average fuel temperature.

(3) A typical "chopped cosine" temperature profile was observed axially along each fuel rod. The temperatures at the fuel rod ends were about 100 to 150°F cooler than at the centerline.

Figure 3 is a plot of average fuel, basket, and cask surface equilibrium temperatures, as a function of spent fuel decay heat. The decay heat level is related

to an approximate fuel age. In the current "millieu," fuel will be allowed to "age" in the reactor fuel pools for at least three to five years.

Temperature transients differ between dry and wet casks. Gaseous layers separate the fuel from the basket and the basket from the cask body and serve to thermally "decouple" the response. As shown in Table 2, the fuel temperature rises far more rapidly than the cask body. As a practical consequence, at design rating, the fuel temperature will rise 500-600°F in a few hours. In the same time period, the cask surface temperature will not have changed. As a result, the cask surface (the only observable temperature) presents no indication of internal temperatures until three to four days have elapsed. In an operating situation, the actual duration of many truck cask shipments is expected to be less than a day. Fuel and cask internal temperatures can be estimated from test data by means of Figure 3 and Table 2.

There are two situations where test data does not agree with computer analyses. The reason for these variances are (1) natural circulation patterns of the fuel cavity gas and neutron shield water and (2) the horizontal orientation of the cask during shipment.

(1) Cask Surface Temperatures (Figure 4A) - Natural circulation of the fluid in the neutron shield tank and the ambient air in proximity to the cask results in the upper surface of the cask being hotter than the lower. This general effect has also been observed with the TN-9 cask where a solid hydrogeneous neutron shield is used, and in the NAC-1 cask where a four-chambered, water-filled neutron shield tank is employed.

(2) Fuel/Basket Temperatures (Figure 4B) - The fuel rods directly contact the lower surface of the basket when the cask is horizontal. In turn, the gap width of the gas layer between the fuel and the upper region of the basket is doubled. The lower region of the basket, is hotter than the upper region due to direct conduction between the fuel rod and the basket segments. The double width of gas in the upper region accentuates this effect by acting as an insulator.

The surface temperature effect is well known and presents no structural problem. However, the skewing of the temperatures in the fuel basket was unexpected. The initial tests performed with the "leaker can" basket indicated as much as a 350°F difference between the upper and lower regions. The four segments of the basket were connected with thin aluminum straps at each corner (Figure 4B). In effect, there was no conduction path to redistribute heat. Slight cracking of the welds was observed at the juncture of the basket segment and the aluminum strap. Testing of the normal basket showed that heat redistribution was considerably improved. The normal basket is fabricated as a continuous aluminum extrusion with a minimum ligament thickness of 5/8-inch in the corners. This adequately redistributes the heat from the bottom of the basket to the upper regions by conduction. The maximum differential between any of 18 measured locations on the normal basket was less than 70°F. In addition, the method of mounting the basket within the cask permits 0.5-inches of axial growth. This basket design, when tested, presented no thermal or structural problems.

IV. ABNORMAL OPERATING CONDITION - TEST RESULTS

Additional tests were conducted to obtain general insight on other thermal aspects of dry casks. These included (1) vertically orienting the cask, (2) using air as the cavity gas, and (3) using a single calrod heater.

A. Vertical Casks

The thermal characteristics of a vertical cask are better and more predictable than when horizontal. The natural circulation of fluids (both helium and water) within cask chambers are greatly increased, thereby, improving heat transfer. This results in both lower and more uniform temperatures. Fuel temperatures are as much as 110°F lower. Additionally, since a uniform gap exists between the fuel and the basket, no "skewing" of basket temperatures, as previously described occurs. The driving force for natural circulation of a fluid is proportional to chamber height. This is the so-called "chimney" effect. Hence the improvements in heat transfer in the long, narrow, vertical fuel cavity of the cask is expected.

B. Air Coolant

Helium was experimentally shown to be superior to air. The thermal conductivity of air is about one-forth that of helium (0.03 versus 0.13 Btu/hr-ft-°F). In a horizontal cask, there is little natural convection within the fuel cavity. The modes of heat transfer from the fuel to the basket are only radiation and conduction through the gas. At the design heat level, the average temperature of the fuel was about 90°F hotter in air. Of greater cause for concern was the increased "skewed" temperature variations in the leaker-can basket. The difference in temperature between the upper and lower regions of the basket increased from 350°F to 510°F in air. Evidently, the doubled thickness of air above the fuel assembly was at fault. The insulating effect of the air accentuates this situation.

C. Single Calrod Heater

Many thermal tests have been performed using single calrod heaters, typically 0.75 to 1.0 inch in diameter, because of lower cost. The cask outer surface temperatures will be the same no matter what type of heater is used. However, cask cavity temperatures will vary significantly. This was shown by measuring temperature distributions during both heat-up and cooldown tests.[1] The primary reasons are:

(1) Fuel heat fluxes are lower and stored heat level higher in the 15 x 15 fuel assembly than in a single heater. There is about a factor of ten difference.

(2) The difference in cross-sectional area between the two types of heaters causes significant differences in the natural circulation patterns exhibited by the gas coolant.

(3) The actual fuel assembly contacts the basket directly when the cask is horizontal.

V. CONCLUSIONS

1. Shipment of nuclear fuel in a dry cask should present no operating or safety problems even when shipping "six-month" cooled fuel. Operations with "aged" fuel can be simpler for a dry cask than a wet cask.

2. Thermal concerns for fuel aged 3 to 5 years or more are negligible. The temperatures of the fuel and the cask will be relatively low irrespective of cavity coolant or orientation of the cask.

3. Natural circulation of the gas in the fuel cavity is only of importance when the cask is vertically oriented.

4. Fuel baskets should be designed with sufficient metal thickness to permit proper heat distribution. This will prevent warping or other thermally induced damage.

5. Helium gas results in both lower and more uniform fuel cavity temperatures than air when shipping fuel with a high heat level.

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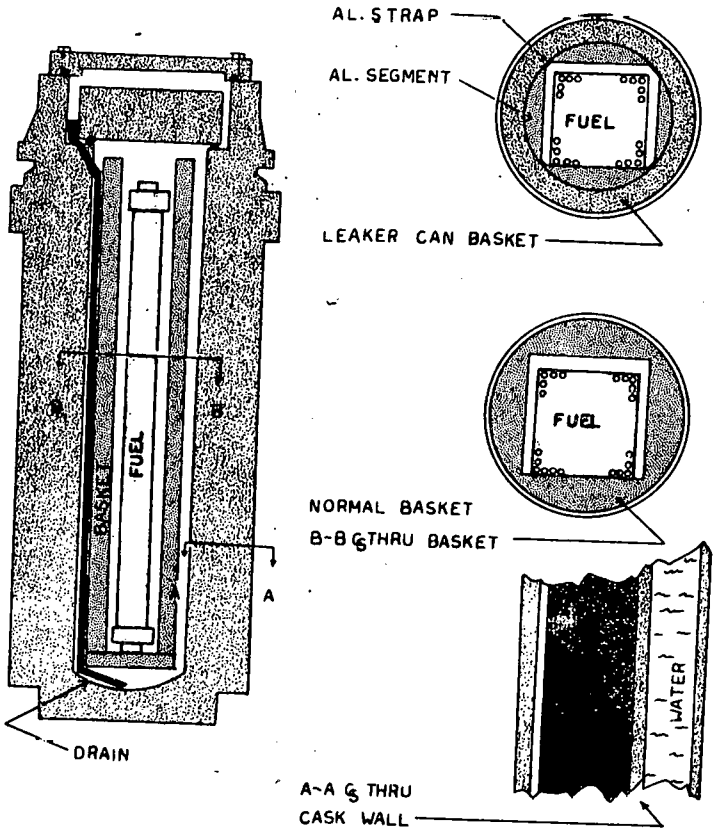


FIG. 1 - NLH2 SPENT FUEL CASK

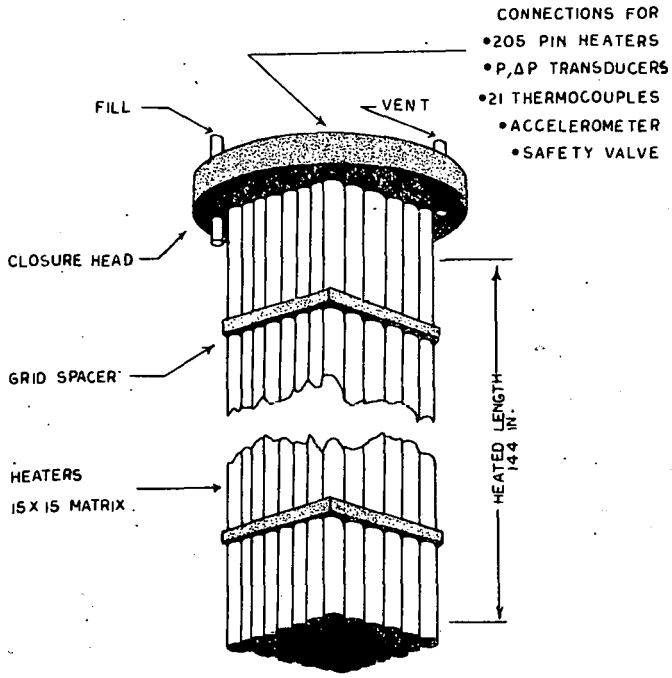
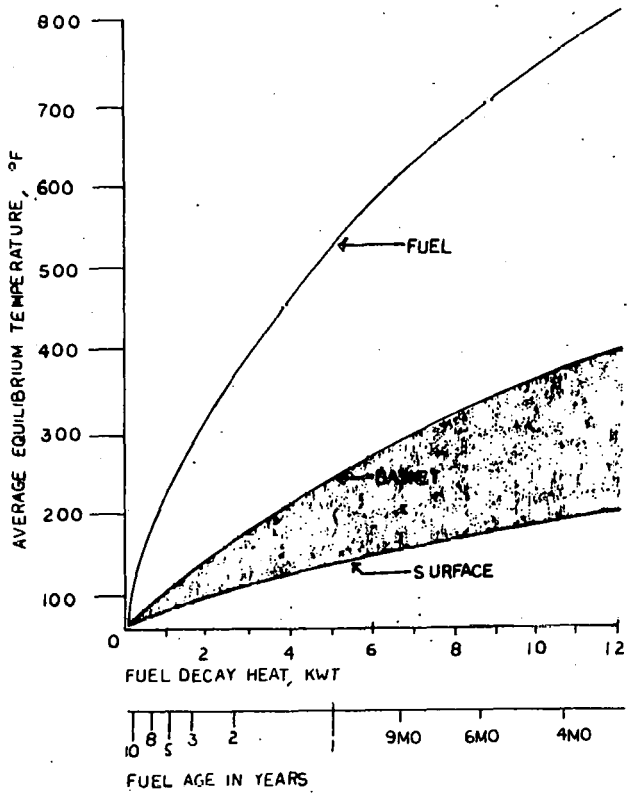


FIG. 2 - DUMMY FUEL ASSEMBLY (Wachter Associates)

FIG. 3 - NL 1/2 CASK TEMPERATURE PROFILE

- 70° F AMBIENT
- NORMAL BASKET
- HORIZONTAL



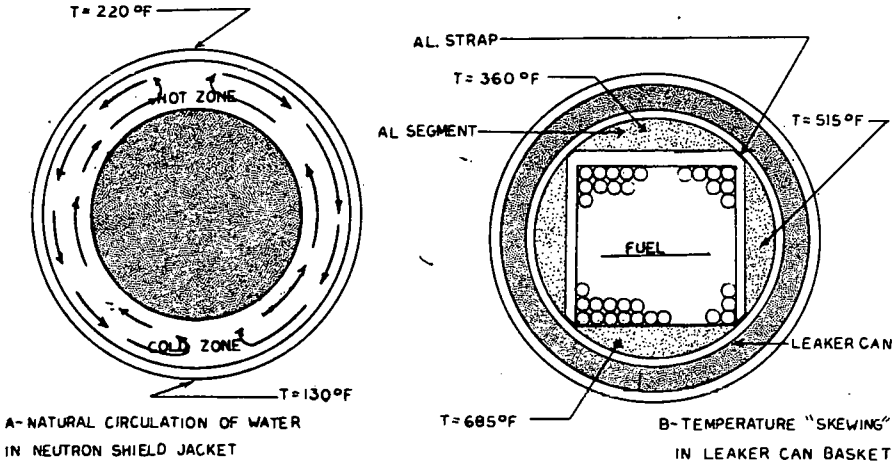


FIG. 4 - TEMPERATURE PATTERNS IN A HORIZONAL CASK

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OPERATIONAL FACETS OF A DRY SPENT FUEL CASK

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I. INTRODUCTION

There are several operational activities unique to a "dry" spent fuel cask. During loading, it is necessary to rapidly displace water and dry out the fuel cavity. During unloading, it is necessary to "cooldown" the hot fuel and cask internals prior to placing the cask in a spent fuel pool. Techniques for rapidly and efficiently performing dryout and cooldown were developed for the NLI 1/2 Legal Weight Truck (LWT) cask at the Barnwell Nuclear Fuel Plant (BNFP). The results of this testing are reported. The techniques developed can be utilized equally well for larger dry casks such as the NLI 10/24 rail cask. The test results indicated that these dry cask operations should not cause problems during loading and unloading in excess of that experienced with a wet cask. In fact, elimination of coolant sampling and the need to meet coolant activity limits is a distinct advantage.

II. DRYOUT

The Certificate of Compliance (COC) for dry casks typically states that water will be removed and the cask cavity be dry prior to shipment. The problem is how to assure that the cask is truly dry after the closure head is bolted shut. It was necessary to develop a technique that was not only safe and rapid but that also gave indisputable evidence of the "dryness" of the cavity. The technique developed involved a high vacuum vaporization of residual water after air pressure displacement of the bulk of the water from the cavity. This vacuum drying technique does not rely on heat from the nuclear fuel and is operable even when the cask body is cold (less than 50-60°F). It can be performed in parallel with other loading operations such as decontamination. Typically, it requires only 15-30 minutes to perform. Evacuation of the cask cavity is also a necessary step prior to filling with helium. Additionally, attainment of cavity pressures in the 1 - 3 torr* range is an excellent check of the leak tightness of the closure seal.

There are several key ingredients to assuring successful dryout:

---

* 1 torr = 1 mm Hg pressure

(1) Prior Removal of Bulk of Water in Cask

It is impossible to boil off a layer of water more than one inch deep in a reasonable period of time. This is particularly true in a cask with a cold body and aged fuel. Only a small amount of the residual water will be "flashed" when the vacuum is first drawn. The drying operation is dependent upon the rate of heat transfer between the cask floor and residual water temperatures. The thermal driving force is increased by lowering the vapor pressure (and hence saturation temperature) of the boiling water. This temperature can be so low that at times, frost has been observed (through a transparent plexiglass head used in an empty cask during test development). A highly efficient drain tube was developed to assure that only a small quantity of water remained. The tube was shaped to draw out all but a thin residual water layer (less than 1/4" deep). The tube shape minimizes swirls and vortices in the vicinity of its entrance.

(2) Properly Sized Vacuum Pump

The specific volume of steam vapor at 5-10 torr is 75-150 times that of steam at atmospheric pressure. The vacuum pump must have sufficient capacity to remove a large volume of steam even though the mass of water is relatively small. A Stokes, Model 148-H 50 CFM industrial vacuum pump performed ideally. For vacuum drying of a rail cask (cavity volume approximately 70 ft^3 versus 7-1/2 ft^3 for the NL 1/2) a 150 CFM pump would probably be used. These pumps employ a "ballast" valve which permits steam vapor to escape directly through the pump seal oil. It is advisable to allow the pump to operate in no-load condition for about 30 minutes prior to evacuating the cask thereby heating the seal oil. This prevents vapor from condensing in the pump.

(3) Sensitive Vacuum Gauge

Total dryout of the cask is assured when the cavity vacuum drops below about 1-2 torr (saturated vapor temperature less than 35°F). Indication of this low vacuum was observed with a "Pirani" type gauge. Figure 1 presents characteristic dryout curves for three different situations. Curve A illustrates proper operation. Curve B occurs when the vacuum pump is slightly undersized or a thin layer of water remains in the cask. Curve C (no dryout) occurs either when the cask closure is leaking; the pump oil is badly contaminated with water; or a deep layer of water still remains in the cask.

III. COOLDOWN

The temperature of the fuel in a dry cask at design heat load may be 200-400°F higher than a wet cask. This does not cause any shipping safety problem; however, it does require that the fuel pin and basket temperatures be lowered below the boiling point prior to unloading the cask in a spent fuel pool. A cooldown system was designed for the BNFP fuel receiving station[1] (see Figure 2). Operation is based on the initial slow gravity flow of hot pressurized water into the cask. This is followed by depressurization of the coolant and then rapid circulation of cool water through the cask. The coolant flow is recycled in this closed system. The system also has provision for filtering and demineralization of the coolant.

For illustrative purposes, the cask system can be divided into three discrete heterogeneous regions at design heat load: (1) nuclear fuel at 700-800°F; (2) fuel basket at 400°F; and (3) cask body at 250°F. The system operation is in three steps:

Step One - Fuel Cooled - Hot pressurized water at 400°F (~235 psig) is admitted to the cask (the water temperature is approximately equal to the average basket

temperature). The coolant is vaporized when it contacts the hot fuel. The steam vapor is piped to a heat exchanger and condensed. The condensate is returned to the hot water heater and recycled. Gravity flow of water is continued until the cask is full. At the conclusion of this step the fuel will have been cooled to the basket temperature.

Step Two - Coolant Depressurized - The cavity is vented to atmospheric pressure and depressurized. The temperature of the water, fuel, and the basket gradually drops to about 212°F.

Step Three - Rapid Flush - Water at room temperature is rapidly flushed through the cask until the fuel, basket, and cask inner wall temperature is lowered to 150°F or less.

The testing was performed using the NLI 1/2 cask and the heated PWR mockup fuel assembly fabricated by Wachter Associates.[2] The tests were run with fuel heat levels ranging from 20% to 100% of design. The operating procedure established matches the inlet water temperature with the average basket temperature in Step One. This reduces cooldown time and prevents thermal shock to the cask body. Under low fuel heat conditions, the inlet water temperature may be as low as 200°F. The tests corresponded to a wide range of fuel "aging" prior to shipping (typically, from six months to about 3-4 years). As the fuel decay heat level in the test was increased upward to the design rating (10.6 kwt), the cooldown time increased from about 20 minutes to one hour. This was due to the hotter temperature of the fuel and cask. Figure 3 illustrates the characteristic cooldown curve. Fuel temperatures derived from thermocouples mounted at three axial positions are plotted as a function of cooldown time. The AGNS cooldown system worked in a smooth and predictable fashion. No pressure transients or boiling instabilities were observed.

One potential operating problem was uncovered that affects fuel unloading after cooldown. The cask cooldown system adequately cools the fuel and the cask internals. However, the cask body is not cooled by the cooldown system. This is due to the physical characteristics of the cask and is not a fault of the cooldown system. The stored heat in the mass of the cask body and neutron shield is considerable, and the rate of heat transfer from the cask body to the coolant is relatively low. The heat from the cask will preferentially flow into the cask cavity water after cooldown. If the cask body temperature is greater than about 220°F, the heat from the cask body supplemented by fuel decay heat will rapidly reheat the cavity water. Boiling will begin in 1.5 hours. Continued rapid flushing of the cask during cooldown (Step 3) for an additional 30 minutes to one hour will not change this, due to the low heat transfer rate. Fortunately, the time needed to place the cask into the pool after cooldown is about 20-30 minutes. This problem has been circumvented with the NLI rail cask. An auxiliary water cooling system mounted on the rail car maintains the cask body temperature below 150°F during shipping.

IV. PORTABLE COOLDOWN SYSTEM

There is a need for a portable cooldown system. This system would be used for cask unloading during emergency conditions or at locations where a permanent system is not available and rapid cooldown is not required. Figure 4 is a photograph of the cooling skid which was developed. The system design differed from the BNFP cooldown system in the following ways:

(1) Feedwater was admitted at a slow rate at room temperature and heated by the cask and fuel (versus preheating with a hot water heater).

(2) Coolant flow was not recycled but directly discharged to the spent fuel pool.

(3) Steam vapor developed in the cask was condensed in the tubeside of a heat exchanger suspended in the fuel pool. The fuel pool water serves as the heat sink.

(4) System pressure was maintained manually by throttling steam issuing from the cask.

The system was tested at three heat levels from 30% to 100% of design thermal rating. Cooldown took from one to two hours. Cooldown was continuous, and no erratic pressure or thermal transients were observed. Subsequent evaluation of the test results did not reveal any areas of high thermal stress to the cask body, cask internals, or the fuel. Figure 5 illustrates the characteristic cooling curve for this system. There are two distinct cooling steps:

(1) Filling the cask with water and cooling the fuel to about 240°F

(2) Slow flushing of the cask internals with cool water until the fuel and basket reaches 150°F or less.

Shipments of "well-aged" fuel are sufficiently cooled to allow regular use of a portable cooldown system. For truck casks, this system should be equally as safe and rapid as the permanent system employed at the BNFP. Cooldown of fuel which has aged for 3-4 years prior to shipping involves little more than filling the cask cavity with cool water.

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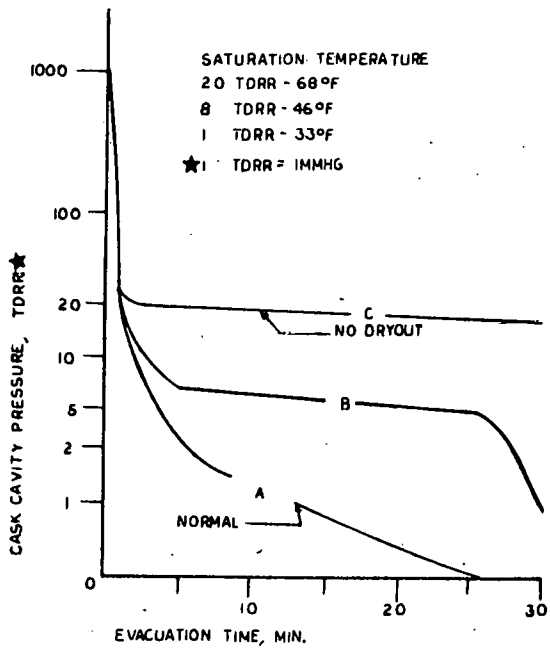


FIG. 1 - CHARACTERISTIC CASK DRYOUT CURVE

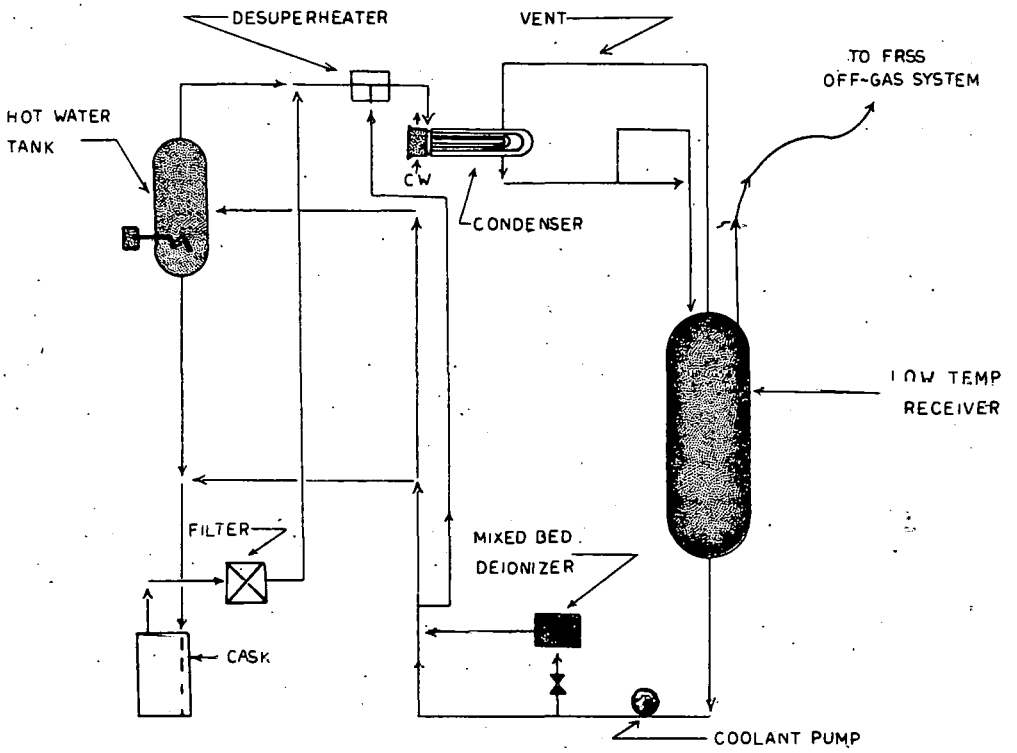
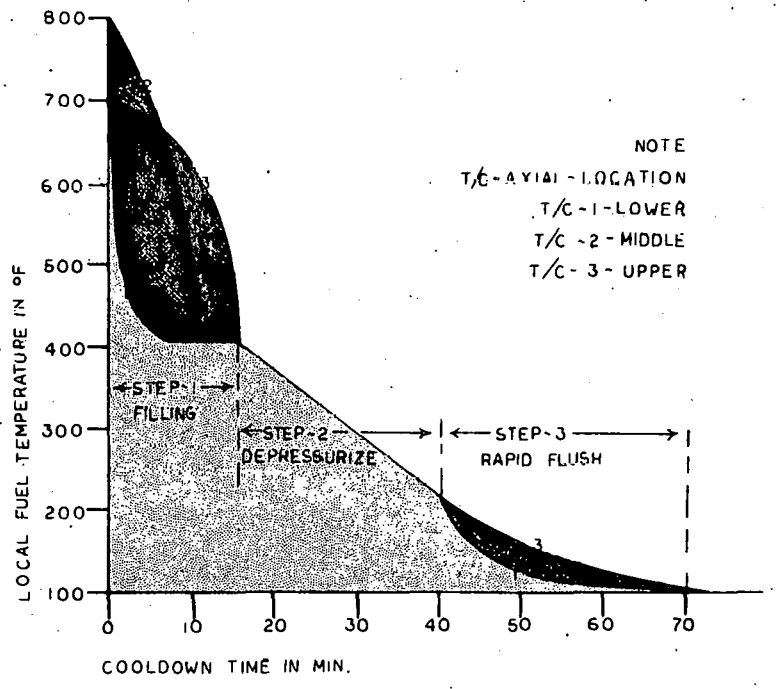


FIG. 2-BNFP COOLDOWN SYSTEM PROCESS FLOW DIAGRAM

FIG. 3—CHARACTERISTIC COOLDOWN CURVE BNFP COOLDOWN SYSTEM — 10.6KWT (NL 1/2 DESIGN RATING)



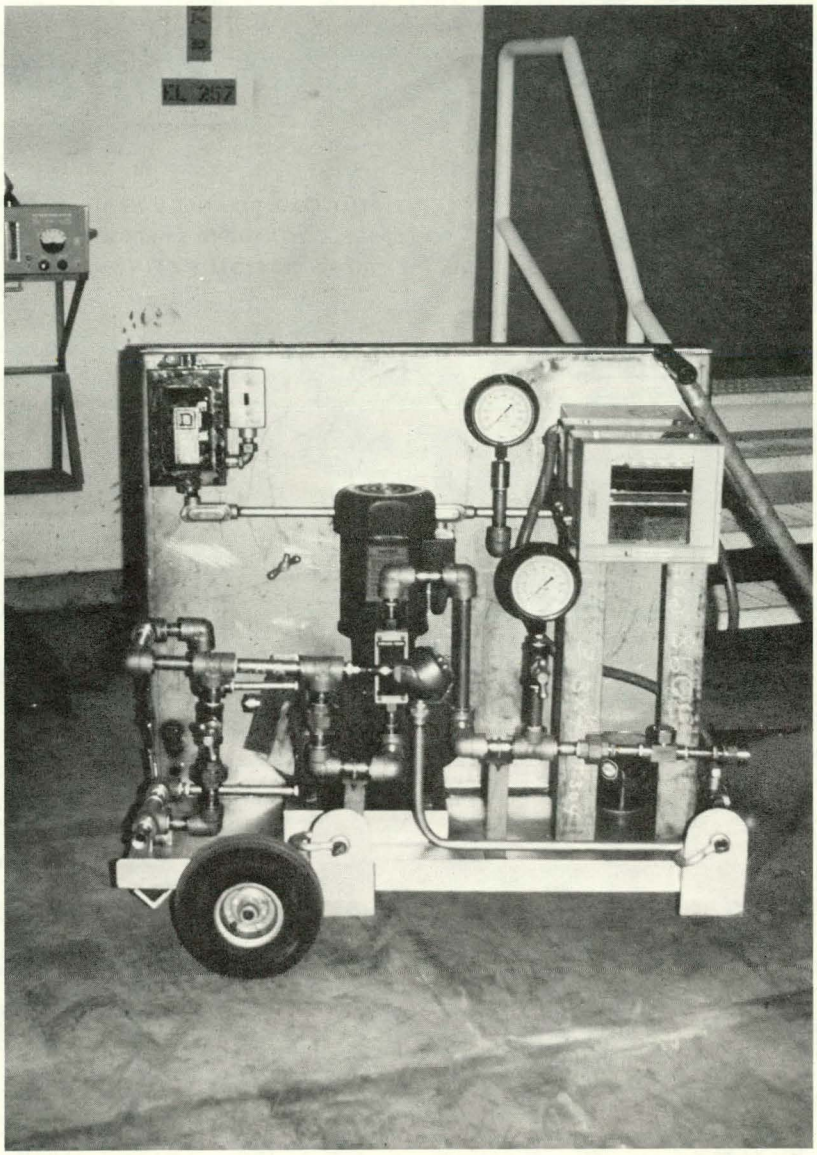


FIGURE 4 - PORTABLE COOLDOWN SYSTEM SKID

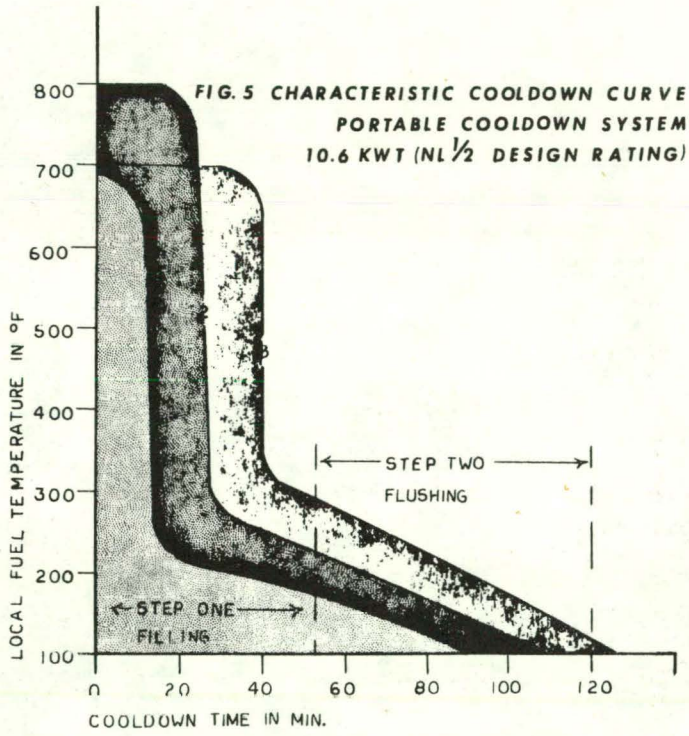


FIG. 5 CHARACTERISTIC COOLDOWN CURVE
PORTABLE COOLDOWN SYSTEM
10.6 KWT (NL ½ DESIGN RATING)

Dup

TEMPERATURE DISTRIBUTION IN SPENT FUEL SHIPPING BASKETS*

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ABSTRACT

An analytical method has been developed for estimating temperatures in solid metal shipping container inserts (baskets) containing irradiated fuel assemblies. The problem is formulated in two dimensions as steady-state conduction over the basket cross section. The fuel assemblies are treated as line sources of heat inside a simply connected region for a solid design or a doubly connected region for an annular design. The method of images is used to construct cylindrical harmonics around isolated singularities. These harmonics are solutions to Poisson's equation, and the boundary conditions are satisfied by linear combination. The fuel arrangement and the rates of heat generation of the fuel assemblies are left in general form. Two solutions are given for baskets containing up to N fuel assemblies. These solutions calculate the temperatures in the basket materials relative to the outside surfaces of the baskets. The solution for the annular design assumes an adiabatic inner surface. The solution for the solid design allows multiple rings of holes.

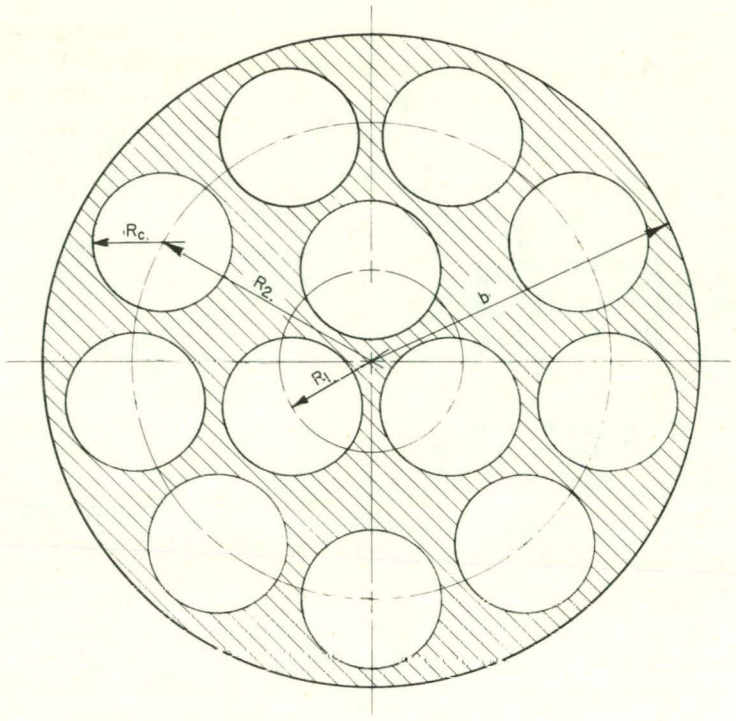
INTRODUCTION

A shipping cask may be considered to consist of two parts: (1) the cask body and (2) the basket insert. The cask body is a cylindrical, composite structure of lead, steel, and other materials that provide structural strength, radiation shielding, and protection from the environment. Items to be shipped are placed in the cavity of the central section of the body. The basket insert is a separate structure designed to fit into this cavity, with holes or framework appropriate to the items being shipped. Because the same cask may carry various types of spent fuel or other material, the basket design may vary.

The basket considered here consists of a solid cylindrical block of metal in which circular holes are bored or cast for receiving the fuel assemblies. Figures 1 and 2 illustrate typical designs. Spent fuel assemblies generate considerable decay heat, and an essential step in cask design and licensing is calculation of the maximum temperatures of the assembly during transport. As part of a heat transfer calculation, a method of estimating the temperature drop between the holes and the outer boundary of the basket is required. The analytical method solves the general conduction heat transfer equation for a solid cylinder with heat generating holes, and presents dimensionless curves for a rapid estimation of temperatures for two typical basket designs.

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* Work done under USDOE Contract No. AT(07-2)-1.



b = 19.25 inches = 48.895 cm

R_1 = 5.25 inches = 13.335 cm

R_2 = 13.75 inches = 34.925 cm

R_C = 4.00 inches = 10.160 cm

FIGURE 1. LMFBR Fuel Basket — Solid Aluminum Design

a = 12.00 inches = 30.480 cm

b = 22.23 inches = 56.464 cm

r_o = 17.11 inches = 43.459 cm

r_c = 3.99 inches = 10.135 cm

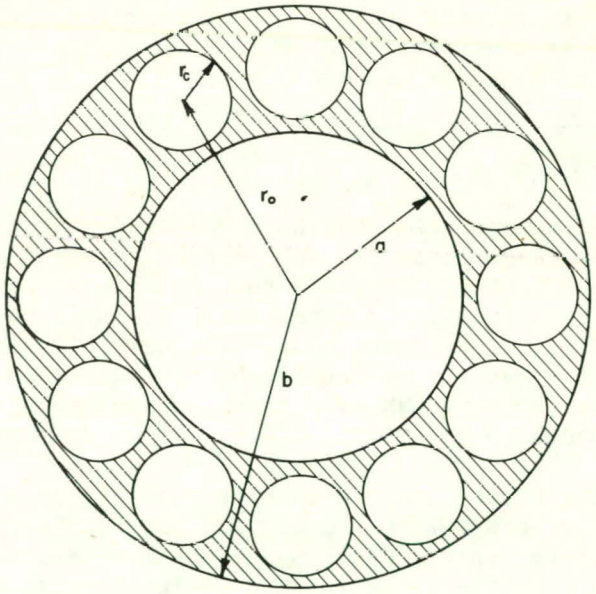


FIGURE 2. LMFBR Fuel Basket — Annular Steel Design

FORMULATION

The calculation is simplified by considering steady-state heat flow in the basket material, and by reducing the three-dimensional problem to a two-dimensional one by assuming that the basket and its fuel assemblies are infinite in length. Furthermore, the fuel assemblies are treated as concentrated line sources imbedded in the basket material. These simplifying assumptions overestimate the maximum temperature by neglecting the axial heat transfer by conduction in the solid and by convection in the cavity holes.

The cross sections of two typical breeder reactor basket designs are shown in Figures 1 and 2. The material regions for both designs are cylindrical in geometry; the solid design is simply connected and the annular design is doubly connected. The fuel assemblies and the cross section intersect at the cavity centers that will be treated as isolated singularities in the plane. In polar coordinates (r,θ), let (r_n,θ_n) for $n = 1, 2, \ldots, N$ be the locations of these sources. Then their power per unit length, Q_n, can be related to temperature gradient by integration along a closed contour S around (r_n,θ_n) as

$$Q_n = -k\int_S \frac{\partial T}{\partial a}\, dS \tag{1}$$

where k is the thermal conductivity, which is assumed to be independent of space and temperature T, and the integrand is the gradient of temperature in the normal direction a. Throughout the region, the steady-state heat flow in an isotropic material is goverened by

$$r\frac{\partial}{\partial r}\left(r\frac{\partial T}{\partial r}\right) + \frac{\partial^2 T}{\partial \theta^2} = 0 \tag{2}$$

Along the circular boundaries of the baskets, we can assume either that there is no heat loss,

$$\frac{\partial T}{\partial r} = 0, \qquad r = r_c \tag{3a}$$

or that the temperature is prescribed

$$T = T_c, \qquad r = r_c \tag{3b}$$

Use of condition (3a) implies complete insulation thus accentuating the temperature rise in the basket material. Only condition (3b) is applicable to the outer boundary of either case. Condition (3a) is applied to the inner boundary of the annular design.

METHOD OF SOLUTION

The method of solution is to construct a set of functions that are harmonic in the material region and singular at the heat source. It can be shown that[1]

$$T_n(r,\theta) = -\frac{Q_n}{4\pi k}\, \ell n\, [r^2 + r_n^{\,2} - 2rr_n \cos(\theta - \theta)] \tag{4}$$

satisfies equations (1) and (2). Based on T_n, the image method[2] is used to find its associated functions which remain harmonic and finite at (r_n,θ_n). For a simply connected region, like a solid basket design, images of the heat sources in the boundary $r = b$ are

$$r_n^* = b^2/r_n \qquad \text{in} \quad r = b \tag{5}$$

By replacing r_n of equation (4) with r_n^*, we obtain the associated image function,

$$T_n^*(r,\theta) = -\frac{Q_n}{4\pi k} \ln [r^2 + r_n^{*2} - 2rr_n^* \cos(\theta - \theta_n)] \tag{6}$$

which is harmonic and finite in the region including (r_n, θ_n). Condition (3a) or (3b) can be satisfied by linear combination of T_n and T_n^*:

$$T(r,\theta) = T_n(r,\theta) + \alpha T_n^*(r,\theta) + \beta \tag{7}$$

where α and β are arbitrary constants. For example, condition (3a) is met if $\alpha = -1$, $\beta = 2 \ln (b/r_n) + T_o$, for $T(b,\theta) = T_o$. As stated in the previous section, condition (3b) cannot be met by equation (7) unless there is a heat sink in the material region. Superposition of N number of heat sources in a solid basket design provides an analytical solution for the temperature rise relative to its surface,

$$T(r,\theta) = -\frac{1}{4\pi k} \sum_{n=1}^{N} Q_n \ln \frac{[r^2 + r_n^2 - 2rr_n \cos(\theta - \theta_n)]\, r_n^*}{[r^2 + r_n^{*2} - 2rr_n^* \cos(\theta - \theta_n)]\, r_n} \tag{8}$$

Application of the image method to a doubly connected region of the annular design requires successive projection of images in the two boundaries. Let $r_{n,o}$ be the radial location of the nth source in the annulus, and let

$$r_{n,1} = b^2/r_{n,o}, \qquad r_{n,o}^* = a^2/r_{n,o} \tag{9}$$

be its images in the circular boundaries r = b and r = a, respectively, then their second images projected in r = b and r = a are

$$r_{n,2} = b^2/r_{n,o}^* = r_{n,o}(b^2/a^2)$$

$$r_{n,1}^* = a^2/r_{n,1} = r_{n,o}(a^2/b^2)$$

Further projection of images provides the following indicial relationships for determination of all image locations:

$$r_{n,2m} = \varepsilon^{-m} r_{n,o}, \qquad r_{n,2m+1} = \varepsilon^{-m} r_{n,1} \tag{10}$$

$$r_{n,2m}^* = \varepsilon^{m+1} r_{n,1}, \qquad r_{n,2m+1}^* = \varepsilon^{m+1} r_{n,o}$$

for $m = 0, 1, 2, \ldots$, $\varepsilon = a^2/b^2$.

Corresponding to these image locations, the associated functions are

$$T_{n,2m}(r,\theta) = -\frac{1}{4\pi k} \ln [r^2 + \varepsilon^{-2m} r_{n,o}^2 - 2rr_{n,o} \varepsilon^{-m} \cos(\theta - \theta_n)] \tag{11}$$

$$T_{n,2m}^*(r,\theta) = -\frac{1}{4\pi k} \ln [r^2 + \varepsilon^{2m+2} r_{n,1}^2 - 2rr_{n,1} \varepsilon^{m+1} \cos(\theta - \theta_n)]$$

$$T_{n,m+1}(r,\theta) = -\frac{1}{4\pi k} \ln [r^2 + \varepsilon^{-2m} r_{n,1}^2 - 2rr_{n,1} \varepsilon^{-m} \cos(\theta - \theta_n)]$$

$$T_{n,2m+1}^*(r,\theta) = -\frac{1}{4\pi k} \ln [r^2 + \varepsilon^{2m+2} r_{n,o}^2 - 2rr_{n,o} \varepsilon^{m+1} \cos(\theta - \theta_n)]$$

Along any one of the boundaries, either one of the conditions (3) can be satisfied by taking linear combination of two functions from equation (11). For example, on r = b,

$$T(r,\theta) = T_{n,2m+2}(r,\theta) - T^*_{n,2m}(r,\theta) + \frac{1}{2\pi k} \ell n \frac{b^m r_o}{a^{m+1}}$$

satisfies conditions (3a), and

$$T(r,\theta) = T_{n,2m+2}(r,\theta) + T^*_{n,2m}(r,\theta) + \frac{1}{2\pi k} \ell n\ r$$

meets the requirement of condition (3b). The term $\ell n\ r$ in the last expression is singular at the origin. Since r = 0 is not in the doubly connected region under consideration, and $\ell n\ r$ satisfies equations (1) and (2), its inclusion is physically equivalent to having a heat sink at r = 0. Or mathematically, r = 0 is the image of $r_{n,o}$ in the inner circular boundary, when $r_{n,o}$ is reflected through both boundaries for an infinite number of times; i.e.,

$$-\frac{1}{2\pi k} \ell n\ r = \lim_{m\to\infty} T^*_{n,2m}(r,\theta)$$

For an annular design with an inner boundary completely insulated, the temperature rise of the basket material containing N heat sources can be shown to be

$$T(r,\theta) = -\frac{1}{4\pi k} \sum_{n=1}^{N} Q_n \sum_{m=o}^{\infty} (-1)^m \ell n\ [f_{n,m}(r,\theta)] \qquad (12)$$

where

$$f_{n,o}(r,\theta) = \frac{r^2}{b^2} \left[\frac{\left(1 - \frac{b^2}{r_o r}\right)^2 + 4\frac{b^2}{r_o r} \sin^2 \frac{1}{2}(\theta - \theta_n)}{\left(1 - \frac{r}{r_o}\right)^2 + 4\frac{r}{r_o} \sin^2 \frac{1}{2}(\theta - \theta_n)} \right]$$

$$f_{n,m}(r,\theta) = \left[\frac{\left(1 - \varepsilon^m \frac{b^2}{r_o r}\right)^2 + 4\varepsilon^m \frac{b^2}{r_o r} \sin^2 \frac{1}{2}(\theta - \theta_n)}{\left(1 - \varepsilon^m \frac{r}{r_o}\right)^2 + 4\varepsilon^m \frac{r}{r_o} \sin^2 \frac{1}{2}(\theta - \theta_n)} \right] \cdot \left[\frac{\left(1 - \varepsilon^m \frac{r_o r}{b^2}\right)^2 + 4\varepsilon^m \frac{r_o r}{b^2} \sin^2 \frac{1}{2}(\theta - \theta_n)}{\left(1 - \varepsilon^m \frac{r_o}{r}\right)^2 + 4\varepsilon^m \frac{r_o}{r} \sin^2 \frac{1}{2}(\theta - \theta_n)} \right]$$

for m = 1, 2, 3, ...

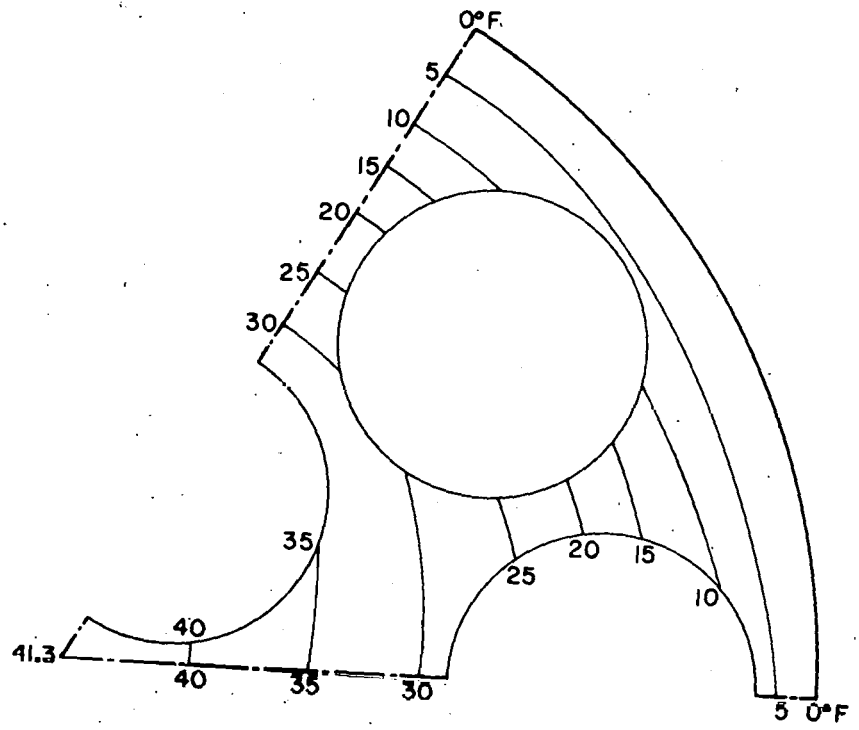


FIGURE 3. Heating of a Fully Loaded Solid Aluminum Basket (Q_n = 3750 Btu/ft-hr = 3.6 kw/m, n = 1 to 12)

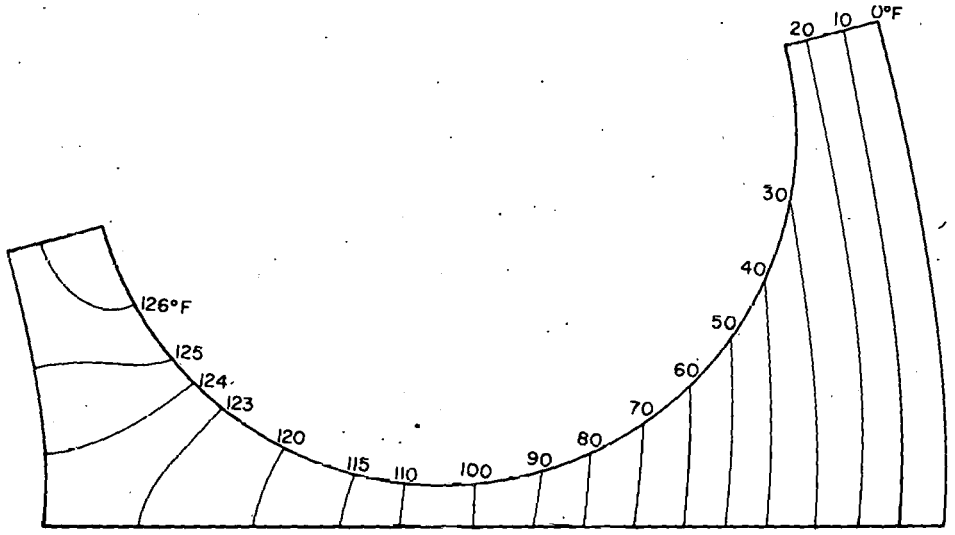


FIGURE 4. Heating of a Fully Loaded Annular Steel Basket (Q_n = 3750 Btu/ft-hr = 3.6 kw/m, n = 1 to 12)

RESULTS AND EXAMPLES

The temperature distributions for two basket designs as shown in Figures 1 and 2 are calculated from equations (8) and (12). Figures 3 and 4 show isotherms for conditions typical of projected breeder reactor fuel. For both designs, the temperature of the outside surfaces is set at 0°F, and each basket contains 12 fuel assemblies of equal power. However, the analytical expressions given by equations (8) and (12) do not require the sources to have equal powers. Aluminum is taken for the basket material of the solid design, and steel for the annular design.

Some generalization of these results is possible, enabling a designer to estimate the temperature gradients in baskets of various dimensions and loadings. Figures 5 and 6 show the effects of the size of the source holes and the number of sources on the maximum temperature rise, $T_{max} - T_o$, in the basket material, where T_o is the temperature of the outside basket surfaces. The ordinates are made dimensionless by multiplying by k/Q_n, where k = material conductivity and Q_n = source power per unit axial length. Equal power for each source is assumed. The size of the source hole is limited by the basket geometry. For the solid design (Figure 5), the centers of the holes are set at $r_n = 0.7b$, hence the maximum hole radius is 0.3b. For the annular design (Figure 6), the sources are located at mid-width of the basket, so that the largest possible hole radius is (b-a)/2. In either case, the abscissa is made dimensionless by dividing the hole radius r_s by the maximum radius. When the number of assemblies N increases, the size of the cavities is again limited due to circumferential interference, as indicated by the truncation of the curves as N increases.

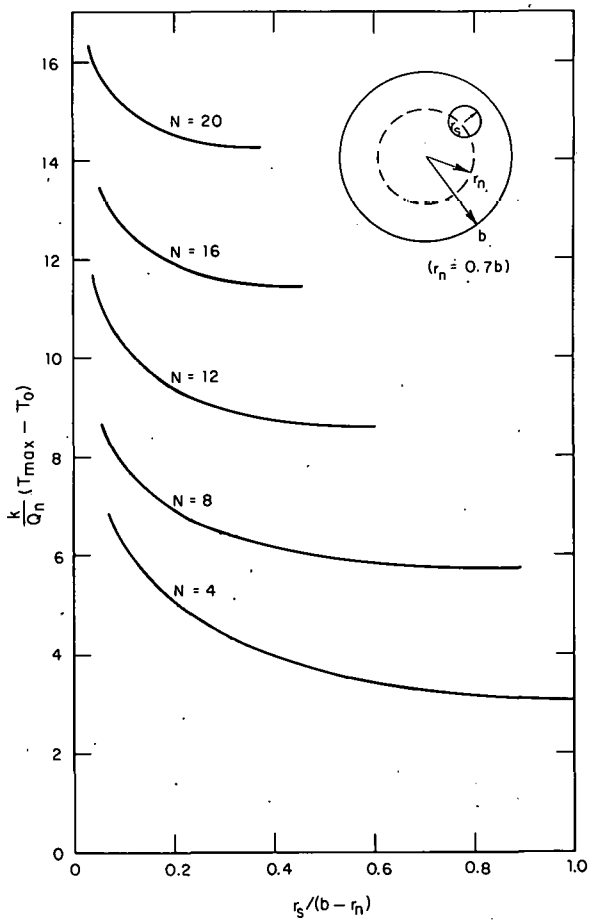


FIGURE 5. Maximum Temperature in Solid Baskets

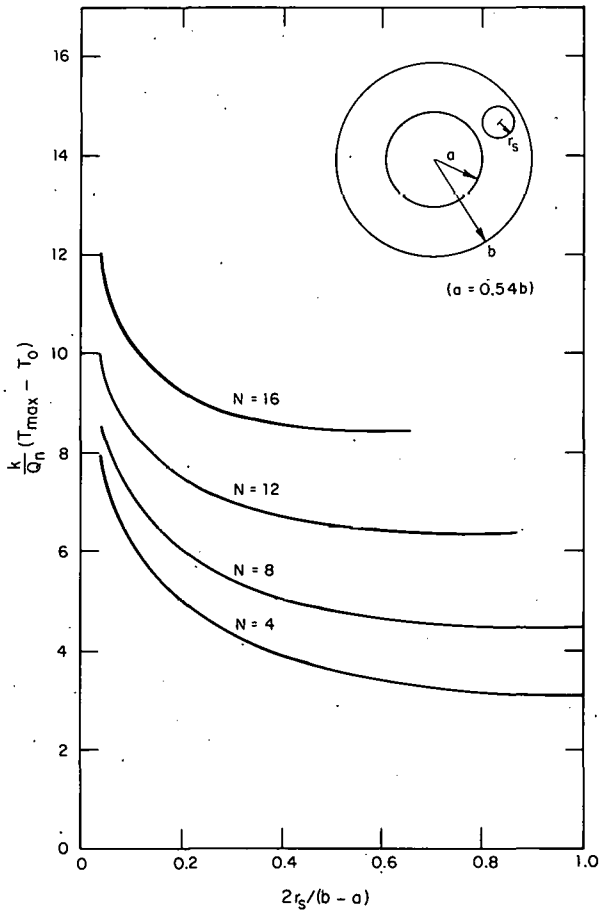


FIGURE 6. Maximum Temperature in Annular Baskets

The solid basket case is shown with only one row of holes, rather than two rows as in Figure 1. The solution for two rows of holes in a solid basket may be developed from Figure 5 by superposition.

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Dup

A PARAMETRIC EVALUATION OF HEAT TRANSFER AND COOLANT OPTIONS FOR BREEDER REACTOR SPENT FUEL SHIPPING CASKS DESIGNS*

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INTRODUCTION

Spent fuel shipping cask (SFSC) conceptual designs have been developed to accommodate spent fuel shipping requirements of both the Fast Flux Test Facility (FFTF) and future plants which may be part of the United States Breeder Reactor Program (BRP).[1]** The transfer of heat from breeder reactor spent fuel assemblies is more difficult than for LWR spent fuel because the volume producing the decay heat is much smaller and is confined by a duct tube. The duct tube is a hollow, hexagonal cylinder made of stainless steel, and is denoted herein as the hex or hex wrapper. The ability to transfer the decay heat from the fuel assemblies to the environment is a major controlling factor in the design of breeder reactor SFSC's. The transfer of decay heat significantly affects fuel basket design, choice of coolants, containment cavity pressure, canister design, plant interfaces, and ultimately the shipping capacity of the SFSC. Furthermore, significant improvements in heat transfer capability are only feasible within the containment cavity. This report provides details of the analysis of heat transfer which is only summarized in Reference 1.

The conceptual designs[1] comply with shipping system requirements of the LMFBR Spent Fuel Shipping System Design Description.[2] Additional considerations which affect the designs are: (1) the internal fuel basket is fixed relative to the cask containment cavity; (2) spent fuel must be adequately cooled while being loaded and unloaded in a one atmosphere argon environment at FFTF; (3) passive cooling of a cask during shipment is required; (4) passive cooling within the containment cavity during loading is desired; and, (5) the capability to handle and transport breeder reactor spent fuel cooled for six-months or less is desired for future breeder reactors.

All conceptual designs consist of an inner containment cavity with a fuel basket, and an outer cask structure. The outer cask structure is composed of an inner containment vessel including the closure, a gamma-shield, a neutron shield, an outer containment vessel, and heat transfer fins.

The following summarizes information from which decisions can be made relative to the different conceptual designs, ultimately allowing the selection of a single design concept for detailed design efforts.

CASK CONFIGURATION

Figure 1 shows the cask design concept considered in this study, which uses a boron carbide (B_4C) neutron shield and a solid basket design (for details see Reference 1). The basket is fabricated from aluminum and clad with stainless steel. The aluminum is protected from exposure to sodium or other caustic materials by the stainless steel cladding and the stainless steel basket inserts. A fuel shipping canister has been designed[1,3] which would be fabricated from stainless

*This work supported under contract by the United States Department of Energy.
**Superscripted numbers denote references.

steel and would use a metal seal for positive closure. The canister is designed to slip into the basket inserts. Details of the design are given in References 1 and 3.

HEAT TRANSFER COEFFICIENT MODELING

Since the conduction process within solid materials is easily characterized, the key to determining the thermal performance of the SFSC's is to realistically model the "pin-to-pin-to-hex wrapper" heat transfer within a fuel assembly which is a strongly coupled radiation, conduction and convection problem that is most easily characterized empirically. A full scale test device was used at Oak Ridge National Laboratory (ORNL) to evaluate the thermal behavior of a fuel assembly mock-up, concentrically located in a pipe[4]. The individual fuel pins were heated electrically using resistive heaters. Tests were performed with argon, helium, sodium and an organic as coolants, as well as with a vacuum. Data for the gases were obtained with gas pressures of 0.5 to 8.0 atmospheres. These data have been used to develop empirical heat transfer coefficients between the hottest pin and the hex wrapper.

Using the ORNL data, it was estimated that with helium and argon coolants only 2 to 3 percent of the heat generated in the core zone of the assembly will be conducted axially away from the zone in the assembly structure (pins and hex wrapper). Equations (1) and (2) were developed for the temperature rise from the hex-wrapper to the hottest pin. No attempt was made to present these data in heat transfer coefficient form. If, in future reactor designs, the dimensions of the fuel assemblies change significantly from those tested, new data and correlations would be required.

For gaseous coolants, a correlation of the ORNL data for the temperature rise from the hex wrapper to the hottest fuel pin gives

$$T_{PIN,MAX} - \overline{T}_{HEX} = 50\ M^{.31}\ Q^{.63} \qquad \text{(Eq. 1)}$$

where M is the molecular weight of the gaseous coolant, Q is the decay heat output of the assembly in kW/m of heated length, and T is temperature in °K. This relationship correlates the data within about ± 20 percent.

For a vacuum in the assembly, the temperature rise is given by

$$T_{PIN,MAX} - \overline{T}_{HEX} = 213\ Q^{.27} \qquad \text{(Eq. 2)}$$

This relationship correlates the data within about ± 15 percent.

The ORNL data from Reference 4 were also used to develop heat transfer coefficients for the annulus between the hex wrapper and the containing cylinder (i.e., the basket insert or the canister). These tests were performed with the mock fuel assembly coaxially located inside an 0.254 m (10 inch) ID stainless-steel pipe. The pipe was allowed to convect naturally and radiate to the surrounding environment. Because of natural convection external to the pipe, significant circumferential temperature variations were frequently observed on the pipe surface -- a condition different from that which will exist in the cask. The effect that other pipe boundary conditions will have on the net heat transfer coefficient is not well defined, although preliminary calculations using a finite-element code for natural convection[5] indicate the effect may be small for gases and low Prandtl number liquids such as sodium.

The measured axial variations of pipe temperature[4] were used to estimate the amount of heat carried axially away from the heated zone by convection within the gas

filled annulus. These calculations were performed by inferring the heat transfer from the external surface of the containing pipe to the surrounding environment using standard convective and radiative heat transfer relationships and the measured pipe temperatures. From these calculations, it was estimated that 30 to 50 percent of the decay heat is convected axially from the heated zone, with the remaining heat being transferred radially by radiation and convection to the containing pipe. Therefore, in developing Equation (3) it was conservatively assumed that 30 percent of the decay heat is axially transferred from the heated zone in the annulus.

To provide the heat transfer coefficient in Nusselt number/Grashof number form, the correlation suggested by Itoh, et al.,[6] was used. It was assumed here that the same characteristic dimensions would apply for a right hexagonal cylinder coaxially located within a right circular cylinder, neither of which was isothermal. Average temperatures were determined for the hex wrapper and the enclosing pipe.

The results of correlating the Reference 4 data in the above-described manner, for the heat transfer from the hex to the basket insert (or canister) with argon and helium coolant are shown in Figure 2, and is given by Equation (3):

$$Nu = 0.43\ (Gr_M)^{.25}\ , \qquad \text{(Eq. 3)}$$

where

$$Nu = h_p[r_p\ \ln(r_p/r_h)]/k\ , \qquad \text{(Eq. 4)}$$

$$Gr_M = (g\beta/\nu^2)_{p=1}(p^2)\,\Delta T[(r_p r_h)^{0.5}\ \ln(r_p/r_h)]^3, \qquad \text{(Eq. 5)}$$

and

$$\Delta T = \overline{T}_{HEX} - \overline{T}_{PIPE}\ , \qquad \text{(Eq. 6)}$$

In these equations, $g\beta/\nu^2_{p=1}$ is the usual buoyancy term, evaluated at one atmosphere pressure, p is the pressure in atmospheres, r_p and r_h are the radii of the pipe and the hex respectively, h_p is the convective coefficient evaluated at the pipe, and k is the thermal conductivity of the gas -- all with appropriate units to render Nu and Gr_M dimensionless. Equation (3) was fit through the lower Nusselt number data to provide a lower estimate of the Nusselt number. Thus the equation overestimates the Nusselt number by no more than 30 percent. Because the data also included the effects of radiative transfer, these effects are included in Equation (3). Previous studies showed that convection is the predominant mode of heat transfer in the annulus[5].

The transfer of heat across gas filled gaps in the cask structure such as between the fuel basket and the containment cavity is modeled using standard heat transfer relationships. The rejection of heat from the external surface of the cask to the environment is a combined convection and radiation process, which must include the effects of solar input. The methods which were used to model this process are outlined in detail in References 7 and 8.

ANALYTICAL RESULTS

The thermal behavior of the different conceptual designs has been predicted, and different basket and coolant alternatives were evaluated for normal loading and shipping conditions. In all cases, an axial peaking factor (peak heat flux divided by average heat flux over the heated length) of 1.2 has been assumed for each

fuel assembly. The maximum allowable pin temperature during normal handling and shipping is 922°K (1200°F). In addition, a temperature of 333°K (140°F) is required for all handling surfaces during loading[2]. Therefore, it has been assumed that forced cooling on the external surface of the cask will be used during loading to maintain the external surfaces at the required 333 °K.

During loading at the reactor site and unloading at the receiving site, the cask will be oriented in the vertical position. Although the heat transfer correlations (Equations (1) through (6)) were developed from data obtained for the horizontal (or near-horizontal) orientation; these relations were used to estimate thermal behavior during loading with the cask and assemblies oriented vertically. The effect of orientation on the heat transfer during loading is being experimentally investigated at Sandia Laboratories.

The base design case involves the loading of nine assemblies in the cask in a one atmosphere argon environment. Results of calculations for this case are shown in Figure 3. These results indicate that the major thermal resistance is in the regions modeled by the empirically derived equations discussed earlier; i. e., the annulus between the hex and the basket insert, and inside the fuel assembly. For the maximum allowable pin temperatures specified, approximately 1.1 kW per assembly of decay heat output can be accommodated. To facilitate the loading of assemblies producing 1.4 kW of decay heat (which is the FFTF requirement), a maximum pin temperature of 1067°K (1460°F) would result if stagnant argon is used as the coolant.

These results indicate that alternate methods must be developed for loading the cask if assemblies producing more than about 1 kW of decay heat are to be shipped. A number of methods have been evaluated, and are discussed in detail in Reference 7. These include using stagnant helium rather than argon as the cover gas, using a liquid in the cask containment cavity, using a metallic "corset" or clamshell device to fill the gap between the hex wrapper and the annulus with a thermal conductor, minimizing the gap between the assembly and the solid basket by using a hexagonal basket opening, filling the gap with powdered copper or copper spheres, canistering the assemblies in helium, and canistering the assemblies in a liquid. These options will be addressed later.

During normal transport, it is assumed that the containment cavity is pressurized with helium. With an average solar load and an ambient temperature of 328°K (130°F) the temperatures within the cask were predicted for different helium pressure and assembly decay heat outputs. The maximum decay heats of assemblies which can be shipped, for different helium pressures and specified maximum pin temperatures, are shown in Table I for the base design case of nine bare assemblies. For a maximum pin temperature of 811°K (1000°F), thereby allowing some margin, assemblies producing 2.5 kW can be transported with 5 atmospheres of helium coolant, and for a maximum of 922°K (1200°F) the allowable decay heat increases to about 3.3 kW. similar to the loading situation, the major thermal resistance is from the fuel basket inward to the fuel assembly and the individual fuel pins.

Table I. MAXIMUM DECAY HEAT CAPABILITY FOR BASE DESIGN CASK.

Maximum Pin Temperature		Containment Cavity Pressure (Atmospheres)				
		1.0	2.0	3.0	4.0	5.0
(°K)	(°F)					
700	800	1.2(kW)	1.4	1.6	1.7	1.8
756	900	1.5	1.7	1.9	2.1	2.2
811	1000	1.7	2.0	2.3	2.4	2.5
867	1100	2.0	2.3	2.6	2.8	2.9
922	1200	2.2	2.6	2.9	3.1	3.3

Many of the needs of both FFTF and future breeder reactors may be met with the base design cask if the assemblies were loaded, either in stagnant helium, with forced argon cooling the assemblies directly, or with a liquid in the cask cavity during loading which would be replaced with helium for shipment. However, currently designed facilities at FFTF, and conceptual designs of the Clinch River Breeder Reactor, are not amenable to these loading methods.

Following an evaluation of the different loading methods, it is recommended that spent fuel assemblies be canistered in helium at relatively high pressures of 5 to 8 atmospheres. A major thermal barrier during loading would then be the argon filled annulus between the canister and the basket insert. To minimize the effect of this gap, the exterior surface of the canister and the inner surface of the basket insert should be maintained clean and should be treated to have high surface emittances of approximately 0.9.

Estimated temperatures are shown in Table II for different loading and shipping options for a cask containing nine assemblies. For the options using canisters, the pin temperatures can be maintained below the required 922°K (1200°F) by maintaining high surface emittances. If surface emittances are not controlled or the surfaces become contaminated with sodium, emittances as low as 0.2 may result[7]. In this situation, it is predicted that the maximum pin temperature during loading of nine assemblies would significantly exceed the required 922°K (1200°F) for assemblies producing 2.5 kW decay heat.

The results shown in Table II also indicate that, if the fuel assemblies are canistered in sodium, temperatures can be maintained at relatively low levels. If sodium filled canisters are used, twelve assemblies could be shipped with decay heat outputs well in excess of 2.5 kW. Further calculations indicate that twelve fuel assemblies, each producing 1.4 kW of decay heat could be loaded and shipped if they are canistered in helium prior to loading.

The space between the canister and the basket insert is small, and the heat transfer therein is dominated by conduction, not convection. Therefore, the cask cavity need not be pressurized to superatmospheric conditions during shipment to attain adequate decay heat rejection capability when the assemblies are placed in canisters.

Table II. PREDICTED TEMPERATURES DURING LOADING AND SHIPPING FOR NINE FUEL ASSEMBLIES.

CONFIGURATION	CANISTER/ BASKET EMITTANCE	ASSEMBLY DECAY HEAT OUTPUT (kW)	MAXIMUM FUEL PIN TEMPERATURE (°F)	(°K)
BARE ASSEMBLIES				
Load in 1.0 atm. argon	N/A	1.4	1457	1065
Ship in 4.0 atm. helium	N/A	1.4	700	644
ASSEMBLIES CANISTERED IN 5.0 atm. HELIUM				
Load in 1.0 atm. argon	0.2	2.5	1556	1120
	0.9	2.5	1124	880
Ship in 1.0 atm. helium	0.9	2.5	1135	886
ASSEMBLIES CANISTERED IN SODIUM				
Load in 1.0 atm. argon	0.9	2.5	393	474
Ship in 1.0 atm. helium	0.9	2.5	399	477

All base design case calculations and the alternate loading method calculations described above were performed assuming that no gaps existed between concentric layers of fabricated structure. In reality, gaps will exist, the layers will be eccentric and the gap dimensions will vary with temperature. A conservative estimate of the gaps that will exist in the baseline cask are:

Fuel Basket/Inner Shell -- 0.76 to 2.29 mm (0.030 to 0.090 inches);
Inner Shell/Gamma shield -- 0.76 mm (0.030 inches);
Gamma Shield/Neutron Shield -- 1.27 mm (0.050 inches); and
Neutron Shield/Outer Shell -- 1.27 mm (0.050 inches).

The effects of gaps in the structure, of eccentricity in the gaps and of thermal expansion were studied using a two-dimensional finite difference model. It was found that the average surface temperature of the fuel basket decreases with gap eccentricity. Results of the finite-difference analysis indicate that the hottest point in the fuel basket is lower for eccentric conditions than for concentric conditions, and the estimates of temperatures will always be conservative if based upon a concentric model.

When the effects of gaps in the structure, including those of thermal expansion are considered, the cask thermal performance may be reduced by approximately 20 to 25 percent relative to the values discussed above. This demonstrates the need to design and fabricate with care to minimize the gap dimensions.

A significant thermal resistance exists during loading in the gap between the basket and the containment cavity when the gap is filled with argon which has a low thermal conductivity. Therefore, the gap width should either be minimized, or the basket should be sealed at the top to the containment cavity, and helium -- which has a much higher thermal conductivity than argon -- should be maintained in the gap during both loading and shipping operations. Considerations of this type for the gap between other stuctural layers may also be of value in enhancing heat transfer.

CONCLUSIONS

In summary, the analysis of heat transfer in shipping casks for spent breeder reactor fuel shows that:

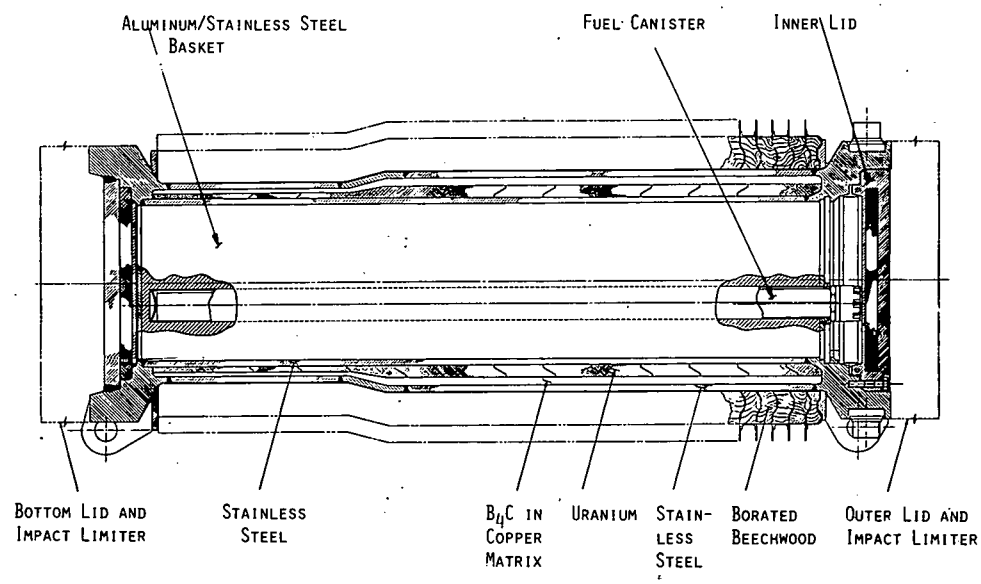
(1) Loading bare assemblies in an ambient argon atmosphere limits the cask capability to about 1 kW per assembly;
(2) Uncanistered assemblies, each producing 2 to 3 kW of decay heat could be shipped in high pressure helium if alternate loading methods can be developed;
(3) Assemblies each producing 2.5 kW of decay heat can be both loaded and shipped if they are canistered in high pressure helium;
(4) Assemblies producing decay heats of 4 kW or more can be both loaded and shipped if they are canistered in a liquid such as sodium; and
(5) Gaps in the cask structure resulting from design and fabrication can decrease the cask capability by as much as 20 to 25 percent.

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Fig. 1
Breeder Reactor Spent Fuel Shipping Cask



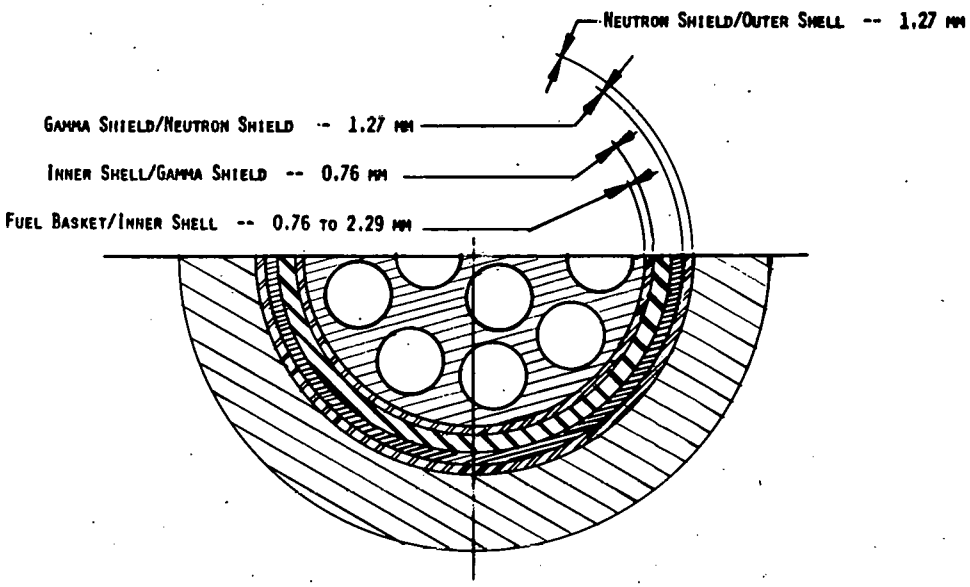


Fig. 2
Nusselt Number Correlation for Hex Wrapper to Pipe Heat Transfer ORNL Data

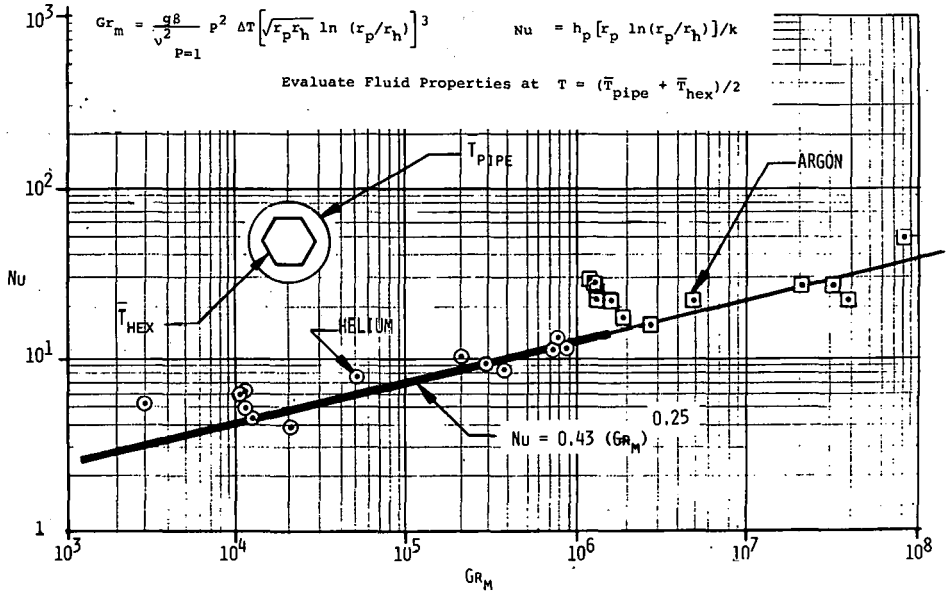
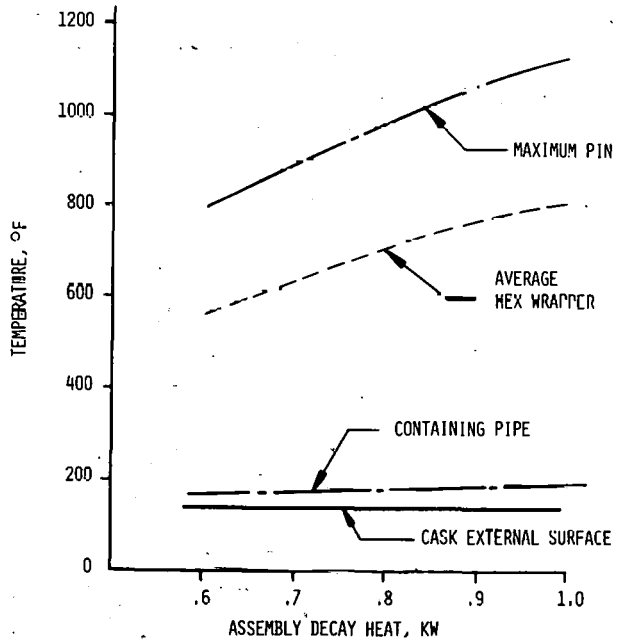


Fig. 3
Estimated Thermal Behavior of Cask with 1 Atmosphere Argon Coolant, Horizontal Orientation



INVESTIGATION INTO THE COMPARATIVE SEVERITY OF PRACTICAL THERMAL TEST PROCEDURES

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INTRODUCTION

In the Advisory Material for the Application of the IAEA Transport Regulations [1], three methods for determining practically whether a Type B radioactive material packaging can withstand the hypothetical fire conditions prescribed in the Transport Regulations [2] are presented. These comprise an open fire test and two tests involving exposure of the packaging in a furnace: the first of which (Method I) is based on using a materials testing type of furnace and the second (Method II), the use of a heat-treatment type. Under Method I, the packaging is heated for 35 minutes, from ambient temperature, by firing the furnace so that its temperature increases in accordance with the "fire-curve" specified in a certain ISO Standard [3]; under Method II the packaging is to be heated for 30 minutes in a furnace maintained continuously at 800°C.

Consideration of these Methods gave rise to two questions concerning them, namely:

(1) are they of comparable severity i.e. would a package gain the same amount of heat from the one test as the other? and

(2) on which temperature should control of the furnace firing in Method I be based? Should it be the average of temperatures measured near the surface of the object being tested, as specified in the ISO Standard [3], or the average of the temperatures of the radiating surfaces, the furnace walls, ceiling and floor, since it is believed that most of the heat transfer occurs by radiation?

The ISO Standard referenced in Method I pertains to the testing of building materials, to determine their flammability and related characteristics; the test is controlled with respect to the proximate temperature to make it independent of the furnace used. This is no doubt quite appropriate for the testing of materials but is it equally applicable in the case of packaging?

To answer these questions, a set of tests using two packagings and two furnaces corresponding to the types contemplated were conducted following Method I and Method II.

FURNACES

The Method I tests were carried out using a materials testing furnace operated by the National Research Council of Canada (NRC). It measures nominally 15 ft long by 12 ft wide and 8 ft high. It is equipped with 12 gas firing burners, 6 along each side, at a height of 4 ft or so. The walls are fire brick covered with a 3/4 inch thick blanket of Fiberfrax®, a very low

thermal conductivity material. The floor consists of firebrick only and the ceiling of concrete.

The Method II tests were conducted in a heat-treatment furnace belonging to Eldorado Nuclear Limited (ENL). It is approximately 12 ft long by 6 ft wide and 4-1/2 ft high at the sides, arching to about 5-1/2 ft at the centre. It is also gas fired with 3 burners along each side. The walls, ceiling and floor are fire brick but the floor was lined with 1/4 inch steel to distribute loads.

SPECIMENS

Figures 1 and 2 show cross sections of the two specimens which were subjected to the tests.

The first (Fig. 1) is basically a cylindrical, carbon steel casing with lead shielding which is bonded to the shell. It is insulated at its top and bottom and is fitted with a fire-shield around its circumference. It is a packaging currently in commercial service and weighs approximately 3450 kg (7600 lb). The specimen had previously been subjected to the prescribed drop tests and thermal tests, consequently it was somewhat damaged - the top cap was severely crushed and the surfaces were well oxidized. The results of a thermal test of a similar specimen without a fire-shield were reported [4] in 1971.

The second specimen (Fig. 2) is also cylindrical in shape and lead-containing; it was constructed from two concentric pieces of carbon steel pipe with well oxidized surfaces and weighs 204 kg (450 lb). The ends were closed by steel plate and covered with Transite® insulation. While this geometry was used primarily to facilitate analysis, this specimen was also intended to simulate a smaller packaging such as a radiography device.

INSTRUMENTATION

All thermocouples, except as noted below, were made from 24 gauge chromel-alumel wire encased in fibreglass insulation rated to 982°C (1800°F). Tips were twisted and silver soldered

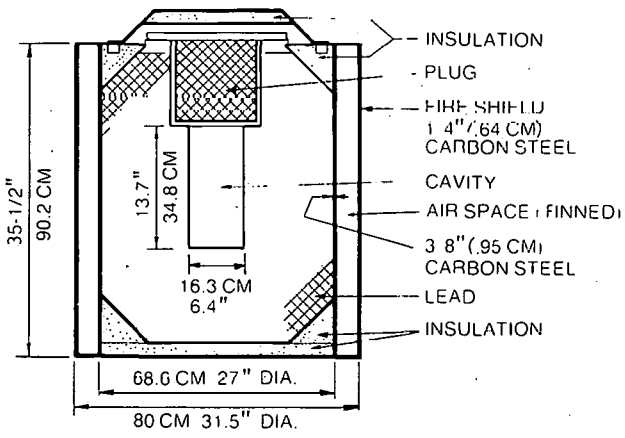


FIGURE 1 LARGE FLASK WEIGHT: 7600 LBS.

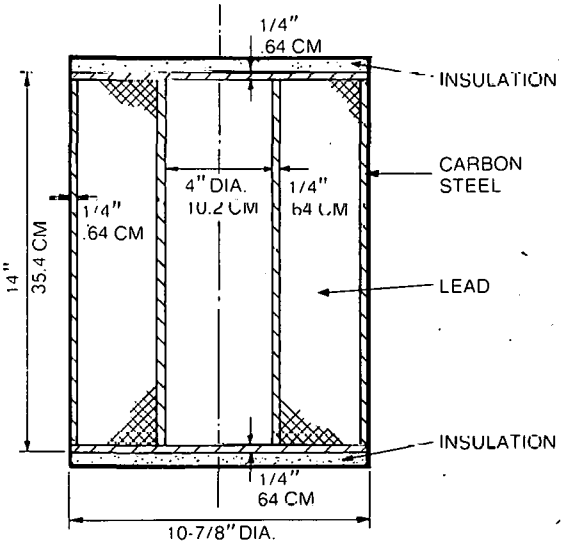


FIGURE 2 SMALL CYLINDER

and good contact was maintained throughout the tests. Care was taken to avoid creating second hot junctions. For measuring internal temperatures of the large flask, wire with insulation rated to 510°C (950°F) was used so that the 8 wires would pass through a hole in the modified c ure plug. Tails, with the higher rating, were connected to run through the furnaces to the r rders; the connections were located inside the thermal shield on top of the flask. The furnace thermocouples for Method I were of 20 gauge wire with welded junctions.

The small cylinder was equipped with 6 thermocouples located on opposite sides at mid-height on the outer surface of the specimen, midway in the lead and on the inner surface.

The large flask was equipped with 25 thermocouples located as follows: on the fire-shield and outer surface of the flask, two sets at 90° to each other at one quarter height, mid height and at three quarter height; on the inner cavity surface, at the top (bottom of the closure plug) and bottom centre points and on opposite sides at one quarter height, mid height and three quarter height. The other five thermocouples were located on the central axis: on the top, at the outer and inner surface of the insulating cap and on the top of the closure plug; on the bottom, at the outer surface and in contact with the steel shell inside the Transite®.

For the Method I test 23 thermocouples were used to monitor temperatures throughout the furnace as follows: 6 measured the surface temperatures at the centre point of each of the walls, the ceiling and floor; 8 provided furnace control, as called for in the ISO Standard, measuring the "ISO temperatures" about 10 cm (4 ins) away from the specimen at one third (approx.) and two third heights on each of the four quadrants as shown on figures 3 and 4; and 9 thermocouples (shielded) as normally used for furnace control using the ASTM Method [5]. These were located in the pipes protruding from the columns shown in the photographs in figure 3 and 4.

For the Method II test 5 bare thermocouples were centrally located and cemented to all furnace surfaces except the door. The furnace temperature was controlled from a shielded thermocouple protruding from the ceiling of the furnace.

TEST PROCEDURES

Method I

The small specimen was placed in the centre of the furnace on a cement block pedestal as shown in Figure 3. Its initial temperature was 7°C (45°F). The furnace was fired for 35 minutes following the prescribed temperature-time curve. After the burners had been shut off the furnace was opened to permit it and the specimen to cool. The temperatures of the cylinder were monitored until the internal temperatures began to fall.

The same procedure was followed for the large flask. Figure 4 shows the large flask in position in the furnace during cool down. Its initial temperature was 20°C (68°F). The ISO thermocouples were twisted around the columns holding the ASTM thermocouples so that the tips were located about 10 cm (4 inches) from the flask.

Method II

For the first test, in which the small cylinder was exposed, to compensate for the unknown response of the furnace during the time the door would be open (to permit insertion of the specimen) the furnace was preheated to 1600°F (870°C), 40°C above the preheat temperature recommended and allowed to soak for approximately 2 hours. This preheat temperature was found to be greater than required so the preheat temperature for the second test was set at 1550°F (843°C). During each test the firing was controlled manually to maintain 1475°F (800°C). Figures 5 and 6 show the small cylinder and the flask in the furnace. The initial temperature of the small cylinder was 22°C (73°F) and the large flask, 21°C (71°F). The start time and end time were taken when the door was at mid-height on closing and reopening respectively. The loading-unloading procedure, from the time the door was opened until reclosed, took less than 1-1/2 minutes. The total length of time the specimens were in the furnace amounted to no more than 31-1/2 minutes.

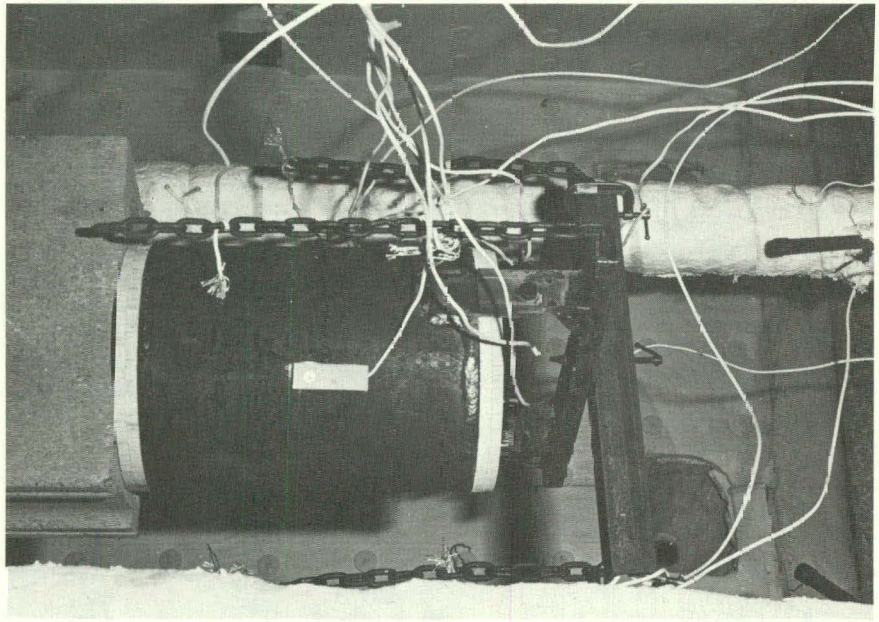


FIGURE 3 Small Specimen - Method I Furnace

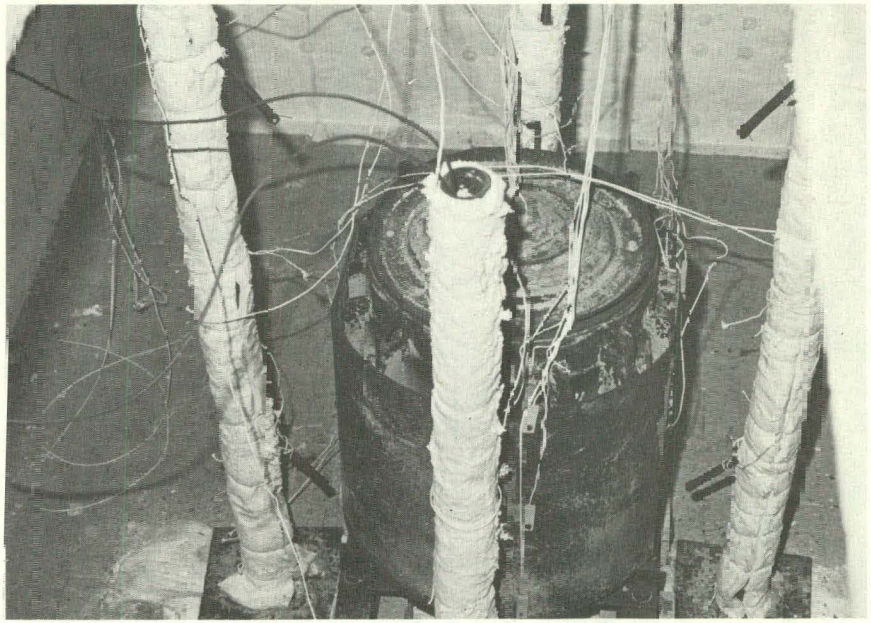


FIGURE 4 Large Flask - Method I Furnace

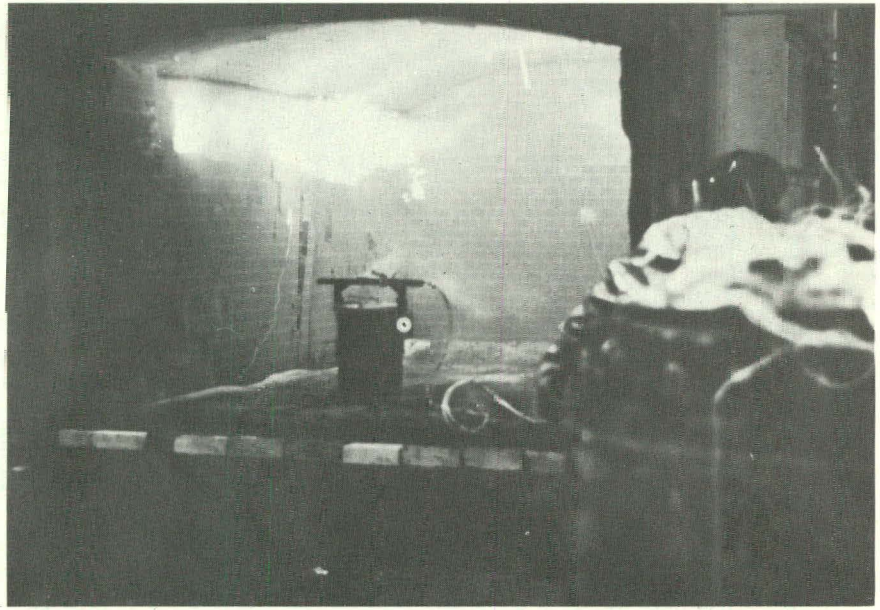


FIGURE 5 Small Flask - Method II Furnace

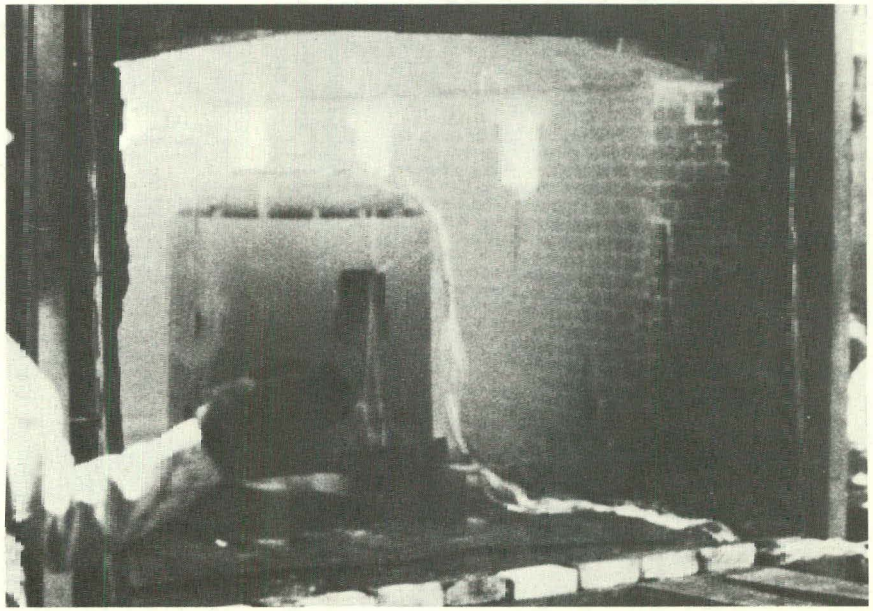


FIGURE 6 Large Flask - Method II Furnace

TEST RESULTS AND DISCUSSION

The temperature data collected was reduced and is presented in graphical form. Arithmetic a ıges of thermocouple readings were used for the furnace "wall" and "ISO temperature"; for t ;mall cylinder (external, midpoint and internal temperature) and for the large flask (fire-shield, external and internal temperature). The time-temperature histories in the Method I and Method II tests are shown in figures 7 and 8 respectively for the small cylinder and figures 9 and 10 for the large flask.

Note that the furnace "wall" temperature in Method I is significantly below the "ISO temperature".

Information derived from analysis of the above data are given in Table 1. The total heat input was calculated for the small cylinder from the observed temperature change and heat capacity of the material using the standard relationship.

$$Q = MC_p\Delta T \quad \text{Btu} \qquad \text{equation 1}$$

where M = mass of material - lb.
C_p = heat capacity - Btu/lb.°F
ΔT = temperature change - °F

It is assumed the mid-height temperatures are applicable over the entire length because the insulated ends effectively limit heat exchange to the radial direction. A correction would not affect relative values. A similar calculation has not been made for the large flask due to the complex geometry and unknown temperature distribution within the body.

The radiative heat input, Q , is obtained by numerically integrating the radiative heat flux during the test interval i.e.

$$Q = \sum_{i=1}^{n} \sigma(T_1^4 - T_2^4)_i.(\Delta t)_i \ \text{Btu/ft}^2 \qquad \text{equation 2}$$

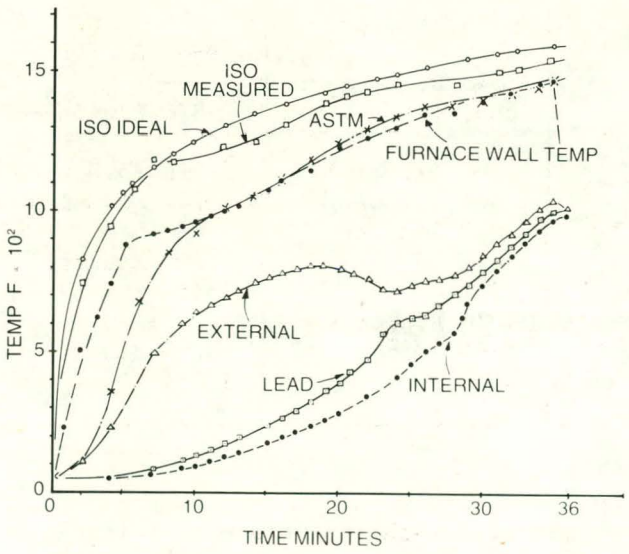


FIGURE 7 — SMALL CYLINDER METHOD I

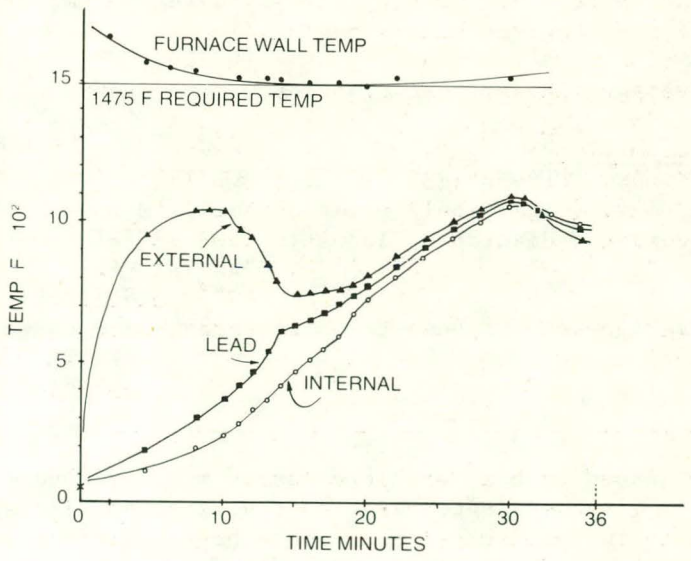


FIGURE 8 SMALL CYLINDER METHOD II

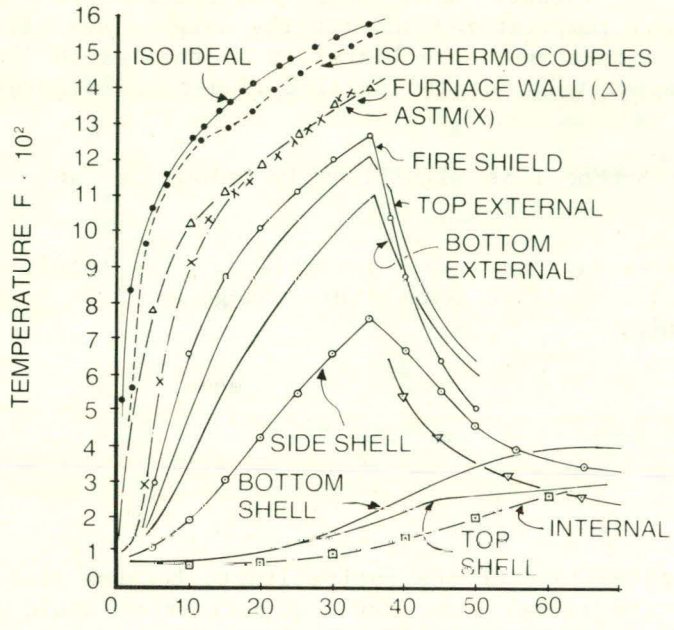


FIGURE 9 LARGE FLASK METHOD I

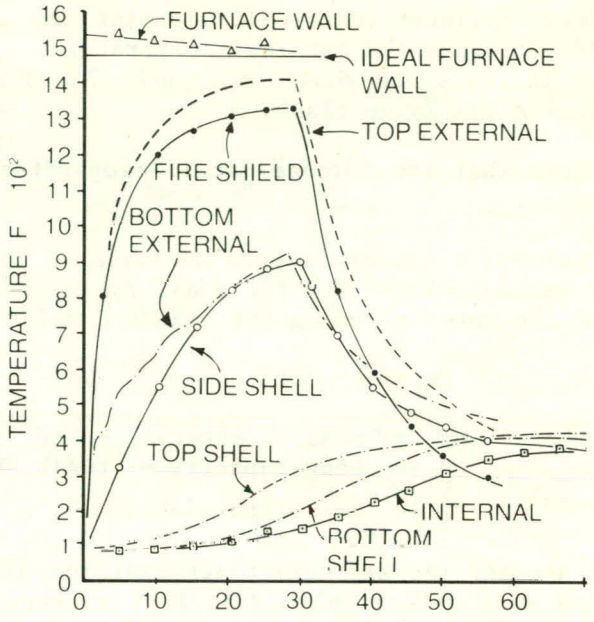


FIGURE 10 LARGE FLASK METHOD II

PARAMETER	METHOD I	METHOD II
Small cylinder:		
ΔT max. (outer shell - inner shell) °F	570 (17 min)	860 (7 min)
Total heat inputs: (ΔT and heat capacity)	22100 Btu	23000 Btu
Total heat input (ISO → surface only)	20900 "	---
" " " ("Wall" → surface only)	14400 "	27000 "
Estimated heat input (partially developed model)	24000 "	---
Average radiant heat input*: ISO	6968 Btu/ft^2	---
" " " " "Walls"	4020 "	9938 Btu/ft^2
Average radiant heat transfer rate*: ISO	11945 Btu/ft^2-hr	---
" " " " " : "Walls"	6891 "	18876 Btu/ft^2-hr
Large flask:		
ΔT max. (Fire-shield, - outer shell) °F	594 (17 min)	723 (5.8 min)
ΔT max. (outer shell - inner shell) °F	641 (35 min)	747 (29 min)
Average radiant heat input*: ISO → FS (Fire-shield)	5878 Btu/ft^2	---
" " " " : "Walls" → FS	3071 "	6609 Btu/ft^2
" " " " : FS** → outer shell	2367 "	5157 "
Average radiant heat transfer rate*: ISO → FS	10076 Btu/ft^2-hr	---
" " " " " : "Walls" → FS	5264 "	13220 Btu/ft^2-hr
" " " " " : FS** → outer shell	4057 "	10314 "

* Based on heat exchange factor = 1.0 in equation 1

** T between outer and inner surface of fire-shield = 0

Both these assumptions produce high estimates but comparative values, furnace to furnace, are not affected.

TABLE 1
COMPARISON OF DERIVED DATA FROM FURNACE TESTS

where: $\sigma(T_1^4 - T_2^4)$ = heat flux - Btu/ft^2 h
σ = Stephan Boltzman constant
T_1, T_2 = temperature of hotter and cooler surface
Δt_i = time interval i, h
n = number of intervals

The heat exchange factor is assumed to be one because it is common and would not affect relative values.

Attempts to calculate heat input to the small cylinder from the measured time-temperature data and assumed values for the relevant parameters have given widely different results from input based on heat capacity. One trial, based on the assumption that the furnace contained combustion gases uniformly at the "ISO temperature" and an absorptivity of 0.8 for the outer surface gave 20,900 Btu, which is within 6% of the heat gain computed for the Method I test by equation 1 but this does not allow for convective heat transfer which has been reported to amount to as much as 31% of the radiative component[6]. Another estimate, predicated on assumed emissivities of 0.95 and 0.15 for the furnace surfaces and gases respectively, the gases radiating at flame temperatures equal to the ISO temperature, an absorptivity of steel of 0.8 and an allowance for convection, resulted in 18,500 Btu. As this is low, an estimate of the effect of increasing the gas temperature by 300° was made. This produces an estimated input of approximately 24000 Btu. This represents fairly good agreement but three other considerations must be taken into account:

(a) In a previous test flame temperatures in the NRC furnace were found to be higher;

(b) the flame is only present intermittently and with varying intensity;

(c) An estimate, based on radiant heat transfer only to a small body (ε=0.8) enclosed by a large surface emitting at the observed furnace wall temperatures in Method II gave 27000 Btu whereas the computed value was 23000 Btu using equation 1.

The latter casts doubt on the value of the absorptivity, 0.8, assumed for the small cylinder. To resolve these matters would require more data than is available: thus it is concluded that there is too little information known to be able to calculate, from first principles, the response of specimens in either furnace. As a consequence only relative judgements can be made. Further development of the model outlined above, including forced convection from the swirling gas and a firing history, may improve the theoretical model for furnace analysis but this would be futile unless basic properties of the specimens are known accurately.

In spite of these theoretical difficulties the severity of the two Methods can be compared in a number of ways. One is to compare the temperatures at the end of the test which are indicative of the total heat input. For the small cylinder the temperature differences are marginal but for the large flask they are significant. This can be attributed to their relative heat capacities. A body with a very small heat capacity tends to "saturate" and approach a constant temperature in a Method II test. However, the temperature of the body would continue to rise along with the environment in a Method I test so that near the end it could exceed that in Method II. This would explain the minor difference between the total heat gains by the small cylinder. The small cylinder and the circumferential fire-shield on the large flask both exhibit the behaviour expected of a body with a small heat capacity.

On the other hand a body with a large heat capacity tends to remain at a lower temperature i.e. not saturate; in this case differences in the total heat input are more evident. This is demonstrated by the large flask. Although the heat input cannot be calculated it is possible to estimate relative values if it is assumed that the radial temperature gradient on the centreline is similar for the two tests (though different in magnitude). By taking the geometric mean value of the outer and inner shell temperatures at the end of the test, and subtacting the initial temperature, an estimate of the temperature rise is obtained, i.e. 109°C (228°F) for Method I and 154°C (309°F) for Method II. Although crude, this indicates that about 1/3 more heat was gained in Method II than in Method I. Another very crude measure is to compare the temperature rise for the two tests at the time when the inner temperatures begin to fall, about 1 hour after firing. Even though the flask was left inside the furnace to cool down in the Method I test,

thus reducing the rate of heat loss below that occuring after the Method II test in which the flask was removed from the furnace, the temperature rise of the inner shell was 21% greater in Method II than the temperature rise in Method I.

Another measure of the severity of the Methods is the difference in temperatures between the outer and inner surfaces of the specimens i.e. the thermal gradient. The maximum gradient would generally result in the greatest thermal stress. If a linear relationship holds, stresses on the small cylinder would be 1.5 times greater in Method II than Method I, and for the large flask 1.17 times greater.

Other indicators of severity are the radiant heat input (eq^{n} 2) and the average radiant heat flux, although time must be considered along with the flux. Values of heat input from the furnace to the specimens in Method II exceed values in Method I whether the basis of calculation for Method I is the "ISO Temperature" or that of the furnace "wall". The difference is greater if the furnace "wall" temperature is used but to compare only the heat input from the walls could be deceptive because other elements of the model mentioned earlier would not be taken into account, even though the furnaces are similar. To obviate this problem another indicator is presented in Table I: the radiative heat input from the fire-shield to the outer surface of the body of the large flask. This provides a good basis for comparison because differences would be a direct indication of the relative severity, independent of which measure of furnace temperature is dominant. On this basis Method II is clearly more severe than Method I.

No temperature drop from the outer surface of the fire-shield (measured) to the inner surface of the fire-shield (not measured) was included in arriving at the radiant heat input. To do so would reduce the values so they would be closer to the theoretical value of 1/2 the heat incident on the shield, but it would not have a significant affect on the comparison.

It is interesting that the only consistent comparisons of radiant heat input between Method I and II occur on the large flask i.e. the ratio of the two test methods is 2.15 (6609÷3071) based on heat input from furnace "walls" to the fire-shield and is 2.17 based on heat input from the fire-shield to the side of the shell. This provides additional evidence that the "walls" are the dominant environmental factor.

To demonstrate this, the temperatures of the outer surface of the small cylinder and the fire-shield of the large flask were compared to furnace temperatures. The strongest correlation was found with "wall" temperatures as explained below.

To bring the data to a common base, the outer surface temperatures are expressed as a percentage of the furnace "wall" temperature and time is started at the point closest to saturation i.e. at the end of the test. Consequently in Figure 11 the data is plotted in reverse to the normal order of time.

Note the expanded scale which emphasizes differences. Even though two different methods are being compared, the correlations in trend and magnitude are remarkable, so much so that they extend through the transient period of lead melting in the small cylinder and even to the early stages of the tests in both specimens. From this it is concluded that the dominant furnace characteristic is the temperature of the furnace walls.

In Figure 11, the secondary variations from the general trend are a result of the method of control. For Method II the burners operated "full on" and full off". The lower points coincide with the burners "on" and the upper points with the burners "off". For Method I the ripples result from adjusting the rate of firing.

FACTORS AFFECTING THE TESTS

There were a number of factors which could cause the tests to deviate from the ideal test procedures described in the Advisory Material [1]. Some are minor and others tend to balance so they would have no significant effect on the conclusions.

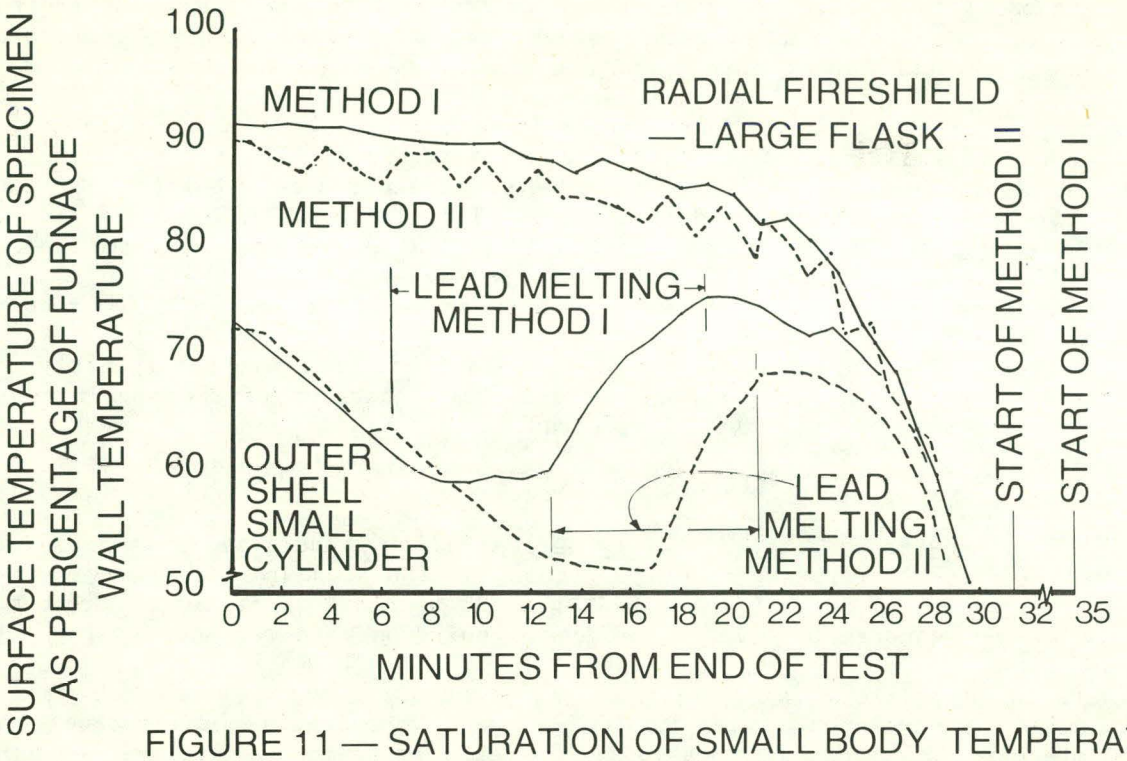


FIGURE 11 — SATURATION OF SMALL BODY TEMPERATURES

The starting temperature for ISO standard fire curve is 30°C (86°F). In anticipation of actual ambient temperatures, the curve followed was extrapolated backwards to 15.5°C (60°F). Assuming the ideal ISO curve were followed and the response of the small cylinder would have been the same as observed, an upwards correction of less than 5% to the radiant heat input would result. This is an upper bound based on the point of maximum radiation potential.

For Method II the furnace walls exceeded the specified temperatures during the early stages of the tests. Assuming a constant wall temperature of 802°C (1475°F) and an identical response to that observed on the small cylinder in Method II, a correction of less than 8% to the radiant heat input would result.

In the Method II tests the specimens could not be placed in the exact centre of the furnace. They were put on the centreline of the floor but closer to the door end of the furnace. At this location heat input would be less than at the centre of the furnace but the difference could not be estimated for two reasons. First, it was not possible to measure the temperature of the door. Secondly, the floor temperature, although known to be cooler than the walls and ceiling at the central point, was not measured near the specimens. This accounts for some of the discrepancy in the two calculated values of heat input to the small specimen and is known to result in lower temperatures on the bottom of the large flask than those measured in the Method I test.

In the Method I test, both specimens were engulfed in the flames from the burners; this would increase the heat transfer. In the Method II test the flames from the burners did not impinge directly on the small cylinder; consequently it would be expected that the convective heat transfer would be less. On the large flask, however, the flame from one burner did impinge on the upper few inches of the flask on the side towards the centre of the furnace. Some localized heating was observed in the measured temperatures, but this was offset by the reduction in heat input due to its non-central location and the reduced "chimney effect" of convection in the space between the fire-shield and body. The latter effect was larger in the Method I test, because the specimens were raised off the floor. These conclusions are based on the measured t eratures.

CONCLUSIONS

1. A test using Method II is more severe than a test by Method I following the ISO procedure. With the few exceptions noted and explained above, all parameters such as temperatu , thermal gradients, radiative heat input, radiant heat flux and total heat input sup : this conclusion.
2. Furnace wall temperatures are the dominant factor in determining heat exchange to specimens in the furnaces used. This is shown by the correlation in Figure 11.
3. Furnace wall temperatures in Method I tests conducted in accordance with the ISO Standard are significantly below the ISO standard fire-curve. This is shown in Figures 7 and 9.
4. From conclusions 2 and 3, the furnace walls should meet or exceed the standard fire curve set out in the ISO Standard, to meet the intent of the Advisory Material[5].
5. The furnace wall temperatures equal or exceed those measured by the ASTM thermocouples. This is also shown on Figures 7 and 9.
6. The characteristics of the furnaces and specimens under test were not sufficiently understood to be able to predict, from first principles, the behaviour of the specimens under test. Further development of the model is required.

RECOMMENDATIONS

1. Paragraph 727 of the Advisory Material, Safety Series #37[1], should be revised to require that the furnace surface temperatures meet or exceed the ISO standard fire curve. As this is a different requirement from that set out in the ISO Standard it is further recommended that reference to the standard be deleted, to avoid confusion, and replaced by the formula specifying the curve.
2. As an alternate to the first recommendation a new standard to meet the IAEA Thermal Test requirement should be developed. This could be based on suitable modifications to the existing ISO standard or it could be based on the ASTM Standard [5] with an appropriate time period, to account for minor differences in the standard fire curve specified.
3. Benchmark experiments that could be used to calibrate the performance of computer codes used to analyse the thermal behaviour of packages should be carried out. From this a validation standard could be developed much like that developed for nuclear criticality safety [7].

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PERFORMANCE OF SMOKELESS GASOLINE FIRE TEST FACILITY

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INTRODUCTION

The DOE Manual Chapter 0529 requires packaging for radioactive materials to perform satisfactorily when subjected to a series of four tests simulating accident conditions. Escape of radioactive materials and reduction of shielding must be below defined limits and the package must remain subcritical. The free drop, puncture, thermal, and water immersion tests must be performed in the listed sequence.

Work reported in this paper has evolved as a result of requirements for the thermal test. These requirements include exposing the package for 30 min to heat which is not less than that which would result from exposure of the whole package to a radiation environment of 1475°F (800°C) with an emissivity of 0.9, assuming the surface of the package to have an absorption coefficient of 0.8. It is further required that the package be naturally cooled for 3 hr after the test period, unless it can be shown that the temperature on the inside of the package has begun to fall in less than 3 hr.

Several techniques can be used to provide this thermal test environment. These techniques can be based on the use of electric or gas fired ovens, or open gasoline fires. We prefer the open gasoline fire since it more closely simulates transportation accident conditions, which form the basis for the thermal test requirements. In addition to providing the necessary thermal radiation temperature and heat flux, the open fire is characterized by strong convection currents and a generous air supply. These latter characteristics produce a degree of realism which cannot be achieved in an oven or other apparatus which has no open flame and a limited air supply.

However, the hypothetical accident thermal test can no longer be performed by simply burning gasoline in an open pit. The uncontrolled open pit technique creates thick, dense, black clouds of smoke which, by present environmental standards, are completely unacceptable. In 1972, an effort was initiated at Mound Facility to develop techniques for reduction of such smoke. The techniques included limited fuel distribution and forced air supply. The result was a substantial, but insufficient, reduction of smoke. Figure 1 shows such a test. This work was described in a paper presented at the 4th International Symposium which dealt with the development of the design of the thermal test facility. [1]

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FIGURE 1 - Partial smoke reduction in developmental test prior to permanent thermal fire test facility.

This is a follow-up to the previous paper and describes the final design and actual performance of the facility. Techniques were developed and implemented in the new test facility for an open gasoline fire which does not produce smoke in any significant quantities. This paper reports the successful results of these techniques in eliminating the smoke plume as well in providing a controllable fire which meets the requirements of the thermal test.

THE THERMAL TEST FACILITY

In the testing facility now used, two techniques are applied to abate smoke. These techniques are the addition of excess air to the fire and the spraying of a water mist at the base of the fire. An excess air technique had been used successfully to reduce smoke in the experimental thermal tests prior to construction of the present facility. The water spray technique, however, is the concept on which the design of the new facility was primarily based. It had been successfully used by others in similar applications. They had demonstrated the effectiveness of the system, and we had studied the "NAVTRADEVCEN 71-C-0083-4" report after meeting and discussing the system with key people in their program. The water is believed to interact in the combustion reaction to provide more complete combustion.

The fire test facility which was completed in 1974 is seen in Figures 2, 3, and 4. It consists of the following major features:

1. Combination wind and heat shield enclosure
2. Fire pit containing water
3. Water spray system
4. Forced air delivery system
5. Fuel delivery and distribution system
6. Hoist.

The combination wind and heat shield is a concrete block enclosure lined with firebrick and located over a 6 in. deep pit. The base of this fire pit is poured concrete measuring 10 ft x 10 ft x 0.5 ft deep. Three sides of the enclosure are 8 ft high, and the remaining side is 4 ft high for observation and ease of handling the shipping container. The enclosure prevents the flame from being blown away from the container and also serves as a shield to protect surrounding equipment from heat.

Figure 5 schematically illustrates the water spray, air supply, fuel feed and the water in the pit. These are the primary features related to elimination of the smoke plume.

The water spray system consists of seven spray nozzles uniformly distributed in the pit and located approximately 4 in. above the surface of the pool of water, which is 5 in. deep. This system provides a fine mist of water approximately 4 in. above the base of the fire. The nozzle spray is directed horizontally and provides complete coverage of the burning fuel surface.

A 5-horsepower fan and air ducts provide approximately 8,000 ft^3/min of air to the enclosure from two opposing sides of the enclosure just above the fire pit.

The fuel delivery and distribution system continuously delivers 100-octane aviation gasoline to the pit from a 5000-gal storage tank buried approximately 100 ft from the fire pit. The fuel is gravity-fed to the pit and released below the water level in a well distributed pattern. The gasoline floats to the surface of the water and burns.

TEST PROCEDURE

In preparation for the thermal test, a hoist is used to place the package on a stand within the enclosure approximately 2 ft above the water surface. Fifty-five gal drum packages are centered within the enclosure approximately 3 ft from the sides. Thermocouples are attached to the package at the required locations and leads are run to a recorder located approximately 30 ft from the fire test enclosure. Approximately 300 gal of water is then fed into the 6 in.

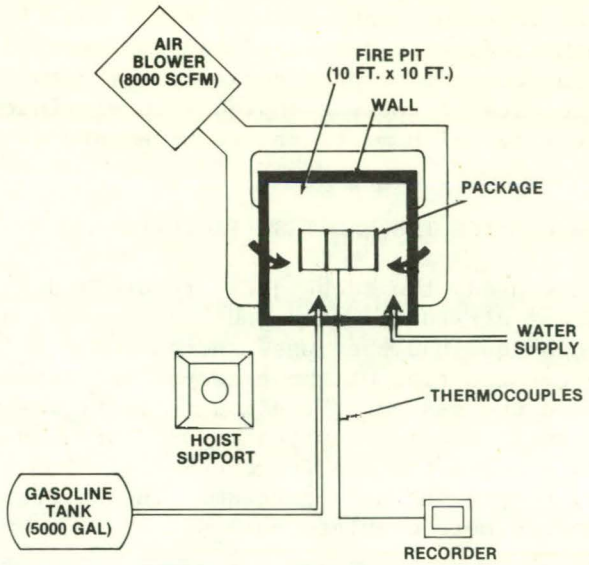


FIGURE 2 - Major features of fire test facility.

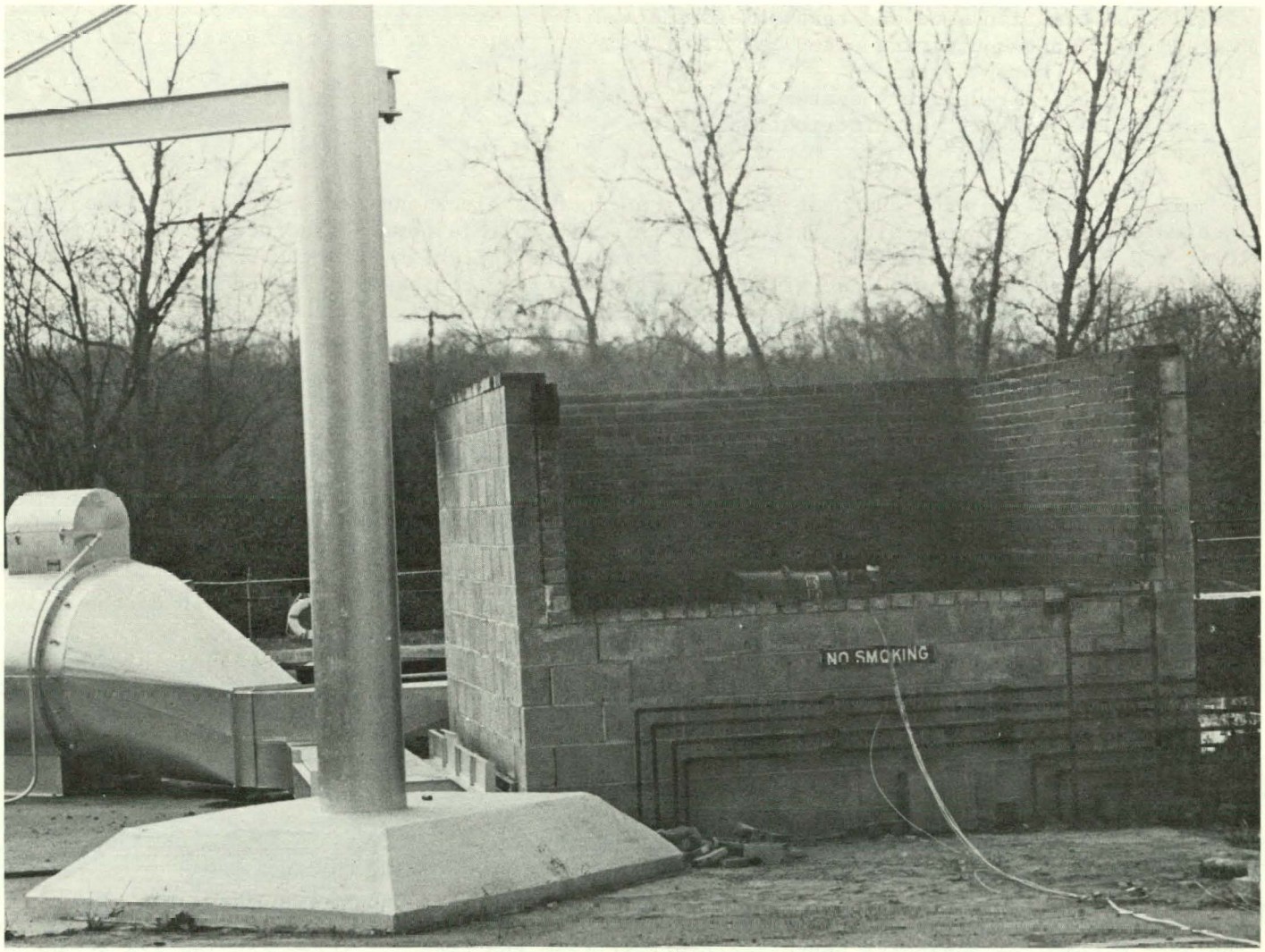


FIGURE 3 - Fire test facility showing air blower, package hoist, fire pit enclosure, and water pipes.

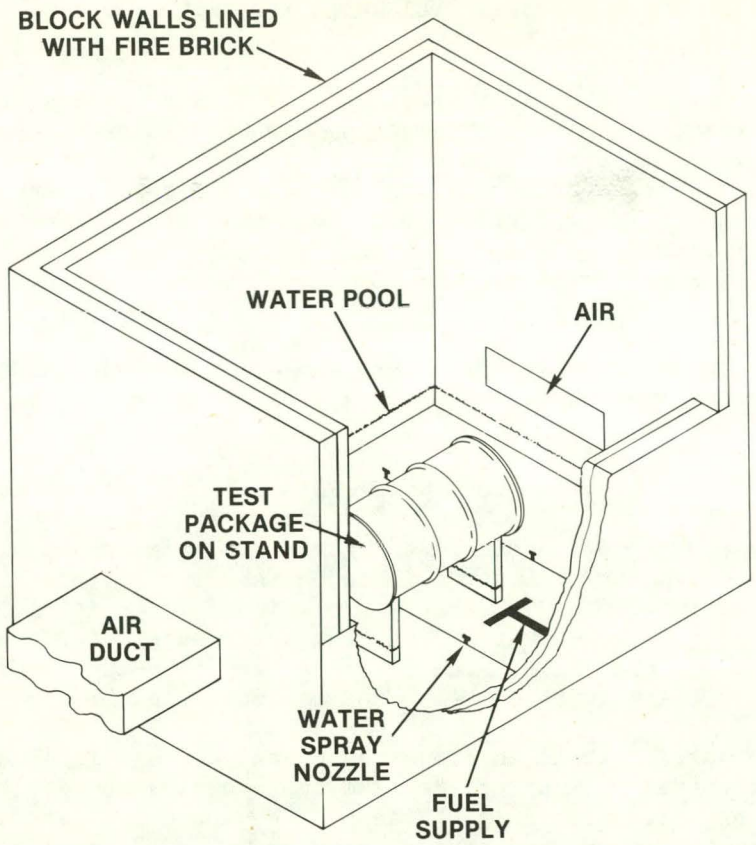


FIGURE 4 - Fire test facility schematic.

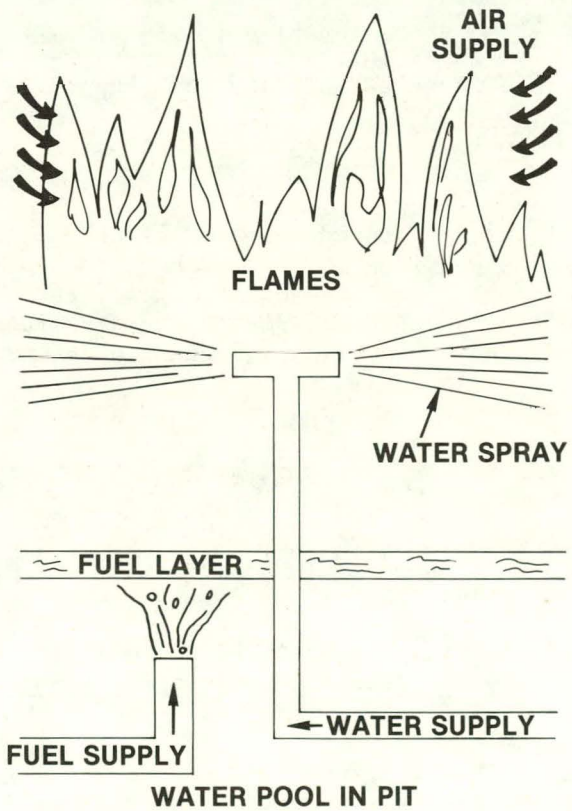


FIGURE 5 - Schematic showing water spray and air techniques for elimination of smoke.

deep pit, resulting in a water level of 5 in. Gasoline is then fed into the pit to form a layer of fuel on top of the water surface. The spark ignition system is used to start the fire remotely.

After ignition of the fire, the water spray and air supply systems are turned on. The water and air supply adjustments are preset, whereas manual adjustments are made to the fuel flow as required. The temperature measured on the surface of the test container is increased to 1475°F. (This is equivalent to conservatively assuming emissivity and absorption coefficients of 1.) The test is run for 1/2 hr. During that time period, approximately 200 gal of fuel is consumed and approximately 700 gal of water is used. The water generated by the water spray system is drained from the bottom of the pit via an overflow system to maintain a constant water level.

After the test, all excess fuel is burned off the water surface, and excess water and carbon particles are removed by a drainage system to a separate area for removal.

RESULTS OF TESTS

To date, 16 tests have been performed in the facility. Figure 6 shows one such test. The results of some of these tests are documented in three SARP's. [2,3,4] Usually, during the course of the 30-min tests, the smoke is virtually eliminated. It is estimated that air accounts for approximately 30% of the smoke reduction, and the water spray, for the remaining 70%. Periodically, under the worst conditions, the smoke may approach the upper limit of a Ringelmann number of 1 (20% opacity), which is still below the maximum allowable.

A thick, continuous wall of flame envelops the container to maintain sufficiently high temperatures for successful testing. Most of the time the exterior surface temperature of the shipping container is controlled within +150°F and -50°F of the desired 1475°F temperature. The temperatures obtained throughout one of the tests are plotted as a function of time in Figure 7. The drum skin temperatures were measured at the side of the package. Chromel/alumel thermocouples and a multipoint recorder were used to monitor the test. The drum surface reached the required 1475°F less than 5 min after ignition. Throughout the course of the test, it was necessary to closely monitor the temperature and manually adjust the fuel flow rates. The fire burned out within a few seconds after the fuel flow was stopped.

CONCLUSIONS

The new thermal test facility has proved to be a reliable method for satisfactorily performing the required test. The flame provides sufficient heat to assure that the test is valid, and the temperature can be controlled satisfactorily. Also, the air and water mist systems virtually eliminate any smoke and thereby exceed the local EPA requirements. The combination of the two systems provides an inexpensive, low maintenance technique for elimination of the smoke plume.

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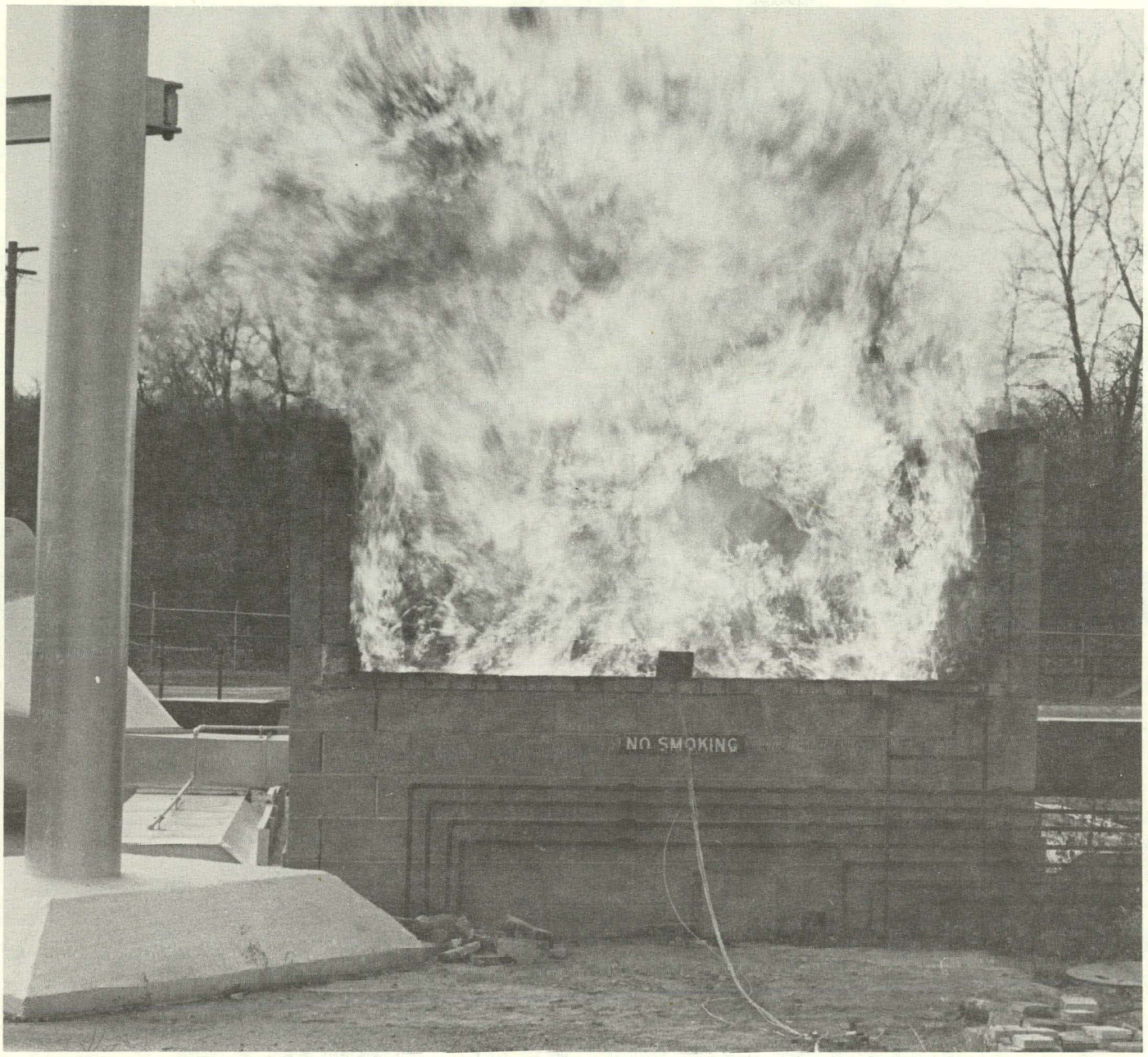


FIGURE 6 - Smokeless fire test in progress.

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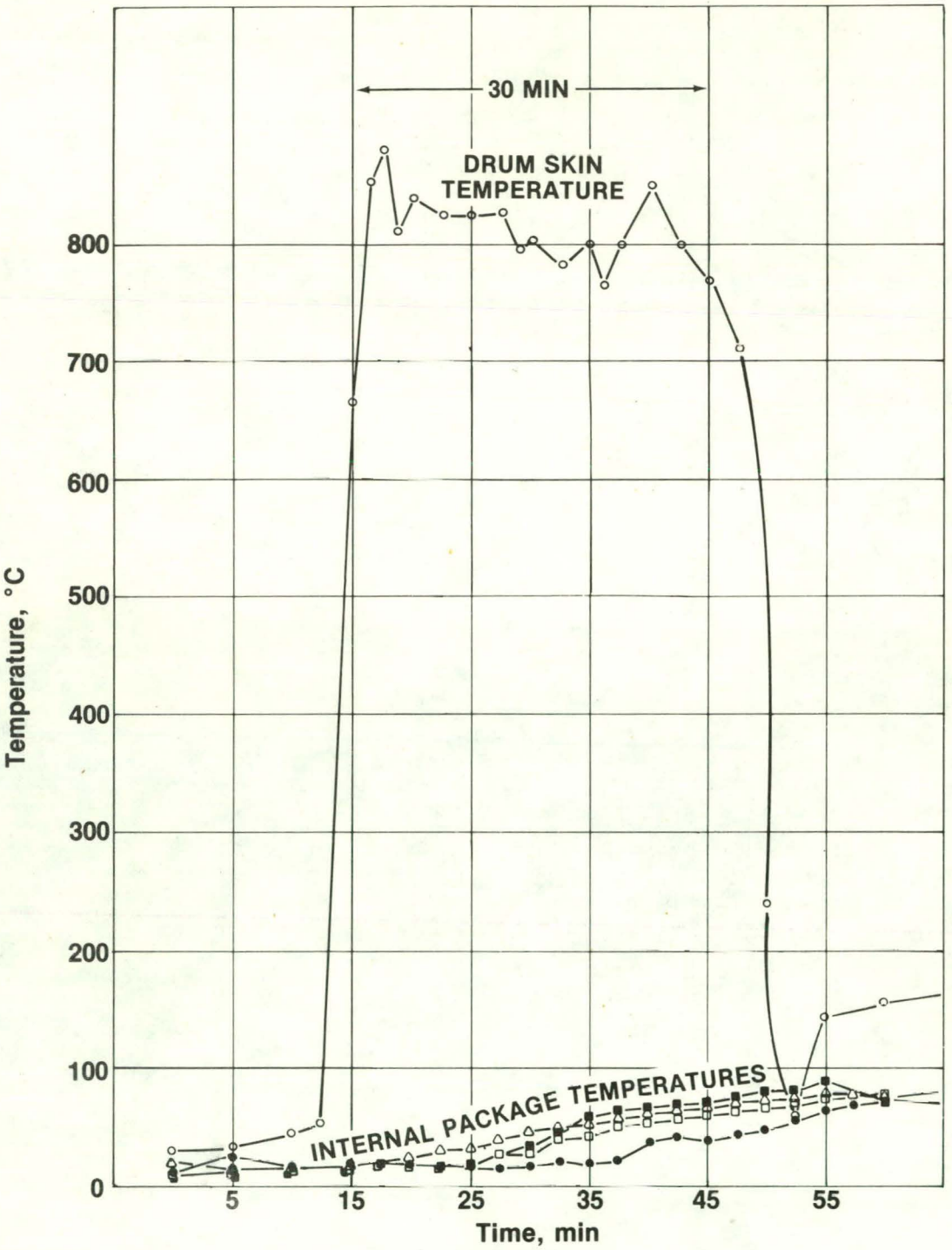


FIGURE 7 - Package temperatures as a function of time in fire test.