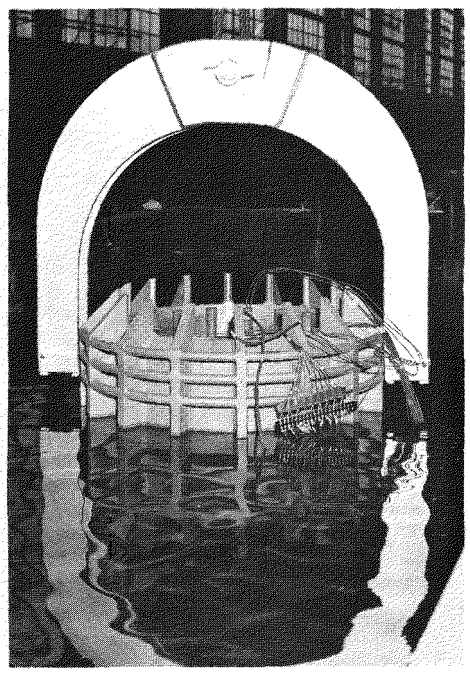
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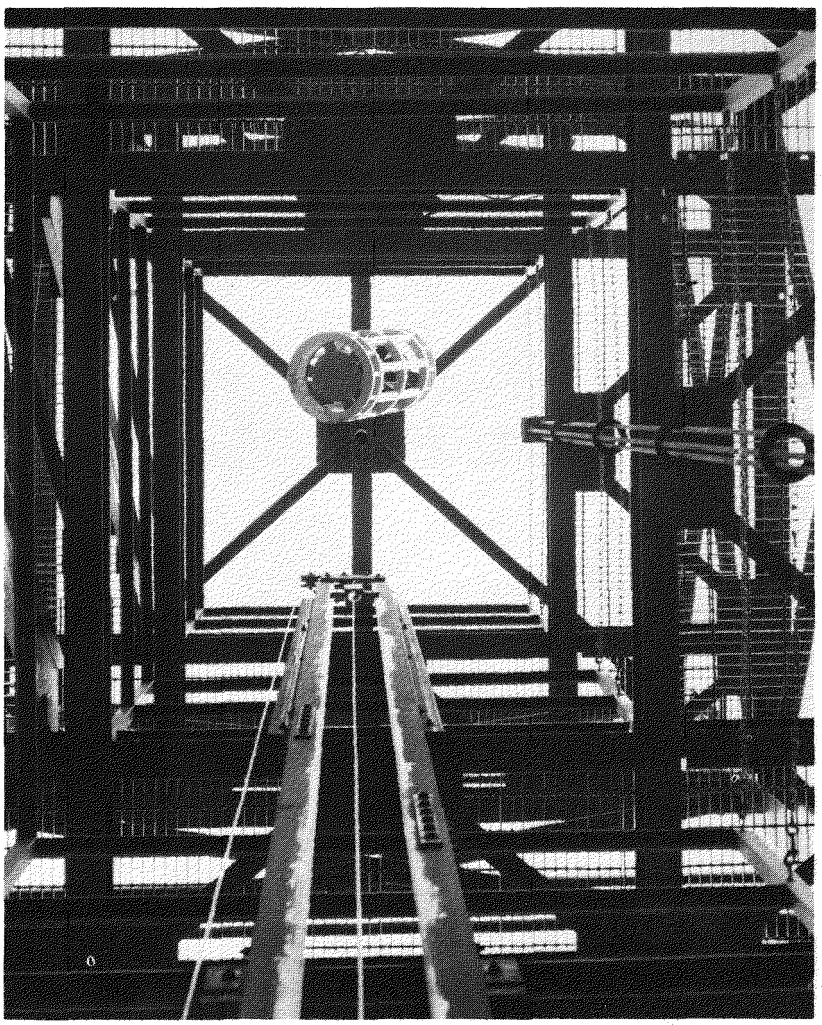
Proceedings of the 4th

INTERNATIONAL SYMPOSIUM

ON

PACKAGING AND TRANSPORTATION OF RADIOACTIVE MATERIALS





MASTER

SEPTEMBER 22-27, 1974
Miami Beach, Florida

sponsored by UNION CARBIDE CORPORATION · NUCLEAR DIVISION
and the U.S. ATOMIC ENERGY COMMISSION

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HARMONISATION OF INTERNATIONAL REGULATIONS AND THE STANDARDISATION OF TRANSPORT DOCUMENTS

E J Wilson

A H Partridge

ABSTRACT

The United Nations Committee of Experts on the Transport of Dangerous Goods recommends that all international and national organisations produce regulations which conform with the IAEA Regulations for the Safe Transport of Radioactive Material. International organisations have already begun work to implement this recommendation in respect of the 1973 Revised Edition of the IAEA Regulations. The paper outlines problems which have arisen in achieving harmonisation. In the second part of the paper the standardisation of the various documents in relation to approval and transport is discussed. Sufficient experience has now been gained to enable the actual format of competent authority certificates to be standardised. Such standardisation is beneficial to consignors, carriers and other competent authorities in easing the interpretation of information and the validations of certificates originating abroad.

INTRODUCTION

The International Atomic Energy Agency's Regulations for the Safe Transport of Radioactive Material have played a major role in the formulation of transport regulations, both national and international, aimed at ensuring adequate standards of safety in the transport environment.

The growing use of radioactive materials in the industrial and medical field make it highly desirable that the transport of such materials should be carried out with the minimum of hindrance and the development of harmonised international regulations clearly facilitates this goal.

Parallel with the need for the development of harmonised safety and administrative provisions is the need to standardise the format of the various documents relevant to the approval of package design and shipment operations. If this could be achieved on an international scale the benefits to competent authorities, carriers and consignors would be considerable. Much delay can occur when a chain of competent authorities are concerned with multilateral package design and shipment approvals due to the lack of information on the relevant approval certificates, obscurity of detail and language difficulties. Standardising the general format of certificates would serve to ensure that essential detail would not be omitted and make for easier interpretation. National regulations may require additional information, but this could be kept separate from the standardised clauses.

REGULATORY HARMONISATION

The international shipment of radioactive material is controlled under the regulations prepared by the organisation responsible for the particular mode of transport. The various principle international organisations and their appropriate regulations, are familiar to everyone in the transport of dangerous goods field viz IATA (air), IMCO (sea). RID (rail in Europe), ADR (road in Europe).

Advantages of Harmonisation

Invariably a shipment operation involves more than one mode of transport, hence inconsistencies between the various modal regulations are undesirable. In certain cases, there are differencies. For example, in relation to transport by air, the surface temperature of both Type B(U) and Type B(M) package is limited to 50°C even when transported as a full load, whereas by other modes the limit for Type B(U) packages under full load conditions is 82°C and for Type B(M) packages there is no upper temperature limit. Such differences are to be expected and can be taken into account by the person responsible for programming the shipment operation, but differences between one set of regulations and another on such items as labelling, particularly in relation to multi-modal units such as freight containers would be irksome and in

many cases impossible to comply with.

Type B(M) packages and shipment thereof, together with packages shipped under the terms of a Special Arrangement are subject to the multilateral approval of the competent authorities of all countries through or into which the package is transported. In certain cases many countries may be involved. The various competent authorities will approve the package and shipment against the regulations governing the mode of transport to be used in their respective country. From experience in the United Kingdom, it has been found very much easier and quicker to validate a foreign certificate of approval if it is known that the regulations against which approval has already been granted are in line with the current IAEA regulations. This is because the UK transport regulations fully implement those of IAEA including the unilateral principles. This means in practice that in the case of a Type B(M) package or shipment operation it is only necessary to consider those items, non-compliance with which, has prevented the package from being categorized as Type B(U).

Problems of Achieving Harmonisation

Although the process of harmonising regulations in line with IAEA is on face value a simple exercise there are problems which take many forms. The IAEA regulations have in the past been revised at intervals of approximately five years and since of necessity, amendments to other national and international regulations to implement the revised prescriptions lag behind, there is a natural reluctance to carry out further amendment after a relatively short period of time. This situation should be eased in the future with the decision of IAEA to change the period between revisions to ten years.

In some countries there are legal obstacles associated with changes in national regulations and this has resulted, in certain cases with national regulations quoting IAEA regulations out of date by two revisionary periods. Although this may cause no problem for shipments exclusively within the relevant country, the problems of import and export operations are numerous. The responsibility must fall on the

shoulders of the experts in the atomic energy industry of such countries to endeavour to make the legal wheels grind more quickly and thus achieve parity between national and international regulatory procedure.

Since the 1964 and 1967 IAEA Regulations were published, problems over practical interpretation of the details have arisen, enhanced when translated into national regulations. Problems of this nature led to IAEA producing Safety Series No 37 - Advisory Material for the Application of the IAEA Transport Regulations. Safety Series No 37 will be the subject of revision at intervals of two and a half years; it should therefore adequately reflect procedural and technical developments.

Discussion on international harmonising would not be complete without reference to the achievements of an ad hoc body of competent authorities called the Radioactive Transport Study Group (RTSG), who operate under the auspices of the IAEA. The RTSG were largely responsible for the comprehensiveness of Section VIII of the 1973 IAEA Regulations.

Programme for the Implementation of the 1973 IAEA Regulations by International Organisations

The work on revision of the various international transport regulations to be in line with the requirements of the IAEA regulations is well underway and the tentative dates for their implementation are as follows:

IATA - June 1975	ADR - Early 1977
IMCO - Early 1975	RID - Early 1977

The United Kingdom will probably revise her land regulations by the end of this year, and follow this with sea and air changes within a few months of changes to IMCO and IATA.

Although differences between the 1967 and 1973 IAEA Regulations are not so great, there will certainly be problems over a period of 2 or 3 years as a result of international and national regulations being so out of step. It will be a task of the RTSG to propose some transitional arrangements.

STANDARDISATION OF TRANSPORT DOCUMENTS

During the thirteen years that the IAEA Transport Regulations have been in existence considerable experience has been gained with regard to the information required on the various documents associated with the transport of radioactive material. In the United Kingdom as probably is the case elsewhere, carriers, consignors and consignees and other interested parties have become accustomed to the format of the various documents produced nationally and as a result queries to the competent authority of an administrative nature have tended to reduce. The same cannot always be said with regard to foreign documentation very largely because in some cases the necessary information is not given and in other cases is given but obscure. With the publication of IAEA Safety Series No 37 means now exist whereby specimen documents for guidance can be brought to the attention of those concerned with producing such documents. The 1973 IAEA transport regulations now list the information required on approval certificates largely as a result of deliberations by the RTSG. The next logical step would be to have a standardised format for the documents on which the information is to appear.

Transport Documents

The following list shows the various documents required by regulation. The regulations allow the package and shipment certificate to form one document if necessary.

1. Type A - Fissile Contents
2. Type B(U)
3. Type B(M)
4. Criticality Clearance Approval Certificate
5. Shipment Approval Certificate
6. Special Arrangement Approval Certificate
7. Special Form Approval Certificate
8. Consignor's Certificate

In general the Criticality Clearance Certificate will constitute a part of Package Design Approval Certificates 1. 2. and 3. in the list and there is thus no need for a separate approval certificate.

Information Required In Certificates

The IAEA Regulations includes in para 824 a list of relevant information to be included on all certificates issued by a competent authority.

Appendix 1 shows a typical format for an approval certificate suitable for Type A - Fissile, Type B(U) and Type B(M) packages.

In the case of shipment certificates, both for large sources and special arrangements, experience in the United Kingdom has shown that the use of a printed format imposes limitations on the space available for the necessary information. Additionally, not all clauses are applicable to all shipments. To overcome the problem it has been found better to adopt a comprehensive list of clause headings to be used as applicable. In this way the content of any clause may be given the amount of space it merits. The applicable clauses will always follow the same general order in the certificate.

Appendix 2 illustrates the format for a Shipment Certificate. That for a Special Arrangement Certificate is similar apart from two important additional clauses, appearing as Clause 4 and 5 after "Consignment Specification", Clause 4 is "REASON FOR SPECIAL ARRANGEMENT" and Clause 5 is "COMPENSATORY SAFETY ARRANGEMENTS".

The certificate of approval for Special Form material can be a fairly simple document and the necessary information concerning the design specification, radioactive contents and maximum permitted activity can be accommodated on a single sheet. Appendix 3 shows a typical certificate.

A restriction in the length of this paper, prevents illustration of the consignment note recommended in the United Kingdom. The recommended format will appear in a revised British Standard No 3895. In addition to the consignor's declaration as recommended by IAEA and the list of appropriate regulations, the note will contain details of the consignment, type of package and radioactive material being carried, etc, a list of accompanying certificates and instruction for carriers

including, in particular, stowage for the safe disposal of heat.

CONCLUSION

It is a fact of life that in the eyes of many, the correct paperwork is the most important aspect in any project. Although many of us may not agree with this, it probably becomes nearer to being true in the transport field than any other, whether we are considering passports, airline tickets, cargo manifests or those documents to which reference has already been made. There is no doubt that lack of, or incorrect certification, is responsible for much of the delay which from time to time we have all been aware. If this paper helps in reducing such delays and easing the load on all those concerned with the transport of radioactive material it will in some measure have achieved its goal.

APPENDIX 1

/Page 1 of Certificate/

APPROVAL REFERENCE .....

Issue .....

CERTIFICATE OF APPROVAL OF
PACKAGE DESIGN FOR THE
CARRIAGE OF RADIOACTIVE MATERIALS
SECTION 1

Packaging	Design No	Quantity per Assembly
Outer	-	-
Intermediate	-	-
Inner	-	-

THIS IS TO CERTIFY That /Name of responsible department/ being for the purpose of the Regulations of the International Atomic Energy Agency, the Competent Authority of /Name of Country/

has approved the above-mentioned Package submitted by /Name of applicant/

as /Type of Package/

for the transport of radioactive material as listed in Section 2(d) of this certificate.

Packages manufactured to this design meet the requirements of the regulations and conventions overleaf, subject to the contents being limited as above and subject to the conditions of approval specified in Section 2.

THIS CERTIFICATE CANCELS ALL PREVIOUS ISSUES

COMPETENT AUTHORITY

IDENTIFICATION MARK:

/Signature(s) of responsible body relevant to the mode(s) of transport/

Date of Approval .....

/Page 2 of Certificate/

Here is listed the international and national regulations:
IAEA, IMCO, IATA, RID, ADR, UK road, rail, sea and air regulations, followed by the following notes:-

NOTES 1. Type B(U) means that the package design is in accordance with the requirements of IAEA 228-241.

2. Type B(M) means that the package design is in accordance with the requirements of IAEA 242.

3. Any changes in the make-up of packaging will be the subject of a separate certificate of approval.

4. Attention is drawn to the labelling and marking requirements of IAEA Section V, one provision of which is that the outside of each package shall be clearly and durably marked with the Competent Authority's Identification Mark.

5. Any alteration to the package design without prior approval from the Competent Authority will automatically invalidate this Certificate.

/Page 3 of Certificate/

SECTION 2

CONDITIONS OF APPROVAL

(a) PACKAGING

/Drawings and/or specification reference/

(b) INSTRUCTIONS ON USE OF PACKAGE

/Pre-shipment, transit and maintenance/

(c) CRITICALITY

/Details of criticality clearances/

(d) CONTENTS

/List of radionuclides and maximum contents together with chemical and physical form/

(e) ASSUMED AMBIENT CONDITIONS

/International or otherwise/

(f) SUPPLEMENTARY OPERATIONAL CONTROLS

/Package preparation, loading, transport, stowage and handling/.

(g) /For Type B(M) a statement of those prescriptions in IAEA 234-241 with which package does not conform/.

(h) RESTRICTIONS ON MODE OF TRANSPORT

/Page 4 of Certificate/

SECTION 3

Sketch and Brief Description of Packaging, Dimensions and Weights.

APPENDIX 2

APPROVAL REFERENCE .....

Issue .....

CERTIFICATE OF APPROVAL FOR A RADIOACTIVE SHIPMENT OF

/Irradiated Fuel, Plutonium Oxide, etc, as appropriate/

UNDER MULTILATERAL APPROVAL*

REQUIRING

SUPPLEMENTARY OPERATIONAL CONTROLS

NOT REQUIRING

CONSIGNOR .....

1. CERTIFICATION

This is to certify that /Name of responsible department/, being for the purposes of the Regulations of the International Atomic Energy Agency the Competent Authority of /Name of Country/ approve the above mentioned RADIOACTIVE SHIPMENT comprising the movement during the period .... consignment of packages from ..... to ..... by /Mode of Transport/.

The Competent Authority Identification Mark on the package is:-

2. RELEVANT TRANSPORT CONDITIONS

The shipment shall comply with the following regulations /List of Regulations/

3. CONSIGNMENT SPECIFICATION

The/Each consignment shall consist of not more than .... package(s) complying with the Certificate of Approval for Package Design No .....

CONDITIONS

4. SUPPLEMENTARY OPERATIONAL CONTROLS

No supplementary operational controls are required during transport, /or specify the operational controls/.

*"Multilateral Approval" means that the radioactive shipment requires the approval of the competent authority of the country of origin of shipment and of the competent authorities in all countries through or into which the shipment is to pass (IAEA 815).

5. ACTION PRIOR TO DESPATCH

The/Each package shall be held before despatch until the temperature of the system has reached equilibrium before it is finally closed. /Not required if satisfactory equilibrium conditions have been demonstrated to the competent authority of the country of origin of shipment./

6. LIFTING AND ANCHORAGES

/Specify, if applicable, together with instructions regarding inspection certificates relating to lifting gear./

7. STOWAGE

/Specify stowage arrangements relevant to the mode of transport/

8. FULL LOAD CONDITIONS

/Insert reasons for full load conditions./

9. LABELLING AND MARKING

Labelling shall be in accordance with the regulations listed in Clause 2. /Insert anything additional to routine regulatory requirements./

10. RESTRICTIONS ON MOVEMENT

/Restrictions with regard to climatic and other conditions./

11. ADMINISTRATIVE ARRANGEMENTS

The consignor shall be responsible for notifying the /driver/ of the identification number of the packages, for fulfilling the conditions set out in this certificate and for the following:-

(1) Information
He shall include with the transport documents accompanying the consignment the following information /List the appropriate details as necessary/.

(2) Certificates of Approval by Competent Authorities
/List of appropriate certificates to be attached to transport documents/

(3) Notification of despatch
/List of appropriate people to be informed together with relevant information to be given/

(4) Stowage arrangements on ships

/The consignor to notify the ship's master of relevant stowage provisions as specified in 7./

12. EMERGENCY ARRANGEMENTS

/As applicable to mode of transport and include any arrangements required by foreign competent authorities./

13. ADDITIONAL INFORMATION

/If not quoted in any package/packaging approval certificate used in association with this document/

- Brief description of package
- Weight and overall dimensions
- Details of vehicle where necessary

14. PREVIOUS ISSUES

This certificate cancels all previous issues.

15. RENEWAL OF CERTIFICATES

The master certificate should be returned to the Competent Authority before the date of expiry if the period of validity is required to be extended.

/Signature of responsible body relevant to the mode of transport/

Date of Approval .....

APPENDIX 3

APPROVAL REFERENCE .....

Issue .....

CERTIFICATE OF APPROVAL OF SPECIAL FORM DESIGN OF RADIOACTIVE MATERIAL

TITLE	
Drawing Nos and/or Specification	
Radioactive Material	
Maximum Activity	

This is to certify that the /Name of responsible department/ being, for the purposes of the Regulations of the International Atomic Energy Agency, the competent authority of /Name of Country/, has approved the above-mentioned radioactive material manufactured to the above-mentioned design as special form radioactive material and as such will meet the requirements of the regulations overleaf.

This Certificate of Approval applies only to the design as set out in the above named drawings and/or specifications ..... dated ..... submitted by ..../Name of Applicant/.....

THIS CERTIFICATE CANCELS ALL PREVIOUS ISSUES

COMPETENT AUTHORITY

IDENTIFICATION MARK:

/Signature of responsible body relevant to the mode of transport/

Date of Approval .....

(Overleaf appears list of relevant international and national regulations)

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AN UPDATE ON THE STATUS OF RECENT AND PROPOSED CHANGES TO THE U.S. REGULATIONS FOR TRANSPORT OF RADIOACTIVE MATERIAL

A. W. Grella

ABSTRACT

During 1972-1974, extensive amendments to the U.S. Regulations for the Safe Transport of Radioactive Materials have taken or are in the process of taking place. The substance of three major rule-making dockets of the DOT Hazardous Materials Regulations Board which have been completed or are nearing completion are discussed. The major aspects of the recent revisions of the IAEA regulations as issued in its 1973 Edition of Safety Series No. 6 are also discussed. These changes are expected to be included in a major notice of proposed rule making which is expected to be issued shortly to both USDOT and USAEC regulations.

I welcome very much this opportunity to address the Fourth International Symposium this morning. Since my previous address before the Third Synposium in Richland three years ago, there has been a great deal of regulatory activity relating to transport of nuclear materials--on both the domestic and international scene. Most of you, of course, are well aware by now of the major revision to the International Standards of the International Atomic Energy Agency (IAEA) which took place in 1973, with the issuance of a revised edition of IAEA Safety Series No. 6 "Regulations for the Safe Transport of Radioactive Materials." This action followed the approval in September 1972 by the Agency's Board of Governors of the recommendations of two regulatory review panels which had been convened in 1970 and 1971.

My mission today is to provide you with an update on the status of the U.S. regulations, with respect to the adoption of the revised international standards, as well as to summarize the other major amendments to the U.S. regulations which have taken or are about to take place since 1971.

First, in 1972, several important changes were accomplished in regulatory Docket HM-98 of the DOT Hazardous Materials Regulations Board. Following an earlier notice of proposed rule making issued on March 17, 1972, a final amendment was issued on September 2, 1972 under the heading: "Radioactive Materials: Preparation of Packages for Shipment," which accomplished the following:

1. New definitions were established for the "containment system" of a radioactive materials package, as well as the "maximum normal operating pressure."

2. Several additions were made to the general packaging requirements, spelling out more exactly those quality assurance-related shipper requirements prior to the first shipment of a package as well as prior to each shipment of a package. One of these requirements included the performance of a pre-shipment leak test on the containment system of <u>each</u> liquid-containing package of radioactive material exceeding a Type A quantity.

These amendments were proposed initially as an outcome of a serious air transportation incident involving spread of radioactive contamination from a package of radioactive liquid which had been improperly prepared for shipment.

Following Docket HM-98, another set of amendments which were completed in Docket HM-73, entitled "Design Approvals for Radioactive Materials Packages" was issued on February 14, 1973, following earlier notices of rule making which had been issued on January 8, 1971 and November 20, 1971. The major changes in this docket included:

1. DOT special permits are no longer issued for packages containing in excess of a Type A quantity, and for which the package safety analysis report has been reviewed and approved by the U.S. Atomic Energy Commission and found to be in compliance with the applicable regulatory performance standards and an appropriate USAEC package approval has been issued. DOT special permits continue to be issued, however, for any packages or shipments which might not meet all of the regulatory performance standards, and for which a "waiver" has been issued by the Hazardous Materials Regulations Board pursuant to the formal petition procedures.

2. The administrative procedures for prescribed shipper and carrier controls for fissile Class III shipments and shipping paper requirements for such shipments were clarified.

3. The administrative procedures and conditions applicable to competent authority certificate issuances by the DOT Office of Hazardous Materials, pursuant to the requirements of the International Atomic Energy Agency regulations were clarified.

4. The FAA regulations in 14 CFR Part 103 were amended to incorporate an international requirement prescribing that packages containing a large quantity (large source) of radioactivity are not authorized on passenger-carrying aircraft.

In conjunction with the above amendments, two other related actions took place. On March 22, 1973, the USAEC-USDOT Memorandum of Understanding for the regulation of safety in transportation of radioactive materials was signed, and also on April 25, 1973, the USAEC made several changes to 10 CFR 71 of its regulations. The major impact of all of the above was to expand the area of USAEC's direct involvement in package approvals from large source/fissile packages into Type B/fissile packages.

The next regulatory docket concerning radioactive materials was issued as a notice of proposed rule making in Docket HM-111, on October 25, 1973 under the caption: "Miscellaneous Proposals Relating to Radioactive Materials." In this proposed notice, a number of amendments, based on experience since the major regulatory revision of January 1969 were proposed. These proposals at the time of this writing (May 1973) have not been published as final amendments, but may have been by the time of the 4th International Symposium. Among the proposals are the following:

1. Packages containing liquid radioactive materials exceeding Type A quantities would be required to meet both the 30' drop test and liquid absorbing provision or a requirement for doubly-sealed inner containment.

2. The method of listing "families" of certain packagings as authorized Type A packagings would be deleted. In their place, reference would be made to only the DOT Spec. 7A, Type A general performance specification, with an added requirement imposed, however, that any user of Spec. 7A packaging would be required to document and maintain on file a record of his determination of packaging compliance.

3. Newly constructed DOT Spec. 55 packagings would no longer be authorized for shipments of limited Type B quantities (20 curies to 300 curies) of special form radioactive materials.

4. Changes to the design requirements, as well as the authorized fissile radioactive contents of the DOT Specifications 6L and 6M were proposed.

5. Design modifications to the DOT Spec. 2R inner containment vessel (a component of both the DOT 6L and 6M) were proposed, providing more definitive requirements for flanged closures.

6. Several new DOT specification packagings were proposed to replace several widely used special permit packages, i.e., DOT SP 4909 (UF_6 in protective overpacks); DOT SP 5800 (Spec. 55/wooden "overcoat"); DOT SP 5725 ("Sugarman" Spec. 55/wooden overcoat); and DOT SP 5021 (fissile uranium-235 loadings in steel drums) and DOT SP 6273 (fissile UF_6 "heels" in bare cylinders).

7. A requirement was proposed that shippers of special form radioactive materials maintain on file a written certification and safety analysis of their special form determination.

8. Proposed requirements were made to clarify the method for "averaging" of multiple wipe samples in the assessment of "significant removable surface contamination", as well as propose a modified level of removable surface contamination in "full-load" surface vehicles.

9. Proposals were made to add to the motor vehicle and rail carrier regulations provisions for segregation and stowage controls for multiples of packages totalling 50 Transport Indexes or less.

10. Guidance for shippers in completing the written entries on "Radioactive" package labels was proposed.

11. Provisions were proposed for shipments of Fissile Class II packages as a fissile Class III shipment, as well as co-mingling of fissile Class II and III packages as a fissile Class III shipment.

The largest and most significant proposed rule making in the history of the Hazardous Materials Regulations Board took place on January 24, 1974, with the issuance of two major notices in Dockets HM-103 and HM-112, entitled " Hazard Information System and Miscellaneous Proposals" and Consolidation of Hazardous Materials Regulations and Miscellaneous Proposals," respectively. In the final paper of this morning's session, Barry Devine will discuss the proposed Hazardous Information System, specifically with respect to the incorporation of certain provisions for radioactive materials.

As of this writing, another major notice of proposed rule making is nearing completion, to be issued hopefully in the near future. I shall for purposes of brevity refer to it as the "Part 187" notice. It will be the proposal to adopt the substance of the 1973 revised IAEA regulations into the U.S. regulations. During the competent authority panel discussion later this week, we shall hear about the status of adoption of the 1973 revised IAEA standards into the domestic regulations of several other nations.

In the following notice, which I referred to as the "Part 187" notice, the provisions of the DOT Hazardous Materials Regulations which relate to radioactive materials will be inserted into a new Part 187 in conjunction with a new Plan of the Regulations which would generally

reorganize the regulations as follows, concurrently with the final amendments in the aforementioned Dockets HM-103 and HM-112:

Part 170 General Information, Regulations, and Definitions.

Part 171 Rule-Making Procedures of the Hazardous Materials Regulations Board.

Part 172 List of Hazardous Materials and Hazardous Materials Communications Regulations.

Part 173 Shippers; General Requirements for Shipments and Packagings.

Part 174 Carriers by Rail.

Part 175 Aircraft Operators.

Part 176 Carriers by Water.

Part 177 Carriers by Public Highway.

Part 178 Shipping Container Specifications.

Part 179 Specifications for Tank Cars.

Part 180 (Vacant).

Part 181 Explosives.

Part 182 Gases.

Part 183 Flammable and Combustible Liquids.

Part 184 Flammable Solids.

Part 185 Oxidizing Materials and Organic Peroxides.

Part 186 Toxic Materials, Irritating Materials, and Etiologic Agents.

Part 187 Radioactive Materials.

Part 188 Corrosive Materials.

Part 189 Other Regulated Materials (ORM), A, B, C, and D.

Parts 181 to 188 would contain the specific packaging requirements for hazardous materials which are now located in Part 173.

Finally, I shall summarize very briefly some of the major changes included in the 1973 revised IAEA transport regulations. These can be considered as indicative of the major thrust of the changes which can be expected to be proposed in the forthcoming "Part 187" notice by DOT and also the anticipated notice of proposed rule making by the U.S. Atomic Energy Commission to its 10 CFR Part 171.

1. A_1/A_2 System - A new method of classifying and limiting radionuclides for purposes of transport, known as the A_1/A_2 system has been established, replacing the "transport group" system. In place of the seven transport groups currently used, the A_1/A_2 system will provide a list of A_1 (maximum quantity of special form material in a Type A package) and A_2 (maximum quantity of non-special form material in a Type A package) values for some 260 specific radionuclides. Procedures are also prescribed for determining A_1 and A_2 values for unlisted radionuclides and mixtures. Except for certain arbitrary upper limits, the A_1 value is based on external radiation level if the contents were to escape from the package and A_2 is based on the hazard of inhalation or ingestion (except where external radiation is more restrictive and then $A_2 = A_1$). While the replacement of the "transport group system" by the "A_1/A_2" system results in lower package limits for some nuclides, for the majority it results in a relaxation of the previously restrictive package limits which were set on the basis of the most radiotoxic nuclide in a particular transport group.

Effective use of the A_1/A_2 values is also used to set the package content or activity limits for small or exempt quantities, radioactive devices, low specific activity materials, packages for postal shipments,

and for a new category: low level solid radioactive material.

2. Activity Limits for Containment of Type B Packages - In recognition of the fact that "zero loss" for a Type B package is technically unattainable, the 1973 standards provide quantitative allowable releases after the normal and accident damage tests. These limits are expressed in very small fractions of the A_2 values for Type B(U), Type B(M) packages not designed for continuous venting, and Type B(M) packages specifically designed for continuous venting. It is expected that these quantitative standards will enable the various competent authorities to apply the package integrity standards in a more uniform fashion.

3. Establishment of an LLS (Low Level Solid Radioactive Material) Category - Comments by several member states to the IAEA indicated a need for the revised regulations to provide for a category of material, which in itself could not meet the precise technical prescription of LSA materials, wherein dependence for safety is not placed upon any specified packaging requirements. Subject therefore to the prescription of certain minimal packaging requirements of the UN Committee on Dangerous Goods, "LLS" category, directed at low and medium activity solid wastes was established. The parameters for qualification of a material as "LLS" are as follows:

(i) The activity under normal transport conditions is and remains distributed throughout a solid or a collection of solid objects or is and remains uniformly distributed in a compact binding agent and does not exceed an average of 2×10^{-3} A_2/gram and that even under loss of outer packaging the activity loss per

package per week resulting in total immersion in water does not exceed 0.1 A_2 Ci; or

(ii) For objects of non-radioactive material contaminated with radioactive material, the contamination is in a non-readily dispersible form (i.e., all loose contamination removed by thorough cleaning) and the contamination level averaged over 1 m^2 of the surface does not exceed 2 uCi/cm^2 for high toxicity alpha emitters and 20 uCi/M^2 for all other radionuclides.

The new IAEA regulations also clarify and greatly limit the range of materials which might qualify as "low specific activity," using the A_1/A_2 conceptual basis. This limitation led to the establishment of this new category "LLS" or "low level solid radioactive material."

4. Point of Measurement of the Transport Index - Another important change to the 1973 IAEA regulations is the point of measurement of the transport index. Whereas in the 1967 regulations the transport index is defined as the dose rate in millirem per hour at one meter from the center of the package, in the 1973 regulations it has been changed to 1 meter from the external surface of the package. For the U.S. Regulations, very little change in its definition will therefore be required since at the present time the transport index is defined as the dose rate at 3 feet from the external surface of the package. This change in the point of measurement by the IAEA is a reversion to the definition used in the original 1961 IAEA Regulations. It is expected that the practical determination of the transport index will be greatly simplified by this change.

5. Change in Criteria for Use of Radioactive YELLOW II Label - Another change involves an upward revision in the allowable radiation at the surface and at one meter from packages allowed to bear the Radioactive-Yellow II category label. The change allows for an increase from 0.5 mrem/hr at 1 meter (transport index) to a limit of 1.0 mrem/hr and also an increase from 10 mrem/hr at any point on the accessible surface of the package to 50 mrem/hr.

6. Change in Format - A frequent criticism of the 1967 IAEA Regulations has been its format, which has been very confusing and time consuming for the occasional user. As a result of the numerous comments, the IAEA Secretariat has rearranged the format of the 1973 edition into a much more readily usable form and one which facilitates the incorporation of IAEA Regulations into other regulatory documents. In addition, a set of 12 schedules or annexes listing in abbreviated form the requirements to be met for the transport of specified types of consignments have been appended to the regulations. These schedules are intended to aid the user and do not contain any requirements in addition to those appearing in the main body of the regulations.

7. Package Test Procedures - The 1973 revisions to IAEA Regulations are not imparting any major changes to the procedures specified for the assessment of the performance of special form material or for Type A or B packages. The few changes which have been made, however, include:

A. The introduction of a 15 meter immersion test for the criticality assessment of fissile packages wherein reliance is placed on the assumption of no water in-leakage due to package integrity;

B. A requirement that each of the Type A performance tests is to be preceded by the water spray test;

C. The introduction of a penetration test for Type A packages containing liquids and gases; and

D. The grading of drop heights for the free drop of very heavy Type A packages.

In a major decision, the Review Panel also decided to retain the requirement for competent authority certification of "special form" radioactive materials. The U.S.A. had sought vigorously at each Panel meeting to obtain the deletion of this requirement.

8. Deletion of the "Large Radioactive Source" Terminology - It should be noted that with the adoption of the A_1/A_2 concept, the "Type B quantity" and "large radioactive source" as finite specified package activity limits has been deleted. As pointed out earlier, the design assessment of Type B packages has been established on the multilateral/unilateral principal, with respect to the competent authority approval of the package design. The Revision Panel, however, felt that it would be desirable to retain a requirement for notification to other competent authorities for international shipments, prior to those shipments which exceed certain levels, i.e.,

($3X10^3A_1$, or $3X10^3A_2$, as appropriate, or $3X10^4$ curies, whichever is the least).

This "notification" level in a sense parallels the previous large radioactive source threhold.

9. Establishment of an "Advisory Document" - Immediately following the second Revision Panel in October 1971, the IAEA convened a follow-on

one-week "Revision" Panel, with the chartered objective of developing a companion "advisory" document to the basic regulations. This Panel prepared a draft of such an advisory document consisting of;

A. Material of an advisory nature which had been included in the 1967 regulations and removed by the Regulatory Panel;

B. Material in documents considered by the Regulation Revision Panel and referred to the Advisory Panel and accepted by it; and

C. Other material provided by the members of the Advisory Document Panel.

The companion advisory document comprises a very helpful "how to" type document, supplementing the "what to" context of the regulations, per se. Whereas the next regulatory revision of Safety Series No. 6 is not contemplated for 10 years, the IAEA is planning on a review and update of the advisory document at, roughly 2-1/2 year intervals. The first issuance of the advisory document, as IAEA Safety Series No. 37, took place in late 1973 (Document STI/PUB/324).

Many of the IAEA regulatory changes discussed above reflect proposals which had been originated and proposed by the Panel participants from the U.S.A. The major and significant impact of the overall changes is expected to be in two major areas, i.e., the elimination of the "transport groups" and conversion to the "A_1/A_2 system" for classifying radionuclides, and secondly, the reassessment of Type B package designs to quantify the allowable loss and leakage rates after both normal and accident damage tests, as well as to determine whether such packages meet the unilateral or multilateral criteria with respect to international movements.

As of this date, the 1973 revised IAEA standards are well along the process of adoption by most major countries, as well as the two international transport associations, i.e., IATA (International Air Transport Association) and IMCO (Inter-governmental Maritime Consultative Organization).

RADIOACTIVE MATERIAL PACKAGE DESIGN REVIEWS

C. E. MacDONALD

U. S. ATOMIC ENERGY COMMISSION

The abstract and/or paper is printed in Volume 3.

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STATE AND FEDERAL ROLES IN REGULATING THE TRANSPORTATION OF RADIOACTIVE MATERIALS

W. M. Rogers, Jr.

ABSTRACT

The current state and federal regulatory processes over shipments of radioactive materials are discussed. Principal findings and recommendations of a study of radioactive materials transportation by the Western Interstate Nuclear Board are described. The paper also stresses the need for a new atmosphere of cooperation among industry, the states, and federal government in developing a sound regulatory structure.

INTRODUCTION

With accelerated expansion and dispersion of the nuclear industry, state governments are becoming increasingly aware of their growing responsibilities in both the development and regulation of nuclear-related activities. Among these areas of increased interest is the matter of transportation of radioactive materials in and through the states. Much of this new awareness (and in some cases, serious concern or alarm) is the result of observations and experiences of state officials in dealing with nuclear activities. The states have also been sensitized to their emerging responsibilities by the regional nuclear compacts, such as the Western Interstate Nuclear Board (WINB).

Recently, increased pressure by certain environmental and "public interest groups" has caused many states and the federal regulatory agencies to re-examine the current regulatory system.

Unfortunately, some of these groups have shed more heat than light on the subject. An example is a controversial report entitled, "Fallout on the Freeway", prepared and widely disseminated by the Public Interest Research Group - Michigan (PIRGIM)[1], a Ralph Nader - sponsored organization. Other "sensational" newspaper articles describing possible horrendous consequences of radioactive materials thefts, diversions, or accidents in transit as well as accounts of the Delta Airlines and other shipping incidents involving radioactive substances have led to public confusion and heavy lobbying for new, restrictive legislation in some states.

THE WINB TRANSPORTATION STUDY

In an effort to assist its twelve (12) Western member states in implementing appropriate regulatory programs for radioactive materials transportation, the Western Interstate Nuclear Board (WINB) created a task force in May, 1973 to study the problem. The study involved an assessment of the magnitude of the radioactive materials shipping problems in the West, a review of current federal and state regulatory practices, development of guides for emergency response, and preparation of recommendations for state and federal action. As a part of the study, an extensive legal analysis was conducted under the direction of Dr. David E. Engdahl of the University of Colorado Law School in an effort to better understand the constitutional and statutory foundations for state and federal programs for regulating the transportation of radioactive materials. The results of the study were published by WINB earlier this year.[2]

One of WINB's objectives in undertaking this effort is to foster to an appropriate extent uniformity of regulations

by the state governments. The Board recognizes that non-uniform and possibly contradictory regulations by various state and federal agencies may result in unwarranted economic burdens on private industry and, in some cases, may actually be detrimental to public health and safety.

This discussion of state and federal roles in regulating nuclear materials transportation is based largely on the WINB study.

FEDERAL REGULATION

Federal regulation over shipments of radioactive cargoes is shared by several federal agencies. The Department of Transportation (DOT) has general responsibilities for transportation safety by sea, air, motor vehicle, and rail. DOT is responsible for promulgating and enforcing safety standards governing all radioactive material packaging and shipping containers, and the classification, labeling, and marking of all radioactive material packages, except for those types of materials which, by mutual agreement, are administered by the U. S. Atomic Energy Commission.

DOT also implements safety standards for the mechanical conditions of carrier equipment and qualifications of carrier personnel, carrier loading and unloading, handling and storage, and other special controls. Also, DOT requires notification and reporting of accidents, or incidents involving radioactive material package if such occurs during transit. In such instances during shipment, DOT is the lead agency for investigation. Further, DOT is the "national competent authority" in administering the Regulations for the Safe Transport of Radioactive Materials of the International Atomic Energy Agenc

Under the U. S. Atomic Energy Act, the Atomic Energy Commission (AEC) has responsibility for safety in the possession and use, including transport, of byproduct, source, and special nuclear materials. Except for certain exemptions and for the intrastate shipment of such materials subject to control by "agreement states", AEC requires a license for possession and use of these nuclear materials.

AEC also assists and advices DOT in establishment of national safety standards and in review of packaging designs. For the more hazardous types or quantities of radioactive materials, AEC also develops standards for container design and preformance. Moreover, AEC requires reporting of incidents involving radioactive material packages if such occurs prior to delivery to a carrier or after receipt.

Three other federal agencies have some jurisdiction over transporting radioactive materials. Postal shipments are regulated by the U. S. Postal Service under Title 39, Part 123, of the Code of Federal Regulations. The Interstate Commerce Commission (ICC) exercises authority as to the economic aspects of land shipments through the issuance of operating authorizations to carriers and through control of freight rates. The Civil Aeronautics Board has similar authority over air shipments.

The authority for these regulatory programs at the federal level is derived from numerous statutes, the most significant of which include the Atomic Energy Act of 1954, as amended, the Transportation of Explosives Act, the Motor Carrier Act of 1935 (Part II of the Interstate Commerce Act), the Dangerous Cargo Act, and statutes pertaining to railroad equipment.

REGULATION BY THE STATES

Although the federal agencies in many ways dominate the regulatory process for transporting radioactive materials, state governments also exercise a significant degree of control. The specific regulatory practices and agencies involved vary somewhat from state to state; however, a general pattern has developed over the years.

Highway departments regulate gross vehicle weights, axle weights, vehicular dimensions, and other parameters. Special permits must be obtained in most states for overweight or oversize loads. Also, special routing may be imposed on such shipments. For large quantity shipments such as nuclear spent fuel, several states such as Florida, Oregon, and New York have imposed advance notification requirements on shippers. New York also requires state inspection of such shipments.

Regulation of carriers is usually performed by state public utility commissions which certify carriers as to intrastate routes, rates, etc. These state commissions do not exercise much direct control over interstate shipments; however, they do have considerable residium of power to impose legal sanctions in the event any shipment, interstate or intrastate, poses a clear threat to public health and safety. In addition, many state carrier agencies have entered into agreements with DOT under Part 388, 49 CFR, for adoption by the states of DOT safety rules and for joint enforcement programs.

State radiation control agencies exercise authority over possible radiation hazards within the state and, in the case of "agreement states", regulate by licensing the possession, use, and intrastate transport of byproduct, source, and small

quantities of special nuclear materials. Generally, these agencies are equipped to cope with radiation incidents and, thus, must play a large role in any state actions related to transportation of radioactive materials.

Some states have created power plant and nuclear facility siting agencies which in some cases have jurisdiction over both the siting of plants and transportation of radioactive materials associated with these facilities. Thus, these agencies may be authorized to promulgate and enforce certain rules concerning intrastate shipments.

FEDERAL-STATE PROBLEMS IN REGULATING RADIOACTIVE MATERIALS SHIPMENTS

As described above, both the federal government and the states exercise considerable controls over transportation of radioactive materials. Some may argue that the field is already over-regulated and that any additional controls would place an unwarranted burden on interstate commerce that inevitably would be borne by the consumers and taxpayers. Others contend that the burgeoning nuclear power industry and other mushrooming uses of radioactive materials are on the verge of getting out of control and thus new restrictions are needed. Some state governments have recently imposed new regulations and others are considering similar measures. Federal controls are being tightened, but AEC and DOT have opposed more state controls holding that federal statutes and powers preempt the states in many areas.

Moreover, rather than attempting collaboration in studying the problem and in adopting compatible regulations, the nuclear industry, federal agencies, and the states have been embroiled in disputes as to state versus federal powers and other issues.

An example is the testimony by federal agencies opposing proposed new state rules in Oregon last year.[3]

RECOMMENDATIONS

After studying the problem for almost a year with the assistance of many knowledgeable people, the WINB study concluded that the current regulatory system has done a reasonably good job in minimizing incidents involving radioactive materials and the consequences of such incidents. However, the study recommends that additional studies and new cooperative efforts between federal agencies and the states be undertaken in several regulatory areas.

Reducing the Likelihood of Accidents

- A more vigorous program for enforcement of State and federal regulations for radioactive material shipments is needed. An approach to improved enforcement may be for the federal government and the States to consider cooperative programs to establish standards and to provide the resources for effective implementation.

- The requirements for routing of radioactive material shipments should be reviewed by the DOT in concert with the States to determine whether any more definitive guides can be developed on routing through populated areas and to determine the applicability of these requirements to rail shipments.

- In cooperation with the nuclear and transportation industries, AEC & DOT should evaluate the feasibility of the "unit train" concept for rail shipments of spent fuel and high level radioactive waste.

- It is proper and reasonable for States, which so desire, to impose requirements for notification of certain types of shipments. As a desirable alternative to individual State action, it is suggested that the federal government and the States jointly develop uniform notification requirements.

- DOT and AEC should formulate guidelines on the necessary training and on-board equipment for transporting of the more hazardous amounts of radioactive material.

- Although rail transportation accident rates have historically been lower than those for motor carriers, there has been a recent apparent upsurge in railroad accidents. The DOT should determine whether this recent experience indicates a need for more stringent regulations.

Reducing the Consequences of Accidents

The study concurs with the current federal regulatory philosophy in which competent containment design is the foundation for minimizing possible consequences of accidents. It also recognizes that federal agencies must take the lead role in establishing and enforcing standards for container designs and handling systems on a nation-wide basis.

The study further recommended that:

- Studies be undertaken to develop transportation scenarios upon which emergency response planning can be based.

- DOT and AEC modify existing systems for notification of accidents involving hazardous materials to provide for concurrent reporting to States.

- State radiation emergency plans should be reviewed (and upgraded as necessary) to determine their adequacy in coping

with transportation incidents involving radioactive materials.

- For shipments of sufficient size to warrant exclusive use of motor vehicles or rail cars, DOT and the States should require the driver or conductor to carry a written emergency plan.

- The States should insure that the proper radiation control equipment is on hand in law enforcement, rescue, and fire protection vehicles.

- The States need to be more involved in planning for physical protection of special nuclear material in transit. For example, the AEC and the States need to define how State law enforcement and other agencies should be involved in communication and response measures associated with special nuclear materials which require physical protection in transit.

- Congressional deliberations on the extension or replacement of Price-Anderson indemnification should include explicit consideration of the transportation process both as to high-level and as to other kinds of radioactive materials.

- Radioactive material transportation incidents may necessitate substantial State expenditures in connection with cleanup, decontamination and repair of public as well as private property and facilities. Reimbursement for such expenditures seems not to be adequately assured at the present time. NEPIA, MAERP, NELIA, MAELU should be consulted concerning the degree to which insuring agreements may be made to provide reimbursement for such costs.

SUMMARY

It appears that the federal role in regulating the transportation of radioactive materials must continue to be that of promulgating and enforcing standards for competent container designs and transport systems. While authority over interstate commerce regulation rests principally in the federal agencies, this should not be interpreted as sole authority. Engdahl[4] and other legal analysts[5] contend that states do have considerable powers in this area.

States may be preempted from some regulatory actions; however, present federal law and various court decisions appear to leave substantial room for state regulation. Where State regulations are designed to supplement and reinforce federal controls, constitutional questions of State powers should not be considered a substantial reason for state failure to deal with radioactive material transportation problems. Thus, States should adopt policies which allow the flow of radioactive materials in commerce consistent with traditional State responsibilities to safeguard public health and safety.

As a subject of national concern, regulating the transportation of radioactive materials should involve federal standards consistent with economic progress as well as public health and safety. At the same time, it is necessary and desirable that state governments be involved in the regulatory process, since they and their citizens are most directly affected by any transportation accidents and since much of the burden in coping with such incidents falls squarely on state and local governments.

Clearly, the time has come for a new atmosphere of cooperation in resolving jurisdictional issues and for getting on with the job of building a regulatory structure that recognizes the legitimate interests of industry, the states, the federal government, and above all, the citizen and his welfare.

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THE STATE OF STATE REGULATIONS

by
William A. Brobst

ABSTRACT

Public fear of nuclear shipments often goads State regulatory agencies into action, proposing restrictive regulations on nuclear shipments, especially spent fuel and waste. Different agencies within a State have overlapping jurisdictions and regulations. Some of the popular rules include shipment notifications and approvals, unit trains, escorts and monitors, routing and speed restrictions, limits on hours of travel, driver training, inspections, and emergency response. Similar restrictions are rarely imposed on other hazardous materials with much worse safety records. Some restrictions are counter-productive. Most are passed in haste, based on emotional reactions rather than factual studies. Rules vary between States, causing confusion and interference with free flow of commerce. Compliance is often expensive, without a commensurate gain in safety. Federal Government shipments are exempt from State regulations. The State regulations themselves are often unlawful, being prohibited in effect by the Atomic Energy Act, but they still exist, posing problems and challenges for both shipper and carrier.

Recommendations are made for appropriate State action other than regulatory control of shipments: surveillance, enforcement, investigation, accident response, and participation in developing Federal regulations of AEC and DOT.

WHAT'S THE PROBLEM?

One aspect of nuclear energy that strikes fear into the hearts of many is the transportation of nuclear materials. Fear is the basis of many actions in this matter. A consumer group in Michigan issued a report in January 1974, alleging gross safety hazards and deficiencies in safety analyses by the AEC.[1] In May 1974, the Joint Committee on Atomic Energy set up a blue ribbon panel on nuclear shipments, based largely on a few minor airline incidents and the potential for theft of shipments. People are writing their Congressman. Why? Is it <u>really</u> that bad?

Fear brings about reactions that are not always logical or rational, but the fear is real. Concern over nuclear shipments is not always logical or rational, but it, too, is real. Those of us in the nuclear fraternity make a grievous error when we tend to pooh-pooh the public's concern, and we spend a great deal of our time trying to counter the resultant reactions. We must change our outlook, and deal with this fear and concern on a reality basis, not a theoretical basis.

One contributing factor for the intense fear is the high degree of visibility of nuclear shipments. Reactors are located "somewhere up the river." Burial grounds or storage facilities are out in the boondocks,

far out of sight. But nuclear shipments move on the same highways that we drive to work, and move on the same railroads that move our orange juice and overalls. Nuclear shipments are visible. The lonesome train whistle in the night takes on a new meaning when you're scared of what's being moved. And the public is not convinced by the claims of the Federal Government and the nuclear industry that the nuclear packages will stand up to any credible accident, and that no catastrophe will ever come from such shipments.

We are all part of the public. We all expect our Governments -- Federal, State, local -- to protect us against those things which we can't individually protect ourselves against. Foreign enemies, industrial polluters, criminals, and nuclear shipments. When those who purport to be knowledgeable claim that the Governments aren't doing their job, we get concerned. The Constitution provides for a balance between Federal and State Government responsibilities for protection of the public. Each has its rightful role to play. But when a hue and cry arises over questions of whether the Federal Government is doing its job, then pressure grows for the States to take action to solve the "problem." The news media get wound up and start writing stories. Nuclear critics press for a moratorium. Shippers and carriers begin fretting over possible interference with their operations. When the pressure gets great enough, the regulators -- Federal, State, local -- swing into action, creating a plethora of purported regulatory solutions to the "problem."

"DO SOMETHING"

When State regulators and legislators propose action, often in haste, they act out of a genuine concern for public safety. They feel their duty keenly. They're being pressured to "Do Something!" Do anything, but do something! The political pressures on the State regulators are unrelenting. If they don't do something, then they're obviously not doing their job. Their haste is understandable, but problematical.

Although State regulators are often accused of trying to establish jurisdiction over questionable areas, and of trying to set up empires for themselves, I think that most of the people are sincerely pursuing their job of providing for public safety. If we fight them, we'll find ourselves engaged in numerous long battles, for there are 50 States, and only one nuclear industry and only one Federal Government.

Granting then, that the State Government people are sincere in their actions, it is interesting to note some of the conflicts within the State Governments. A recent report of the Western Interstate Nuclear Board[2] outlined the conflicts quite clearly. Most nuclear regulations are passed by a State radiation control agency. But when that agency makes a rule applying to a trucker, it is often lost in the wind, for the carriers don't read the health department regulations. They read the State Public Service Commission rules, for that's the agency that regulates the carrier. It may be a Public Utility Commission in some States. Those agencies have a great deal of power over carriers, but rarely get involved in regulating nuclear shipments. And the shippers rarely read their

regulations. Interestingly enough, the PSC's or PUC's are usually the ones who adopt the U.S. Department of Transportation Hazardous Materials Regulations for intrastate shipments, but they leave the regulation of nuclear shipments to the State health departments, who are not often known for their transportation expertise. On top of that, we have seen many times that the two agencies in a State don't even check with each other before passing transportation rules! Not only that, they don't even tell each other what they've done! No wonder shippers and carriers get confused as to which rules to follow. Now, add to that the fact that such things as highway load limits and speed limits usually come forth from the State Transportation Department or Highway Commission. How on earth can the shippers and carriers be expected to keep up with these conflicts, and for 50 States?

The United States has been participating for years in the development of international standards for nuclear shipments. The United States' efforts in this area are based upon the principle that individual countries should try to minimize occasions where national regulations would diverge from the international standards. A few years ago, the U.S. Department of Transportation modified its regulations to bring them into essential conformity with the international standards in order to facilitate both domestic and international traffic in radioactive material shipments. Parallel efforts are under way, through the United Nations, to provide similar harmony in the regulations for transportation of other hazardous materials. Shippers in Great Britain, France, or Canada can hardly be expected to be aware of both the international standards and also the individual standards of 50 states within just one of the member countries. For that matter, it would be a great burden on shippers in other states in the United States to have to maintain up-to-date regulations of all of the different states through which they might be shipping radioactive materials.

COMMON STATE REGULATIONS

Over a period of several years, we have seen a number of proposals come out of the States which are claimed to provide significant increases in public safety in nuclear shipments:

1. Blanket prohibitions against certain nuclear materials moving through the State, or portions of the State, to completely eliminate the problem.
2. Advance notification to the States of certain shipments, mostly spent reactor fuel and nuclear wastes, to allow the States to "prepare" for the shipment.
3. Prior approval of each individual shipment, to provide the State with positive control over the shipment.
4. Inspection of shipments, either on the shipper's premises, or on the vehicle at State borders or weigh stations, to ensure compliance with the rules.
5. Minimum track standards for rail shipments, often conveniently excluding controversial nuclear facilities, to ensure relative

freedom from derailments.

6. Exclusive use of the vehicles, to prevent any other cargo in the transport vehicle.

7. No transfers of packages from one vehicle to another within the State, to minimize the chance of a handling accident.

8. Special or unit trains only, with no other freight, to allow total control of the shipment and to eliminate accident exposure to other hazardous materials. Sometimes, a prohibition of spent fuel casks on passenger trains.

9. Pilot trains or pilot trucks/cars, to precede the shipment to ensure no roadbed or road hazards, with appropriate communications between vehicles.

10. Routing restrictions or requirements, to avoid highly populated or dangerous areas.

11. Escorts -- police, shipper, carrier, Government -- in separate vehicles (as is now required for certain strategic shipments for national security purposes), to take necessary action in the event of a crash.

12. Speed restrictions, to reduce the severity of any accident which might happen.

13. Specific hours of movement, to avoid rush hours.

14. Accompanying monitoring personnel, with radiation detection equipment, to measure levels periodically en route and to monitor the area after an accident.

15. Special training of drivers so that they know what action to take in case something goes wrong.

16. Possession of emergency instructions and plan of route by truck drivers or train conductors, so that proper action will be taken in case of an accident.

17. Truck drivers and train conductors to carry a copy of the State regulations, so that they will know just what they have to do.

18. Limitations on radiation exposure of transportation workers, (often classing them as radiation workers) in order to restrict their exposure.

19. Notification by carrier/shipper to State authorities of accidents, to assist in State response and investigation.

20. State inspectors to visit nuclear facilities and carrier terminals, and to conduct en route inspections, in order to assure compliance with State and Federal rules.

21. State enforcement of Federal regulations, to increase the punitive coverage beyond that provided by the Federal Government.

22. Accident investigation by State officials, in order to find causes and to develop remedies.

23. Emergency response capability by State officials, in order to quickly respond to incidents.

A detailed discussion of each of these common proposals is found at the end of this paper, along with the obvious advantages and disadvantages of each. On the surface, most of them seem to have some tangible value for safety. But do they really?

PERSPECTIVE

It is interesting to note that, in most cases, the States proposing restrictive rules on nuclear shipments do not have correspondingly strict rules on shipments of other hazardous materials, such as chlorine, propane, or explosives. Nuclear shipments are usually singled out, and, if one looks at the actual risks involved, he must conclude that the basis is emotional, rather than factual. Now, in those cases where some regulatory body rules against all hazardous materials, then nuclear materials fall right in line with the others, and rarely are any complaints heard about discrimination against nuclear shipments. Even the Federal Government, which claims a Constitutional exemption from such regulations, usually goes along without question. But, where discriminatory rules are passed or proposed against nuclear shipments, and not against other hazardous materials, which are responsible for hundreds of deaths and injuries each year, people start taking sides and gird their loins for a fight.

Because so many of the antinuclear proposals are not based on fact, or do not consider all aspects of the problem, we often see counter-productive proposals. For example, shipments are sometimes required only to move in the wee hours of the morning on weekends, to avoid rush hours. Yet the highest percentage of fatal accidents occur during just those hours.[3] Truckers normally avoid rush-hours anyway, since they are paid to cover miles of highway, not sit in traffic jams. Local agencies often require circuitous routing on side roads, which will produce more accidents than if the safer Interstate highways are used.[4] Speed restrictions often result in trucks or trains becoming obstructive hazards, thereby increasing the potential accident rates. Other rules have the appearance of increased safety, but not the real effect -- they just cost more.

THE PANDORA'S BOX OF STATE RULES

There seems to be a widespread feeling amongst those who are regulated that there are too many rules already, and the rules are too complicated. If the various regulatory systems are made even more complex by a multiplicity of different State rules, one likely result will be less compliance with more rules. Simple rules are the easiest to follow and the easiest to enforce. But when each of the States heads off in a different direction with different rules and different State agencies involved, the result can be chaos, as we have seen in other State regulatory areas, such as highway load limits, licensing of drivers, and road traffic signs.

The question of need for consistency in State rules has many facets. There needs to be a high degree of uniformity between States in transport regulations, in order to avoid restrictions on the free flow of commerce between States. There needs to be a high degree of consistency between

State and Federal regulations to avoid catching shippers and carriers in the crack between the two sets of regulations. Both interstate and intrastate shipments should be subject to the same basic safety rules. Jurisdictional squabbles between agencies within a State will only result in less safety, not more. Conflicts between Federal and State rules cause confusion, not order.

The Constitution of the United States effectively prohibits States from enacting rules which would interfere with the free flow of commerce between States. The Federal right and responsibility to regulate in interstate commerce is clearly stated in the commerce clause of the Consititution and its interpretations over the years. The supremacy clause in Article VI effectively prohibits a State from passing a rule, compliance with which would put a person in violation of a Federal rule. Also, States can clearly regulate intrastate activities. The more interesting questions arise when one considers the rights of the States to co-regulate with the Federal Government, or to pass regulations in addition to the Federal rules.

The question of practicality of regulations must also be considered. When rules are proposed or passed in haste, they are often unsupported by studies of the overall effects of their compliance. It's easy to say "Don't move that stuff through my back yard --- send it somewhere else." But if it has to move, it has to move through somebody's back yard. A simple rule that would appear to provide for more safety may be so difficult to comply with that it might cost a million dollars to prevent a thousand dollar accident. That's not good safety engineering. It's popular these days to require an applicant for a license to perform a cost-benefit analysis to support what he wants to do. But the same cost-benefit analysis requirement should be applied to rule making. Before imposing a rule that would keep someone from doing something, or restricting how he would do it, the total people/system cost-risk-benefit of that rule should be determined. Increased safety always carries a price, and responsible State, or Federal, regulators should be aware of what that price is, and who's going to pay it, or else they're not doing their job. Too often we see safety regulations that apparently disregard the total cost of the rule to the public.

GOVERNMENT SHIPMENTS

The applicability of State regulations to shipments of nuclear materials by, for, or to the Federal Government is another interesting aspect of the problem. A well-known case[5] involving State driver's licenses and postal workers resulted in a U.S. Supreme Court decision that is still generally considered to control. The principle of that case is that no State may regulate any activity undertaken by the Federal Government itself. Subsequent cases have extended this immunity to contractors employed in doing business for the Federal Government, as well. From this, one may conclude that shipments for which the AEC, or one of its prime operating contractors, or the Department of Defense, is either the shipper or carrier, are not subject to State regulations. Indeed, the AEC recently notified the States of Oregon and Florida that their newly imposed State regulations "would be an unwarranted restriction on Constitutional immunity inasmuch as such State requirements for notification, routing,

and approval of shipments of radioactive materials in interstate commerce constitute an impediment to the free flow of such commerce." AEC cannot guarantee to any State that it will comply with State rules on transportation.

FEDERAL PREEMPTION OF STATE REGULATORY AUTHORITY

An even more critical aspect of the problem is the question of how much authority the States have at all to regulate the transportation of nuclear materials. The AEC made its feelings known at a public hearing in Oregon on May 21, 1973, in which it testified that States are prohibited from regulating the radiological health and safety aspects of shipments of by-product, source, and special nuclear material unless the State is an Agreement State under Section 274 of the Atomic Energy Act.[6] Even if such an AEC-State agreement has been entered into, the State is still prohibited from regulating shipments of special nuclear material in quantities sufficient to form a critical mass, unless the AEC specifically delegates its Federal authority to do so to the State. So far, the AEC has made no such delegations. This means that no State may lawfully regulate shipments of such things as new or spent reactor fuel elements or plutonium. The congressional intent to preclude State regulation of this type seems clear.[7]

How about those States that have already passed such regulations? If it is unlawful, how could they do it? Their regulations have been enacted, and are on the books. The Federal Government says that its shipments aren't subject to State law, but how about licensees? Do they have to comply with State regulations on nuclear shipments? Well, whether a State regulation is lawful or not, if you don't follow it, you might be arrested or cited by State authorities. We expect that any conviction arising from violation of such a State regulation could be successfully appealed through the courts, just as in the case of Northern States Power vs. the State of Minnesota,[8] which involved the authority of a State to impose additional restrictions on reactor effluents above and beyond those already imposed by the AEC under the Atomic Energy Act. In that case, the U.S. Supreme Court upheld the U.S. Court of Appeals opinion that State regulations "might conceivably be so overprotective in the area of health and safety as to unnecessarily stultify the industrial development and use of atomic energy for the production of electrical power." That case is very instructive as regards to the current problem of State regulation of nuclear shipments.

The Atomic Energy Act is not alone in recognizing the need for Federal preemption. In the Railroad Safety Act of 1970,[9] we find a concise declaration by the Congress for the regulatory scheme which preempts the field of railroad safety: "The Congress declares that laws, rules, regulations, orders, and standards relating to railroad safety shall be nationally uniform to the extent practicable. A State may adopt or continue in force any law, rule, regulation, order, or standard relating to railroad safety when necessary to eliminate or reduce an essentially local safety hazard, and when not incompatible with any Federal law, rule, regulation, order, or standard, and when not creating an undue burden on Interstate commerce." (underlining added) This Congressional declaration is also evident in the statute -- creating the DOT and in the implementing regulations issued by the Secretary of Transportation.

This subject was discussed at some length at a meeting of the Southern Interstate Nuclear Board at Oak Ridge in May 1973.[10] At that meeting, the Attorney General of the State of Arkansas pointed out the shakiness of the State's claim to regulatory jurisdiction over nuclear shipments.[11]

What surprises me is that, during the time since NSP vs. the State of Minnesota, no one in the nuclear industry has yet challenged any of the existing State regulations through the courts. Perhaps that is because the nuclear industry is not yet sufficiently well organized to take such a stand.

WHAT SHOULD BE DONE?

Clearly, there is public concern. The concern is based on fear -- fear of the consequences of an accident, and fear that Governmental authorities aren't doing an adequate job to protect the public. Something has to be done. What alternatives does a State have to passing more regulations?

When one considers the scope and complexity of the existing Federal regulations on nuclear shipments, it is apparent that there is already an extensive body of regulations. No State should feel compelled to start regulating nuclear shipments on the basis that there is a shortage of regulations. If anything, there is an overabundance of them.

What can the States do? To me, the first step would be to adopt the U.S. Department of Transportation Hazardous Materials Regulations,[12] in toto, for all intrastate transportation activities. Nearly half the States have done this already. This provides a framework of regulating hazardous materials shipments, both intrastate and interstate, within a State, to which no one can logically object, and for which there has already been great support from the Federal Government and numerous recommendations of industrial and intergovernmental study groups. Adoption by reference is the most practical way to go, rather than verbatim repetition or paraphrasing. Since the State regulations may not conflict with Federal regulations, and because the Federal regulations are revised several times each year, frequent revisions to State regulations would be necessary if they were quoted verbatim. In the interim, there would in fact be conflict due to the time lag involved. Paraphrasing involves the inherent hazard of inadvertently changing the meaning or scope of the original regulations. Shippers and carriers should not have to carefully examine the regulations of many different States in order to detect possible subtle differences between them and the Federal rules.

Next, the States can take a more active role in helping to develop adequate Federal standards to apply throughout the Nation on a uniform and totally pervasive basis. There are lots of Federal regulations, but are they adequate? If not, change them. But it's poor economy of manpower and resources for individual States to perform the studies and investigations necessary to solve the problems. Why do it 50 times, when by pooling efforts, we can do it once, and better? Nuclear shipments present no more serious a hazard to the citizens of Florida or Oregon than they do to the citizens of Wisconsin or Maryland. If it's a problem in one State, it's probably the same problem in all the other States. National problem solutions are

the only sensible answer. Perhaps threatened State action is a good prod for the Federal Government to sit up and take notice, and perhaps the Feds should be actively seeking State input more than at present. But that's the avenue that's most likely to lead to a balanced national safety program for nuclear shipments.

Another very fruitful area for State action would be in making sure that the existing regulations are followed. The AEC believes that the needs and concerns of the people of any State and its neighbors would be better served by having the State employed in a cooperative working relationship with the AEC and the Department of Transportation in an inspection-enforcement-educational effort to assure safety in the transportation of all nuclear material in the States. Cooperative efforts of State and Federal agencies having an interest in nuclear shipments would result in the most efficient and safe system possible. It would provide for improved communication and cooperation amongst shippers, carriers, and receivers, and State officials, so as to provide a knowledgeable, coordinated response to any accident or incident which might affect State residents. The AEC has negotiated such agreements with several States, and is working on several more. The Department of Transportation has made even more progress in negotiating State agreements for the enforcement of the DOT Motor Carrier Safety Regulations by State authorities.

Such Federal-State agreements are likely to be much more productive than just relying on Federal inspectors and enforcement actions. DOT views a scene of hazardous materials transportation which in 1972 produced 70 deaths and hundreds of injuries. The total nuclear losses in the same year were no deaths and no injuries. DOT concentrates its efforts on those nonnuclear materials and their resultant many millions of dollars worth of property losses. One out of twelve cars moving on the Nation's railroads is carrying a hazardous material, in bulk, in a tank car that is likely to fail in a serious rail accident. One out of ten trucks carries hazardous materials. There are over 20 billion shipments of hazardous materials each year in the U.S. A thousand spent fuel shipments each year is a very small piece of that pie. In 1973, the Department of Transportation recorded about 2,000 cases of hazardous materials leakages in transport; three involved nuclear materials. Tens of thousands of tank cars of chlorine move through the U.S. each year. In perspective, the actual hazard to the public of nuclear shipments is but a mouse in a field of elephants. The mouse traditionally has the role of creating quite a lot of fuss and alarm, but not much real threat. The elephants do the damage. DOT would rather spend its time on the elephants. I can't blame them.

We encourage States to set up emergency response capabilities at all levels -- State, county, borough, city. There have been, and will continue to be, nuclear incidents of all kinds, and governments should be equipped to handle them, just as they set up to handle any other kind of potential health problem. Should there ever be a transportation accident which releases enough material to cause a health hazard, that capability could be brought into play on short notice.

We encourage States to investigate accidents, to determine what went wrong, and how to avoid it happening again. After all, they are Johnnie-on-the-spot, not the Federal agencies, and can provide valuable insight and input into the development of Federal standards as a result of their investigations. We see their role as implementing the Federal standards, not supplementing them.

I strongly recommend better communications within the States in their regulatory programs. The State Radiation Control people, the PUC/PSC people, and the Highway Commission people have got to get together and work together in whatever nuclear shipment safety programs the States set up. We recently attended a "State-wide" meeting of State officials on nuclear shipments, and the State Health Department never even invited the State Public Utilities Commission people to the meeting! We think the States can do much better than that.

One other thing they can do is to draw up some guidelines as to who's in charge of an accident scene in the State. There is often a lot of confusion on that point, whether nuclear materials are involved or not.

SUMMARY

Participation by State agencies in the total regulatory scheme for nuclear shipments is vital if we are to preserve the phenomenally good safety record we have enjoyed so far. Rather than adding to the already complex body of regulations, the States could play a more important and contributive role by becoming active in surveillance and enforcement programs, inspecting to be sure the rules are being followed, and by taking punitive action when they aren't. Rather than becoming embroiled in Federal-State jurisdictional disputes, the States would perform a real public service by establishing capabilities to handle both routine and emergency surveillance duties at the local level. Closer cooperation between Federal and State agencies will result in more compliance and more safety. More participation by State agencies in Federal rule making efforts, and more encouragement by the Feds for State participation, will make for better regulations. All of those things together will provide a national system for nuclear shipment safety which will be reasonable, practical, enforceable, economical, and most importantly, effective.

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9. Federal Railroad Safety Act of 1970, Public Law 91-458, Title II, Sec. 205.

10. W. A. Brobst, "Nuclear Materials Moving in Commerce," Legislative Workshop, Southern Interstate Nuclear Board, Oak Ridge, Tennessee, May 16-17, 1973.

11. J. G. Tucker, "The Constitutional and Legislative Authority of State Government in the Transportation of Radioactive Materials," Legislative Workshop, Southern Interstate Nuclear Board, Oak Ridge, Tennessee, May 16-17, 1973.

12. Hazardous Materials Regulations, U. S. Department of Transportation, Title 49, Code of Federal Regulations, Parts 170-189.

POPULAR REGULATIONS - A LOOK AT THE PROS AND CONS

1. Blánket prohibitions against certain nuclear materials

   Pros: - Ensures absolute safety to citizens against loss or injury
   - Ensures minimum expenditure of State resources on shipment safety programs

   Cons: - Unlawful under Atomic Energy Act
   - Unconstitutional by unduly interfering with free flow of interstate commerce
   - Severely restricts development of nuclear energy
   - Discriminatory by selecting just one of many hazardous materials

   Discussion: This is more of an avoidance of the problem, rather than a solution. It is only occasionally suggested, rarely formally proposed, and never yet passed.

2. Advance notification to the State of certain shipments

   Pros: - Advance planning and rapid identification of potential problems in case of accidents
   - Statistical data on certain shipments for planning purposes and identification of magnitude of problem
   - Better scheduling of en route inspection of shipments
   - Advise shipper/carrier of possible en route problems

   Cons: - Each State has different cognizant agency; hard for shippers to keep up with whom to notify
   - Administrative burden of multiple notifications
   - Incorrectly implies unusual hazard beyond other hazardous materials shipments; causes unnecessary alarm
   - Information required is often not available

   Discussion: On its face, the concept of notification appears to be both reasonable and necessary. In practicality, it is not so simple. It is often not clear just what the State or local authorities would do with this information on individual shipments. They might file it with the thought that some day they might need it for statistical or emergency use, but the total systems cost would probably far outweigh any possible benefit that might derive from the collection of this data. They might take some specific surveillance or investigatory actions for each of these shipments, but this might well establish a precedent which would be unwarranted from a safety viewpoint, and one

which could have serious impact on the transportation of nuclear materials.

Many years ago the Atomic Energy Commission required shippers to notify appropriate State agencies in advance of shipments of spent reactor fuel through the State. After considerable discussion with the States, and some experience under this regulation, the AEC learned that such notifications served no useful purpose to the States and proved to be totally impractical for the shippers to comply with. The practice was discontinued. The basic feeling of the States was that: (1) if the shipments were safe, notification was unnecessary and unwarranted, and (2) if the shipments were not safe, they should not move through the State, notification or not.

The practical aspects of prior notifications of routing and schedules present some serious problems. Most shippers deliver the material to be transported to a carrier, rather than transporting it themselves. The shippers have essentially no information as to the routing the carrier might use. The initial carrier may not have firm information as to the routing successive carriers would use. The same is true of operating schedules. The same problem would apply to air shipments, where routing and scheduling data would be impractical for a shipper to know, particularly in the case of overflights (aircraft passing over a State, but not landing in that State).

Aside from the question of Federal preemption, there does seem to be considerable justification for a State to have advance notification of any shipment which does not meet Federal or State regulations. For example, if a spent fuel shipment were being made in a cask which for some reason had unusually high radiation levels or did not meet the Federal cask integrity standards, someone in the State ought to know. This is akin to rules for handling overweight or overheight shipments.

3. Prior approval of certain individual shipments

Pros:
- Provides State with positive control over routing, hours of movement, police escorts
- Asserts State authority over shipments
- Provides State with opportunity to review packaging and transport conditions

Cons:
- Unlawful under Atomic Energy Act
- Unconstitutional by interfering with free flow of commerce
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring approval; causes unnecessary alarm
- Duplicates AEC approval of shipments
- Each State has different approval criteria; hard to keep up with
- Potential delays due to communications breakdowns

Discussion: The same points apply here as for item 2 on notification. In addition, such a requirement is a clear infringement on Federal prerogatives.

4. Inspection of shipping/receiving facilities, packages, and vehicles

Pros:
- Determine degree of compliance with regulations by first-hand observation
- Ensure higher degree of compliance by shippers/carriers because of knowledge of inspections ("the policeman is watching"); DOT effort is minimal

Cons:
- Could result in delays at borders or weigh stations, thereby interfering with free flow of commerce, causing increased exposure to theft or sabotage, and causing small, but calculable, increases in radiation exposures of persons in area.

Discussion: It should be noted that Federal regulations include comprehensive requirements on shippers for proper preparation of containers prior to shipment. AEC supports inspection by State officials, provided that there is adequate staffing to avoid delays in transportation. Scheduling of inspections would have to consider the practical difficulties of irregular hours of movement on weekends and holidays, last-minute route and schedule changes, and locked trucks or railcars. It might well be impractical for a State to require that each shipment be inspected, particularly if there are large numbers of such shipments moving through a State.

5. Minimum track standards for rail shipments

Pros:
- Ensure relative freedom from derailments by eliminating movements on low-class tracks
- Political tool to exclude rail shipments in and out of certain facilities

Cons:
- Unlawful under Atomic Energy Act and Railroad Safety Act
- Unconstitutional by interfering with free flow of commerce
- No real increase in nuclear safety since likelihood of release is already so low
- Differing State standards would be burden on railroads to keep up with
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring such restrictions; causes unnecessary alarm

Discussion: This requirement usually goes along with requirements for speed restrictions. This presents some conflict, for the higher class track is usually associated with higher speed trains. A

slow train could present a bigger hazard, with an increase in the overall public risk.

6. Exclusive use of vehicles

Pros:
- Prevents loss to other cargo due to leaking packages
- Reduces risk to nuclear packages by eliminating potential environmental stresses from other materials
- Eliminates need for in-transit transfers; speed the shipment through the State

Cons:
- Unlawful under Atomic Energy Act
- Unconstitutional by interfering with free flow of commerce
- Unnecessary, since essentially all shipments of concern move by exclusive use anyway, for operational, economic, or security reasons
- Could result in delays if required
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring such restrictions; causes unnecessary alarm

Discussion: When exclusive use is required for safety purposes, the DOT regulations and AEC Certificates of Compliance will provide for it based on well-known and clear criteria. Under conditions other than those specified by DOT or AEC, no real gain in safety would be realized.

7. No transfers of packages from one vehicle to another

Pros:
- Speeds the shipment through the State
- Eliminates potential handling accidents

Cons:
- Unlawful under Atomic Energy Act
- Unconstitutional by interfering with free flow of commerce
- Potential excess delays in event of vehicle breakdown
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring such restrictions; causes unnecessary alarm

Discussion: While an extra transfer operation certainly includes an additional risk, that risk is already extremely small. What would happen in the case of vehicle breakdown? Delays. Any package designed to withstand severe transportation accidents could surely withstand any conceivable handling incident. Other hazardous materials are not so selectively restricted.

8. Special or unit trains, carrying no other cargo

Pros:
- Reduces exposure to other hazardous materials, especially flammables and explosives, in the event of an accident. In case of single train derailment, would eliminate the exposure.

- Reduces number of movements through State
- Easier for State to inspect and control
- Potential for more direct route, fewer stops

Cons:
- Unlawful under Atomic Energy Act and Railroad Safety Act
- Unconstitutional by interfering with free flow of commerce
- Seriously interferes with rail operations, causing a burden on railroads
- Enormous extra cost
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring such restrictions; causes unnecessary alarm

Discussion: The prime driving force for this requirement is the fact that if an accident did occur, there would be no danger of the nuclear material becoming involved in an explosion or conflagration caused by explosives or flammables on other railcars in the same train. Also, by reducing the number of stops, the shipment could move through the State faster. However, the proposal is patently impractical. Special train service, although it brings some extra revenue to the railroads, is not welcomed by the railroads. Such trains seriously disrupt routine rail schedules and operations. States usually also would require several buffer cars on both ends of the train. This is not an efficient or economical way to run a railroad, and would aggravate shortages of railcars. There is no basis for assuming that pulling empty cars ahead of and behind the loaded cars will add to the safety of the operation, except in the case of head-on or rear-end collisions, which account for only a few percent of the rail accidents. Special trains usually present some extra hazards to the railroads because of the interference with normal operations, and the usual accompanying speed restrictions.

The cost would be astronomical. The least of the costs would be the freight charge for the special train -- perhaps $40,000 per train. Many more casks per reactor would be required, perhaps 4 or 5, at a cost of $500,000 each, in order to make up a train. A higher inventory of fuel would be required in order to provide for the shipment delays at the reactor end. Reactor storage facilities would have to be increased. Casks would sit loaded for days, not moving, at both ends of the trip, waiting to be handled. The costs would be borne by the consumer in his electric bill. This is akin to paying a fantastically high insurance premium to protect oneself against a highly unlikely accident, such as being struck by a jet plane crashing into the ground.

9. Pilot trains or pilot cars, to precede the shipment

Pros:
- Provides check of roadbeds or road conditions prior to passage of the shipment
- Potential for reducing the likelihood of an accident, particularly those caused by driver error

Cons: - Unlawful under Atomic Energy Act and Railroad Safety Act

- Unconstitutional by interfering with free flow of commerce
- Increases the chance of a vehicle accident, by doubling the total vehicle miles involved
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring routing control; causes unnecessary alarm
- Increase in administrative burden without significant gain in safety

Discussion: Pilot cars are normally required for those loads and movements over State highways which might interfere with the normal flow of traffic, or which, by reason of size or speed, would significantly increase the probability of an accident if the public was not warned that such a load is approaching, or which might entail some unusual public hazard requiring advance warning. This is simply not the case with nuclear shipments. The shipments move readily with the normal flow of traffic.

Derailments are often caused by faulty cars, none of which would likely be found either on a pilot train or special train. The pilot train would therefore not detect those potential accidents. Bad switches might be detected by a pilot train, or by the preceding regularly scheduled train.

10. Routing restrictions

Pros: - Allow shipments to be routed around strike or riot areas, or dangerous road conditions to protect the shipment from the environment
- Allow shipments to be routed around highly populated areas, tunnels, bridges, to protect the area from the potential hazards of the shipment
- To make it easier for State inspectors to schedule en route shipment inspections

Cons: - Unlawful under Atomic Energy Act and Railroad Safety Act

- Unconstitutional by interfering with free flow of commerce
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring routing control; causes unnecessary alarm
- May put carrier in violation of ICC-designated routing
- Potential delays due to circuitous routing
- Potential increase in hazard due to use of less-safe roads and longer routes

Discussion: The current Federal regulations (49 CFR 397.9) require truck drivers to avoid populated areas, tunnels, narrow streets,

etc., where practical, and precludes operating convenience as a justification for determining practicality. It might well be appropriate for State agencies to work with DOT to improve that language to provide a more definitive guide, since the regulation presents a problem because of its general nature. For example, is it preferable to travel through a metropolitan area on multi-lane divided highway or to circumvent it on poorer roads through suburban areas? Also, to what extent does this philosophy of avoiding heavily populated areas apply to rail transportation? Rail shipments are not so easily diverted. Routing of aircraft flights is solely within the jurisdiction of the FAA and cannot be controlled by the States.

Probable routing of shipments is often listed in environmental impact statements. Shipments should not be restricted solely to those routes, however, since the safety standards of the AEC and DOT do not rely on routing restrictions for assuring safety in transport. The regulations of the States impose controls on weights of loads on roadways and bridges. Also, in some cases, cities and bridge, tunnel, and turnpike authorities place restrictions on travel at specific periods of the day or night and over certain sections of routes; those restrictions may also affect the choice of routes.

Routes for nuclear shipments could be selected to avoid populous areas, special risk areas due to local road or rail conditions, areas of high accident frequency, extremes in heat or cold, high elevations, and delays. Such restrictions could reduce the probability of an accident occurring in many cases. However, if shipping distances were increased thereby, the accident frequency could be increased. Carriers normally avoid such conditions anyway in order to avoid delays in their shipments. A truck caught in a traffic jam earns no money for its owner. Anyway, most rush hour accidents are minor; its hard to drive 70 mph in a traffic jam.

Requiring nuclear shipments to move over less populous routes would reduce the radiological consequences of accidents which might cause a release of material from the package. However, the overall public risk from the common aspects of traffic accidents far overshadows the risk from nuclear causes, and if increased mileage results from the routing requirements, it would probably increase the total deaths and injuries from those shipments due to those common causes. Such special routing would have only a very small effect on radiological risk in any case.

There are no specific Federal regulatory requirements with regard to routing of hazardous materials shipments by rail. Severe rail accidents involve high speeds and frequently occur because of faulty roadbeds or equipment. Roadbeds connecting centers of population are used more frequently than off-line roadbeds, and generally are better maintained for that reason. Further, rail accidents occurring inside city limits are unlikely to be as severe as those outside the city since speeds are restricted. City accidents are better handled because emergency equipment is more readily available. For these reasons, it appears likely that for rail shipment, the frequency of

severe accidents may be greater for shipments made on routes chosen to avoid centers of populations that if those same shipments were made on main-line routes between population centers.

11. Escorts in separate vehicles

Pros:
- The presence of escort vehicles may help to prevent accidents
- Escorts are not likely to be involved in the same accident as the cargo, and so would be available to keep people away from accident scenes, make notifications, direct traffic, perform radiation monitoring, etc.

Cons:
- Unlawful under Atomic Energy Act
- Possibly unconstitutional by interfering with free flow of commerce
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring escorts; causes unnecessary alarm
- Potential delays if State provides escorts
- Potential for increased accidents due to increase in total vehicle miles
- Likelihood of need is so remote
- Local civil officials can perform same function

Discussion: Escorts in separate vehicles could be required to accompany the shipments. They could be equipped to monitor the area and to take corrective action in case of an accident. Escorts who survive could assist in control of any accident, but probably could not reduce the immediate effects of any releases of nuclear materials. It does not appear likely that a requirement for escorts can be justified in view of the low probability of a severe accident occurring in which an escort would be effective.

Although a highway escort might reduce the already small probability of the shipment vehicle being involved in an accident, a rail escort could not serve that purpose. A separate escort vehicle has a probability of being involved in an accident at least equal to that of the shipment vehicle. Because injuries occur in 13% of all motor vehicle accidents, the increased number of injuries due to highway accidents involving the escort vehicle outweighs the small benefit to be gained from its presence.

For rail, the escort involves a practical problem of passenger transportation. There are no facilities available in a freight train for transporting persons other than members of the operating crew. If the escort person is to accompany the special train at a position in the train to "watchdog" the nuclear shipment, he would necessarily be placed in close proximity to the shipment. He is thereby likely to be involved in any accident involving the shipment.

Even if located distant from the shipment car, it would require the placement of passenger-carrying equipment, such as a caboose or passenger car somewhere in the train. Cabooses are in short supply on the railroads. Few passenger cars are available for this purpose. The cost would be significant.

It is not clear that any escort would have the authority to do anything. Normal accident response procedures would take precedence -- divert traffic, fight fires, keep onlookers away. The escort's usefulness to do those things is very limited. A better way of providing these controls is through State emergency response capability and its public safety officers who have the authority to assume control and to act in the event of an accident.

Many of the comments on the use of pilot cars also apply here.

12. Speed restrictions

Pros:
- Reduce both the consequences and the likelihood of an accident

Cons:
- Unlawful under Atomic Energy Act
- Unconstitutional by interfering with free flow of commerce
- Incorrectly implies unusual hazard beyond other hazardous materials shipments not requiring special speed restrictions; causes unnecessary alarm
- Delay in shipment transit time; increase in potential for theft or sabotage
- Increase in hazard by obstructing traffic flow
- Interference with routine railroad operations and resultant cost

Discussion: Statistics clearly show that fewer accidents happen when all traffic is flowing smoothly at the same speed. This is true of both rail and highway. Slower speeds will reduce the damage to shipments, but they are designed to withstand accidents at higher speeds than are allowed by law. In any event, the decreased damage is offset by the increased likelihood of an accident, in terms of overall public risk.

13. Specific hours of movement

Pros:
- Reduce likelihood of accidents
- Minimize interference with high volume traffic if an accident occurs; including potential losses to toll roads or bridges/tunnels
- Make it easier for State inspectors to schedule inspections

Cons:
- Unlawful under Atomic Energy Act and Railroad Safety Act
- Unconstitutional by interfering with free flow of commerce
- Discriminatory against nuclear materials

- Administrative burden on carriers
- May tend to increase serious accidents by diverting traffic into early morning weekend hours when fatality rates are high

Discussion: The primary reason given for this restriction is to minimize the possible interference with traffic flow, particularly where there might be a significant revenue loss to the State. It would eliminate regular train service, since trains operate on set schedules.

14. Accompanying monitoring personnel, with radiation detection equipment

Pros: - Instant monitoring service in the event of an accident

Cons: - Unlawful under Atomic Energy Act
- Unconstitutional by interfering with free flow of commerce
- Discriminatory against nuclear materials
- Low likelihood of need is a waste of manpower
- Potential exposure to injury from accidents

Discussion: Many of the comments on the escort requirement apply equally here. Monitors aren't needed unless there's a serious accident. A truck driver can't usefully serve as a monitor since he's likely to be killed or injured in a serious accident. A monitor is useful only if he's in a separate vehicle.

Most local public health officials have radiation detection equipment. It would be better to rely on them than to rely on a monitor who is likely to be involved himself in the accident.

15. Special training of drivers

Pros: - Driver can take preplanned and knowledgeable action in event of accident
- Special training may result in more care on part of driver, thereby avoiding accidents

Cons: - Unlawful under Atomic Energy Act
- Discriminatory against nuclear materials
- Already required by DOT regulations

Discussion - DOT regulations (49 CFR Part 177) already require each employer to instruct and train his drivers and other personnel in the handling and care of hazardous materials shipments. The same regulations require the carrier to tell the driver what he is carrying, and for the driver to carry a copy of the shipping papers describing the cargo. Other regulations (49 CFR 397) require the driver to be aware of the nature of the hazardous materials contained in his

vehicle, and to be aware of the specific DOT driver's rules in Part 397. State regulations to this same end are unnecessary, particularly if the States adopt the DOT regulations for intrastate operations.

16. Possession of emergency instructions and plan of route by drivers or train conductors

Pros:
- Driver can take preplanned and knowledgeable action in case of an accident
- Requires thought and consideration in determining best routes

Cons:
- Unlawful under Atomic Energy Act and Railroad Safety Act
- Discriminatory against nuclear materials
- Already required by DOT regulations

Discussion: DOT regulations (49 CFR 397) requires that drivers be instructed on the procedures he must follow in emergencies. The regulations do not require that he have a set of emergency instructions on his person, and this might be a matter worthy of specific consideration by DOT.

DOT regulations require that dirvers of Class A or B explosives must possess and follow a written route plan. That provision could be extended by DOT to other hazardous materials, such as poisons, flammable gases, nuclear materials, etc.

17. Truck drivers and train conductors carry a copy of the State Regulations

Pros:
- Compliance is more likely if the drivers or conductors have a copy of the rules in their possession
- Enforcement is easier because knowledge of the rules may then be assumed

Cons:
- Unlawful under Atomic Energy Act
- Discriminatory against nuclear materials
- Would require some means of letting all users know of rule changes
- Could require a monstrous sheaf of papers for a long move involving many States
- Drivers and conductors not likely to read them

Discussion: This appears to be a well-meaning but totally impractical requirement. The task of keeping up to date on all the State regulations is burdensome enough, but to require a carrier to obtain

enough sets of the rules to distribute them to all potential drivers or conductors is even more so. Considering the stacks of papers that drivers and conductors are already required to have, the addition of this burden can hardly be justified. The likelihood that they would read through the rules of many States is even more remote than the likelihood that the shipment might cause a public health problem.

18. Limitations on radiation exposures to people

Pros:
- Controls the exposures of carrier personnel and the public
- Measures the exposure for legal purposes (workman's compensation, lawsuits, etc.)
- Provides for penalties in case of overexposure

Cons:
- Unlawful under Atomic Energy Act
- Already provided for in AEC and other Federal regulations
- Control already effected by limitations on packaging in AEC and DOT regulations

Discussion: Federal standards already limit the amount of radiation that various classes of people may lawfully receive. States are not permitted to set any different limits, and duplication of the standards is unnecessary. Guidelines for determining whether transport workers should be considered as radiation workers, and whether they should be provided with dose-measuring systems, already exist.

19. Notification to State authorities of accidents

Pros:
- To allow State to provide emergency response action
- To allow State to investigate and prosecute as appropriate
- To allow State to plan for future resource needs
- To allow State to answer public and media inquiries

Cons:
- Reporting requirements for shippers and carriers are already burdensome and complex
- Requirements usually extend down to insignificant levels, thereby exaggerating problem out of perspective
- Lack of uniformity between States as to whom to report to causes burden on reporters.

Discussion: It is understandable that the States would want to know what accidents are happening within their States. They should know, and have a right to know. The question is how should that be brought about? Agreements already exist between AEC and any State that requests it for the AEC itself to notify an appropriate State agency of any accidents brought to AEC's attention, either by an AEC licensee or contractor, or by a call from a civil official to the AEC asking

for help. Where radiological assistance is needed, the AEC is usually called, although some times the local public safety official may call a State agency. It is hard to conceive of a serious accident involving a nuclear shipment not coming to the State's attention, even without a specific State regulation imposed on the shipper or carrier.

For notifications of more routine accidents or incidents, other reporting requirements apply. All accidents involving leakage, contamination, or unusual radiation exposures must be reported by interstate carriers to the DOT. Such accidents are usually well-publicized and are known to State personnel. Accidents not involving any leakage or breakage of the packages do not have to be reported to DOT, nor should they be under these considerations, for they do not really have public health significance.

It therefore appears that States already are notified of any nuclear shipment accident which has any public health significance, and further reporting requirements are not really necessary.

20. Inspection of nuclear facilities and shipments

Pros:
- To determine the degree of compliance with Federal and State regulations
- To correct, stop, or prevent movements of shipments which do not meet the regulations or which present some unacceptable degree of public hazard.
- Education of State personnel into procedures and practices being followed
- Augments Federal inspections activities to provide better coverage

Cons:
- Too much inspection would tend to magnify problem out of perspective
- Other hazardous materials shipment operations not similarly inspected
- Requires expenditure of State resources

Discussion: Inspection provides a means of finding out just what and how big the problem really is. Federal inspection programs are minimal, and both AEC and DOT support State augmentation, particularly when the State provides for followup in the way of corrective action and prosecution. Joint Federal-State agreements allow a State to enforce the Federal regulations. Although many hazardous materials shipment operations are not inspected, many others are, providing some consistency in philosophical approach to public safety.

21. State enforcement of Federal regulations

Pros:
- To improve the degree of compliance with Federal and State

regulations and thereby enhance public safety

- Augments Federal enforcement activities to provide better coverage

Cons:
- Too vigorous a program would tend to magnify problem out of perspective
- Requires expenditure of State resources

Discussion: This requires agreements between States and AEC and/or DOT. Federal enforcement programs are minimal, and both AEC and DOT support State augmentation. A good program here is likely to be much more effective than just a Federal program, since Federal efforts are aimed more at other hazardous materials which have a bad accident record and present a much higher risk to the public.

22. Accident investigation by State officials

Pros:
- State officials are "Johnnie-on-the-spot" and more immediately available at accident scene
- States can follow up on causes and violations, especially if they have a State-Federal agreement
- State officials more familiar with local situations
- Provide assistance to Federal investigators

Cons:
- Possible jurisdictional conflicts
- State training programs for investigators generally not as thorough as Federal programs

Discussion: There is a lot to be said for investigation of accidents by local authorities. For one, witnesses are more likely to be candid with a local citizen than with a "Fed". Local investigators are more likely to know where to turn to ferret out necessary data than a stranger. Federal investigators are limited in number, and cannot investigate all serious accidents, much less those less serious.

23. Emergency response capability by State officials

Pros:
- Immediate local capability, rather than having to wait for several hours for Federal assistance
- Local public health problems are local responsibility, and should not rely totally on Federal help
- Capability useful for nontransportation-related accidents - hospitals, laboratories, schools, factories

Cons:
- Cost and diversion of duties makes it harder to provide comprehensive training and good equipment

Discussion: This is a fruitful area for local and State activity. There needs to be some coordination of response between the different local authorities, and between State and Federal officials, and this coordination is best done by other than Federal people. An emergency response program should not be aimed at just transportation, but shipment accidents should be handled in a total context of a public safety program.

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THE DOT HI SYSTEM

B. D. Devine

ABSTRACT

The U.S. Department of Transportation has proposed a Hazard Information System which would entail the assignment of a two digit number to each package of hazardous material offered for shipment. The system will result in several changes to labeling, marking, placarding and documentation requirements for radioactive materials. Hazard Information cards, distributed to local emergency organizations, will provide advice on recommended emergency procedures.

One of the most obvious requirements in dealing with an emergency situation is the ready availability of information. Incomplete or inaccurate knowledge of the circumstances surrounding a particular incident can easily result in compounding the dangers encountered by emergency personnel and the general public.

In the transportation field, this particular problem is further enhanced by the conditions peculiar to the system. An accident can very possibly take place in a location which is readily accessible to the public and yet can be sufficiently remote as to render direct communication links difficult to establish. Most frequently, local emergency organizations are called upon to make decisions regarding the appropriate immediate actions. As opposed to the case

involving a fixed facility, knowledgeable individuals with information regarding the materials involved in the incident and the hazards which may be associated with the particular situation are not readily available for consultation. As a consequence, fire fighters and local enforcement agencies are forced to rely on information accompanying the shipment, and in conjunction with advice available from organizations such as CHEMTREC or the shipper, if he can be reached, take emergency action.

Experience has shown that the transmission of information can easily introduce serious errors. As a case in point, a few years ago, a shipment of acetone cyanohydrin, which is classified as a Poison B, was involved in an accident. In the course of identifying the leaking material, the last word of the name of the product was dropped, and the personnel dealing with the incident proceeded on the basis that they were concerned with acetone, a flammable liquid. Both materials are hazardous, of course, but the recommended handling procedures differ considerably.

Several identification systems consisting of numerical designators, color codes, or combinations thereof, have been developed and proposed as a solution to the problem of rapid and positive classification of hazardous materials. After careful consideration of several identification proposals, in June of 1972, the Hazardous Materials

Regulations Board of the U.S. Department of Transportation published an advance notice of proposed rule making under Docket HM-103 proposing a Hazard Information System. In the preamble to that advance notice the Board stated in part:

> "The need for improved hazard communications has been the subject of considerable controversy and debate during recent years. It has been pointed out that the communications requirements of the regulations (1) generally are not addressed to more than one hazard; (2) do not in all instances require disclosure of the presence of hazardous materials in transport vehicles; (3) are not addressed to the different hazard characteristics of a mixed lod of hazardous materials; (4) do not provide sufficient information whereby fire fighting and other emergency response personnel can acquire adequate immediate information to handle emergency situations; and (5) are inconsistent in their application to the different modes of transport. The Board believes there are deficiencies in this area and that certain changes are necessary to provide for the adequate communication of hazards for materials in transportation. However, the Board also believes that it must consider

the complexity of any regulations it adopts in this area and what is to be imposed on the personnel who will be required to follow them."

The salient features of the system as proposed are: (1) the assignment of a hazard information number to each hazardous material which will identify not only the primary hazard to be considered but also any other significant hazardous characteristic to be taken into account; (2) the publication and dissemination of hazard information cards for each number to emergency personnel and local authorities with emergency procedures to be followed (as well as those to be avoided) in dealing with the specific identified material; (3) a coordinated labeling and placarding system to be used, incorporating the hazard information number; (4) a requirement that the hazard information number be entered on the shipping documents; and (5) a prescribed method for the assignment of hazard information numbers.

On January 24, 1974, the Department published in the Federal Register Docket HM-112; Notice 73-9 and Docket HM-103; Notice 73-10. In Docket HM-112, proposed Section 172.101 lists hazardous materials and the hazard information number associated with each. Docket HM-103 is the proposed rule making delineating the marking, labeling, shipping paper and placarding requirements

to be followed. In each case the first digit in the two digit hazard information number is the same as the United Nations class number for that commodity, i.e., for flammable liquids, the first digit would be 3; for poisons, the first digit would be 6; and for radioactive materials, the first digit would be 7. The second digit of the hazard information number is arbitrarily assigned and denotes the significant secondary hazard or hazards associated with the item. For example, acetone will carry the number 30 as a flammable liquid; acrylonitrile will carry the number 35, indicating a flammable liquid which is also a poison; and acetone cyanohydrin, which is highly toxic and is also flammable would be designated as 67.

With the implementation of this proposal, a booklet of about 60 pages, one for each H.I. number, would be published. This document would provide early information on the hazards of each identified regulated material to persons at the scene of a transportation accident. The booklet will be distributed to fire, police, civil defense, and other emergency response agencies as well as to carriers. To the extent practical, these instructions have been developed on a generic basis to provide coverage of a broad range of specific materials with the minimum number of instruction pages. Hopefully, the booklet may prove useful in the development of emergency response plans as

well as providing guidance in actual accident situations.

In addition to changes in labels and placard to accommodate the incorporation of the H.I. numbers, the proposal will require the inclusion of the H.I. number in the shipping paper description and also that all hazardous materials be listed first on shipping documents for mixed loads.

Of particular interest to those in the nuclear field, of course, is the effect of these proposals on the transportation of radioactive materials. The most obvious change is the requirement to use a hazard information number in the transportation activities. The numbers 70, 71, 72, 73, 74, 78 and 79 have been assigned to radioactive commodities. For radioactive materials in general, where radioactivity is the only hazard, the numbers 70 and 71 are used dependent upon quantity and external radiation levels. For quantities within the Type A limits and bearing a radioactive White I or radioactive Yellow II label, the H.I. number 70 is assigned. This particular designation does not require placarding of the vehicle. For Type B quantities with any labels and for Type A quantities requiring a radioactive Yellow III label, the number 71 is to be used. An examination of the proposed hazard information cards in Docket HM-103 will reveal the differences in instructors for dealing with these two materials. It will be apparent that the H.I. number 71 card recommends

somewhat more restrictive emergency procedures. The remainder of the 70 series numbers are assigned to specific radioactive items. For example, number 72 is the designation for solid thorium or uranyl nitrate and the H.I. card recognizes the oxidizing quality of these compounds; number 73 is used for low radiotoxicity nitrate solutions, taking into account the corrosive properties of acid solutions; number 74 indicates pyrophoric radioactive materials and the concomitant hazards; uranium hexafluoride because of its multiple characteristics as a poison, a corrosive material and an oxidizer in addition to its radioactivity has been assigned the specific H.I. number 78; and last, the number 79 denotes extremely hazardous radioactive material. At present, the number 79 is limited to plutonium nitrate solutions, however, we are considering the use of this number for additional materials of high radiotoxicity which require extreme caution when encountered.

There are also several changes in label requirements as a result of these proposals. The most significant is the introduction of the hazard information number block at the bottom of the label, replacing the block previously used for the transport index entry. The transport index will appear as a line entry below the identification of contents and the activity. This change will undoubtedly cause some confusion during the transition period, however, to obtain consistency, it is a necessary change, requiring a justi-

fiable educational effort. To minimize the obscuring effect of the red bars on the information entered by the shipper, the bars have been moved to the same line as the "radioactive" imprint. This modification is expected to improve legibility of the entries. Authorization is proposed to use accepted symbols for radionuclides and to use abbreviations in entering the curie content of the package. A significant change is proposed in the requirements for use of the radioactive Yellow II label with respect to radiation levels. At the surface of the package, a limit of 50 millirem/hour is proposed rather than the present limitation of 10 millirem/hour for use of the Yellow II label. A change is also proposed for the authorized radiation level reading to 1.0 millirem/hour at three feet from the package, as compared to a limit of 0.5 millirem/hour as at present.

For shipping paper information the hazard information number will be required to be entered after the proper shipping name and before the classification. The use of symbols and abbreviations for radionuclides and activity of contents will be authorized. An important additional requirement will be the insertion of the transport index of each package in the shipment.

The proposed hazard information system involves extensive revisions in placards and placarding requirements. A single set of placards will be used for all modes of trans-

port. For transport by highway, certain designated materials will not require placarding unless 1,000 pounds or more (gross weight) of the material are carried, while other commodities will require placarding for any amount transported. In addition, the required display of certain placards may be canceled by the display of other specifically listed placards. It should be noted that no exception is proposed in the case of radioactive placards. In particular, radioactive materials assigned the hazard information number 70 require no placards; the H.I. numbers 72, 73, and 74 require a placard, except for highway transport no placarding is required for less than 1,000 pounds gross weight; and H.I. numbers 71, 78, and 79 require placards in any quantity. A proposed requirement that placards be supplied by the shipper is included in this notice.

Another much needed revision to the regulations is proposed in these notices. Within the past few years, containerization of freight to facilitate handling has increased rapidly. In recognition of this trend, labeling and placarding requirements for transport containers have been developed. Any transport container with a volume of less than 640 cubic feet must be labeled as required for the hazardous material involved unless it is placarded for air transport. All transport containers with a volume of 640 cubic feet must be placarded as required for a transport vehicle. Those transport containers intended for air

transport, and having a capacity of less than 640 cubic feet must be placarded or must bear the appropriate labels. It should be noted that with the incorporation of 1973 IAEA revisions, all freight containers will require labels if radioactive materials are present, regardless of placarding.

At the same time as the proposal incorporating the hazard information system was published, Docket HM-112; Notice 73-9 was also issued. The main thrust of this proposal is the consolidation of all hazardous materials regulations for transport by any mode. It will amount to a major reorganization of the present Title 49 to include material presently in Title 14 and 46, the regulations of the Federal Aviation Administration and the U.S. Coast Guard. While this will require effort to become familiar with the new format, it is hoped that the net result of this change will be a simplification of the application of Federal requirements for transportation.

With these changes, coupled with anticipated amendments to recognize the revised IAEA regulations, we expect the regulations for transporting radioactive materials to be fairly stabilized for the immediate future. We will, of course, attempt to make whatever modifications are appropriate as changing conditions and circumstances may dictate. We believe that these changes, and in particular, the hazard information system, represent a significant step in

the improvement of safety of transport of not only radioactive materials, but all hazardous materials. It must be realized, however, that no matter how comprehensive a set of regulations may be, the effectiveness of those regulations is dependent upon their application. Your compliance with these requirements and your control of your shipping procedures will dictate, to a large degree, the amount of surveillance and enforcement effort the Department of Transportation is forced to expend to ensure the safety of the transportation system expected to result from these changes.

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SAFE - SECURE TRANSPORTATION, WHO IS RESPONSIBLE?

R. F. Garrison

ABSTRACT

This presentation reviews the imperfections of the worldwide transportation network, our dependence upon it, the author's views on why there has been no loss of life nor clinically detectable injury due to the hazardous nature of nuclear shipments in over a quarter century, present legal and moral responsibilities of shippers and carriers, and some thoughts about what shippers should consider prior to release of a nuclear materials shipment into today's transportation environment.

Transportation, briefly defined, is an act, process, or instance of transferring something (persons or things) from one place to another. Very simple. The world transport network of railroads, motor carriers, airlines, and water carriers provides the mode, the means, and the way to conquer vast geophysical barriers (mountains, oceans, deserts and space), thereby creating a free flow of commerce and interchange of ideas. Without transportation, you and I would not be in this room today, nor would we be enjoying the luxury that surrounds us, the clothing we wear, the food we eat, and the interchange of the latest technological developments in our profession.

This vast world transport network is not happenstance, nor is it without risk of loss to property and life. It is all too often inefficient, unsafe and too costly. There are too few alternatives for users of this imperfect system upon which we must so heavily rely, and by which we continuously thrive.

I possess firsthand knowledge that much concern and frustration can exist among we shippers who spend millions of dollars annually to buy a safe secure journey for our hazardous materials and other commodities in a transportation environment where one U. S. railroad alone had 2,129 derailments during the first four months of 1974. Reported U. S. hazardous materials shipping accidents via motor carrier average 2,300 annually, kill about 70 persons, and injure 450 causing property damage of about 8 million dollars. Also we have a worldwide environment where terrorists and activists are said to be lurking behind every large tree, bush and boulder, supposedly awaiting to pounce on any shipment which may serve their cause. Therefore, "Safety-security - Who is responsible?" seems a rather ludicrous subject, and probably too much for one mere mortal to rationally address.

A baffling endeavor to the uninitiated is the safe secure transportation of nuclear shipments. The person in the middle is the transportation manager who must live in a world of total compliance with all the requirements of both safety and security and at the same time effectively control transportation economics and scheduling to meet program needs.

It apparently remains too incredible for most of the general public and even many persons engaged in the nuclear field to believe that the many safety regulations for hazardous and radioactive materials were contrived by talented, technically skilled, and dedicated people. Many of these people presently occupy this room. It is also apparently too incredible to believe that not only

packaging integrity, but personal integrity is the reason we have passed the quarter century mark without loss of life or clinically detectable injury due to the hazards of nuclear shipments during transportation. This is a fine record and plainly speaks for itself. At the same time we know of no successful diversion of nuclear materials during transport. Again this is a remarkable record. We do know, however, that this is no time to rest on our laurels and cheerfully hope that existing potential and real problems will diminish without increased effort at each turn of the road.

We only have time in this paper to oversimplify some of the complexities that face nuclear shipments in the future. As you may or may not know--the carriers are charged with the responsibility to serve, deliver, charge reasonable rates, and provide safe, secure transportation with reasonable dispatch. The degree to which this applies depends upon the mode of transport, and the regulatory agency involved. He (the carrier) possesses no license to carry nuclear materials, although he does operate as a bailee and must deliver the goods to its intended destination in the same general condition as when tendered for transport. Due to the package design and integrity for nuclear shipments, this should be about the least troublesome type of shipment offered to the carrier for transport. Unfortunately many carriers seem to disagree with this point of view.

The shipper is charged with a number of requirements under the regulations such as properly packaging, marking, labeling, sealing, properly describing and providing detailed documentation, plus a written certification to the carrier that all these hazardous materials

regulations are met. It is most foolish for anyone to take this certification lightly, and no prudent person (shipper) should offer, nor prudent person (carrier) accept such a certification unless to the best of his knowledge and ability the certification is indeed factual and not fictional. Before this certification is written by a shipper, a designated individual or individuals should have thoroughly inspected the package, properly loaded it per the packaging requirements, properly closed and sealed it, marked and labeled it, and monitored it for radiation. All regulations should have been met, and everything humanly possible should have been done to assure 100% effectiveness during the transport cycle. This takes well-planned, redundant procedures to offer the necessary assurances. This shipping person who acts on your behalf must be highly responsible and aware of the civil penalties imposed for noncompliance (fines and imprisonment or both) before he turns the package loose in today's transportation environment. A great amount of care and precision has gone into the design and building of the package to meet the rigid packaging requirements of DOT for Type A packages and AEC for Type B and large quantity packages. Type B and large quantity packages have been certified safe by the AEC after meeting exceptionally stringent requirements including evaluations for structural integrity, thermal resistance, radiation shielding, radiological safety, nuclear criticality, and quality assurance.

Now we are finally ready for the Murphy's law part of the transportation cycle, "anything that can possibly happen probably will". What can we do to guard against Murphy's law? Or is there anything? This individual says you can and must do something. Again this is

where professionalism plays an important role. We have already implied that if a nuclear package is designed and built properly, and then loaded properly, nothing short of a series of incredible circumstances under the most extreme hypothetical accident conditions can cause loss of contents and result in injury or death to a member or members of the general public. We have placed our faith in "packaging integrity" and the integrity and professionalism of the individual or individuals assembling the package with the authorized contents. So what else is there to be done? I offer a long list of carrier and mode selection considerations which should be considered to further negate Murphy's law". The several modes available are as follows:

Air Express

Proprietary Truck, Aircraft, etc.

Air Freight (passenger)

Air Freight (cargo)

Air Mail (registered - return receipt) exempt quantities

Truck (less-than-truckload)

Truck (exclusive use)

Rail (less-than-carload)

Rail (carload - exclusive use)

Waterways

Freight forwarders, etc.

SOME PROFESSIONAL TRANSPORTATION MANAGEMENT CONSIDERATIONS IN SELECTION OF THE MODE AND PARTICULAR CARRIER WITHIN THE MODE FOR NUCLEAR SHIPMENTS

The main features we are concerned with are:

1. Special safety security requirements.

2. Carrier route of travel advantages.

3. Service advantages and in-transit controls.

4. Other administrative and cost controls.

   a. Package adaptable for certain modes only.

   b. Special equipment such as covered or roll top van for low profile.

   c. Special tiedowns.

   d. All weather capability for aircraft.

   e. Scheduled arrival date.

   f. Overhead crane loading or unloading requirements.

   g. Special features on equipment, such as cushioned underframe and drawbar on rail cars, or air ride equipped motor vehicles.

   h. General condition of carrier equipment being loaded.

   i. Avoid high risk or congested areas en route.

   j. Minimize transfers and handling en route.

   k. Minimize dwell time.

   l. Oversize in dimension or quantity (can or will the carrier or mode handle it?).

   m. Carrier acceptance of Price Anderson Indemnity.

   n. Special permits required for overweight or oversize (State Highway Departments, etc.).

   o. Special state or local regulations in addition to federal regulations.

   p. Does carrier offer armed guard, hand-to-hand service, security-cleared drivers, etc.

   q. Is carrier a consistent on-time performer?

r. Does carrier respond promptly to inquiries for progress of shipment by computerized tracing, etc.

s. Will carrier pay freight claims in a timely manner.

t. Does carrier report incidents and shipment irregularities as required.

u. Does carrier have safety and security programs in place for employees.

v. Is special monitoring in terminals required.

w. Are AEC couriers required by AEC Manual Chapters 2401 or 2405.

x. Are guards required by Title 10 CFR Part 73, etc.

y. May we consider low transportation costs without sacrifice of safety or security.

z. Should we pay premium costs when advantageous, such as hiring a mode or carrier who can shorten the dwell time during transport or provide special safety or security features.

The foregoing is an oversimplification and probably will do a minimal job of convincing any of the "but whatiffers", that nuclear materials shippers leave little and hopefully nothing to chance prior to release of any shipment (however small) into today's transportation environment. When the hypothetical accident imagination machine is actuated, it is next to impossible to turn off; however, in our attempts to turn it off, we shippers become increasingly efficient in protecting human life and property from all hazardous shipments including those that are radioactive. Our task is to continue to

respond with development of the right kinds of regulations, highly knowledgeable people, to implement these regulations, and effectively redundant procedures to attain an elusive 100% perfect shipping record, in an all too imperfect shipping environment. After all is said and done, we are responsible.

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APPROPRIATENESS OF REGULATIONS FOR AIR SHIPMENT OF RADIOACTIVE MATERIALS

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ABSTRACT

The regulations for air shipment of radioactive materials are out of date and the permitted exposure levels to passengers and airline employees are much too high. The regulations are not properly enforced and the government agencies responsible for their enforcement have not adequately discharged this responsibility.

REGULATIONS ARE OUT-OF-DATE

The Interstate Commerce Commission shipping regulations were adopted on October 24, 1947 and went into effect on January 21, 1948.[1,2] I was a member of the seven man National Research Council Committee (R. D. Evans, Chairman) that prepared the technical specifications for these first regulations and it never occurred to me and probably not to the others that these early regulations would be perpetuated with little or no improvements or feedback from experience for the next several decades. When these regulations were written relatively little was known about the magnitude of risks from chronic exposure to ionizing radiation. At that time the only official exposure limit was the March 17, 1934 occupational level of 0.1 R/day or 0.5 R/week as had been recommended by the Advisory Committee on X-Ray and Radium, the forerunner of the National Council on Radiation Protection and Measurements (NCRP). The reader will recognize that this guide which we used in developing these shipping regulations in 1947 is higher than the present occupational exposure limit of NCRP (i.e. 5 rem/y) by a factor of five and is higher than the present

limit set by the International Commission on Radiological Protection (ICRP), the NCRP, the AEC, and the Federal Radiation Council (FRC) for individual members of the public (i.e. 0.5 rem/y) by a factor of 50. Furthermore, the quality of these early recommendations has in some respects deteriorated through the years in that the regulations have become less restrictive and permit higher exposures to passengers and airline employees than originally contemplated, even though radiation risks were not at the time recognized to be as severe as we know them today. Also, it should be recalled that, at the time these regulations were prepared, our principal concern was not exposure to the passengers and crew but prevention of exposure to undeveloped photographic film to more than 14 mR at a distance of 30 feet in aircraft such as the DC-3 or Lockheed Lodestar.

At the time these shipping regulations were first formulated many people believed there was a threshold dose below which no damage would result and that so long as one did not exceed this threshold no radiation damage would be manifest. Today, however, it is generally believed there is no dose so low that the probability of consequent radiation damage such as life shortening, various forms of cancer, terotogenetic changes (malformations, microcephony, monstrosites, etc.) and genetic mutations is zero. Thus, all the international and national radiation standards setting bodies state that all unnecessary radiation exposure should be avoided and all exposures should be kept as low as practicable. For some unknown reason this philosophy seems to have been applied in many areas but not in the case of shipping of radioactive materials and it is especially noteworthy that these admonitions have been ignored in the case of air shipment of radioisotopes.

EXPOSURE LEVELS PERMITTED BY AIR SHIPPING REGULATIONS ARE MUCH TOO HIGH

As explained above, the present shipping regulations are a hangover from the DC-3 days when it was estimated that it was very unlikely a person would be a passenger on a plane carrying radioisotopes more than once or twice a year. On this basis and in view of the fact that at the time many of the presently known radiation risks could not have been contemplated, dose rates to the passengers as high as 10 mR/hr were considered acceptable. Today, however, there are many passengers like myself who are going on their second million air miles and on those flights we often are accompanied by shipments of radioisotopes--sometimes on long overseas trips.

Mr. D. M. Davis[3] of Oak Ridge National Laboratory collected some data for me on potential exposure of passengers who may have flown all the way with radioisotope shipments from Knoxville, Tenn. during the period of November 12, 1971 and December 25, 1972 as shown in Table 1. Although dose rates were within the limits permitted by the Department of Transportation (DOT) on these flights and it is probable no single person actually received doses as large as those shown in the last column of Table 1, it seems to me improper to have regulations so lax that such exposure is possible.

More recent studies (1974) supported by the AEC and DOT in the Chicago and Boston Airports indicate the potential exposure to passengers flying from these two cities may be about the same as Davis found for the Knoxville Airport. For example, these studies indicate a maximum single passenger dose of 12 mR from Chicago to Miami and 80 mR from Boston to Denver.

An air passenger permissible dose rate of 10 mrem/hr seems to me to be too high by an order of magnitude when we realize the ICRP, NCRP, and FRC limit for all man-made ionizing radiation (except medical exposure) is 500 mrem/y to an individual and 170 mrem/y on the average.

Table 1. Exposures Passengers May Have Received on Scheduled Flights from Knoxville, Tennessee

Date	Quantity	Product	Destination	Flying Time	Estimated Max. Dose (mrem)
Nov. 12, 1971	100 Ci	Kr-85	Tokyo, Japan	9:58	29.9
Nov. 12, 1971	50 Ci	Kr-85	Brussels, Belgium	11:23	19.3
Nov. 16, 1971	10^{-3} Ci	Eu-152 & 154	Rio de Janeiro, Brazil	17:39	26.5
Nov. 30, 1971	100 Ci	Kr-85	London, England	10:26	24.0
Dec. 17, 1971	100 Ci	Kr-85	London, England	10:26	28.2
Dec. 17, 1971	395 g	Np-237	Denver, Colorado	5:30	46.8
Dec. 22, 1971	10 μg	Cf-252	Tokyo, Japan	9:58	89.7
Jan. 07, 1972	5×10^{-3} Ci	Zr-95	Pasco, Washington	11:43	21.0
Jan. 11, 1972	5×10^{-3} Ci	Zr-95	Tokyo, Japan	9:58	24.9
Jan. 21, 1972	1.5×10^{-2} Ci	Zr-95	Tokyo, Japan	9:58	23.0
Jan. 21, 1972	6×10^{-3} Ci	Zr-95	Stockholm, Sweden	11:33	26.5
Jan. 25, 1972	33 Ci	Kr-85	Frankfurt, Germany	9:51	22.6

The NCRP limit for pupils in school is only 50 mrem/y. This air passenger limit of 10 mrem/hr appears to be out of line especially when compared with the new levels set by the AEC of 5 mrem/y to an individual or 1 mrem/y to a sizable group living in the vicinity of a 1000 MWe light water cooled nuclear power plant when the operation is carried out in conformance with the AEC regulatory guide of maintaining exposures as low as practicable. Also, it seems a bit peculiar to permit a level at the surface of a package of radioisotopes shipped by air of 200 mrem/hr when the NCRP has, for example, set a limit of only 0.5 mrem/hr at any accessible surface of a TV set as measured at 5 cm (the distance at which surface doses are commonly measured).

One of the surprising observations about the government agencies (DOT and AEC) charged with the enforcement of air shipping regulations is that they seem to feel that because there are over one half million shipments of radioisotopes each year without what they consider any serious accidents that all is fine and well. They forget two important considerations: (1) perhaps many accidents--some serious--have never been discovered or reported and (2) all radiation exposure is potentially harmful and should be avoided; for example, the risk from one rad of exposure is considered to be about one chance in 10,000 the person will as a consequence die of a radiation induced cancer. Thus, if one person received 100 rad of radiation, he would be expected to have one chance in 100 of dying of a radiation induced cancer or if 100 persons each received one rad, each has one chance in 10,000 of dying of a radiation induced cancer or the total overall risk is the same in each case. This attitude of the AEC and DOT that all is well was expressed by representatives of the AEC[5] and DOT[6] in testimony at St. Paul, Minn. on April 5, 1974 in opposition to Mayor Lawrence D. Cohen's proposed ordinance to provide radiation monitoring at the Minneapolis-St. Paul Airport.

Another shortcoming of the Shipping Regulations is that none of the agents who handle air shipments are treated as radiation workers.

A 1958 survey[7] of the British European Airways cargo personnel at the London Airport over a two month period indicated an average dose of 182 mrem/y or a maximum of 770 mrem/y for transport workers. This dose received by workers at the London Airport is about the same as that presently received by employees at the National Laboratories under the supervision of the AEC and yet these Laboratory employees are provided with personnel monitoring meters (e.g. average dose at Oak Ridge National Laboratory is about 250 mrem/y). I have reason to suspect that doses received by transport workers at some of the U. S. airports are even higher than those received at the London Airport.

A LACK OF ENFORCEMENT OF AIR SHIPPING REGULATIONS

Not only are the air shipping regulations far too lax, but those we have are not enforced and to some extent are incapable of adequate enforcement. It seems to me there can be a conflict of interest when compliance with the regulations depends entirely on self policing of the shipper and receiver. If the shipper reduces the shielding of the shipping container so that it is marginal or substandard, he may exceed the permitted dose limits but if no one checks up on him, why worry, because his shipping costs are less and he may not have to buy a larger shipping container. Likewise, the receiver or addressee has little to complain about if he checks the package and finds a high reading because his freight bill is less than he expected or he has gotten more radioisotopes than he paid for. It seems to me rather inconsistent to have all the multiple safety checks of an aircraft and to employ a pilot who is one of the highest skilled individuals in our society and yet have it possible that an inexperienced technician (maybe one who is intoxicated) package and ship a potentially lethal package of radioisotopes without any airport checks to see whether the technician did his job properly. The check takes place only when the package is received by the addressee. In the meantime, after the package is received by the addressee, all the potential damage has been done to the truck drivers, the terminal agents, and the passengers and

crew of the aircraft.

It would take far more than the 10 pages allotted me for this lecture to list all the known major infractions of the air shipping regulations and undoubtedly many more pages to list those that are unrecorded. A study by Davis et al.[8] in 1961 indicated at that time a maximum dose rate of 400 mR/hr in passenger areas. A more recent (1969) joint study by the AEC, DOT, and HEW[9] indicated there are many infractions of the shipping regulations, a few of which were: (1) 25% of the packages were marked with a lower transport index than actually was measured (i.e. the mrem/hr at 3 feet from the package was greater than marked on the shipping label), (2) packages exhibited radiation levels at their surface in excess of the allowable 200 mrem/hr, (3) packages had a transport index greater than the allowable limit (i.e. the TI was greater than 10), (4) 29% of the packages was incorrectly labeled, (5) three packages had no labels, (6) requirements for a security seal which is not readily breakable were seldom complied with, (7) stowage and loading restrictions for radioactive materials were not observed, and (8) the radiation level in aircraft occasionally exceeded 10 mR/hr and in one case measured 25 mr/hr.

A review of the many accidents with radioactive shipments reveals many infractions of the regulations that in some cases contributed to the accidents. Of these I will mention only the two most recent ones of which I am familiar at this writing (May 10, 1974), i.e. the Ir-192 accident on DAL on April 5-8, 1974, and the I-131 accident on DAL on May 3, 1974. Regarding the Ir-192 accident, it seems evident the shipping container was not checked properly with a survey meter before it left Value Engineering in Alexandria, Virginia on April 5. In fact, the 32 Ci source was essentially sticking out the side of the shipping container with no shielding during the entire shipment. It was reading 15.5 R/hr at 3 feet when it arrived in Baton Rouge, La. or 15,500 TI when the maximum allowable by the DOT regulations is only 10 TI. In fact, the radiation level about this shipping container was so high that it set off all the alarms in the Gamma Industries building in

Baton Rouge when the delivery truck was still over 100 feet away from the building. It requires only simple arithmetic to estimate the dose received by the poor chaps that sat 3 feet from this source for the 1.8 hour flight of the 727 from the Washington National Airport to Baton Rouge! The exact dose in various sections and seats of these aircraft is being reported elsewhere in a separate paper by R. L. Zimmerman, Niel Wald, B. J. Porter, and myself and they will not be discussed here other than to say they were much too high especially for pregnant women and children. This accident never should have happened and likely would have been prevented if radioisotope packages were routinely checked at airports.

The second recent accident with I-131 which I will mention was a classic example of infraction of the DOT shipping regulations. We were lucky in this case because the source and its shipping container apparently were not turned upside down until the package was delivered to the hospital in Alabama. In this case it is evident two important DOT regulations were broken: (1) the shipment was not packaged so the radioactive liquid could not escape the inner containment and (2) there was not enough adsorbant material in the shipping box to contain all the liquid. We were much luckier here than with the December 1971 Mo-99 Delta accident when some of my employees (I was at that time director of the Health Physics Division of Oak Ridge National Laboratory) had to track down and survey baggage, shoes, clothing, people, etc. in homes in several states to find and remove the radioactive material that had been carried into many public and private places.

HAVE THE RESPONSIBLE GOVERNMENT AGENCIES CARRIED OUT THEIR RESPONSIBILITY?

It seems to me the government agencies responsible for the shipment of radioactive materials (the DOT and AEC) have two major health physics responsibilities: (1) to see that radioactive materials are shipped safely and (2) to see that persons who may be involved in a high radiation exposure or are contaminated as a result of a spill of

radioactive material are properly identified, informed, and treated.

In the first instance, I have been impressed with the prescribed tests to which these shipping containers must be subjected in the way of temperature changes, drop tests, leak tests, etc. and yet it seems never to have occurred to anyone in these agencies to see that the source holders are designed in such a way that the source cannot be pushed through and locked into apparent proper position outside of the radiation shield. When I examined the C-10 shipping container involved in the April 5-8, 1974 Ir-192 Delta accident, I was shocked and appalled to find the responsible government agencies apparently had approved the use of a container for shipping of very dangerous, highly radioactive industrial radiographic sources such that by a very simple and likely error in loading of the source it would actually be locked in place completely outside the shielding material of the container. This C-10 shipping container has an S-tube running through the central depleted uranium shield but is so designed that, if an absent-minded shipper happens to place the locking plunger in the wrong end of the tube, the source can be locked in place but completely outside the protective shield (i.e. no shielding provided). I would guess anyone with a grain of imagination could have in a short while redesigned the C-10 container in such a way that the source could be loaded only in the proper manner (i.e. so that it is safely in the center of the shield).

We all make mistakes, but it seems to me this mistake on the part of the government approving agencies is so serious that they should assume the responsibility and take the primary blame for any of the consequences of this Ir-192 accident. Unfortunately in this accident and in most others, the airlines have been the innocent victims, but in the eyes of the public are made to appear guilty. They have been forced to accept DOT and AEC approved shipping containers on good faith and are forced to assume the shipper who delivers the radioactive material to them at the shipping office has properly packaged and surveyed the shipment prior to its delivery at the airport.

It is indeed ironical and perhaps ominous that, when the AEC representative[5] was testifying in St. Paul on April 5, 1974 regarding Mayor Cohen's proposed ordinance, "We cannot support the proposed local ordinance because it does not appear necessary" and the DOT representative[6] was saying "There exists insufficient data to support the need for the ordinance," it was this very hour that this unshielded 32 Ci Ir-192 source was being delivered to the National Airport in Washington.

Often good can come of a misfortune. In this case I am pleased that Delta, which seems to have borne the brunt of most of the serious radiation shipping accidents, is no longer sitting by and waiting for the next accident or again begging DOT and the AEC to correct this situation. In essence it is setting up its own monitoring program--similar in many respects to that proposed by Mayor Cohen--but in this case for all Delta terminals.

If time permitted, I would discuss in detail what I consider a second major health physics responsibility of the Government agencies, viz. to identify and see that proper care is provided persons who may be contaminated or injured as a result of accidents in the shipping of radioactive materials, but time will permit only stating that this part of the program has much to be desired. These agencies, for example, could have been of some help in determining the dose received by each individual passenger on the DC-8 and 727 flights with this 32 Ci Ir-192 source or as brought out in the St. Paul hearings on Mayor Cohen's proposed ordinance, Mr. Roger Welsch an employee of Northwest Airlines that was involved in the April 1972 contamination incident could have been given more appropriate decontamination treatment. It seems to me in his case, for example, urine and fecal samples should have been collected and analyzed to better estimate his internal dose from inhalation and ingestion of radioactive material.

In conclusion I would like to state that, during the past 18 years, we should have learned much from our mistakes, our trials and errors, and the accidents in the air shipment of radioisotopes. So

far, we have been lucky in not having had a major accident and I hope these regulations will have some long needed revisions before our luck runs out.

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ADEQUACY AND EFFECTIVENESS OF FEDERAL AVIATION ADMINISTRATION REGULATIONS IN CONTROLLING RADIATION DOSE TO PASSENGERS AND CREW IN PASSENGER CARRYING AIRCRAFT

Robert F. Barker
Anthony N. Tse
Donald R. Hopkins

ABSTRACT

Any discussion of how well the safety regulations of the Federal Aviation Administration control radiation exposures of aircraft passengers and crew should include information on both the adequacy (expected control) of the regulatory provisions, on the effectiveness (degree of conformance) of the implementation of those provisions, and on the end result which combines both the adequacy and effectiveness. If the end result is not satisfactory, you must, of necessity, return to an evaluation of the adequacy and effectiveness separately in order to determine how to correct the deficiency.

A recent study, conducted jointly by the Atomic Energy Commission and the Department of Transportation's Federal Aviation Administration (FAA) and Office of Hazardous Materials (OHM), followed this pattern. The adequacy of the radiation safety provisions of the FAA regulations were evaluated by physical testing at the FAA facilities in Oklahoma City during February 1973. Standard packages containing cesium-137 were mocked up in two aircraft at that facility and the radiation levels produced in the passenger compartment from those packages were measured. It was thought that packages stowed in the aircraft cargo compartment, with the packages separated from the passenger compartment by the minimum distances allowed by the regulations, would produce the highest allowable radiation levels in the passenger compartment, and that radiation levels found in actual operating aircraft would be significantly lower.

An effort to measure the effectiveness of the air carriers' implementation of the FAA regulations was made in May-June 1973. Graduate students from Northwestern and Harvard Universities were allowed, through the cooperation of member airlines of the Air Transport Association, to board operating aircraft at Chicago O'Hare and Boston Logan Airports respectively, and to

measure radiation levels in the passenger compartments produced by the normal loading of radioactive material in the aircraft cargo compartments. Chicago and Boston Airports were chosen for this portion of the study because of the large number of radiopharmaceutical packages originating near the airports, and transported through them. The aircraft measured were selected on the basis that they were known to be carrying packages of radioactive material. While most of the radiation levels found were as anticipated, some levels were found in excess of those maximum allowable levels measured in Oklahoma City, along with other evidence of carrier noncompliance with the FAA stowage rules.

An effort to measure the end result of the FAA regulatory system was made in November 1973. Again, through the cooperation of the member airlines of the Air Transport Association, statistical data were gathered by those airlines at 33 airports across the country, chosen to be representative of all the nation's airports. The statistical data collected related to the total transport index of the packages carried aboard passenger-carrying aircraft, and to the number of passengers associated with such flights and the flight duration. The total transport index on each flight was translated into passenger compartment radiation levels, and the data then used to statistically determine maximum and average annual radiation exposures to passengers and crew, and to determine total radiation dose to the population resulting from the transportation of radioactive material by passenger aircraft.

Studies such as these have been useful in determining whether individual exposures are being adequately controlled, and whether the risk associated with the total population dose is adequately balanced by the benefit derived from using passenger aircraft for the transportation of radiopharmaceuticals. These studies should also provide some basis for determining if radiation exposures are as low as practicable.

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EFFECT OF REGULATORY REVISIONS ON DESIGN AND OPERATION OF LIGHT WATER REACTORS

Ken Gablin

ABSTRACT

In order for the architects and engineers to properly design light water reactors, it is necessary to indicate trends and basic changes that make future planning as flexible as possible. The designer must understand current thinking as well as proposed revisions from Agreement States, United States Atomic Energy Commission, and United States Department of Transportation. It is the purpose of this writing to point out the most significant of these changes and point out methods in which a designer can prevent his concepts from obsolescence on the drawing board.

LEACHABILITY

Until a few months ago, the term called leachability was not discussed much in terms of rad-waste. Up to that time, the burial containment was felt to be the total containment, with a high percentage of aqueous rad-waste shipped from light water reactors in an absorbed condition. It is obvious to the trained observer that policy is changing very fast in the manner and methods of storing, shipping and burial of rad-waste.

The first agency to put numbers on leachability was the International Atomic Energy Agency. This agency has published a booklet entitled Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition, Safety Series No. 6. If this booklet is not in your library, it should be purchased for reference from UNIPUB, Inc., P.O. Box 433, New York, N.Y. 10016. At the time of this writing, no other agency has put limits on the actual percentage of material

in a radioactive form that can leach from a larger source, so the International Atomic Energy Agency statements shall act as a reference for future regulations. In due process, the United States Department of Transportation's Office of Hazardous Materials will produce their regulations for review in the Federal Register. After some review period, our transportation regulations for rail, air, truck and barge will have the same regulations as most of the nuclear nations do today.

Two important changes from current regulations should be pointed out.

In the older regulations that are currently being used by the United States Department of Transportation, levels of toxicity divide the various radionuclides into broad classifications called Transport Groups. In the new and proposed regulations, each a value in curies, called A_2 is used. The value of A_2 in curies is for all radioactive materials that are not in special form, and varies for each radionuclide. For example, the common radionuclides are shown in Table No.1.

Table No. 1

Symbol of Radionuclide	Element and Atomic Number	A_2(Ci)
Co-60	Cobalt (27)	7
Sr-90	Strontium (38)	0.4
Cs-137	Caesium (55)	9

From Table No. 1 it can be seen that the A_2 value for Strontium is very small, stating that this radionuclide is more toxic than either Cobalt-60 or Caesium-137.

In the form of Low Specific Activity rad-waste, most of the material coming from the light water reactor was grouped

in Group III or Group IV. Now the value of Low Specific Activity varies with each radionuclide. The upper limit of Low Specific Activity is now defined numerically as $10^{-4}A_2/g$. In Table form, this can be seen in Table No. 2.

Table No. 2

Symbol of Radionuclide	Upper Limit of LSA in micro Curies/g.
Co-60	700
Sr-90	40
Cs-137	900

When the activity per unit weight exceeds the upper limits of Low Specific Activity, the rad-waste falls into a category called Low Level Solids. The upper level of Low Level Solids are still given in terms of activity per unit weight, but its upper limit is twenty (20) times higher than the upper limit of Low Specific Activity, and is defined numerically as $2 \times 10^{-3}A_2/g$. In table form, this can be seen in Table No. 3.

Table No. 3

Symbol of Radionuclide	Upper Limit of LLS in micro Curies/g.
Co-60	14,000
Sr-90	800
Cs-137	18,000

The graphical presentation of both Low Level Solids and Low Specific Activity are shown on Chart No. 1 for three common radionuclides, namely; Cobalt, Strontium, and Caesium. Refer to Chart No. 1.

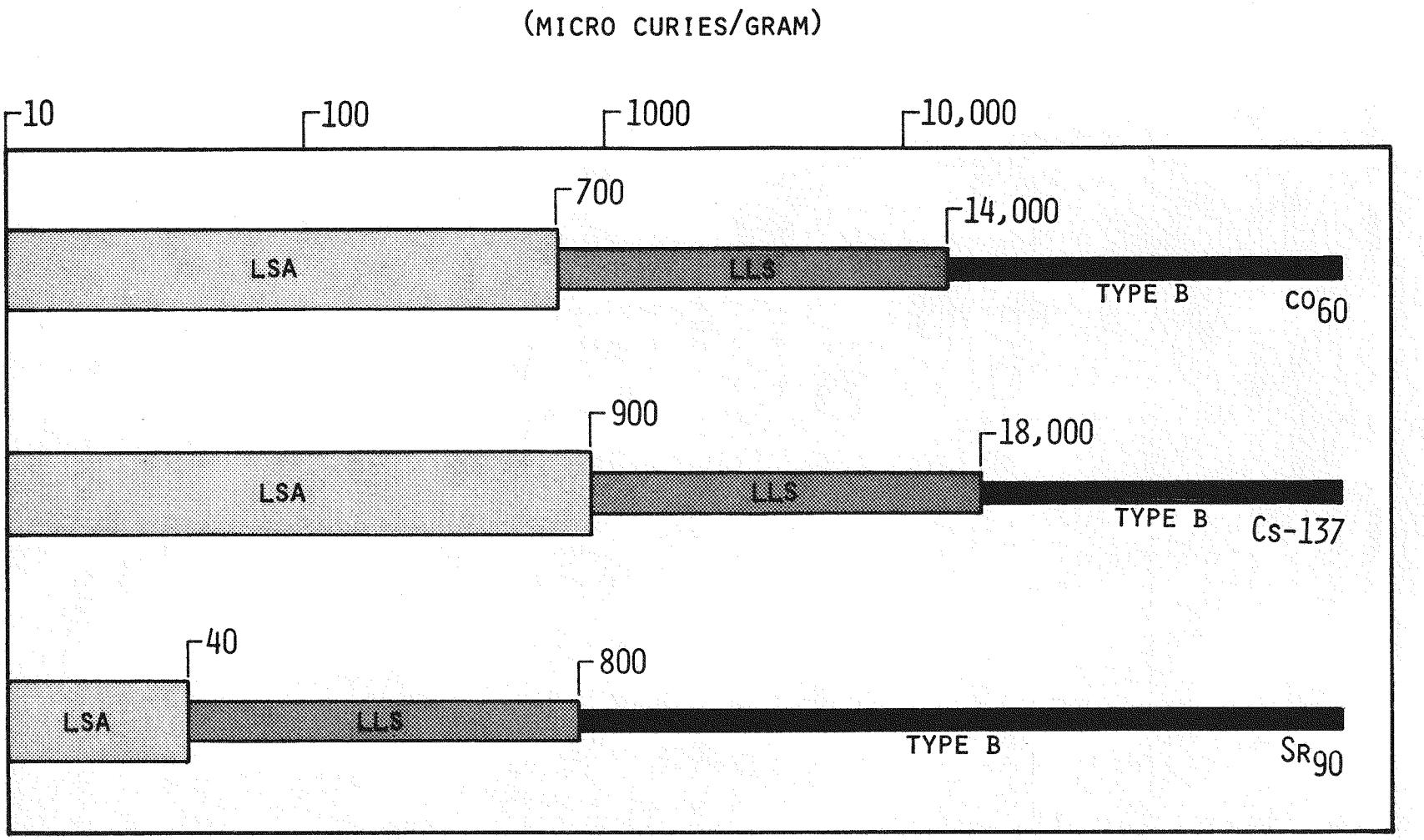


Chart 1. Graphic Presentation of LSA and LLS Limits for IAEA Regulations.

The leach rate for Low Level Solids is limited to 0.1 A_2 in a period of one week or 168 hours. Again, it should be pointed out that the total curie content of a LSA or LLS package is not normally limited by activity per unit weight, but by weight itself, due to shielding requirements. The leach rate is an absolute value of curies, as pointed out in Table No. 4.

Table No. 4

Symbol of Radionuclide	Maximum Loss of Activity from 168 Hour Leach Test
Co-60	0.7 Curies
Sr-90	0.04 Curies
Cs-137	0.9 Curies

As an example, if the package were subjected to immersion in water under a loss of packaging situation, the total loss of activity is not a function of a percentage of the initial or original activity but only the 0.1 A_2 value as stated in Table No. 4.

Effects of Leach Rates on Other Regulations

As the International Atomic Energy Agency regulations are absorbed into the United States Department of Transportation's Office of Hazardous Materials, other agencies, both federal and state, will review the general concept of leach rates and interpret the values.

Much of the study in leach rates seems to consider itself with the problem of solidification of liquid rad-waste. In the future, the subject of container life and solidification of ion-exchange resins will be equally important.

As risk to the public increases from shipments of low level rad-waste, it may be that public opinion forces the industry into engineered storage of rad-waste at the point of origin. It is a dynamic industry with many forcing functions shaping the outcome of the nuclear industry and its shipment and disposal of rad-waste.

SOME UNIQUE FABRICATION TECHNIQUES EMPLOYED ON THE GENERAL ELECTRIC IF-300 SHIPPING CASK

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The abstract and/or paper is printed in Volume 3.

A CURRENT ASSESSMENT OF THE SECURITY OF CARGO IN THE NATION'S TRANSPORTATION SYSTEM

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Theft of cargo from the Nation's transportation network continues to be a serious problem which costs this country something in excess of $1 billion annually. The cargo security program undertaken by the Department of Transportation in 1971 is now beginning to produce many isolated cases where significant reductions in theft have been achieved. In all cases, the common ingredients for success have been personal interest and determination by transportation management that cargo can be protected and will be protected from theft. In operational terms, this means close supervision of transportation personnel and facilities plus the use of tightened accountability procedures. Some carriers do better than others in protecting the goods they have been entrusted to move. The movement of nuclear materials demands the best protection available with flawless accountability. Such protection is achievable and the nuclear industry must not settle for less.

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PROTECTION OF SPECIAL NUCLEAR MATERIAL IN TRANSIT

James A. Powers

ABSTRACT

It is essential that the nuclear industry continue to maintain its record of never having a shipment of strategic nuclear material stolen. Regulations promulgated by the U. S. Atomic Energy Commission are directed toward requiring certain shipments of plutonium and uranium containing materials be protected against hijacking. These regulations were amended in November, 1973, and are under continuing review to assess the need for further requirements in light of the ever changing threat of hijacking.

INTRODUCTION

At the end of May, 1974, there were 44 operating nuclear power plants in the United States. All but one of these were light water reactors which use uranium enriched to 3-5 percent in the fissile U-235 isotope as fuel. A by-product of the operation of light water reactors is plutonium, a material which is used to make nuclear weapons. Another material used to make nuclear weapons is the fuel for the remaining one operating nuclear plant, uranium enriched to greater than ninety percent in the U-235 isotope. Both of these materials fall into the category of what the Atomic Energy Commission calls special nuclear material.

During the last few years the Atomic Energy Commission (AEC) has initiated and completed a substantial upgrading of the safeguards (security) afforded shipments of certain special nuclear material. I would like to give you an overview of what has happened in this area and what further developments are under consideration. For purposes

of this paper, special nuclear material (SNM) is defined as plutonium, uranium-233, or uranium enriched to 20 percent of more in the U-235 isotope. My remarks will be addressed to the shipment of special nuclear material in the private sector.

Prior to 1969 there were no safeguards requirements on the transportation of special nuclear material. Beginning in 1969, the potential for theft or diversion became more apparent when misrouting led to several instances of shipments being temporarily lost. This resulted in the issuance of 10 CFR Part 73 for the protection of SNM in transit. Under Part 73, as it existed at that time, shipments containing more than 5 kilograms (kg) of plutonium, uranium 233, or high enriched uranium were required to be transported: (a) in the custody of a person designated by a licensee, or (b) under the established procedures of a common or contract carrier which provided for the protection of valuable material in transit and which required the exchange of hand-to-hand receipts whenever there was a change of custody. Because the transportation cost was lower, almost all of the shipments were made under the "signature service" rather than the personal custody option. Licensees were also required to notify consignees in advance of names of carriers and the estimated time of arrival of a shipment; to conduct a trace investigation of any delayed shipment; to report to the AEC any lost or delayed shipment which results in a trace investigation, along with results of the investigation; and to keep appropriate shipping records.

CURRENT SHIPMENT SECURITY REQUIREMENTS

Following the initial publication of 10 CFR Part 73, the need for further protection of special nuclear material was under continuous study and on February 1, 1973, additional requirements were published for public comment and certain restrictions were placed on air shipments. These restrictions, still in effect, stipulate that no shipment of special nuclear material in excess of (1) 20 grams or 20 curies whichever is less, of plutonium or uranium 233, or (2) 350 grams of high enriched uranium shall be made in passenger aircraft.

Following a public comment period of sixty days on the proposed new requirements, and consideration of the comments received, new transportation security measures were published on November 6, 1973, and were effective on March 6, 1974. The rules applied to any person who is licensed pursuant to the regulations in Part 70 of the 10 CFR who imports, exports, transports, delivers to a carrier for transport in a single shipment, or takes delivery of a single shipment free on board at the point where it is delivered to a carrier, either uranium-235 (contained in uranium enriched to 20 percent or more in the U-235 isotope), uranium-233, or plutonium, or any combination of these materials which is 5,000 grams or more computed by the formula, grams = grams contained U-235 + 2.5 (grams U-233 + grams plutonium). The difference in the threshold quantities of the revised Part 73 is that shipments of plutonium and uranium 233 at or above 2,000 grams are now covered under the new regulations whereas before the threshold was 5,000 grams or more. The significant requirements of the amendments to Part 73 are as follows:

a. If a common or contract carrier is used, the special nuclear material must be transported under the established procedures of the carrier which provides a system for the physical protection of valuable material in transit and requires a hand-to-hand receipt at origin and destination and at all points enroute where there is a transfer of custody. Transit times of all shipments must be minimized and routes selected to avoid areas of natural disaster or civil disorders. Special nuclear material must be shipped in containers which are sealed by tamper-indicating type seals. The outer container or vehicle is required to be locked and sealed. No container weighing 500 lbs. or less may be shipped on open vehicles, such as open trucks or railway flatcars.

b. All shipments by road must be made without any scheduled intermediate stops to transfer special nuclear material or other cargo between the point of origin and destination. All motor vehicles are required to be equipped with a radiotelephone. Calls must be made at predetermined intervals normally not to exceed 2 hours and if calls are not received when planned, the licensee or his agent must immediately notify an appropriate law enforcement authority and the AEC. Shipments by road must be accompanied by at least two people in the transport vehicle and must be kept under continuous visual surveillance. If the transport vehicle is a specially designed vehicle with penetration resistant and immobilization features, armed guards are not required. An armored car with two armed

guards may be used instead of a specially designed vehicle. Alternatively a common carrier or licensee vehicle can be used provided it is escorted by an armed escort consisting of at least two guards in a separate escort vehicle. In addition, transport vehicles are required to be marked on the top, sides, and rear with identifying letters or numbers.

c. Shipments on cargo aircraft are required to be arranged so as to minimize the number of scheduled transfers and these must be monitored by armed guards.

d. Rail shipments must be escorted by two armed guards in the shipment car or in an escort car. Continuous on-board radio-telephone communications capability must be provided with conventional telephone backup. Periodic calls are required to the licensee or his agent.

e. Shipments by sea must be made on vessels making the minimum ports of call. Transfer at domestic ports from other modes of transportation must be under continuous visual surveillance by a guard. Shipments must be placed in a secure compartment which is locked and sealed. Export shipments must be escorted by an authorized individual who may be a crew member, from the last port in the U. S. until it is unloaded in a foreign port. Ship-to-shore communications must be made every twenty-four hours to relay position information and the status of the shipment as determined by daily inspections.

f. A licensee who makes a shipment must notify the consignee of the shipment schedule and details, including the estimated time of arrival of the shipment. A shipment which fails to arrive at the destination on time must be traced and the AEC must be notified.

One of the requirements of the revised 10 CFR Part 73 is that, before licensees can transport SNM, a plan outlining the procedures that would be used to meet all the requirements must be submitted to and approved by the Atomic Energy Commission. There were 10 licensees who submitted plans to ship SNM above threshold quantities, and seven who submitted an interim plan which stated that they would not ship over threshold quantities. In addition, three firms who are not licensees submitted plans which were approved to transport SNM for licensees. In many cases the licensees included one of these firms' plan into their own transportation plan.

RESEARCH AND DEVELOPMENT

The in transit protection requirements are aimed at preventing the successful theft of special nuclear material by persons or groups who might wish to either sell this material or use it as a basis for a terrorist threat or action. The preventive nature of the regulations is either immediate or deferred, depending upon the size of the attacking force. A small attacking force could be repulsed by the armed guards present at the more vulnerable points of the transportation cycle. In the event the protection afforded the shipment is not adequate to ward

off the attacking force, the communications system and the use of secure vehicles with penetration resistant and immobilization features are to summon assistance from appropriate law enforcement agencies and delay the theft attempt unitl assistance arrives, respectively. Two programs in the research and development area are the improvement of the present communications capabilities and the design of a secure vehicle, both of which are expected to greatly improve our in transit protection requirements for special nuclear material.

Communications

A mobile radio system is under development for protection of long distance shipments of special nuclear material.[1] Experiments conducted this spring demonstrated the technical and practical feasibility of this continuous communications system. Messages have been transmitted for distances of more than a thousand miles from a test vehicle to five widely separated ground stations. It provides for digital message traffic between trucks and dispatchers with capability for the truck to send out an immediate emergency call for assistance. The nature and results of the experiments were described at the 1974 Summer Meeting of the American Nuclear Society.[2]

Secure Vehicle

A specially designed truck is being developed which will provide penetration resistant and immobilization features. A licensee using this truck under the present regulations would not be required to use armed guards. Penetration tests were conducted this spring on prototype walls of a cargo compartment for a secure vehicle.[3] Laboratory tests have also been conducted on equipment the truck crew could use to immobilize the vehicle if threatened or attacked. The tests were conducted to establish how long these design features can resist attempts by attackers to obtain possession of cargo or vehicle. Knowledge of this time and the time needed for a protective reaction to any such an attempt is needed for preplanning routes along which protection is adequate. Initial results have been promising and a prototype truck is being designed and built. Publication of the details of the design and testing of the secure vehicle would increase vulnerability to possible attacks. Information will be made available to those having a need for this information. It is mentioned to give assurance that serious effort is being devoted to the development of this vehicle.

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NOTE

Pages 116 through 127 are intentionally omitted.

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THE MANUFACTURING ENGINEERING ASPECT OF EFFECTIVE ECONOMICAL CASK DESIGN

F. J. Bush, Jr.

ABSTRACT

This paper is intended to highlight the contribution of the Manufacturing Engineer to effective and economical cask design. All too often the use of this discipline is not considered when engineering schedules are planned. The resultant waste of time and money are most unfortunate and should be avoided.

Possibly one of the most, if not the most, overlooked aspect of cask design is Manufacturing Engineering. All too often the input of sound fabrication practices and the effects of process limitations are factored into the design too late or not at all. Many times the design parameters and physical characteristics of the cask are set and then the Manufacturing Engineer is called in to determine the best method for fabricating a fixed and final design. This procedure cannot possibly result in either optimal design or the most economical cost.

Whether the design is being done by an organization with or without its own fabrication facilities, the people who specialize in the production work should be called in early in the design effort. Their input will contribute greatly to the economy and effectiveness of the design in many areas. Obviously, they can best provide comparative

data of the costs of alternate methods of fabrication, the cost effect of the use of alternate grades of material, the effect on cost and fabricability of forgings, weldments, castings, etc. as well as advising practical limitations achievable with various production procedures such as degree of bond of lead shielding to cask walls. It is the intention in this paper to provide a few basic parameters of the role of the Manufacturing Engineer in cask design and to suggest means of assuring the most effective use of this discipline.

At the outset let me say, although there are inherent dangers in generalizations such as will be presented in this paper regarding the effectiveness and economies represented in various types of material and methods of fabrications, it is my belief that these can be avoided by judicious use of the underlying principles discussed and recognition of the problems resulting from generalization.

The specific areas which I will attempt to touch upon in this paper are the following :

1. Material selection as affects fabricability and cost
2. Material type - plate, casting, forging, etc.
3. Welding design and its effect upon cost
4. The cost effect of tolerances
5. The trade-off in cost of acceptance criteria for various production procedures
6. Miscellaneous items

7. The effect of all these items upon delivery cycles and thus upon cost.

Obviously, it would be impossible to cover these items in depth in anything less than a complete book, but we can discuss some of the basic principles and the most commonly encountered problems in the areas having the most direct application to Shipping Casks for Spent Nuclear Fuel as well as other type "B" containers.

Two key words relating to the proper application of manufacturing engineering are "planning" and "timing". In today's hectic world there is never adequate time from the date a commitment is made that a cask is required to the time when the hardware is needed. The tendency, therefore, is to eliminate all unnecessary steps in the cask design problem and all too often this includes bypassing manufacturing engineering. Invariably, such a procedure results in a loss of time rather than a gain. Designers are seldom experts in the discipline of the manufacturing engineer and more often than not the details must be revised later to make the design one that can be built or else the omission is reflected in higher fabrication costs.

Let me discuss in a little further detail some of the areas enumerated above.

1. Material Selection

Selection of the proper material obviously can have a great effect upon the economy of a cask design. This is particularly true of

some of the more unusual materials used in shipping casks. For instance, often overlooked in the detailing of larger shipping casks is the possibility of using carbon steel forgings for the bulkier strength members and applying a cladding or weld overlay of corrosion resistent material. This technique developed in the earlier reactor vessel program is seldom applied to shipping casks but could present substantial economies.

High strength material commonly used in supporting or mounting components for shipping casks very often require highly specialized welding and/or heat treating procedures which are both difficult and expensive and normally do not lend themselves to long life and high reliability factors. Very often designers overlook such high strength materials as titanium which often present fewer fabrication problems than many of the heat treated alloy steels more commonly used.

One material problem which is particularly sensitive today is that of delivery. Deliveries on some materials can extend as long as one year from placement of order to delivery of the basic plate or forging material. Very often economy must become secondary to the need for delivery. This situation is probably not going to improve for some materials within the next couple of years.

2. Material Type

The type of material selected can have a significant effect on the cost of the cask. Where flanges or rings are cut from stainless steel plate, for instance, the cost of the center drop out plus the lost

corners often results in a part which could be reduced in cost significantly by use of casting or forging. Forgings can also often add economy to a design where a flat bottom with a welding lip is required rather than resorting to corner welds. Often overlooked in the use of flat plate design is the tolerance to which plate can be obtained as regards flatness. For instance, a 1" plate 4 ft. sq. has a flatness tolerance of 1/2" from the mill. If this plate must subsequently be machined to achieve a tolerance, only 1/2 the thickness of the plate is actually being used. Very often special flattening or substitution of forged material or some other alternative would result in considerable economy of design.

It is equally true, however, that in some instances welded fabricated plate can be considerably less expensive than other material types.

Castings, of course, present their own particular problems when they are made part of a weldment. The tendency of castings to result in porosity problems may necessarily require special inspection techniques such as radiography to assure acceptable material prior to welding.

Unfortunately, no firm rule of thumb can be applied to material selection but each case must be evaluated on its own merits.

3. Welding Design

The design of weldments is, needless to say, an art in itself and with today's designs pushing limits of material as regards tensile strength, ductility, etc., this area is becoming more and more critical.

The use of various available welding procedures can have a very marked effect upon cost. In general, automated procedures or semi-automatic procedures are the most economical. Not all cask weldments lend themselves, however, to automatic or semi-automated welding.

One of the more critical aspects of proper welding design is control of or elimination of distortion during the welding procedure. Very often the type of weld joint design and the welding process selected plays a large part in control of distortion. Additional control may often be obtained by the proper use of jigs and fixtures.

In today's casks where shipping weight is becoming more and more of a factor resulting in the use of depleted uranium for shielding rather than lead, the application of welded joints for the uranium can result in substantial savings by taking credit for the strength of the welded uranium shield for the strength and impact resistance required for the accident conditions and resultant stresses.

4. Tolerances

It is surprising how much economy can be achieved through the judicious application of tolerances. The temptation is to standardize

tolerances and apply these standards to all dimensions. Invariably, this results in excessive costs. For instance, such a small item as a 1/8" radius expressed as a two-place decimal and carrying a tolerance as a two-place decimal of .02 can cost ten times as much as the same radius expressed as a fraction in carrying a tolerance of $\pm$ 1/16". Very often such dimensions are not at all critical and can easily be relaxed. It is also important to understand just what a tolerance is expected to achieve. For instance, a tolerance on the ID of a cask does not in itself assure that the expected contents will fit into the cavity. An inside diameter can be within tolerance at any point measured on the diameter but might be effected by the perpendicularity of the cylinder of the base plate or the concentricity to other related components. More than one cask has had to be modified after it was built because of improper application of tolerances.

5. Production Procedures

In order to properly achieve economy in cask fabrication, it is necessary to understand the effect of some of the procedures used. For instance, it is quite common in cask design to specify that the lead shielding be bonded to inner and outer walls of the cask in order to provide for thermal heat transfer. Merely specifying bonding, however, isn't enough. A reasonable degree of bond can be achieved quite economically in some casks by merely pouring the lead against a cask surface adequately prepared. Far more expensive in large casks

is the achievement of a 95% bond of the lead to materials such as stainless steel. This can only be accomplished by the build-up of hand bonded lead to thicknesses of 1/4" to 1/2" on the stainless surface prior to lead pour. This can be many more times expensive than the minimal first coating described earlier. Very often the heat transfer advantage gained is not worth the cost incurred.

One additional item which I will include in this section that can result in excessive costs is surface finish requirements. Decontamination costs for casks can very easily become a major operating cost. As a result the tendency is to require superior surface finishes. I will merely note that it can be far more costly to specify a surface finish as a particular polish or a particular RMS finish rather than to develop an acceptable sample which can often be duplicated using a far less expensive production procedure.

6. Miscellaneous Items

One of the most overlooked facets of the cost of a cask can be the tools, handling equipment, or miscellaneous attachments to the cask. I have seen cask designs where the miscellaneous equipment equalled the cost of the basic cask itself. Special valves and fasteners procured to today's standards can carry very high price tags. Special seals and seal surfaces can also add considerably to the cost. Very often these costs are unavoidable but many times an adequate cost analysis would result in a reduction of cost for these items.

Accessory equipment such as special tools, handling equipment, and other items of this nature are very often handled as an afterthought with little or no consideration being given in advance to the cost of these items. Built to today's stringent requirements, specifications, and standards, such items can become excessively expensive and major economies await the designer who will make sure that the proper principles of manufacturing engineering are applied to the designs in this area.

7. Delivery Cycles

To properly apply the principles of manufacturing engineering for the items listed above will obviously result in an additional span of time which must be added to the design cycle. Off hand it would seem that with today's stringent schedules in cask design that this would become a luxury one could not afford. Quite the opposite is invariably true, however. To eliminate or avoid the manufacturing engineering review during the design cycle will always result in an extension of overall schedule. The time saved by elimination of manufacturing engineering review will be more than used up in problems during material procurement or in an extended fabrication cycle.

Summary

It is not my hope that the items discussed have created any experts in manufacturing engineering. Rather, the purpose of this paper is to convince all of you design engineers that proper use of the manufacturing engineering discipline can provide for you very definite and specific benefits and economies to your design. I would hope that it is this thought you will take away with you.

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DESIGN, MANUFACTURING, AND OPERATIONAL EXPERIENCE WITH THE NFS-4 SPENT FUEL SHIPPING CASK

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ABSTRACT

The NFS-4 was the first spent fuel shipping cask licensed and fabricated to handle the larger physical sizes and higher fuel burnups associated with the second generation nuclear reactor plants. Its design capacity of one PWR or two BWR fuel assemblies is normally transported by truck without restriction under legal gross vehicle shipping weights. The very successful operation of the cask has been well documented by over 200 transports and typical turnaround times at reactor facilities of under four hours. The background leading to the selection of this design is discussed along with the planned development of an advanced design with double the carrying capacity.

DISCUSSION

A detailed study was made in 1971 to consider the effect of various cask types and modes of transportation on the shipment of second generation reactor fuel. This evaluation indicated that truck transport would provide the most attractive approach for the widest variety of reactor installations. Some of the basic considerations originally emphasized and developments to date which further enhance this choice were presented by the author at the AIF Topical Conference Uranium/Fuel Cycle '74 (March 1974) in a paper entitled "Truck Transport of Spent Nuclear Fuel".

The intial optimization study resulted in the selection of a uranium shielded cask having a capacity of 3 PWR or 6 BWR spent fuel assemblies. This cask, designated as NFS-3, would normally be shipped by truck under special permit due to the resulting overweight configuration. A detailed design was prepared and the Safety Analysis Report (SAR) was submitted to the AEC in August 1971. During the preparation of the SAR and while performing the detailed design of the NFS-3 it became apparent that the cask would weigh significantly less than the scoping calculations had originally indicated. Then it also became apparent that a 2 PWR/4 BWR cask of similar design could be developed that would meet the normal gross vehicle shipping weight limitation of 73,280 lbs. imposed by most states. Because overweight shipments frequently had special transport restrictions applied in many areas (i.e., no night and/or weekend movements,

frost restrictions, etc.), the 2 PWR/4 BWR cask was much more attractive from an economic and operational point of view. A difficult decision was then made to indefinitely delay the NFS-3 development and displace it with the new 2/4 design (now designated the NFS-5). Unfortunately, the schedule for initiating contracted spent fuel shipments did not provide ample time for the detailed design, licensing and fabrication of the new NFS-5 cask.

Therefore, in order to proceed with the NFS-5 it was necessary to defer its development and proceed with the design, licensing, and fabrication of a relatively small, simple, and conventional cask to satisfy the early shipping schedule. This cask, designated the NFS-4, had a capacity of 1 PWR or 2 BWR fuel assemblies and its design, licensing, manufacture and operational performance provides the basis for this paper. The NFS-5 design was subsequently completed as planned and is currently under evaluation for licensing by the AEC.

The NFS-4 detailed design drawings and a 260 page Safety Analysis Report were completed and submitted to the AEC in January 1972. As a result of additional questions raised, five additional submittals were made which eventually were combined into a consolidated SAR (345 pages) and a final submittal made in September 1972. The license was issued in November 1972 for an overall lapse time of approximately 10-1/2 months. Due to the intentional simplicity of the design, this time period was significantly longer than originally anticipated. However, since the material procurement and fabrication period for the cask was also correspondingly longer than contractually committed, licensing was not a critical path item.

A fabrication contract for two NFS-4 casks was awarded to Stearns-Roger, Inc., Denver, Colorado, in February 1972 and ran in parallel with the AEC licensing activity. These casks were the first of their type to require a detailed quality assurance program, in-process inspection by AEC Directorate of Regulatory Operations, and an extensive acceptance test program. This test program involved hydraulic checks, verification of thermal performance (both normal operation and simulated accident conditions), shielding acceptability, temperature and pressure tests of valves and O-ring seals. All phases were completed without major problem with the first cask delivered in January 1973 and the second the following month (i.e., eleven and twelve months after the contract award date). The casks were checked out and a dry run made at NFS's West Valley, New York, reprocessing facility. They were then shipped to Rochester Gas and Electric's Ginna reactor plant for their first operational usage. This and subsequent shipments will be discussed later in this paper.

First, to obtain a perspective of the cask and its operating features one should examine the cask presented as a schematic in Figure 1 and as operational hardware in Figure 2. It has a cavity 178 inches in length by 13-1/2 inches in diameter. Interchangeable fuel baskets provide it with a capacity of 1 PWR or 2 BWR fuel assemblies from the second generation reactors or up to four assemblies from some of the earlier first generation reactors. Operational characteristics and limitations of the fuel are detailed on Table I. The primary cask cavity consists of a nominal 5/16 inch stainless steel pressure shell surrounded by a 6-5/8 inch thick lead

TABLE 1

NFS-4 Operational and Fuel Limitations

Operations	
Fuel Decay - Minimum	120 Days
Fuel Burnup - Maximum	40,000 MWD/MTU
Decay Heat - Maximum	11.5 KW

Fuel Assembly	PWR F/A	BWR F/A
No. of Assemblies**	1	2
Envelope, Inches*	8.60	5.44
Length, Inches*	177	177
Active Length, Inches*	150	144
Enrichment, w/o U-235*	3.6	3.0
Weight U, Kg*	470	197

* Indicates maximum value

**May transport up to four first generation fuel assemblies

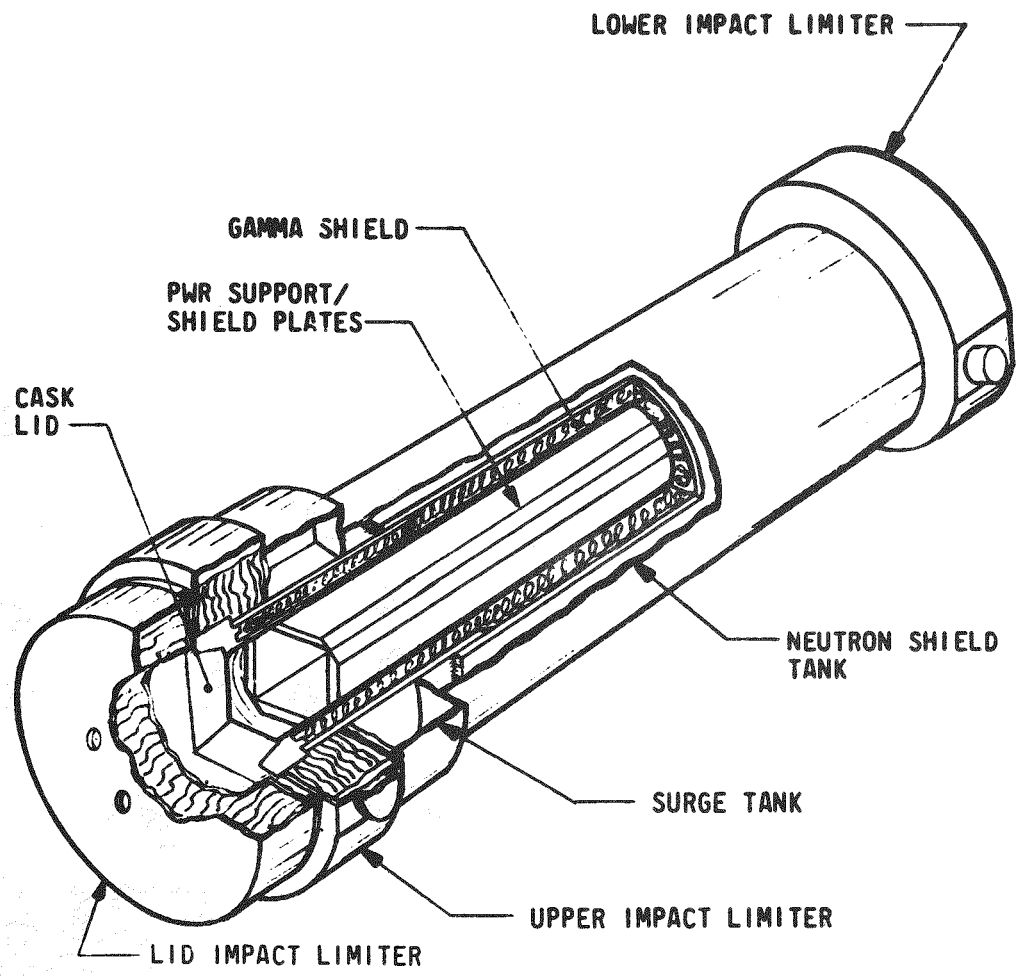


Figure 1 - Schematic of NFS-4 Spent Fuel Shipping Cask

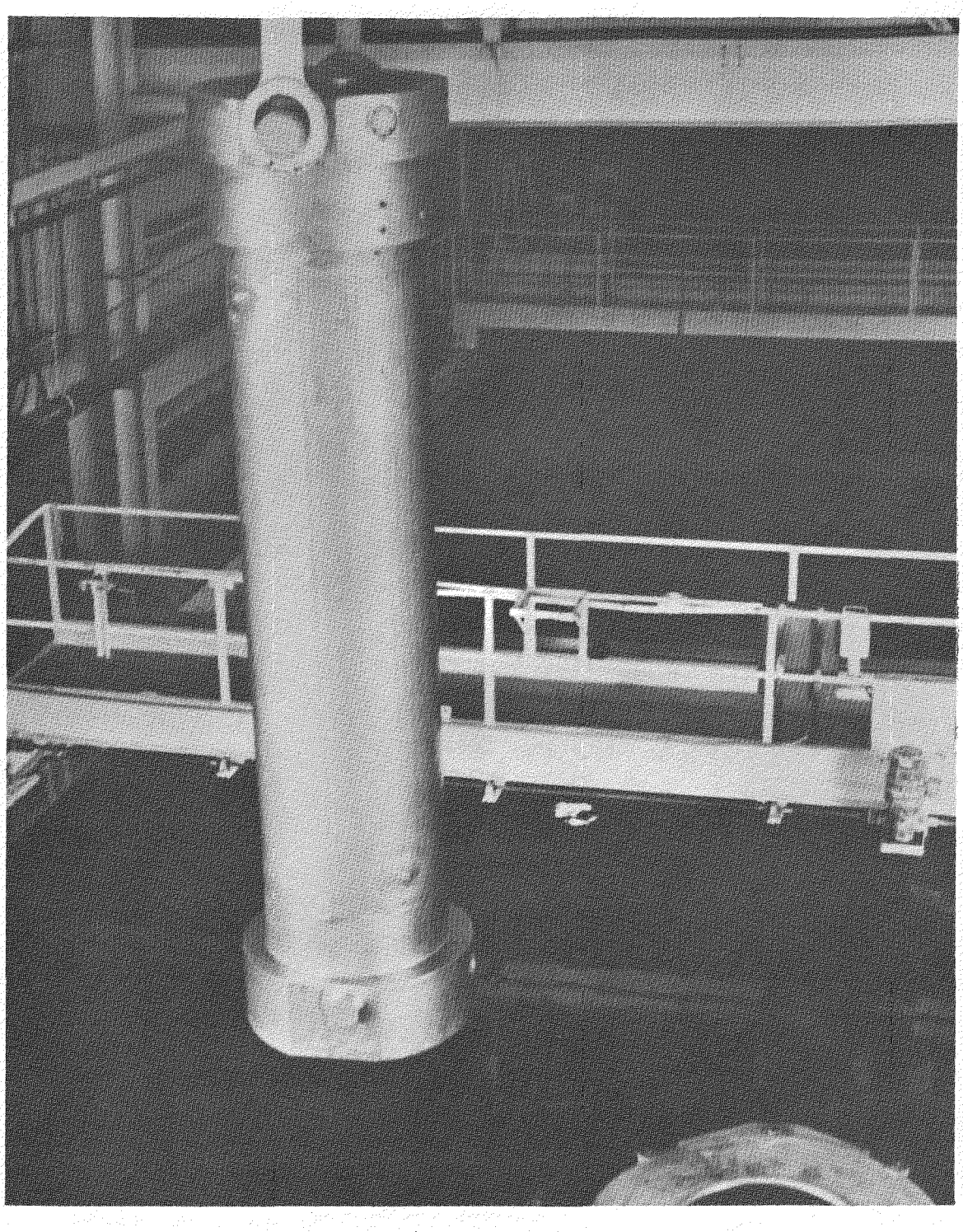


Figure 2 - NFS-4 Spent Fuel Shipping Cask
Over Unloading Pool

gamma shield and a 1-1/4 inch thick stainless steel penetration barrier. A 4-1/2 inch thick compartmentized neutron shield tank containing a borated water antifreeze solution then surrounds the cask. An expansion chamber for the shield tank is built into this section to accommodate temperature changes of the solution.

The cask lid which seals and shields the cask cavity is solid stainless steel and is attached to the cask by six high strength bolts. Two Teflon O-rings, arranged so that each may be pressure tested, provide the head seal.

Balsa wood impact limiters, encased in stainless steel, are permanently located on the sides and bottom of the cask to provide necessary crash protection. The lower impact limiter also serves as an expanded base when the cask is set vertical. A removable impact limiter of similar design is used to protect the cask head. This limiter is normally removed and stored on the transport trailer during the normal unloading operation. Necessary connections to the primary cavity (i.e., vent and drain valves, pressure test connections, and relief valves) are buried within the impact limiter and special structure to maintain integrity under all accident conditions.

Two sets of trunnions are used for normal cask handling and transport tie-down purposes. The upper set, attached to the upper impact limiter, is used for lifting the cask in conjunction with a special "swing arm" type yoke. This yoke is normally permanently locked to the lift trunnions throughout the complete handling cycle at the reactor or reprocessing site. The lower trunnions are offset to provide a gravity pivot from the vertical loading position into

the horizontal transport mode.

The most unique feature of the NFS-4 cask is its ability to handle defective fuel assemblies without the use of special check fixtures or failed fuel cans. This greatly reduces associated manpower requirements at the reactor site as well as the expense of the special hardware. A weight penalty is paid for this "zero release" concept; however, the operational ease and added safety margin more than offset this disadvantage. The primary cavity is designed to withstand temperature and pressure conditions of 532°F and 984 psig under the fire accident condition. Maximum transport conditions for design basis fuel (i.e., 130°F direct sunlight, still air, maximum fuel burnup, minimum fuel cooling period) are 345°F and 150 psig. Operational experience to date has indicated that this is also quite conservative in that the cask has yet to be received pressurized over 5 psig.

As mentioned previously, the over-the-road weight of the cask/vehicle combination must remain below 73,280 lbs. in order to travel throughout the states without restrictions. Because of the desire to simplify both cask licensing and manufacture, lead shielding was used as the main gamma shield instead of a lighter weight uranium or uranium/lead combination. This imposed a weight restriction on the transport vehicle which made it necessary to use a special trailer as illustrated in Figure 3. Since the load was relatively small, well defined, and would be loaded in the exact same position for each shipment, it was possible to vary the design considerably from the more conventional flatbed trailer (i.e., the flatbed being design optimized for a heavy distributed floor load). The weight efficient

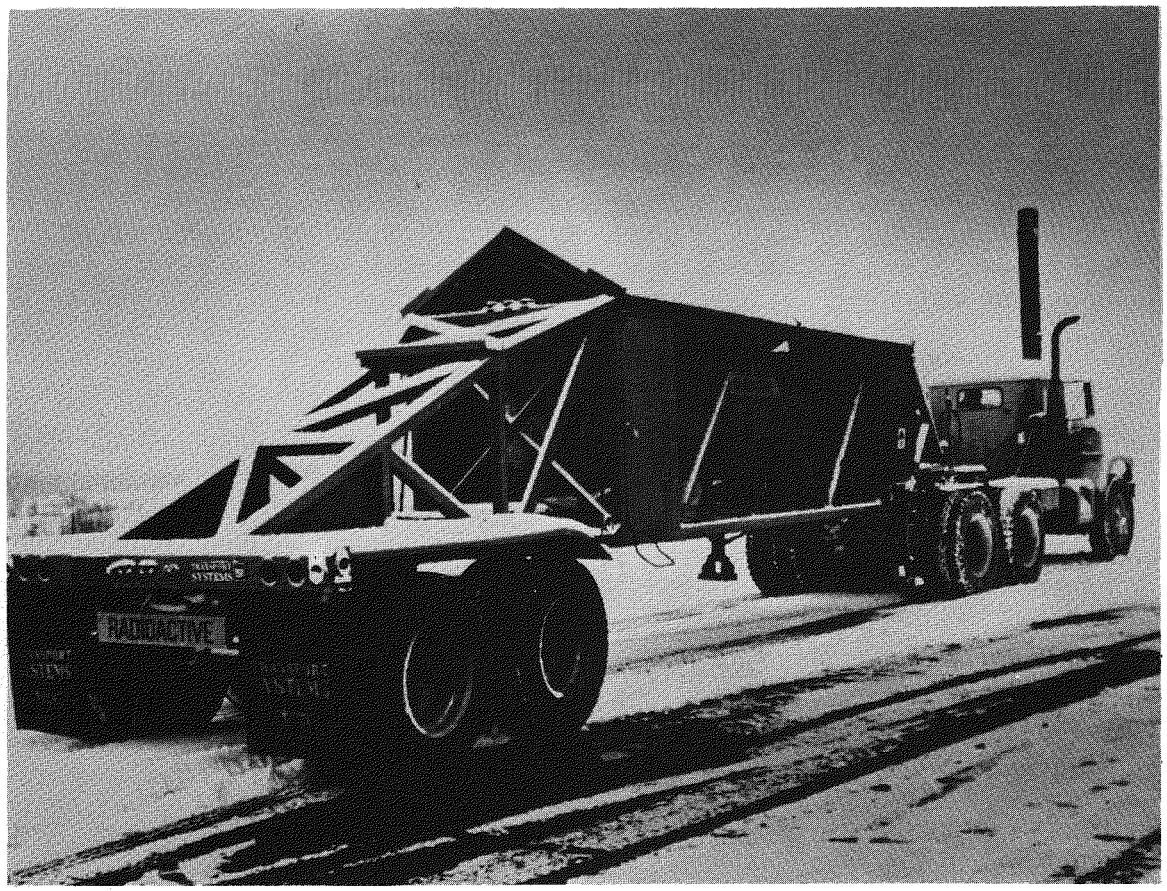


Figure 3 - NFS-4 Cask and Special Trailer

truss structure was then used in conjunction with conventional running gear to provide the desired weight advantage. Further details of this special trailer are being covered by its manufacturer, Transport Systems, Inc., Denton, Texas, later in this program.

Nuclear Fuel Services, Inc. first started receiving fuel in June 1965. Since that time it has handled a wide variety of truck and rail casks from both private transporters as well as with its own cask fleet. This experience was factored into the NFS-4 design to assure its smooth operation at both the power reactor as well as the reprocessing facility. Operationally, the cask loading procedure may be summarized as:

(1) Receive cask and obtain surface smears as may be required.

(2) Release forward and aft tie-downs.

(3) Unbolt lid impact limiter and roll forward on carriage.

(4) Attach cask yoke and remove cask from trailer.

(5) Open vent valve, attach lid lifting spider, remove lid bolts (6), and lid.

(6) Lower cask into pool and load fuel.

(7) Place lid on cask, move cask to decontamination area.

(8) Torque lid bolts and pressure test double O-ring seal.

(9) Decontaminate exposed cask surfaces while draining prescribed amount of coolant from cavity to provide expansion void.

(10) Pressure test cask cavity and valves.

(11) Check cask surfaces for contamination. Obtain H&S release.

(12) Load cask onto trailer and secure.

Since the casks were first delivered approximately 200 shipments have been successfully completed to date. Performance has been outstanding, no significant problems developed and only minor changes were required to the detailed operating procedures. Perhaps the best indication of the cask's smooth operation may be depicted by the time taken to turn the cask around at a reactor facility. The first series of shipments with the cask were made from Rochester Gas and Electric's Ginna Nuclear Power Station. The first shipment took approximately 12 hours for a complete turnaround. As the work crew gained experience and minor improvements were made in the handling procedures and the facility, this time rapidly decreased to about four hours. Their typical work shift would turn one cask around in the morning and the second in the afternoon. Similar turnarounds at the reprocessing plant combined with the rather short transport distance (approximately 110 miles) enabled this cycle to be repeated on a daily basis. The record established by RG&E's fine crew, and likely a world's record, was two complete cask turnarounds in less than four and one-half hours.

This experience is now being extended to the new NFS-5 cask design which has twice the carrying capacity of the NFS-4. While both cask types are attractive in an overall fuel cycle economic evaluation, the new NFS-5 cask obviously provides further advantages due to its reduced turnaround requirements.

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DESIGN AND LICENSING CONSIDERATIONS OF THE NLI 10/24 RAIL TRANSPORTATION SYSTEM FOR SHIPPING SPENT NUCLEAR FUEL

C.E. Williams
H.G. Clarke, Jr.
G.L. Stukenbroeker

ABSTRACT

The NLI 10/24 shielded cask and transportation system has been designed to provide the ultimate in safe shipment of irradiated nuclear fuels, a maximum payload and long equipment life. These objectives have been achieved by designing for a "zero release" system, the equipment having the capacity of 10 PWR or 24 BWR fuel elements within a 100 ton weight and with anticipated equipment life of 20 years.

The design provides for a low pressure, dry shipment with double containment for radioactivity. The cask internals are an aluminum basket with each fuel assembly pocket in the basket having a liner of stainless steel clad Ag-In-Cd alloy plate. The Ag-In-Cd provides reactivity control of the system for fresh fuels in water. Lead and uranium/steel combinations are used for gamma shielding, while water is used for neutron shielding. No cooling of the cask is required for safety. However, auxiliary cooling is provided to the outer wall of the inner container by two redundant water-to-air cooling systems. The cooling systems maintain cask and fuel temperatures as low as possible solely to facilitate cask cooldown and unloading at the reprocessing plant.

The cask and cooling systems are mounted on a specially designed 59 foot rail car equipped with 6-wheel trucks. The car and cargo will be within AAR Plate B dimensions for unlimited interchange. The rail car is designed and will be impact tested in accordance with AAR regulations.

Licensing of the NLI 10/24 system is anticipated in late 1974.

INTRODUCTION

The recent and near future light water reactors being built today place a new set of requirements on the shielded cask and transportation system designers for the shipping of irradiated nuclear fuels from the Utility to the reprocessor. The higher burnup of fuels, higher specific power levels, the use of mixed oxide fuels, the desire for shorter turnaround of the fissile materials, larger payloads and the desire for a "Universal" shipping cask, all exert their influence on current cask and transportation system design efforts.

Coupled with these engineering design criteria is the ever most concern of the federal and state regulatory agencies and the concern of the general public, which re-emphasize the need for the industry to continue with their impressive safety record. Also, the need to design, license, construct, maintain and operate these transportation systems as economically as possible and for as long as possible, is always a force which has much bearing on design. All these factors were evaluated in the design of the NLI 10/24 transportation system.

DESIGN PHILOSOPHY

The basic philosophy followed in developing the NLI 10/24 cask design was to approach as nearly as possible to a "zero release" system. Moreover, it was decided that this system should be passive in nature and not be dependant upon post accident operation of active components to achieve containment. For this reason, dry shipment with

double containment was selected as the preferred mode for large rail shipments of spent fuel. With the containment vessel filled with helium, the cavity pressure under normal conditions of transport is 23.1 psig. If all the fuel rods fail, which is a highly unlikely event, the pressure in the inner container would increase as a result of the release of fission product gas from the fuel. Data on initial rod fill pressure is not specifically reported, but based on available information, an upper limit of 500 psig is assumed for PWR type fuels and 200 psig for BWR type fuels. Although these rod fill pressures are extremely high, the object is to establish an inner container design pressure that would include the hypothetical condition of a gas release from all rods in a fuel assembly and also accommodate future fuel designs which could involve increased fuel rod pressures. During transport, the maximum pressure based on this assumption would be 80.5 psig. The maximum possible cavity pressure is only 104.5 psig, which occurs at the extreme temperatures reached after the hypothetical fire accident, and only if all the BWR fuel pins rupture. The NLI 10/24 containment vessel has a maximum allowable working pressure several times greater than the maximum possible internal pressure of 104.5 psig. The NLI 10/24 dry cask concept does not require a safety relief valve. No release of the helium filler gas is predicted under normal conditions of transport or hypothetical accident conditions. The filler gas is considered

to be part of the package contents, thus eliminating the concern over coolant activity levels.

By contrast, with a wet shipment, normal operating pressure would be on the order of 400 to 500 psig. Loss of primary coolant would occur as a result of the hypothetical accident sequence.

Pertinent design criteria for the NLI 10/24 Spent Fuel Shipping Cask is listed in Table 1.

CASK SYSTEM DESCRIPTION

The NLI 10/24 shipping cask has been designed to provide double containment of either intact or failed fuel assemblies. The package as presented for shipment consists of the cask body, fuel basket, cask cavity closure head (inner closure head), the cask body closure head (outer closure head), as shown in Figure 1. Impact structures, which protect the ends and sides of the cask, are placed over both ends of the cask. The impactors are attached to the tie-down structure after the cask is mounted on the rail car.

The cask body consists of a .75 inch thick inner stainless steel shell and a 2 inch thick outer stainless steel shell which are joined by stainless steel forgings at each end to make a continuous weldment. The annulus between the inner and outer cylinders contains a lead gamma shield which is 6 inches thick. Uranium shielding is used on the ends of the cask and at strategic locations in the wall of the cask. Neutron shielding is provided by 9 inches of water contained

Table 1. NLI 10/24 Spent Fuel Shipping Cask Data

Capacity	10 PWR Type Fuel Assembly	or	24 BWR Type Fuel Assembly
Fuel Data	10 PWR		24 BWR
Envelope, inches	8.60 sq. x 171.5		5.44 sq. x 176.25
Enrichment w/o U-235	3.35		2.65
Weight of Uranium, kg	4,540		4,728
Avg. burnup, MWD/MTU	35,500		29,700
Avg. specific power, kw/kg	36.3		25
*Total Decay Heat, kw	97.2		88.8
Cask weight, loaded	Calculated Design weight		193,000 lbs. 200,000 lbs.
Cask Envelope	88 in. dia. 204.5 " long		
Internal Cavity Dim.	45.0 in. dia. 179.5" long		
**Max. normal operating pressures			
Cask cavity (primary containment)	80.5 psig		
Water Jacket	345 psig **		

** Water jacket relief is set for 320 psig

* Does not include 1.2 axial peaking factor

** Maximum normal operating pressures are based on operation without auxiliary cooling and total fission gas release.

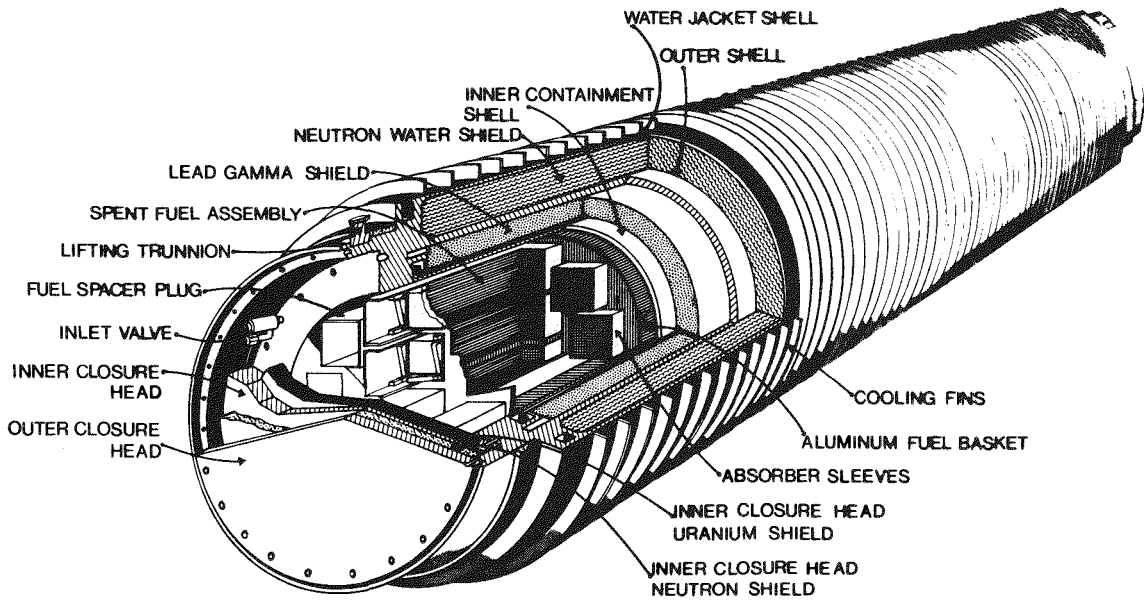


Fig. 1 NLI 10/24 Rail Cask Assembly

in a finned stainless steel jacket surrounding the outer shell. The water jacket extends axially over the full active fuel zone of the fuel assemblies. The closure head end or open end of the cask cavity is machined from a stainless steel forging. The length of the forging is such that it forms part of the cask cavity. This design provides the necessary material from which two closure head flange seating surfaces can be machined. The design also provides the opportunity to have the cask body penetration pass through the solid stainless steel forging, eliminating the need for a penetration through the shielding materials and the inner and outer shells compromising the integrity of the cask structure. The head end forging also provides maximum protection of the inner and outer closure heads, closure head seals and primary containment system valves, since they are all located well within the outer dimensions of the forging. The cask lifting trunnions are located in the head end forging, which eliminates the concern of a puncture of the shielding containment shell by an impact on a trunnion. Four trunnions are provided to accommodate those utilities having redundant crane systems. The neutron shield water jacket is constructed of 3/4 inch stainless steel with a short heavier shell section at each end to provide an adequate transition section from the shell to the upper and lower forgings. The water jacket design was developed to provide structural integrity and resistance to penetration under normal conditions of transport as well

as to contribute to the overall strength of the cask in the hypothetical accident conditions. Expansion tanks are provided in order to maintain a solid water shield. Under maximum normal operating temperatures the water jacket operating pressure is 300 psig. The water jacket design pressure is 345 psig.

There is one penetration through the head end forging of the cask body, which exits into the space between the inner and outer closure heads and is outside the primary containment seal area. This penetration is used to drain the area between the inner and outer closure heads as the cask is removed from the spent fuel storage pool and to pressure test the secondary containment system prior to shipment of the loaded cask.

The cask body closure head (outer closure head) is a flat stainless steel plate 2-1/2 inches thick, held in place by 24, $1\frac{1}{4}$ inch high-strength studs. There are no penetrations through the outer closure head since the single necessary service penetration is made through the cask body end forging. The outer closure head, which is the secondary containment boundary, also serves as a rugged valve box cover for the four service penetrations in the inner closure head cask flange.

The cask cavity closure head (inner closure head) consists of a stainless steel forging, the center section of which is filled with uranium and is covered by a stainless steel plate welded to the forging. The bulk of the required gamma shielding is contained in the inner head to provide operator protection during handling operations involving a

loaded cask. The closure head is held in place by 20-$1\frac{1}{2}$ inch high-strength studs. The inner closure head seal is a metallic ring, which is more than adequate to withstand the temperatures expected as a result of the hypothetical fire accident. There are four penetrations in the inner closure head cask flange which are used for servicing the cask cavity. These penetrations are equipped with $1\frac{1}{2}$ inch globe type angle valves and terminate in the secondary containment volume between the inner and outer heads. The globe valves are designed for maximum leak tightness under fire accident conditions. The internal cask cavity, which is 45.0 inches in diameter, has been sized to accommodate removable aluminum fuel baskets. Separate baskets are designed to carry either 10 PWR type or 24 BWR type fuel assemblies. The aluminum basket covers the active length of the PWR and BWR fuel and are constructed from forged aluminum disc. Axial channels are located in the basket to accept individual fuel assemblies. Criticality control is achieved by means of box structures which line the fuel channel openings in the aluminum basket. The box is constructed of Ag-In-Cd plates encapsulated in 304 stainless steel. This design results in a K_{eff} of 0.852 ± 0.014 for the 10 element PWR arrangement with an U-235 enrichment of 3.35%. The 24 element BWR arrangement has a K_{eff} of 0.901 ± 0.014 with an average U-235 enrichment of 2.65%. The box structures are supported at both ends by means of stainless steel plates to form a self-supporting structure independent of the aluminum

basket.

The aluminum basket serves as a heat conduction path from each individual fuel assembly to the cask cavity wall. Decay heat is removed from the cavity wall by means of cooling water circulated through channels of one inch pipe cut longitudinally in half and welded to the outside surface of the inner shell in an axial direction. The design provides for two (2) separate independent coolant flow paths. There are twenty (20) cooling channels, ten (10) for each system and can be seen in Figure 2. These cooling channels terminate in two (2) separate header arrangements machined into the top and bottom forgings of the cask body. The inlet and outlet penetrations connecting the cask system to the heat exchanger systems pass thru the solid end forging material. With one cooling system in operation and a maximum design decay heat load (including 1.2 axial peaking factor) the cask temperatures in the stipulated Normal Conditions of Transport would be as follows:

Outer Surface	227° F
Inner Shell	268° F
Aluminum Basket (Max.)	451° F
Fuel Assembly (Avg.)	690° F

When the cooling system is not operating, heat dissipation from the inner shell occurs by conduction through the lead and outer shell, natural convection and radiation to the environment from the finned outer surface

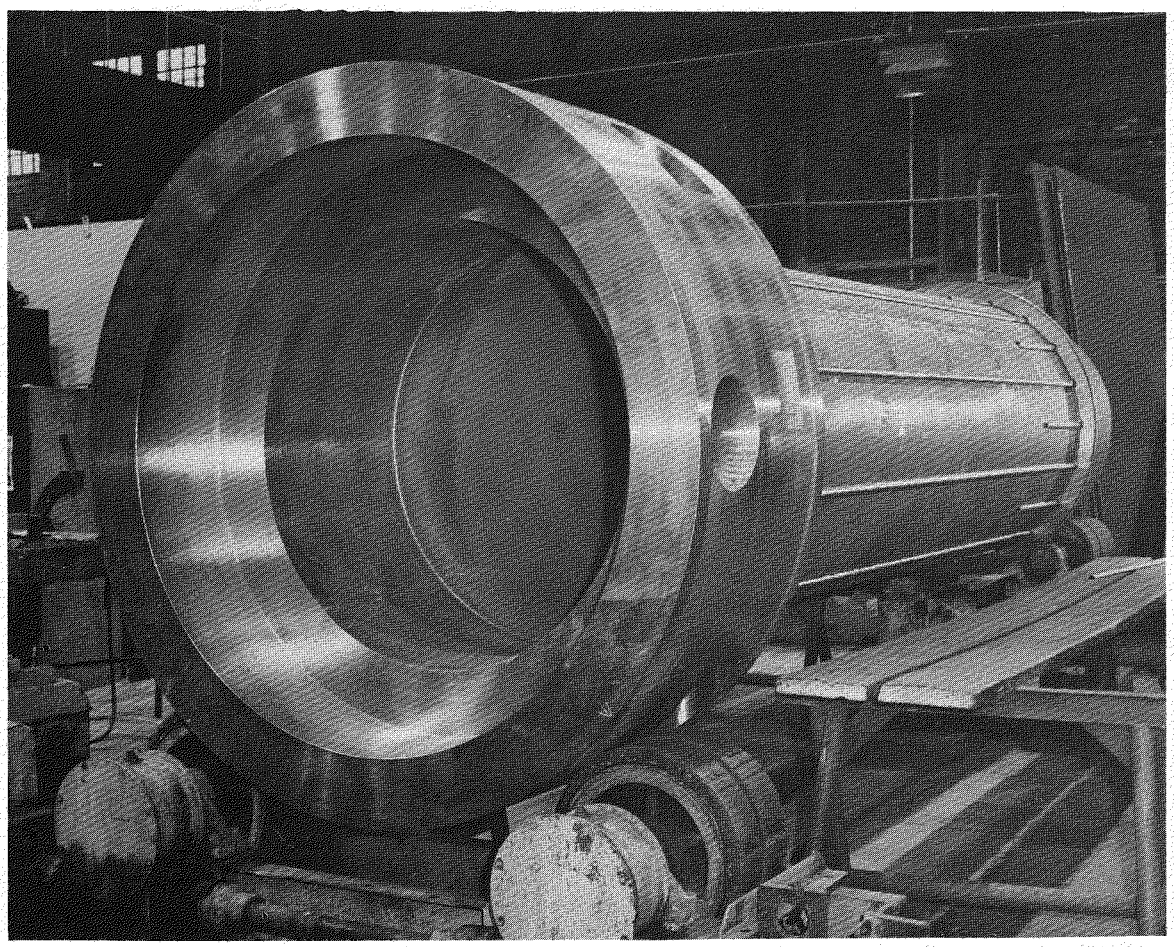


Fig. 2 NLI 10/24 Rail Cask Head Forging and Inner Shell Weldment showing location of cooling channels.

of the water shield jacket. The resulting temperatures in this case would be:

Outer Surface	407° F
Inner Shell	541° F
Aluminum Basket (Max.)	757° F
Fuel Assembly (Avg.)	899° F

Impact limiting structures are integrated into the tie-down structure at each end of the cask. The end impact limiter is a cylinder of balsa wood encased in aluminum sheet. The side impact limiter is a circular ring of white pine encased in aluminum sheet. The impact limiters are designed such that the kinetic energy of the system upon impact is spent in the progressive crushing of the impact structure yet at the end of the failure mode there has been no contact between any portion of the cask body and the impact surface.

The NLI 10/24 Spent Fuel Shipping Cask has a fully loaded weight of 193,000 pounds. The transport system consists of a specially designed rail car with the cask tie-down arrangement built into the car frame. An arrangement drawing of the rail system is shown in Figure 3. The rail car is equipped with redundant cask cooling systems which are used to maintain lower in-transit cask and fuel temperatures to facilitate cask cooldown and unloading operations. The rail car-mounted cooling system is a closed circulating water loop consisting of a pump and a water-to-air heat exchanger in series with cooling channels located on the cask

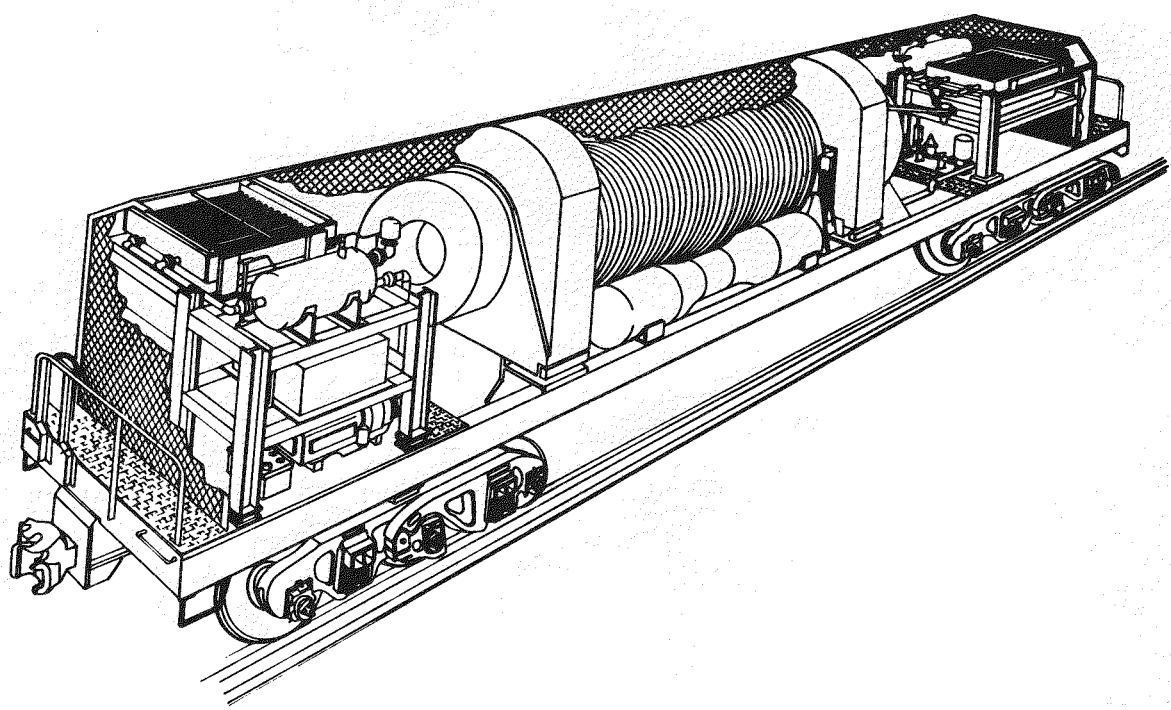


Fig. 3 NLI 10/24 Rail Cask/Car System

inner shell. The system is normally run on forced convection with power supplied to the pump and heat exchanger fan by means of a diesel generator. In the event of loss of power or loss of an active component, automatic transfer to a redundant rail car mounted system takes place. Should power be lost in both systems, decay heat removal will be accomplished by boiling natural convection in both systems. In the ambient condition of still air at 130° F , the average coolant temperature is 181° F with forced convection and 250°F in the boiling natural convection mode.

A personnel barrier completely covers both cooling system packages as well as the cask under normal transport conditions. The personnel barrier is basically an aluminum cage made from expanded aluminum sheet on an aluminum frame.

THE GENERAL ELECTRIC RA-SERIES PACKAGE

B. D. Wilson

INTRODUCTION

Each new General Electric Boiling Water Reactor (BWR) involves the shipment and, normally, the temporary storage of up to 848 fuel assemblies for the initial core loading including spares. After nearly ten years of service, the RA-Series fuel shipping container has proved to be safe, versatile, economical and relatively simple to use. This paper points out the design features of this container system and summarizes the various fuel loadings currently authorized.

RA-SERIES PACKAGE DESCRIPTION

BWR Fuel

The important safety characteristics of the RA package are perhaps more easily understood after a brief description of the fuel assembly for which the shipping package was designed.

The typical GE BWR fuel bundle consists of a square array of fuel rods spaced and supported at each end by means of stainless steel tie plates and by Inconel spacers at intervals along the length. Overall dimensions of the bundle are about 14 feet long and 5-1/2 inches square.

The typical fuel rod of which the bundle is composed consists of a zirconium tube containing a 12-foot cylindrical column of one-half inch diameter UO_2 pellets. The uranium is enriched to a maximum of about 3% U-235. The ends of the rods are sealed by welded Zircaloy plugs in both ends of the tubing.

Future utilization of plutonium recycle fuel in BWR's requires testing of bundles containing some rods loaded with mixed oxide in assorted PuO_2 to UO_2 ratios. Mixed-oxide fuel specifications have not yet been standardized as fully as for low-enriched uranium oxide. For that reason,

the RA-Series package has been licensed for shipment of a specific mixed-oxide fuel model, as further described in the discussion of criticality safety.

Packaging

The GE Model RA (reactor assembly) packaging consists of two containers, an inner metal box which is cushioned within a larger wooden box. (See Figure 1.)

The fuel bundles are accommodated within the inner metal box. The box consists of an outer shell and perforated inner basket separated by structural angle spacers. The outer shell is formed of 16-gauge, carbon-steel plate, with an integral-welded end of the same material. The inner basket is also 16-gauge steel plate with 3/4-inch perforations on 1-3/4 inch centers. This inner basket is welded to the upper edge of the outer shell to form two u-shaped channels about 6-7/8 inches square, which may be lined with cemented, low-density cushioning, so long as the perforations match the size and location of those in the basket. The inner basket is supported within the shell by four 1/8-inch thick angle-iron spacers positioned longitudinally along the entire length of the body.

A cover and an end cap constructed in a similar fashion to that of the box provides a 2-inch annulus around the fuel when the box is closed. Four 1/8-inch thick angle stiffeners are welded on 4-inch centers to the outer surface of the closed end. A manually operated pressure relief valve in the outer shell passes two cubic feet per minute of air automatically whenever there is as much as half pound per square inch pressure difference between the outside and inside of the box. The cover and the end cap are equipped with a half-inch thick neoprene gasket which provides a full cover seal to protect product quality in normal transport. Two alternate means of closure--latching devices or nut and bolt sets--hold the cover and end cap in place.

The outer container is a wooden box fabricated of one-half inch plywood, 2 x 4 cleated, on a 2 x 6 tongue-and-groove platform with bolted 4 x 4 skids. The original dimensions were nominally 207 inches long,

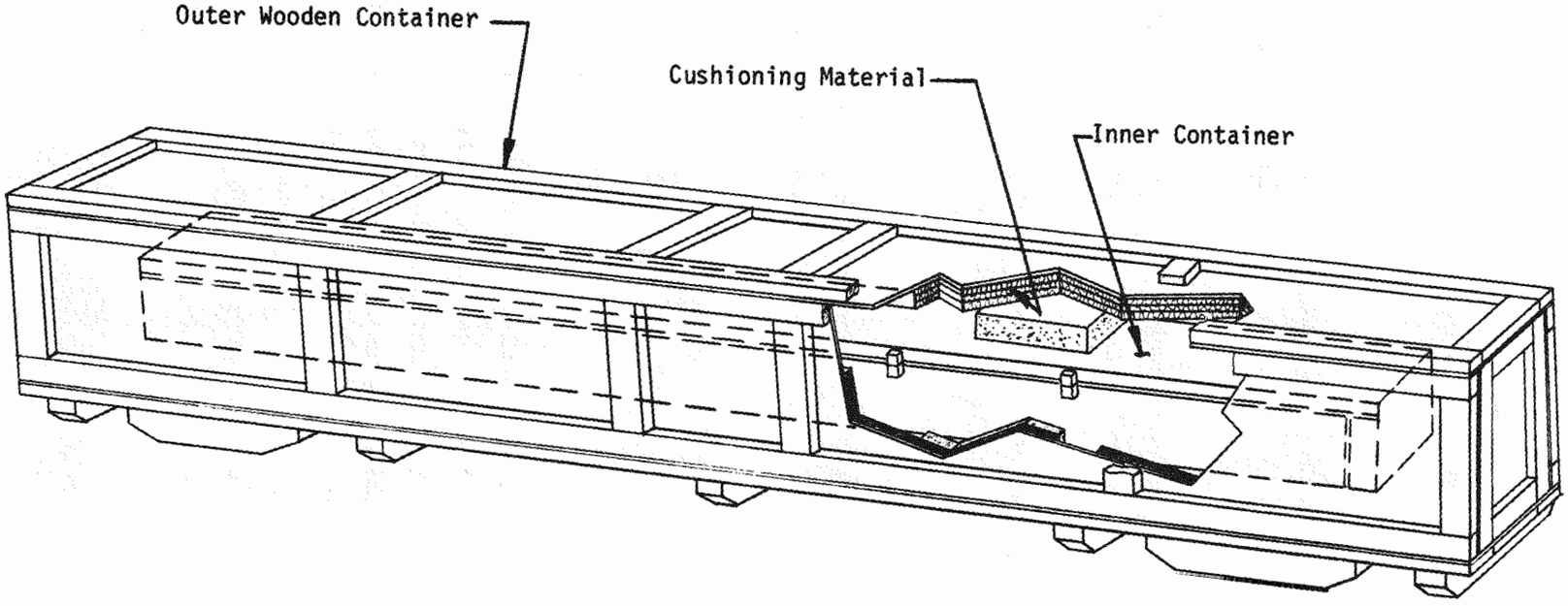


Fig. 1. RA Series Fuel Element Shipping Container.

33 inches high and 32 inches wide. The current design is the same length but is about two inches less in height and breadth. Fiber honeycomb and ethafoam pads are placed strategically inside the wooden box to cushion and support the inner metal container with its two fuel assemblies.

The outer wooden box, with its cushioning, performs two functions. First, it provides a shock-mitigating device for the inner metal box containing the reactor fuel bundles. Secondly, it provides sufficient shock-mitigating properties to retain the fuel bundles intact within the inner box.

TESTS

Any unirradiated fissile material shipping container must perform two basic safety functions during transport:

1. To prevent the loss or dispersal of the radioactive material.
2. To prevent the accumulation of enough fuel in a geometry that could make criticality possible.

Thus, in the normal transport and hypothetical accident test series prescribed by 10 CFR Part 71, these two functions must be demonstrated. It is at once apparent that loss or dispersal will not occur if the fuel remains within the inner metal box throughout the test series. The fuel will not burn nor dissolve in water and therefore needs only to resist the impact testing.

With respect to maintaining subcriticality during the hypothetical accident series, no credit is taken for the outer wooden box which is considered to be consumed in the accidental fire at 800°C. It is also assumed that the inner metal box permits leakage of water. Thus, impact testing alone is required. In 1966, a prototype RA-1 package containing two simulated fuel assemblies, each weighing 632 pounds, was test-dropped from a height of 30 feet. An end edge where the top cover of the inner metal box joins the end cap was selected as the most vulnerable impact area from the standpoint of potential release of the fuel as well as maximum deformation of the inner container.

The impact separated the end of the outer box from the body exposing the inner container but leaving it entirely within the outer box. The end cap had to be pried off with a tool to permit subsequent removal of the cover to inspect the simulated fuel. Otherwise, there was scarcely any deformation.

Over the subsequent eight years, a number of minor design modifications were authorized on the basis of engineering assessments. On March 22, 1974, a second 30-foot drop test verified the quality of those assessments. The integrity of the outer box and its cushioning was again demonstrated to protect the inner metal unit and its fuel against loss and against criticality.

CRITICALITY EVALUATION

Each package accommodates two fuel bundles--either in the form of reactor-ready assemblies or as unassembled individual fuel rods simply bound together into bundles. These bundles fit into the perforated basket of the inner metal container. The basket is a key criticality safety feature of the Model RA design. In effect, it suspends the two fuel bundles within a larger cavity so that at least a two-inch thick surrounding annulus is always present. Thus, if an array of RA packages were to become accidentally flooded, a 2-inch thick water blanket would absorb neutrons thereby overmoderating the array substantially.

As previously stated, the accident-test program demonstrated that the loaded package survives a 30-foot drop with no loss of fuel bundles from the inner metal container and no significant deformation of that container. Since the outer wooden container and its cushioning materials are presumed to be consumed in the accidental fire, only the spacing provided by the inner metal unit is considered in the criticality safety evaluations of the accident conditions.

Subcriticality of a single, damaged and reflected inner container loaded either with 5% U-235 enriched rods or with fuel bundles, is demonstrated by the fact that the geometric buckling is greater than the material buckling. Additionally, the same container fully loaded with 6.5%

U-235 enriched UO_2 rods or with mixed UO_2-PuO_2 rods (MO_2) in a nominal 5-inch pipe product container has been demonstrated subcritical using the KENO[1]-Monte Carlo criticality program.

An array of damaged packages loaded either with 5% enriched rods or with fuel bundles was evaluated some eight years ago by calculative methods incorporating standard GE criticality safety methods. A maximum k-infinite value of 0.93 resulted from calculations over a wide range of interspersed moderation assumptions.

Early this year, it was necessary to analyze the RA package with mixed oxide rods as the contents, as contained within a nominal 5-inch diameter steel pipe with a 1/4-inch thick wall. At the same time, calculations were also made for higher enrichment (up to 6-1/2%) UO_2 rods. A fuel rod model somewhat more reactive than actual designs was used in the calculations.[2] The model is considered to have an outside pellet diameter of 0.508 inches, at a density of 11 grams/cc, as 8 w/o PuO_2 in natural UO_2. The fissile plutonium concentration is taken to be 85 w/o of the total plutonium. The model fuel cladding is 30-mil Zircaloy 2, with a clad-to-fuel gap of 5-1/2 mils. As contained in the pipes, the rods are sleeved in 6-mil thick polyethylene, which gives a 12-mil edge-to-edge spacing on triangular pitch. To simplify the calculations, it was conservatively assumed that all the space within the 5-inch diameter pipe, not occupied by fuel rods, was fully flooded with normal density water.

Using the KENO-Monte Carlo computer code, the package was analyzed[2] under normal transport and three potential situations that could result from the 10 CFR Part 71 hypothetical accident series:

1. "Normal" transport - An infinite array of undamaged packages containing the model fuel rods in flooded pipes has a multiplication constant of 0.7715 ± 0.0052.

2. Accident situations - A 13 x 20 array of 260 damaged packages, i.e., model fuel in flooded pipes remain in the inner metal container, and

   - Case A - Containers in tight, water-reflected array and completely immersed in water (fully flooded). k-infinity = 0.53022 ± 0.00743.

- Case B - Containers dry but separated by a maximum reactivity water blanket computed value 0.15-inch thick between each. k-effective = 0.9076 ± 0.0051.
- Case C - Containers in tight, water-reflected array but all spaces between product containers occupied by water at a density of 0.075 grams/cubic centimeter; that density giving the optimum degree of moderation in the array. Maximum k-effective = 0.92718 ± 0.00587.

The calculations demonstrate the RA-Series package containing authorized fuel loading is correctly categorized as Fissile Class I.

In addressing the MO_2 shipping problem, it was realized that the plutonium 239, 240 and 241 isotope ratios vary considerably in fuel fabrication feed material delivered from fuel reprocessing plants. Since the fissile characteristics of these isotopes are different, it would seem nearly impossible to analyze the RA package for all the possible mixed oxide isotopic combinations that might be shipped. This problem was resolved by deriving a set of curves[3] establishing a relationship between the Pu-241 to Pu-240 percentage ratios as a function of Pu-239 percentage at four different PuO_2 weight percentages in natural UO_2. Figure 2, which is taken from reference 2, shows three of the curves. Each point on the curve represents the Pu isotopic "mix" in fuel which has the infinite multiplication factor of the model fuel in the RA-Series package. Thus, any Pu-241 to Pu-240 percentage ratio less than the values shown on the curves will be less reactive than the model. Values not exceeding those on the associated histograms which were used to tabulate conservative values for the license application may therefore be shipped in Model RA, in accordance with current AEC license authorizations.

QUALITY ASSURANCE

Two separate quality assurance programs are used in connection with the RA-Series package:

1. Inspection of new and returned packaging, and
2. Inspection of the loading and closure of the package.

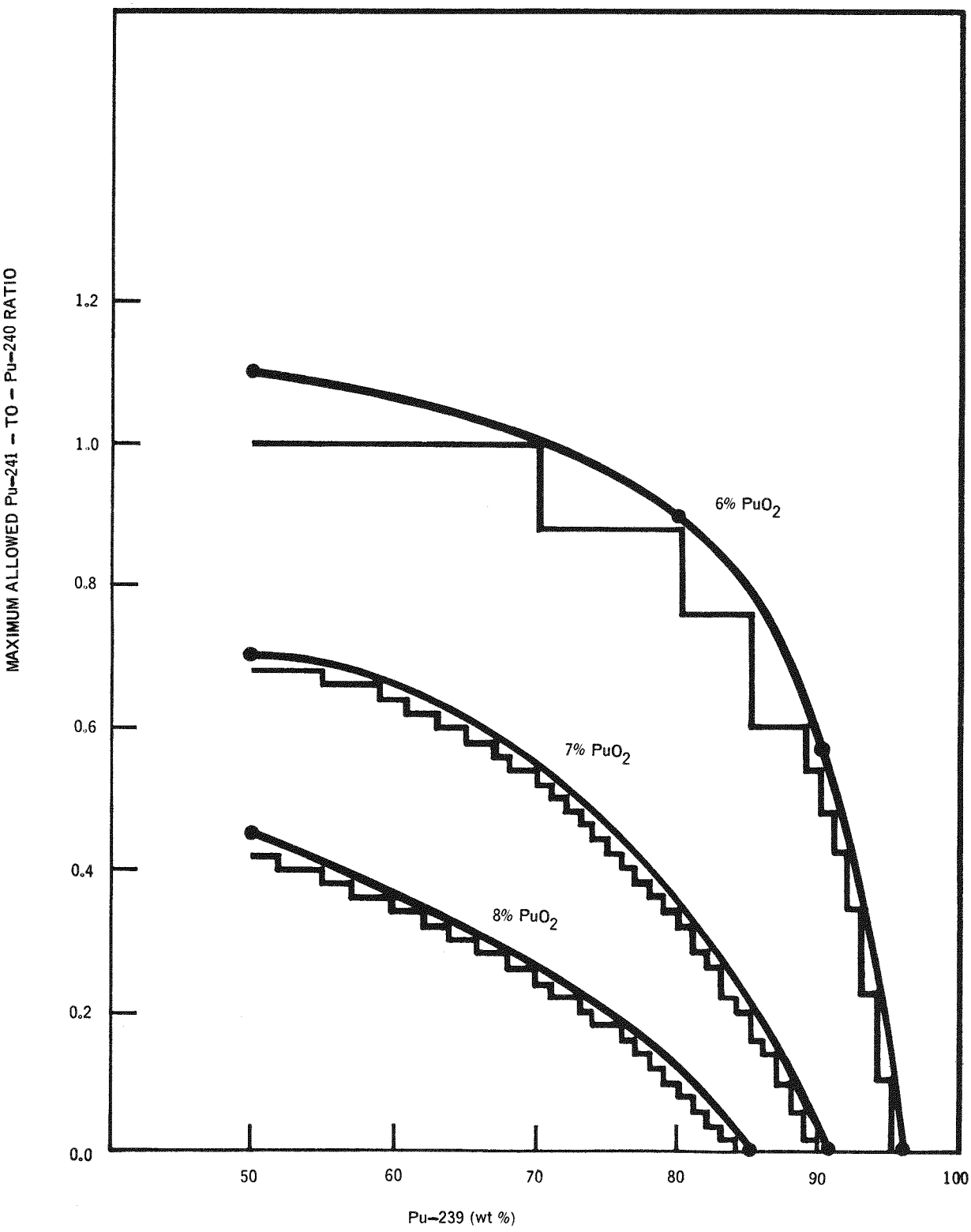


Fig. 2. Maximum Allowed Pu-241/Pu-240 Ratios as a Function of Pu-239 wt% for Different PuO_2 Concentrations in Natural UO_2.

New or returned packaging is inspected prior to transfer into the container loading area of the fuel shop. An Inspection and Repair Record sheet provides a list of some 15 inspection points and the inspector initials the sheet as each point is completed. Included are determinations that the angle separators between the basket and outer skin are in place, and all closure devices, i.e., latches or bolt-down lugs, are present and in good condition, that the ethafoam perforations match those of the inner basket, and that the covers and end caps are in good condition and fit the body. There are similar kinds of check points for the outer container and its cushioning.

A 24-point inspection list covers the loading of fuel bundles in the inner box, its closure and security sealing. In turn, the record calls for inspection of the presence and placement of cushioning and proper placement of the inner container in the outer box and its closure.

SHIPPING AND MAINTENANCE

Although a number of minor design modifications have been made, none have been required on account of transport safety. The inner container of the first system, Model RA-1, was lengthened by an elongated end cap (Model RA-2) and by a longer body (Model RA-3) with appropriate changes in the cushioning within the outer box. Latches on the inner metal box required high maintenance, and bolt-down lugs were substituted.

In order to utilize Seavan transport containerization practically, the outer wooden box was reduced a few inches in girth and retested to verify it still retains the inner box intact. Cleated plywood construction permits simple refurbishing and facilitates forklift errors. The inner metal unit is carefully handled by reactor people who understand the value of its contents. The one outstanding mishap in shipping the RA-Series packages was an accident wherein a tractor-trailer load of fuel turned over on its side as the driver rounded a curve too fast. None of the fuel packages broke open; the trailer was simply detached from the load, righted and towed away. The packages were loaded on another truck and returned to the fabrication plant for evaluation of possible damages.

AUTHORIZATIONS

Currently, May, 1974, the RA-Series is licensed by the Atomic Energy Commission under Amendment No. 71-16 to License SNM-54, Amendment No. 71-57 to License SNM-960, and Amendment No. 71-11 to License SNM-1097, as revised. The package is identified by the AEC as USA/4986/BF. An IAEA Certificate of Competent Authority for foreign shipments was issued by the U.S. Department of Transportation, Office of Hazardous Materials on January 25, 1974.

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2. Ricardo Artigas, "General Electric Model RA-Series Shipping Container for Unirradiated BWR MO_2 and UO_2 Fuel Rods - Criticality Safety Analysis," NEDO-11278, General Electric Company (February, 1974).

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LMFBR SPENT FUEL SHIPPING CASK

Allen H. Flagg

ABSTRACT

The LMFBR Spent Fuel Shipping Cask is a double container package with a separable canister containing a nine position rotary magazine which may be loaded through a single off center port in the canister head. The outer shield assembly may be used for the transport of other radioactive material.

The magazine incorporates a central expansion void with a syphon line from the top of the coolant to the base of the void chamber, in either the loading or the transport orientation. The 30 minute fire accident causes a maximum pressure of 250 psia within the canister with the diphenyl eutectic coolant at 600°F. Containment is assured by means of metal O-ring seals on all closures of both the shield and canister penetrations.

The energy absorber structure is comprised of tapered wall shells of stainless steel extending 12 inches beyond the end closures and three radial rings on the sides. This energy absorption material is sized to provide sufficient sacrificial material to bring the cask to a stop, following the 30 foot drop accident.

Heat transfer analyses have been done with a thermal transient computer program which models the forward end of the cask from the central radial ring to the closure end. Each fuel position is modeled with a coolant flow parameter analagous to the expected natural convection as it effects the axial heat transfer.

Criticality analyses have been done using the KENO program modeling the normal configuration and side and end drop hypothetical accident configurations. The maximum reactivity configuration results in $k_{eff} < 0.9$.

The cask is supported on a skid, with an enclosure over the cask, designed to fit a depressed bed, 125 ton capacity, railroad car. Auxiliary equipment is carried in boxes also mounted on the car.

LMFBR SPENT FUEL SHIPPING CASK

GENERAL DESCRIPTION

The LMFBR Spent Fuel Shipping Cask is designed to meet the transport of radioactive materials regulations and the operational requirements of the fuel handling systems of the Fast Flux Test Facility, FFTF, the Clinch River Breeder Reactor, CRBR, and an as yet undefined processing plant. The cask design is independent of developmental requirements but it does incorporate several interesting features to satisfy the regulations and service requirements.

Figure 1 illustrates the three major subassemblies of the cask. The Shield Assembly incorporates a 4.5 inch thick depleted uranium cylindrical shield within a 3 inch thick stainless steel shell with integral energy absorber structure and the carbon steel fins. The Canister is a separable container which lends versatility to use of the Shield Assembly for other radioactive material transport.

The Magazine Assembly is supported on bearings within the canister to allow rotation for loading through the single port in the canister head. A 12-foot FFTF fuel assembly and magazine adapter is also illustrated in Figure 1. A 14-foot CRBR fuel assembly would use a shorter adapter. The full length of the magazine is 15 feet, sufficient to hold an individual CRBR fuel assembly canister.

STRUCTURAL DESIGN

The 30-foot drop accident analysis dictated the shield structural design features illustrated in Figure 1. A tapered wall shell of stainless steel extending 12 inches beyond the end closures, serves as a sacrificial energy absorber structure for end and corner drop conditions. The energy absorber has been sized to provide sufficient sacrificial material to bring the cask to a stop before the penetration reaches the shield head. The maximum penetrations into the energy absorber are illustrated in Figure 2 for various angles of impact from 0° (end drop) to 90° (side drop). Three radial rings of stainless steel

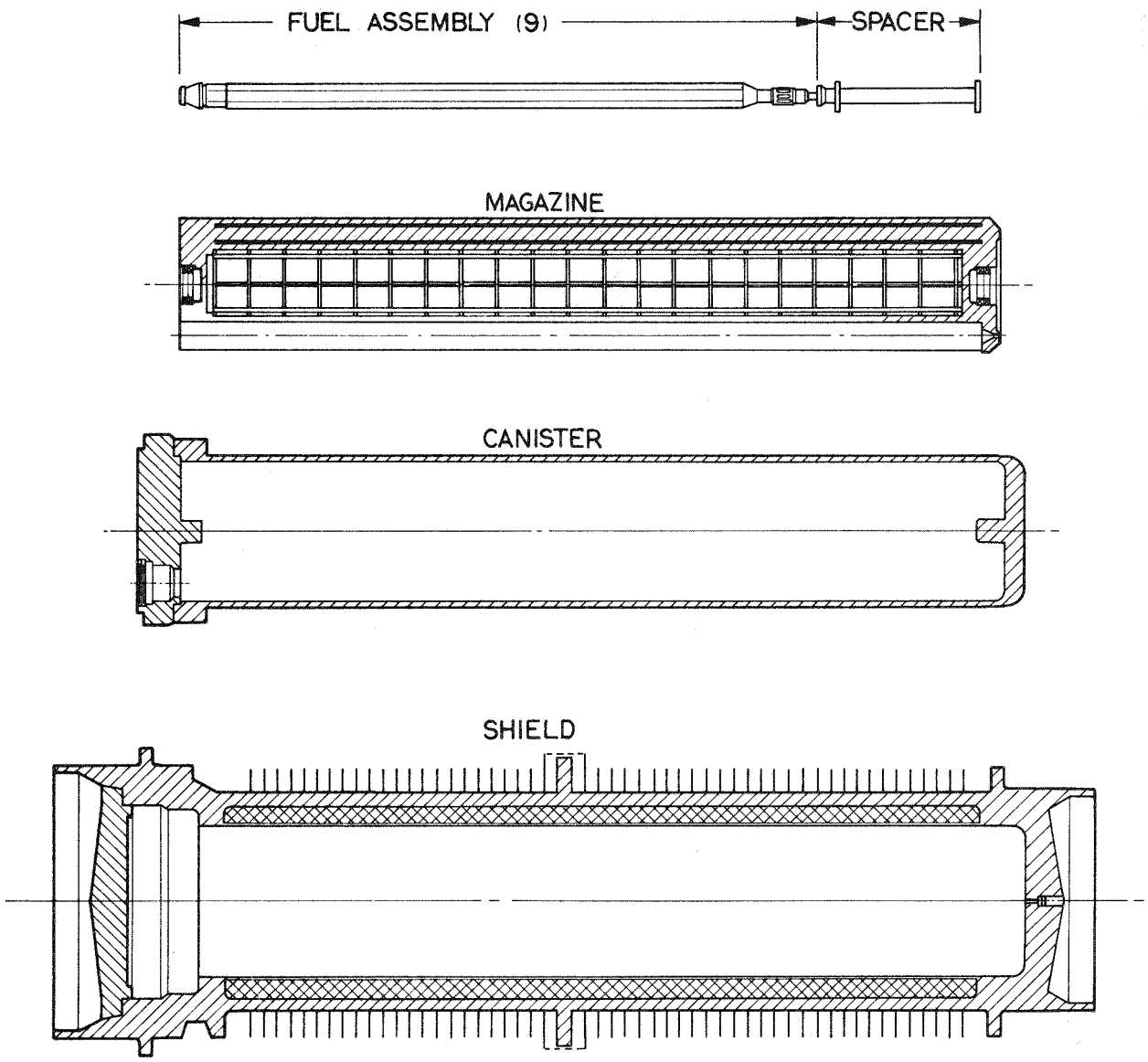


Fig. 1. LMFBR Spent Fuel Shipping Cask.

serve as energy absorbers for a side drop accident. The cooling fins are expected to buckle thus providing negligible energy absorption in a side drop.

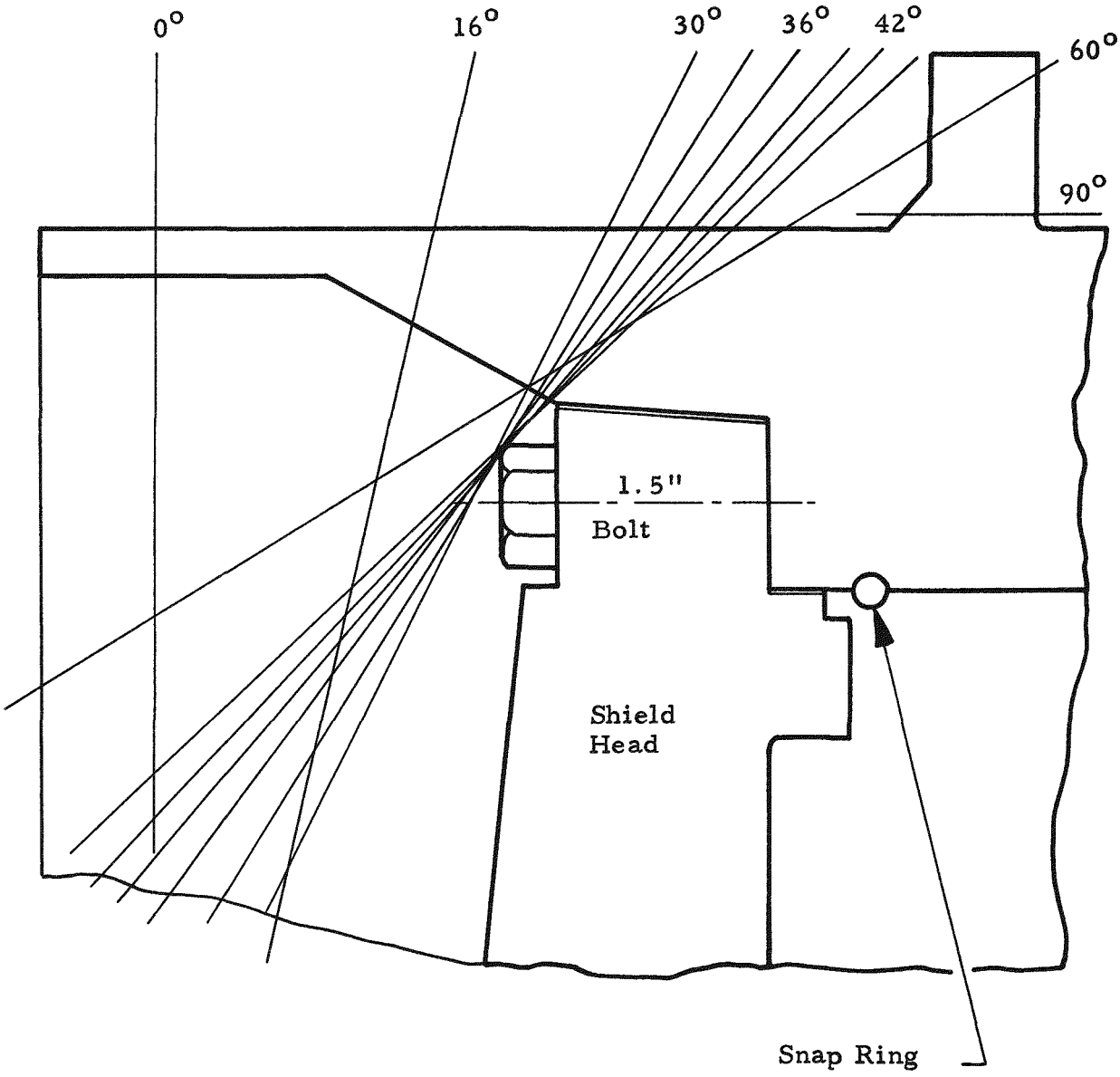


Fig. 2. Energy Absorber.

The canister head is secured to the canister with 36 bolts and to the Shield Assembly with nine of the bolts extending through the canister flange to the shield structure. In an end drop accident these

nine bolts could stretch enough to possibly impose a load on the shield head closure. To avoid this a "snap ring" is installed in the shield structure between the canister and the shield head.

The Magazine Assembly is a nine position rotating structure with a central region serving as an expansion chamber to allow the diphenyl eutectic coolant to expand under extreme temperature rises, as in the 30 minute fire accident. A syphon line extends from the highest point of the magazine in the canister to the lowest point in the expansion chamber. As the coolant expands, it enters the high point of the syphon line and penetrates to the expansion chamber where the helium cover gas pressure forces the coolant in the expansion chamber back into the canister. Thus the fuel is always covered with coolant and there is no need to vent the potentially contaminated coolant to the atmosphere.

HEAT TRANSFER

Heat transfer within the canister is by natural convection of the coolant so that the axial temperature gradient is minimized and the full set of fins along the length of the cask are effective. Heat is transferred to the canister wall by the natural convection coolant currents and by conduction to the carbon steel fins on the cask exterior. Under the extreme conditions of a 26 Kw heat load and an ambient temperature of 130°F the fin tip peak temperature is less than 180°F. The central radial ring may reach a temperature of 220°F. The fins on each side of the radial ring are spanned with a screen to prevent access to this excessive temperature.

A transient thermal analysis computer program has been prepared for analysis of the LMFBR cask as illustrated in Figure 3. An axial coolant flow model has been devised to provide an analog of natural convection currents. This model correlates well with the Oak Ridge Simulator experiments and thus lends confidence to the cask analysis results. The bold arrows in the magazine position indicate coolant flow which picks up heat in the fuel zone and transports it toward the end of the cask.

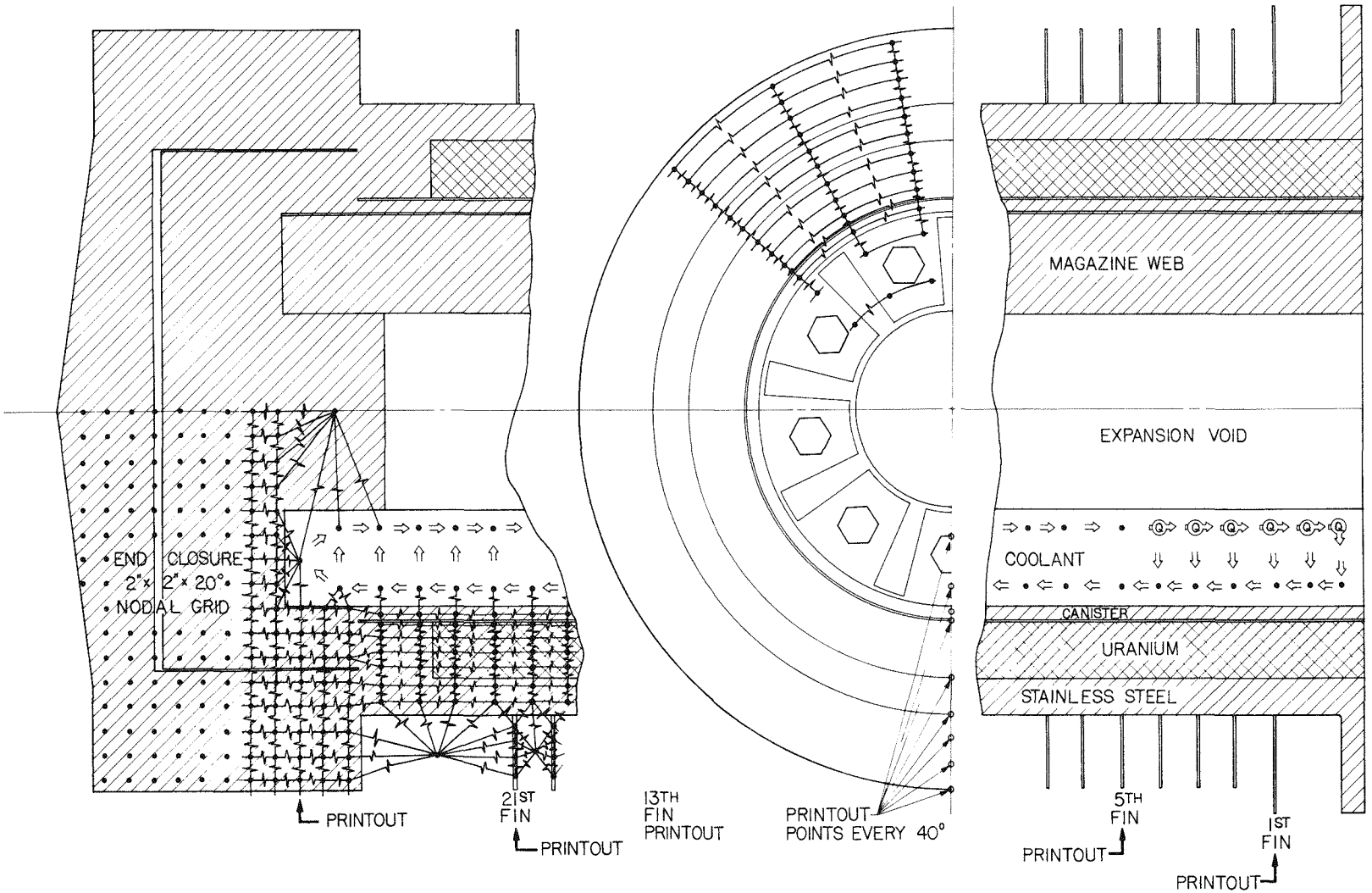


Fig. 3. LMFBR Cask – Thermal Model.

The thermal analysis model has been used to evaluate the effects of the 30 minute fire and immersion in water three hours after the fire. The results are

Peak Steel Shield Temperature	^{o}F
Peak Uranium Shield Temperature	^{o}F
Peak Canister Temperature	^{o}F
Bulk Coolant Temperature	^{o}F
Maximum Internal Pressure	psig

CRITICALITY ANALYSIS

Criticality analyses have been performed utilizing the KENO Monte Carlo computer program. The cask was modeled using FFTF Zone II fuel (25 Pu) in each of the nine magazine positions and maximum density coolant (moderator). The expansion chamber was assumed to be void, a maximum reactivity condition. There are two boron carbide poison elements between each magazine fuel position. The effective reactivity for this configuration was 0.66.

Two accident configurations were also analyzed with the KENO program. One case assumed a radial compaction of the fuel around the central expansion chamber and with the neutron leakage fraction allowed to approach zero. This resulted in $k_{eff} = 0.76$. The other case assumed an axial dispersal of the fuel pins over a length of 52 inches (normally 36 inches). This condition results in better moderation thus giving $k_{eff} = 0.78$. Combining the radial compaction around the expansion chamber with the axial fuel dispersal and the neutron leakage fraction at zero gives $k_{eff} = 0.88$. This extreme is an incredible configuration.

SHIELDING

The cask shielding has been designed to accommodate LMFBR fuel exposed to 100 Mwd/Kgm burnup, 5.76 Mw power for 730 days, followed by 100 days decay. This exposure to new plutonium fuel results in a neutron source which does not require any neutron

shielding. Later, as reprocessed plutonium comes into general use there there will have to be some neutron shielding as described below. The LMFBR Cask gamma shielding, as described before, satisfies the Department of Transportation regulations of a dose rate less than 10 mrem/hr at six feet from the vehicle surface.

It should be noted that the burnup and decay time indicated gives a decay heating rate of 4 Kw per fuel assembly. Nine such fuel assemblies is excessive for the LMFBR Cask, but six fuel assemblies and three blanket assemblies, a typical proportion and shipping load from the CRBR, would yield the design heat dissipation capacity of 26 Kw.

TRANSPORTER (R.R. CAR)

The LMFBR Cask has a shipping skid designed to support the cask on a depressed bed, 125 ton capacity, railroad car. An enclosure is installed on the skid to cover the cask during transport. This system is illustrated in Figure 4. Also shown are auxiliary equipment boxes which contain lifting fixtures, special tools, spare parts and other equipment for handling and operating the cask.

The enclosure is designed with a double wall so as to allow Water Extended Polyester, WEP, to be cast in place with a thickness of three inches. WEP is a mixture of 50 percent water, polyester resin and promoters and catalysts which hardens to provide a solid material equivalent to 95 percent water as a neutron shield. The three inch thickness reduces the neutron flux by a factor of 8 to 2 mrem/hr at six feet from the car.

LOADING OPERATIONS

The LMFBR Cask is designed to be loaded with spent fuel through the off center 10 inch port in the canister head. This may be done while interfacing with a loading port below a fuel handling hot cell as at the CRBR. Alternatively, as at the FFTF, the cask is installed in a pit interfacing with the pit cover so as to allow the Bottom Loading Transfer Cask, BLTC, to deliver fuel assemblies to the cask. This

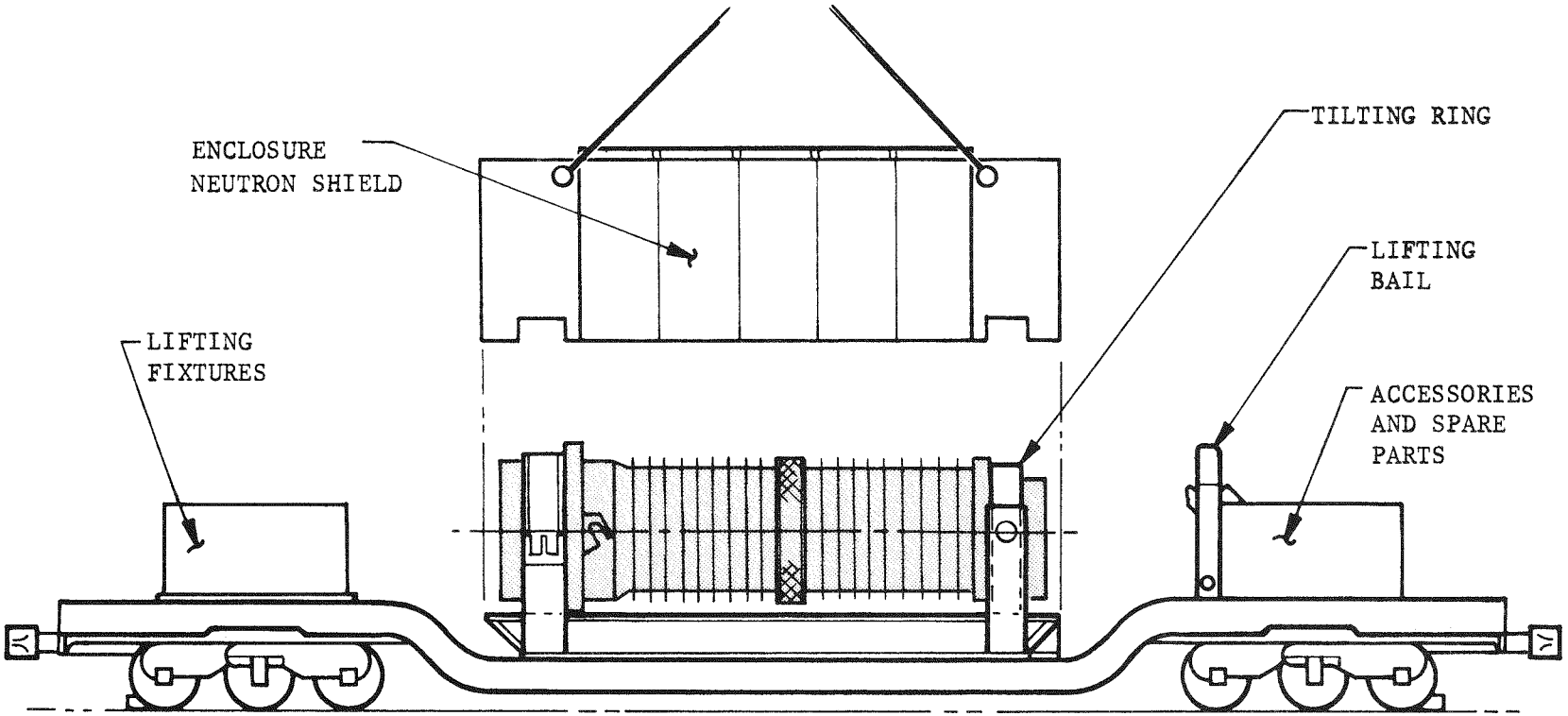


Fig. 4. Model I-(182)-1 Spent Fuel Shipping Cask on Rail Car.

arrangement is illustrated in Figure 5.

In preparation for loading the cask is installed in the cask loading station pit. The shield cover is removed and various items of equipment are installed on the canister head. Fill and vent valves in the canister head are connected to Dowtherm and helium supply sources and the facility waste disposal lines. The magazine drive motor and indexing mechanism are installed and connected to the control system. These two items are carried on the railroad car in the auxiliary equipment box. The interface Adapter, an FFTF provision, is attached to the canister head over the loading pot. Inside of the interface adapter is a mechanism which raises the loading port plug and rotates it out of the way to allow fuel assemblies to be lowered into the magazine.

After all the equipment has been installed the Cask Loading Station pit cover is replaced and the seals are made to contain the contaminated argon atmosphere between the cask and the BLTC. Magazine adapters are placed in the cask; Dowtherm coolant is loaded; fuel assemblies are brought in, one by one, to be loaded into the nine position magazine. Then the loading port plug is replaced and the operations are reversed to prepare the cask for shipment.

CONCLUSION

The LMFBR Cask design has been completed and procurement packages have been prepared to obtain quotations on the fabrication. A Safety Analysis Report has been submitted to the AEC Department of Licensing for design approval. The first model of the LMFBR Cask is planned for delivery late in 1977 for acceptance testing and license approval so as to be ready for FFTF service in 1978.

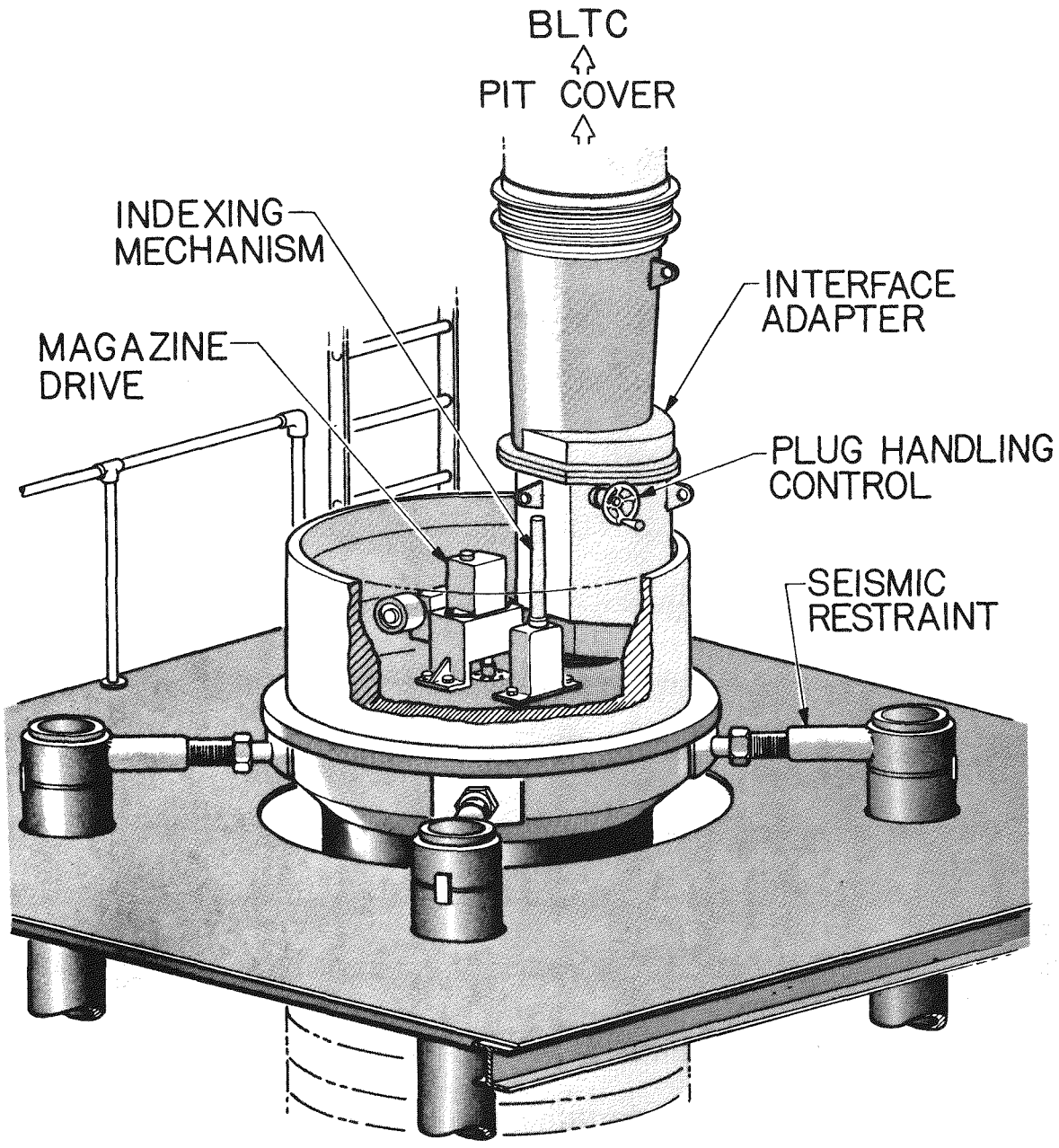


Fig. 5. LMFBR Cask Loading Arrangement at FFTF.

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THE PADUCAH TIGER: A TYPE B OVERPACK FOR TEN TON URANIUM HEXAFLUORIDE SHIPMENTS

L. J. Hansen

ABSTRACT

In order to transport Uranium Hexafluoride (UF_6) at a U235 assay greater than 1.1% within a standard 48A cylinder it became necessary to design an overpack for the cylinder that would meet the TYPE B requirements as established by AECM-0529 Packaging Standards. The package was designed, a prototype fabricated and tested in approximately eight weeks. Full scale testing included two 30 foot drops, two 40 inch puncture tests and a 1475°F exposure for 30 minutes. It successfully withstood the hypothetical accident conditions. After completion of an Engineering Evaluation Report the Paducah Tiger was issued D.O.T. Permit No. 6553.

INTRODUCTION

When it became necessary to ship Uranium Hexafluoride (UF_6) in 48A cylinders at U235 enrichment levels greater than 1.1% a need arose for a TYPE B overpack able to meet all the requirements of AECM-0529 Packaging Standards. Protective Packaging Inc. of Tacoma, Washington was contracted by Union Carbide and the AEC to provide such an overpack.

An exceedingly tight time schedule called for the analysis, design, fabrication and full scale testing of the package in less than eight weeks. This accelerated schedule discouraged creation of a totally new design leaving the only acceptable avenue of approach one of modification of an existing design. Since the payload and estimated container weights approached that of the Super Tiger (D.O.T. 6400), it was felt that a modified version of this container would be the most expedient method.

Design criteria called for compliance with all applicable federal regulations as well as those of the operating contractor. Operational requirements enforced the need for easy loading and unloading of both cylinder and overpack lid. Also of importance was the severe rail environment through which the package must sustain repeated exposure. Still, the prime function of the package was to protect the UF_6 cylinder from loss

of contents when subjected to a:

30 foot drop;
40 inch puncture;
1475°F exposure for 30 minutes.

Protection from these three conditions is provided by the overpack. Basic construction of the overpack consists of a soft external shell containing a fire retardant rigid polyurethane foam. The foam serves to absorb impact energies as well as provide an efficient high temperarure insulation. Puncture protection is provided by means of a "Break-Away Plate" system (patent pending). This system of plates molds itself around the incoming pin bringing large volumes of energy absorbing foam into play as well as providing a steel fire stop.

Solution to the above design criteria called for the fabrication of a prototype which would be tested. The prototype's function was not only to verify the design's integrity but also to provide a means of optimizing the actual production configuration. For that reason a number of repetitive tests were conducted. Test results were recorded and the production design was finalized for the Paducah Tiger.

CONFIGURATION

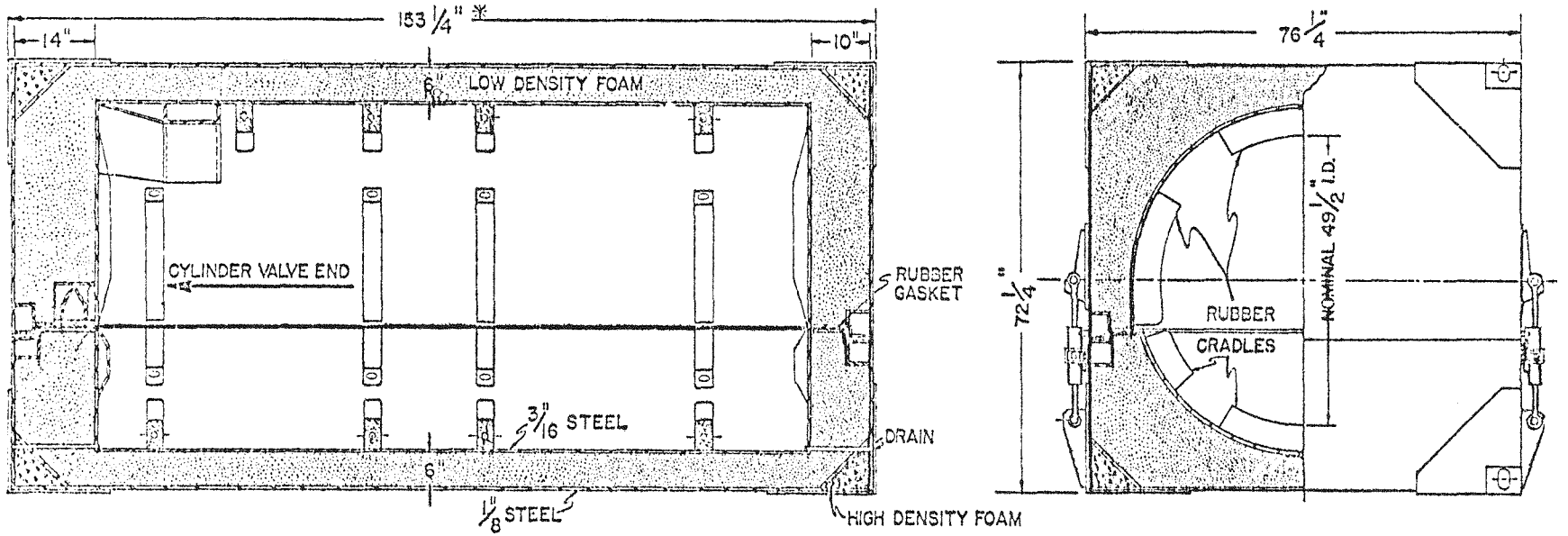
The overpack was specifically designed as a standard shipping container for enriched 48A cylinders of UF_6 and consist of two steel shells separated with rigid fire retardant polyurethane foam. (Ref. Figure 1 & 2)

Outer Steel Shell

The entire outer steel shell is fabricated from very ductile low carbon steel plate 1/8th of an inch thick.

All external edges are backed up with a diagonal gusset plate of 12 gage steel. Should any of the external seam welds fail during impact, the diagonal plate will form not only a redundant load path, but also a secondary seal for the seam.

Set into each corner assembly is a standard I S.O. steel casting This allows it to be handled, stored and shipped in the same manner as any standardized shipping container.



* PROJECTED OVERALL LENGTH OF FUTURE PACKAGES TO ACCOMMODATE MODEL 48X TYPE CYLINDER

APPROXIMATE WEIGHT

SHIPPING CONTAINER	10,500 LBS
FILLED CYLINDER	25,500 LBS
TOTAL	36,000 LBS

FIGURE 1 PROTECTIVE SHIPPING PACKAGE FOR UF_6 CYLINDER MODEL 48X



FIGURE 2 PADUCAH TIGER TEST UNIT WITH 48A CYLINDER

As can be seen from Figure 2, the top half of the container is a removable lid. A number of ratchet type binders are used to clamp and secure the lid. In addition, 1 3/4 inch guide pins are used; these are fitted with 3/4 inch high strength cross locking latch pins capable of reacting considerable shear and tension loads. Each guide pin is rigidly welded to the lower tube and is matched drilled to interface with the upper tube. This not only assures accurate match up of the assembly but also provides an extremely rigid connection capable of reacting and distributing large concentrated loads.

A puncture resistant scheme generated during the Super Tiger development was incorporated. This system calls for a series of thin 1/8 inch "Break-Away Plates" to be placed in critical areas of the package. On contact with the pin, the external skin yields inward until a tension-shear failure is initiated. At this point the pin has penetrated the external skin only absorbing a small portion of energy. Since the break-away plates are not structurally tied to the skin, they tend to mold

around the pin head causing a large area of foam to be distorted or compressed. The distortion absorbs large quantities of energy thereby stopping the pin short of the internal structure. (Ref. Figure 3) The key feature here is the ductility and elongation of the plates.

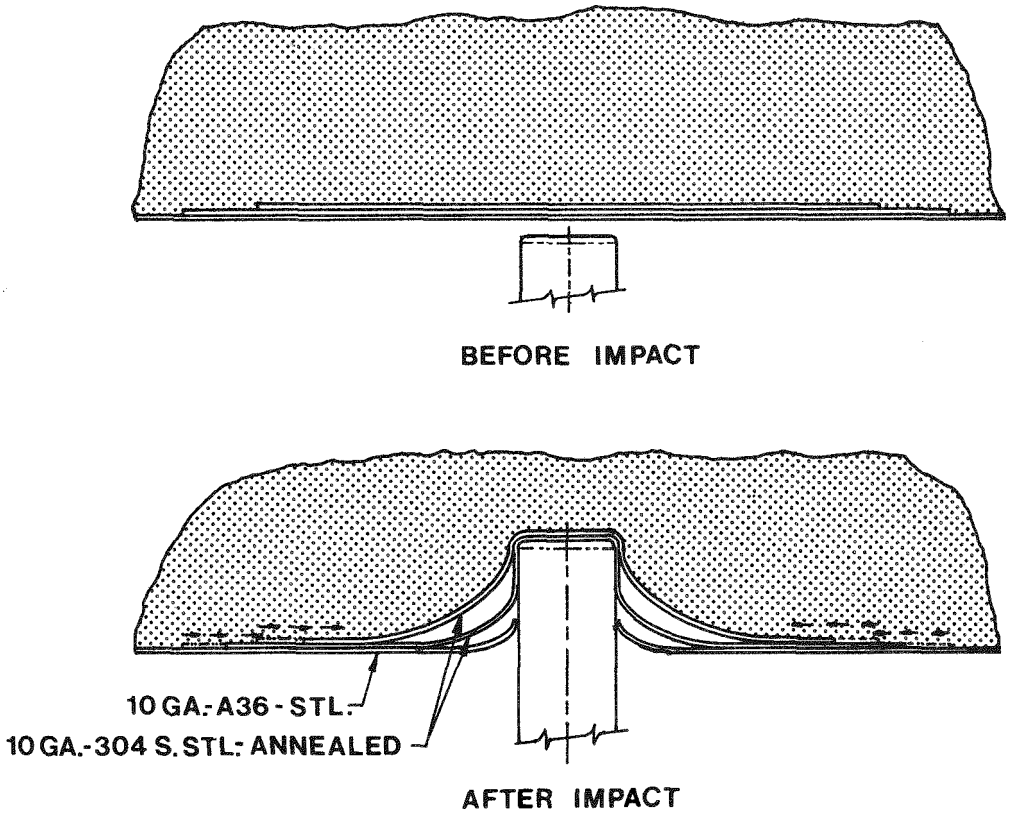


FIGURE 3 BREAK-AWAY PLATE ACTION

Polyurethane Foam

Polyurethane foam is a cellular plastic that is formed by the reaction of two liquids, a polyol and a polyisocyanate in the presence of a gas producing agent such as freon.

This foam is poured in place and allowed to expand between the two steel shells. Since, on expanding, foam rigidly bonds to all surfaces, the inside and outside shells are forced to work together forming a stressed skin type design.

Inner Container

The inner container consists of a cylindrical steel shell approximately five feet in diameter by ten feet long. Contained within the inner shell is a series of pre-curved rubber support pads. These support the cylinder during transit and provide a flexible guiding system for loading.

30 FOOT DROP ANALYSIS

It was concluded that a 30 foot fall onto the valve end of an unprotected cylinder would cripple the skirt, thereby failing the valve's threaded interface. For this reason, it was imperative to protect the relatively fragile valve end from excessive load or deformation. To accomplish this, a built-in valve protector was provided. It consists of a 1/2 inch thick steel plate across the valve end of the internal shell. This plate is re-enforced across the upper 1/3 with a 3/8 rolled plate, ten inch channel and stiffening cone. (See Fig. 4)

The purpose of this localized re-enforcement is to eliminate any potential loadpath onto the cylinder skirt that could cause it to be buckled down into the valve. All longitudinal loads are reacted directly by the 1/2 inch plate. Lateral loads are reacted by direct compression of the cylinder (aft of skirt) onto the channel.

Vibration Analysis

Since this package was to be designed specifically for rail environment it was imparative to provide shock isolation capability for the cylinder. It can be seen that the cylinder is fully isolated from the basic overpack by means of flexible rubber cradles. Much effort went into the dynamic analysis and final supports to insure that UF_6 cylinders would not be subjected to the full rail transport environment.

The isolating system was designed to mitigate the high dynamic input levels of rail environment to a level that will have minimal effect on the package and cylinder. It should be noted that subsequent rail tests demonstrated the operation of the isolator system.

FULL SCALE TESTING

A full scale container was manufactured for testing. The prime

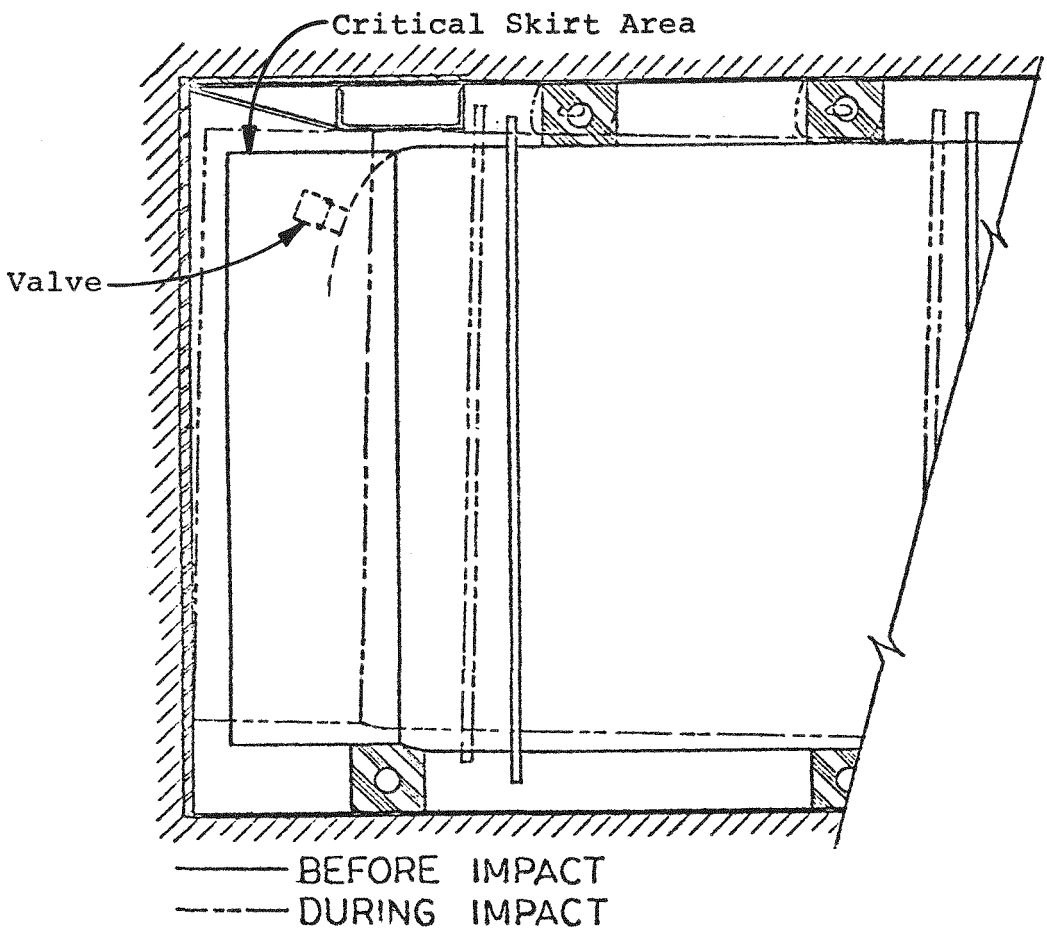


FIGURE 4 VALVE PROTECTOR

purpose of the test was to demonstrate the structural integrity of the design to meeting AECM-0529 requirements. A secondary function was to provide a means of evaluating more than one approach to the solution. A number of different design features were built into the prototype.

Prototype Configuration

Foam - two densities of foam were used. A high density material with a static crush strength of 1175 psi was placed in the valve end while a lower density (140 psi) was used elsewhere. This would allow a comparative evaluation of both crush depths and impact accelerations.

Break-Away Plates - the package sides were constructed differently relative to puncture resistance. The top was fitted with two 1/8 inch thick 304 annealed stainless sheets, while the bottom employed the same configuration of mild steel.

Testing

All tests were conducted in Tacoma, Washington, at Protective Packaging Inc.'s test pad. The pad consisted of a 1 3/4 inch thick steel plate grouted onto a massive concrete pad. Puncture tests were also conducted on the pad using a 16 inch long solid steel pin six inches in diameter.

Prior to actual testing, a UF_6 48A Cylinder (4434 lbs) was filled with 20,011 pounds of steel shot and $BaSO_4$ to simulate an actual fully loaded cylinder.

Numerous mock loadings of the cylinder into body and placement of lid were conducted. In all cases, the interfacing went smoothly with little or no help from a single rigger there for fulfilling the design criteria of the operating personnel.

30 Foot Drop Test No. 1

The purpose of the first 30 foot drop was to evaluate the valve protector end of the package. It was felt that the most critical loading would be experienced if the package was to impact the lid, along the valve edge.

At this point it was raised to a height of 30 feet, assuming an inclination of approximately 20°, and released. On impact, the package's external shell compressed locally, along one edge, coming to rest in the vertical position. Measurement indicates the corner fittings were displaced approximately four inches. Neither I.S.O. fitting showed signs of fracture. Each of the ratchet load binding and safety pins remained securely fastened with no evidence of over-stress or fracture.

40 Inch Puncture Test No. 1

Puncture test No. 1 was conducted on the lid backed up by two 1/8th inch stainless steel (304 annealed) break-away plates. The package was raised to a position 40 inches above the top of a six inch diameter bar. The flat skin of the overpack impacted perpendicular to the axis of the bar, cutting a circular hole through the external skin. The maximum penetration of the bar from the plane of the skin was ten inches.

The stainless steel break-away plates stretched and deformed over the

pin but did not tear, and in so doing preserved the integrity of the overpack. No foam was exposed.

After completion of the first two tests, the package was opened for inspection. No trouble was experienced in removing the lid.

From the 30 foot drop the valve end at the cylinder suffered no severe damage. The bulkhead plate itself was undamaged. No significant distortions to the cylinder or the package interior were present from the 30 foot drop test.

The 40 inch puncture test resulted in some damage to the internal shell. Because of the vulnerable location of the impact, the inner skin was sheared approximately 12 1/2 inches long and opened to a maximum depth of one inch.

The cylinder wall was locally deflected inward a maximum of 1 1/2 inches. It was apparent that the cylinder had not been distorted or stressed to a point that would cause any loss of structural integrity.

Upon completion of the damage survey, the lid was reinstalled and positioned for the second 30 foot drop test.

30 Foot Drop Test No. 2

The second drop was conducted in the same manner as the initial drop. However, the package was dropped this time on the opposite edge, in order to determine the crush resistance of the overpack in an area filled with "low density" foam. After the impact, inspection showed the edge of the external body displaced longitudinally approximately 12 inches. The cylinder had not been recentered after the first 30 foot drop and for that reason was initially located some four to five inches away from the inner-bulkhead. On contact, a much more severe secondary impact from the cylinder was realized. It is believed this led to a very critical test. (Ref. Fig. 5)

The lid remained securely locked to the body by the latching pins. Because of the softer foam, the acceleration loads were much less severe.

40 Inch Puncture Test No. 2

For the second test the package was dropped on the two 1/8th inch

mild steel break-away plates. The pin penetrated approximately 12 inches of its 16 inch length. Both the inner and outer skins of the package were penetrated.



FIGURE 5 SECOND 30 FOOT DROP TEST

At this point the package was opened and the cylinder removed for observation. The bottom skirt received no observable damage from the 30 foot drop. In fact there was no apparent yielding or distortion to any part of the cylinder from this test.

Impact results of the second pin drop indicated a local indentation of 2 1/2 inches. Deformation of the cylinder was small and resulted in no loss of structural integrity.

Fire Test

After the Paducah Tiger had successfully undergone a series of two thirty foot drops and a 40 inch puncture test, a fire exposure test was conducted. Per the requirements of AECM-0529, the package had to be subjected to a 1475^{O} environment for a period of 30 minutes. In order to determine the maximum temperature experienced by the UF_6 test cylinder during the test exposure, over 200^{O} temperature indicating labels were used. The requirements for high temperature as well as the accommodation of the 36,000 pound gross package weight dictated that the test be conducted in a modern steel heat treating furnace.

The furnace was preheated to temperatures exceeding 1500^{O}F, for 1 1/4 hours before the Paducah Tiger was placed inside. The package remained in

the furnace for 50 minutes and was subjected to temperatures ranging from 1475°F to as high as 1820°F for a period exceeding the 30 minutes requirement. In actuality, the package bottom experienced these temperatures for nearly one hour. Thus, the duration and severity of the thermal exposure were nearly twice that required.

Instrumentation indicated that the test cylinder experienced a temperature rise resulting in maximum surface temperatures of 275° to 300°F with a small localized hot spot of 450°F. Based on a thermal analysis and comparison of the physical properties of the tested ballast versus actual UF_6, it was concluded that a cylinder of UF_6, under the same extended test period, would have increased from an original 100°F temperature to no more than 200° F. At this temperature the resultant pressures induced from liquid expansion and vapor would not be sufficient to damage the cylinder. Therefore, the Paducah Tiger was found to be capable of protecting the 48A UF_6 cylinder from excessive temperatures and/or potential rupture while being exposed to the hypothetical accident environment of AECM-0529.

DESIGN OPTIMIZATION

From the results of the two 30 foot drops, it was concluded that the hi-strength foam used in the valve end is largely unnecessary. In light of the high acceleration loads, resultant from the small deformation, its presence even becomes detrimental. Based on the measured drformations, it's easily calculated that the experienced g levels approached 90 g's for the high strength end versus less than 30 g's for the soft end. For this reason it is felt that the production configuration would be substantially improved by the replacement of the hi-density material with that of the more flexible low-density.

With regard to foam thickness, it can be seen that the second 30 foot drop crushed the edge thru approximately 12 inches of foam. For this reason, we feel that the valve end need not be thicker than that previously tested in the non valve end. For conservatism, 14 inches of foam have been added to the production valve end.

The two 40 inch puncture tests revealed the obvious superiority of 304 stainless steel break-away plates over those of mild steel. The 1/8

inch stainless plates stretched and deformed during the first puncture test but did not tear, and therefore effectively preserved the integrity of the overpack.

CONCLUSION

Based on the results of the test program and analysis, a complete Engineering Evaluation was prepared and submitted for approval. U.S. D.O.T. Permit No. 6553 was subsequently issued. Four production unit Paducah Tigers were immediately fabricated by Protective Packaging, Inc. and placed into service in Nov. 1971. Each package has successfully transported enriched UF_6 for hundreds of thousands of miles since that date without incident or malfunction.

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LLL SHIPPING CASK FOR "H" CONTAINERS

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J. Evans
J. Doggett

ABSTRACT

Four shipping cask models of similar configuration have been designed and tested. The primary use of these overpacks is to transport large quantities of radioactive material and to provide total containment for contents during both normal transport and accident conditions. A full-size model was impact- and fire-tested to demonstrate compliance with the accident provisions of AEC Manual Chapter 0529. The casks were specifically designed for air transport and have a gross weight of approximately 4000 lbs. All models are designed for a content weight of 1000 lbs and a thermal decay of 750 W.

INTRODUCTION

In 1973, personnel from the Oak Ridge National Laboratory and the Lawrence Livermore Laboratory designed and tested a family of overpacks for transport of nuclear components and large quantities of radioactive materials. The overpacks meet the Atomic Energy Commission/Department of Transportation Type B packaging standards and can be used to transport up to 4.5 kg of ^{239}Pu or 18.5 kg of ^{235}U. A typical shipment will be a modified AEC type "H" container transported in the vertical or the horizontal position.

The primary use of the shipping cask is to provide containment and impact and thermal resistance for fissile contents during both normal transport and accident conditions. The contents of the package are solid materials of Fissile Class I and Class III with no associated liquids, powders, or other dispersables. The overpacks were specifically designed for transport by air in an unpressurized cabin, but they may also be shipped by rail, highway, or water transport.

We demonstrated compliance with the AEC-DOT regulations by subjecting a fullsize model to hypothetical accident conditions.

SHIPPING CASK DESCRIPTION

There are four models of the basic cask. The model shown in Figure 1 will be used to transport modified "H" container models 1057, 1224, and 1227 in the vertical position. Additional primary and secondary heads, lengthened 6 in., have been provided to transport a modified H1259 container. The shipping cask may be converted for shipment in the horizontal position by changing the forklift base and by adding a lifting bar, as shown in Figure 2. In this configuration modified H815, H862 and XM467 containers can be transported. The fourth model is a shortened version of the vertical model (30 in. high x 38-7/8 dia. internal) and will be used to transport nuclear components and large quantities of radioactive materials. All models are designed to carry weights of up to 1000 lb. The longest model, which is outfitted with the extended heads, will have a gross weight of approximately 4000 lbs.

CASK CONSTRUCTION

The overpack provides double containment of contents. Primary containment is provided by the inner vessel and head (Shown in Figures 1 and 2). Heavy, double O-ring sealed flanges connected to relatively thin shell sections form the primary closure. This design permits large impact loads to deform the shell sections plastically without applying failure-level loads to the flanges. A connection is provided on the vessel flange to pressurize the space between the O-rings for routine verification of the seal effectiveness using a time pressure drop technique. The thirty-two 3/4-in. alloy steel socket head cap screws securing the flanges are drilled for seal wiring to prevent loosening during transit.

Secondary containment is provided by the outer vessel and head. A double O-ring sealed closure with provisions for leak testing similar to the primary closure is provided. The sixteen 1/2 in. cap screws securing the secondary closure are also drilled for seal wiring to prevent

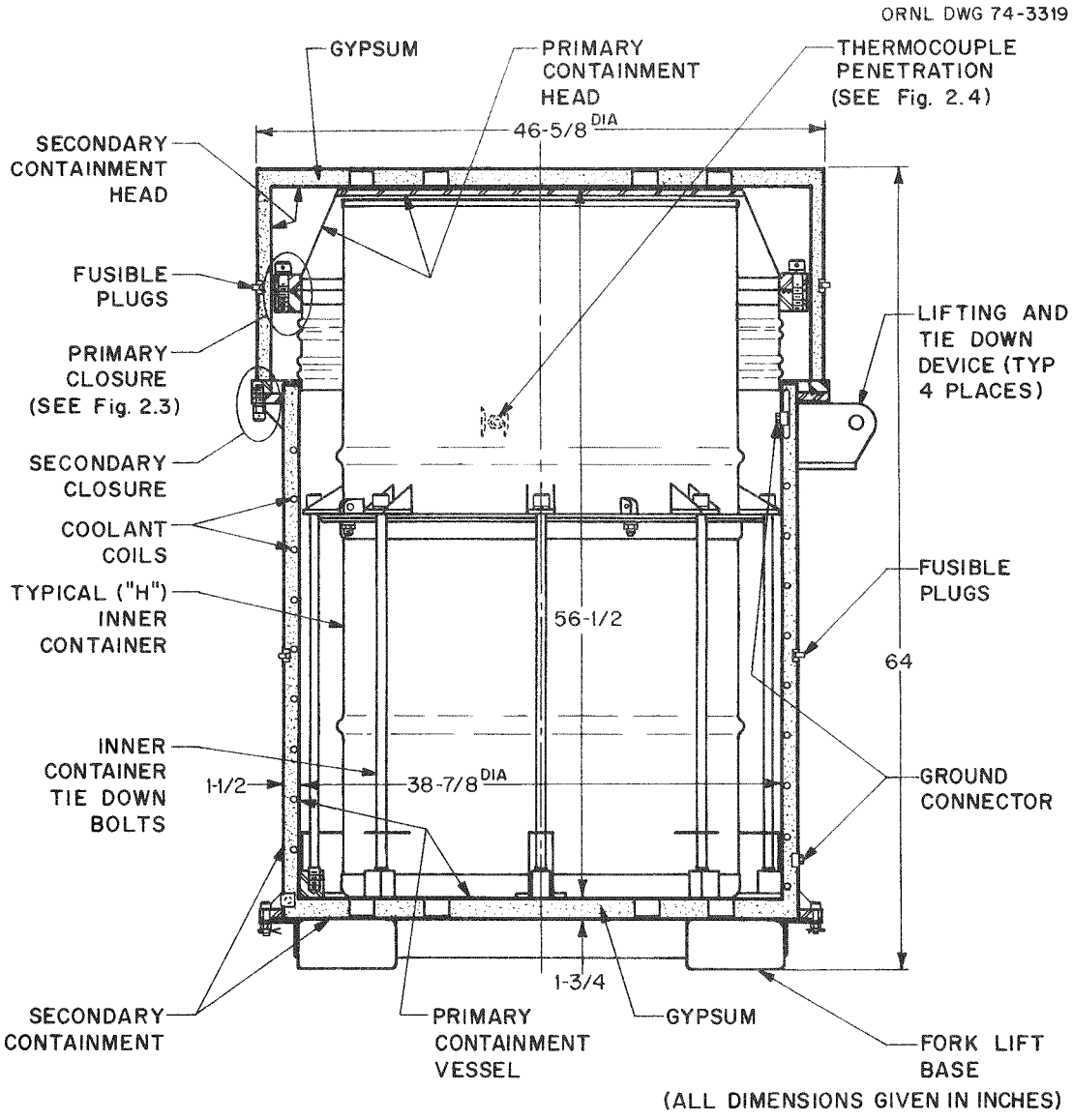


Fig. 1 LLL Shipping Cask in Vertical Position

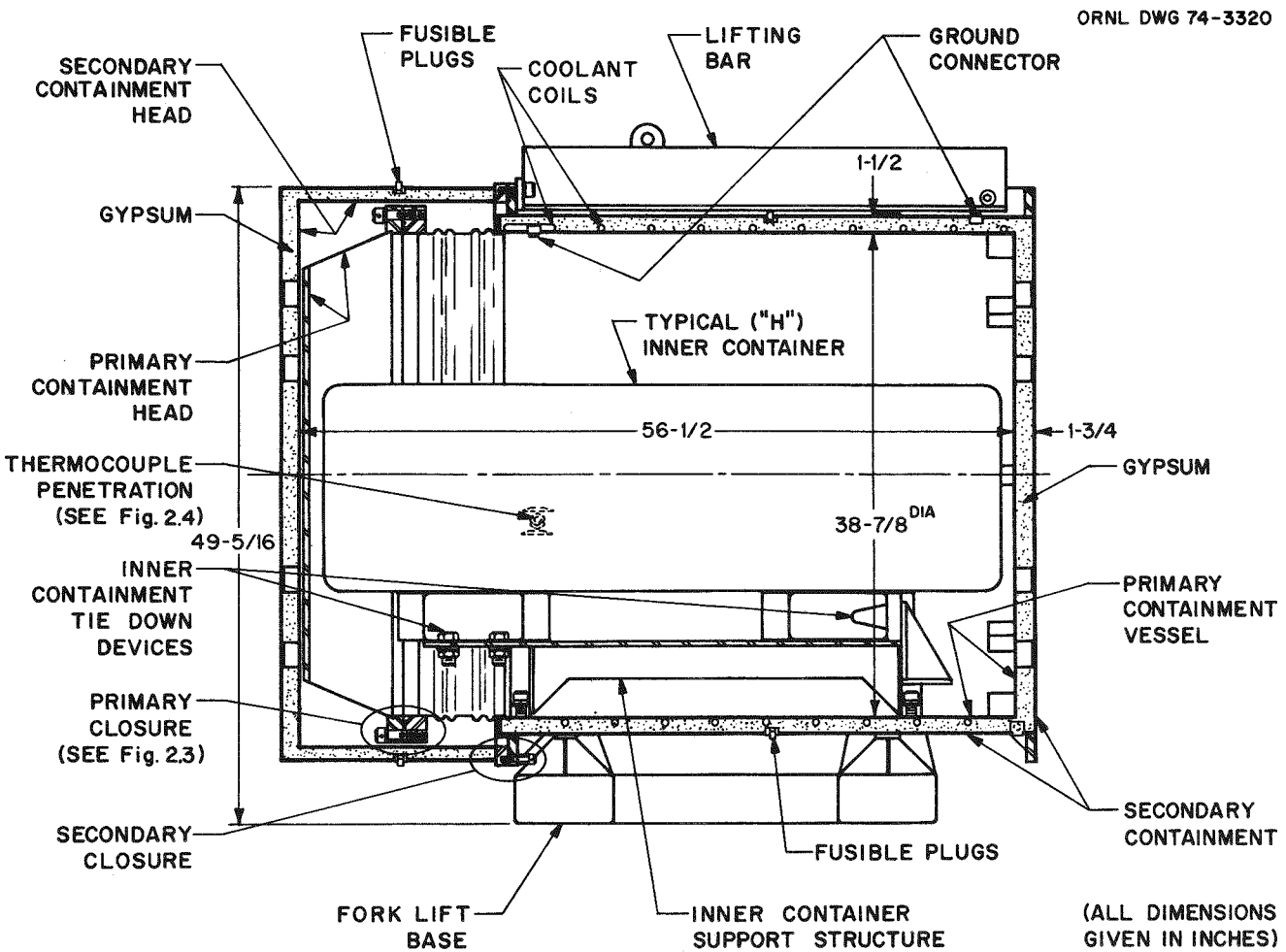


Fig. 2 LLL Shipping Cask in Horizontal Position

loosening and to indicate unauthorized tampering.

The containment vessels, fasteners, and appurtenances are fabricated from stainless and low alloy steels in accordance with ASTM specifications. They have a Charpy impact resistance of at least 15 ft-lb at -40°F. The O-ring seals are molded from a silicone rubber compound that is suitable for operating temperatures up to 500°F. The space between the cask walls is filled with gypsum molding plaster and the exterior of the cask is painted with intumescent paint to provide thermal protection. The cask will safely dissipate 750 W of decay heat without the use of an external cooling system. Cooling tubes embedded in the gypsum and a special thermocouple feedthrough have been provided for cooling and continuously monitoring the temperatures of shipments with concentrated heat sources. This cooling system is for the convenience of the user and is not required for normal transport conditions.

RESULTS OF FULLSCALE MODEL TESTING

A fullscale model of the shipping cask was impact- and fire-tested at the Oak Ridge National Laboratory to demonstrate compliance with the accident provision of the AEC Manual Chapter 0529.

The model was dropped from 30 ft and landed on its corner. The corner was deformed in a wedge shape 4-3/4 in. deep by 7 in. long (Figure 3), and the outer secondary containment shell was ruptured, exposing the gypsum. The peak acceleration measured in this drop was 200g.

The model was then dropped on its side from 30 ft. (Figure 4). The side of the cask was flattened such that the width of the flat was 12-1/2 in. at the base and 16 in. at the head. The secondary shell was ruptured and ten out of sixteen 1/2-in. cap screws securing the outer head failed in the quadrants 90° from the impact plane. The bolts failed in a tensile mode. The peak acceleration measured in this drop was 500g.

The model was then dropped 40 in. onto a 6-in. diameter mild steel bar as required by the regulations. The only damage incurred in this test was a 1-in. deep indentation in the secondary containment shell.



Fig. 3 Test Model after 30 ft. Corner Drop

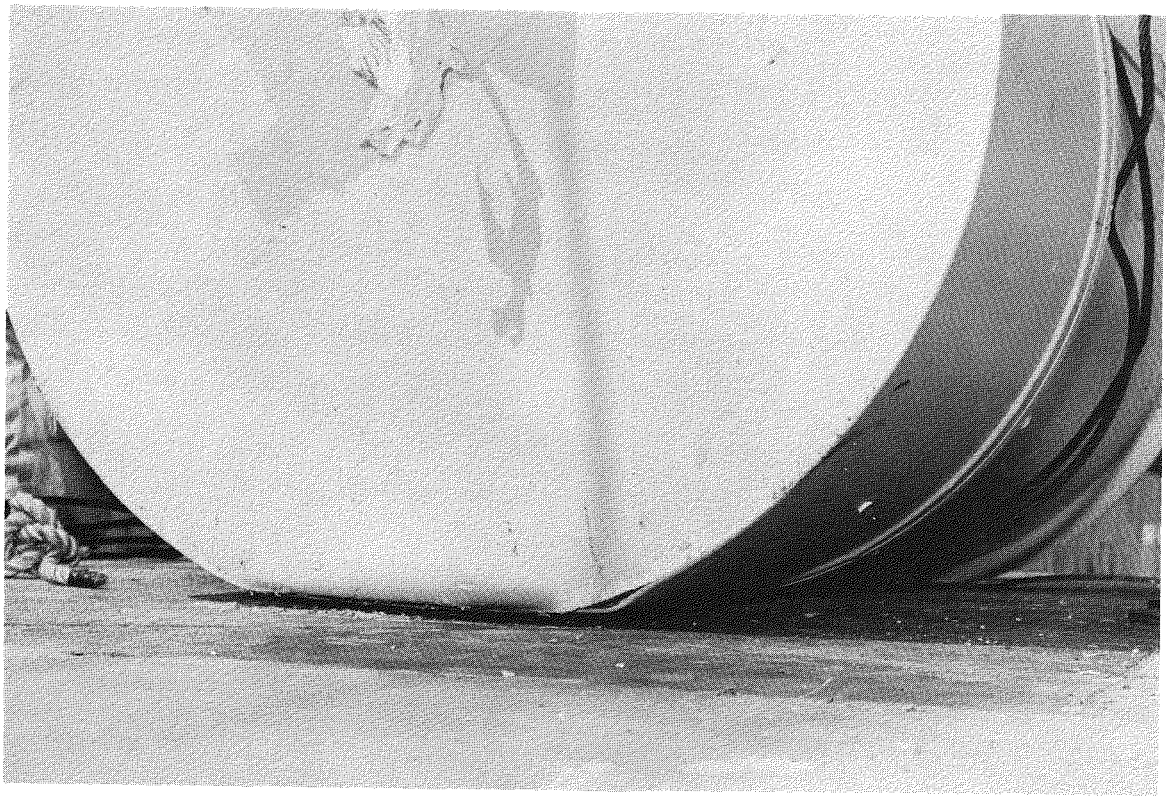


Fig. 4 Test Model after 30 ft. Side Drop

Figure 5 shows the condition of the model after exposure to a 24-min. open-pit petroleum fire. Considerable steam was emitted from the gypsum plaster throughout and after the fire test. The cooling effect of the steam and the swelling of the intumescent paint were effective in keeping the internal contents of the package at a low temperature. "Thermal sticks" located inside the primary containment indicated that the temperature in this area did not exceed 113°F. We analyzed samples of gypsum directly exposed to the fire and found a water content of 24% by weight. This indicates that considerable water still remained in the gypsum to help cool the cask during a longer exposure to fire.

We also noted that, except for a small area connecting the overpack to the inner vessel, surfaces painted with intumescent paint but not directly exposed to the fire (such as inside the secondary head) were unaffected by the heat.

After the secondary head was removed, the primary closure seal was pressurized with argon to 50 psig (Figure 6), and soap bubble tested to determine if the flanges had been damaged by the impact and fire tests. No leaks were found, but there was a pressure drop. To verify further that the primary containment vessel had not been ruptured, we removed the bottom of the secondary containment vessel and pressurized the primary vessel to 10 psig for 8 hr. There was no detectable loss of pressure. After a 48-hr. holding period, the pressure had dropped to 9.75 psig. All welded and mechanical joints were soap bubble tested; we detected no leaks. Apparently the small leakage occured in the primary double O-ring seal and was caused by compression failure of the intumscent paint during the drop tests. In the future, the primary joint flanges, which are fabricated from stainless steel, will not be painted.

VERIFICATION OF FIRE TESTING BY COMPUTATIONAL METHODS

Since the actual fire test of the model did not meet the full 30-min. fire as required by the AEC regulations, we made an analytical evaluation of the specified thermal incident. We assumed the computational model representing half the overpack and used the HEATING -3 computer program[1]

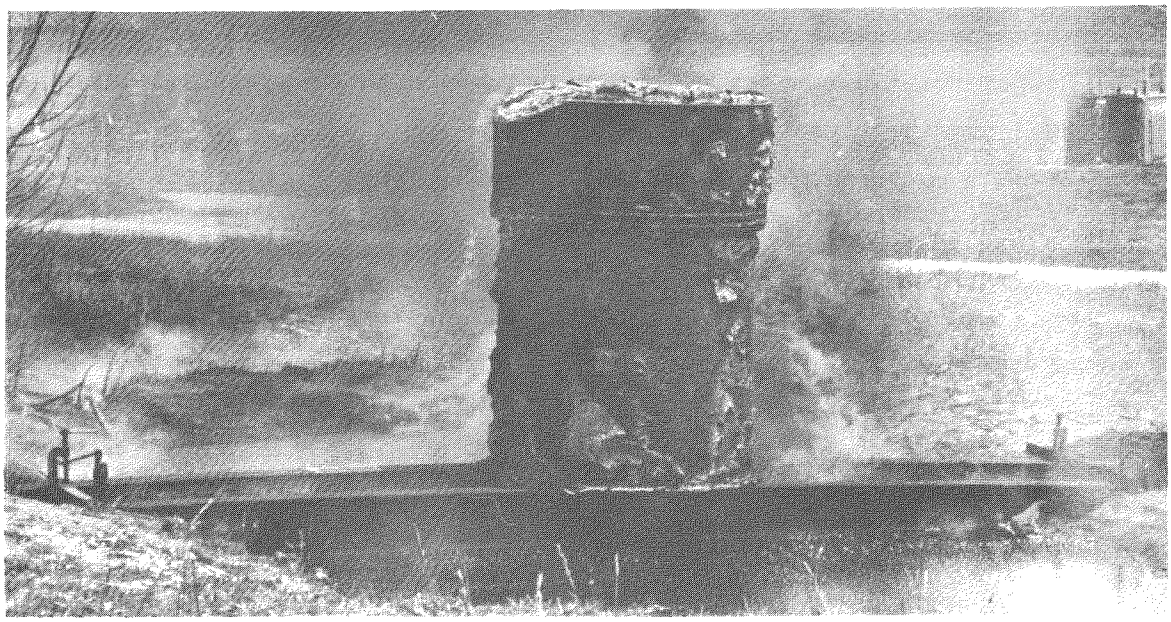


Fig. 5 Test Model after Fire Test

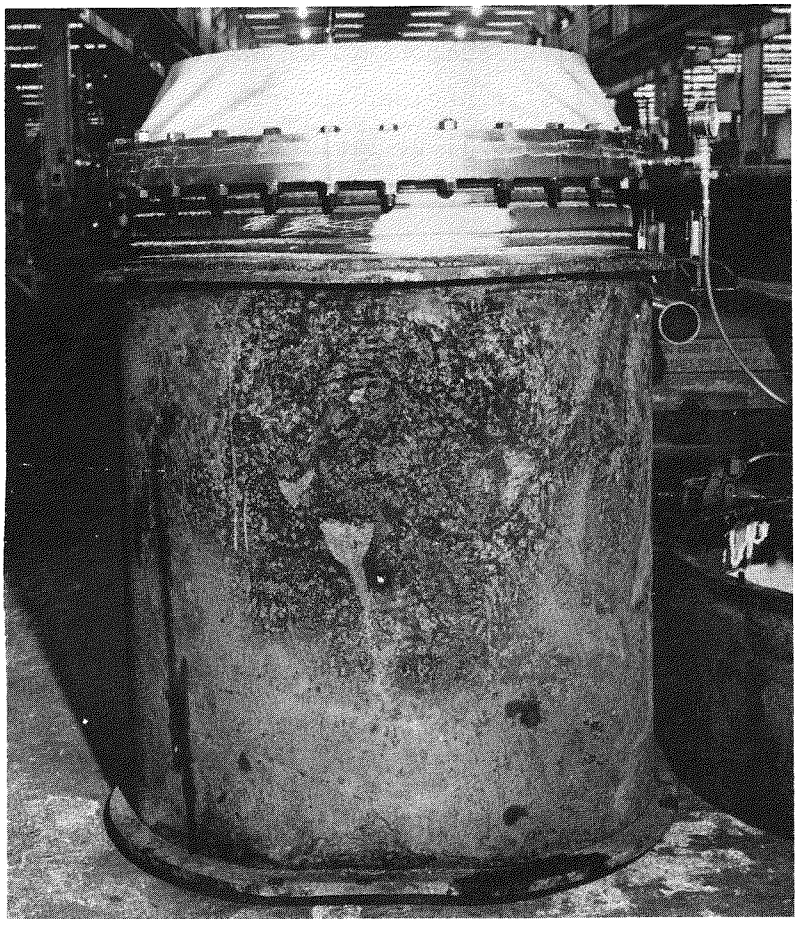


Fig. 6 Test Model with Secondary Head Removed

to determine the temperature distribution in the cask from 30-min. thermal exposure. Though the effect of intumescent paint was not included in the computer program, we found that the temperature of the contents would not exceed 200°F (compared to 113°F in the actual fire test). The analysis indicated the primary O-ring seal would reach 300°F, which is well below the service temperature of silicone rubber. From these calculations and the results from the model tests, we concluded that the 30-min. fire would not affect the containment capability of the container nor the criticality of the contents.

CONCLUSION

We have designed a family of overpacks that will be very useful and versatile in transporting nuclear components and materials. Fullscale model testing has demonstrated that the shipping casks meet all the AEC-DOT type B packaging requirements for transport of large quantities of radioactivity. The overpacks may be used to transport modified "H" containers with nuclear components or may be used to transport large quantities of nonfissile nuclear material or Fissile Class I in solid form. These shipments will be limited to a thermal decay of 750 W and a content weight of 1000 lb.

A safety analysis report for this family of shipping casks was submitted to the AEC in February, 1974. Fabrication of the four models will start when an approval of the SARP is received from the AEC.

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DEVELOPMENT AND TESTING OF OVERPACK-CONSTRUCTIONS BASED ON PHENOLIC FOAM WALLS

R. Christ

B. Schulz-Forberg

ABSTRACT

For shipping liquid or solid plutonium-waste or nitrate solution no special shielding is necessary. In view to transport however, a shock absorbing and fire insulating overpack is required. Attention is to be given for easy handling.

The physical properties of phenolic foam seemed to be an optimum having in mind the safety and the economy. The thickness of foam walls depends on the fact that boiling of liquids has to be avoided and that the temperature of gasket material will stay in the allowed range. Special constructions are needed to prevent high deformetions of the inner container.

The chosen phenolic foam had a chemical constitution which guarantees the fire resistance.

Nowadays we have 7 type B-packagings using phenolic foam in the overpack-constructions. The shipping experience has never shown any problems.

Introduction

Plutonium-containing waste from Plutonium fuel production has to be conditioned and disposed according to the regulations for disposal in the Asse salt mine. A main item of these regulations is that waste has to be mixed with a binding agent such as concrete.

When the ALKEM Plutonium fuel facilities where installed at Hanau, it had to be decided wether the waste should be processed on site or the services of the Karlsruhe Research Center should be used for conditioning. A careful review of economics showed that shipment of liquid and solid waste to Karlsruhe for treatment in the already existing facilities of the Research Center was the best solution.

For this purpose Transnuklear GmbH developed an overpack system which had to meet the following requirements

- type B large source qualification
- use of different containments for solid and liquid waste
- use of 200 l (55 gallon) steel drums for solid waste
- payload up to 400 kg (drum filled with concrete)

Design Features

Overpack

Phenolic foam was chosen as the shock absorbing and insulating material of the overpack (See Fig.1). When defining the thickness special attention had to be given to the fact that boiling of liquid had to be avoided under fire conditions. The foam quality was chosen on the basis of previous tests which have demonstrated the need of proper selection and specification of material and

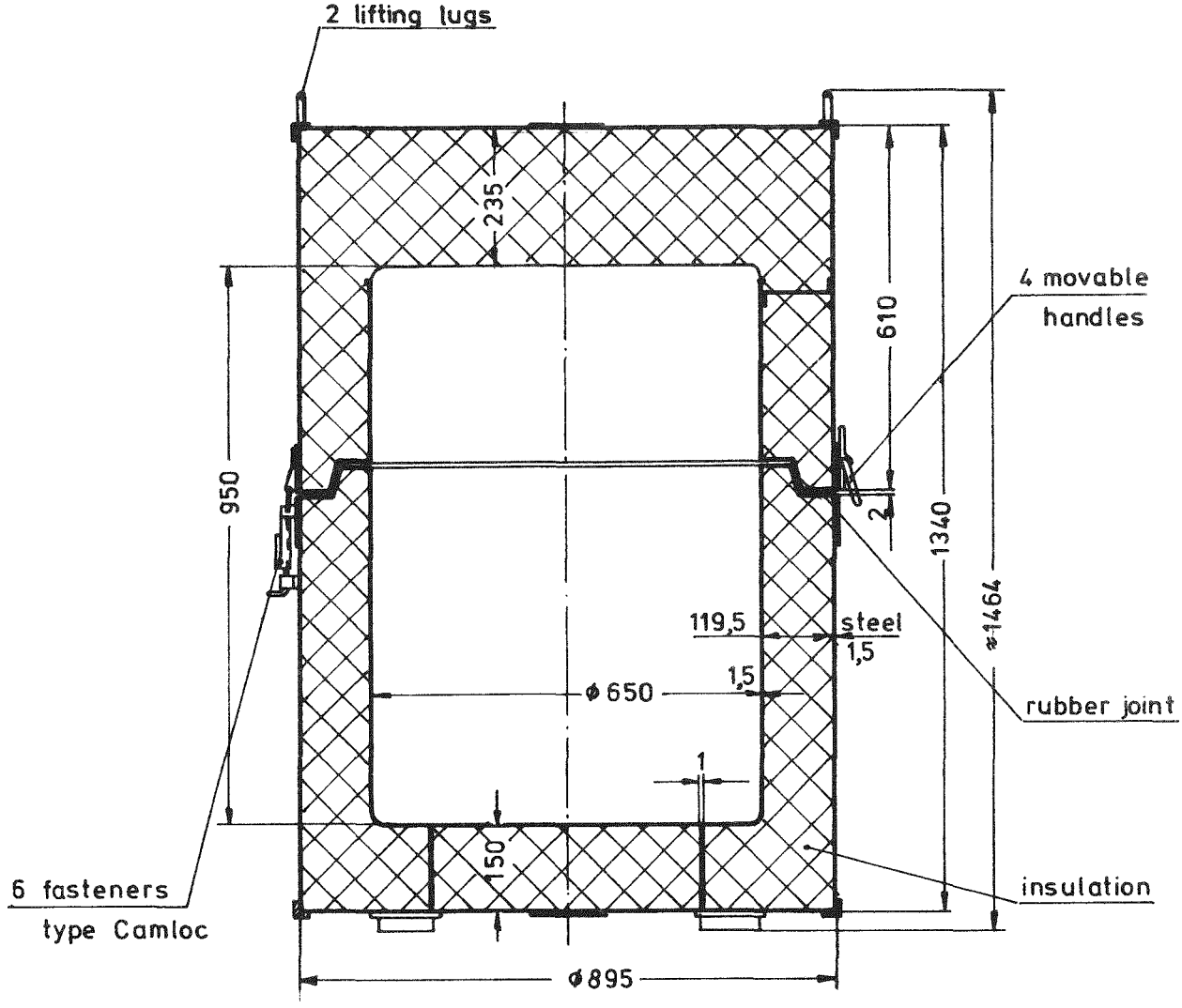


Fig.1. Type B overpack based on phenolic foam

application. The foam is clad with mild steel sheet of low temperature ductility. To avoid deformation of the inner container in the drum head area as a consequence of the 9 m drop the inner steel cladding is omitted at this location. This design feature proved to be very effective in reducing the g-loads on the drum closure. Overall dimensions of the overpack are 900 mm in diameter and 1340 mm in height.

Containment Vessels

- Solid Waste
  The containment vessel for solid waste is a 200 l (55 gallon) steel drum with bolted head (See Fig.2). This drum is the specification container for final disposal of medium level waste in the Asse salt mine.

- Liquid Waste
  For liquid waste containing Plutonium a stainless steel vessel was designed and manufactured (See Fig.3). Its outer dimension is equivalent to the 200 l drum. The vessel is equipped with connections for filling, draining and flushing operations and a level gauge which automatically stops the filling operation. Below sealed valves are used to minimize leakage. The valves are connected to special couplings which are designed for minimum contamination at the coupling interface.

  The vessel was designed and approved according to German regulations for pressure vessels. Design pressure is 7 atü (100 psig). Helium leak tests have demonstrated that the leak rate is below 10^{-8} Torr 1/sec. This limit is requested by the German competent authority for the containment of type B quantities of radioactive liquids.

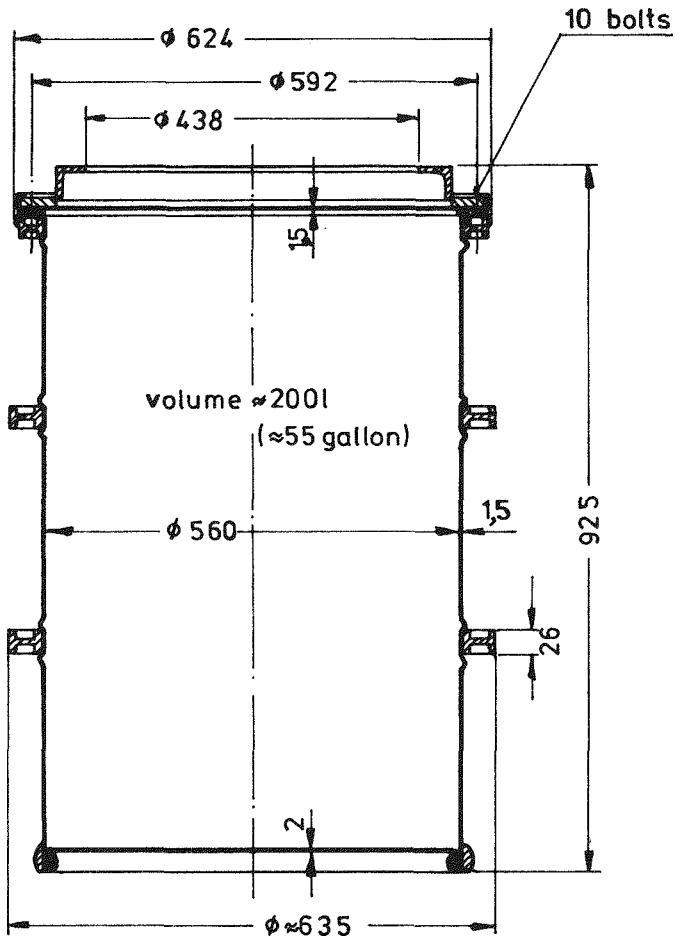


Fig.2. 200 l (55 gallons) steel drum used as the containment for solid radioactive material

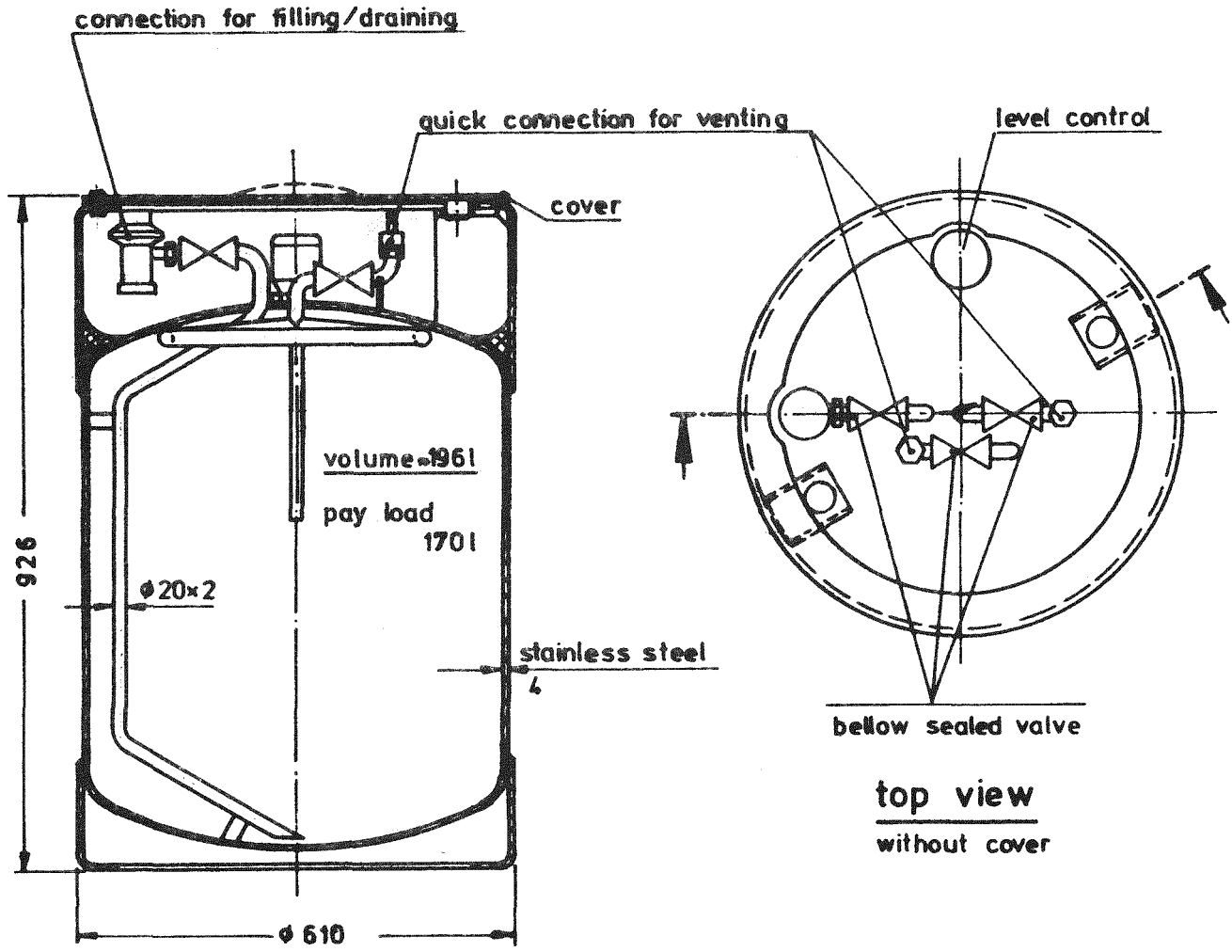


Fig.3. Containment vessel for liquid Plutonium waste

Weight of the Overpack System

Some characteristic data are given in the following table

	overpack + drum	overpack + vessel
Weight overpack	150 kg	150 kg
Weight containment	50 kg	115 kg
Pay-load	200(400)* kg	170 kg liquid
Ratio pay-load/ packaging weight	1 (2)*	0,64

*For waste bonded to concrete

It seems that the ratio pay-load packaging weight is quite economical for a type B packaging.

Testing

Since 1970 phenolic foam was chosen for Type B-Packagings as a shock absorbing and fire insulating material. The first packaging which should be used for transport of pellets containing some Plutonium was tested in actual drop and fire tests. The second packaging was constructed for shipping one MTR-element. Actual tests were made too. The third packaging is a little one with no defined content. It also fulfilled the requirements and may be used nowadays for shipping pacemakers having radioisotopic batteries.

In 1971 the competent authority in the Federal Republic of Germany for packaging - the Bundesanstalt für Materialprüfung (BAM) - issued two certificates for packagings containing Plutonium-Nitrate-Solution.

On the basis of the above mentioned types the over-pack-construction in Fig.1 was developed preventing either a drum with solid waste (Fig.2) or a tank with liquid waste (Fig.3).

During the development we made the experience that special attention has to be given to the chemical constitution of the phenolic foam. Examinations of one type of packaging for Plutonium-Nitrate-Solution were done twice, first with a none-fire-resistant foam (Fig.4) and secondly with a special foam having a good resistance. The chemical constitution belonging to this foam is given by a code, which is necessary to reproduce the qualities needed for type B-packagings. An example of this foam is to be seen in Fig.5.

Some measurements were done in the tests. Putting the results in the table below, we get an approximate survey

G : Total load
Δ_s : thickness of phenolic foam wall [mm]
$\Delta\vartheta$: temperature drop in the wall [°C]
Δ_h : deformation after drop test [mm]

G	Δ_s	$\Delta\vartheta$	Δ_h
620	-	-*	30 (a = 620 g)
450	120	50*	
410	120	100*	30-50
45	125	160	15
270	195	65	-
125	225	40	60-100
135	225	40	

* 1. liquid contents with high capacity
  2. solid contents
  3. small packaging



Fig.4. Packaging after drop and fire test having a non-fire-resistant phenolic foam as insulating material

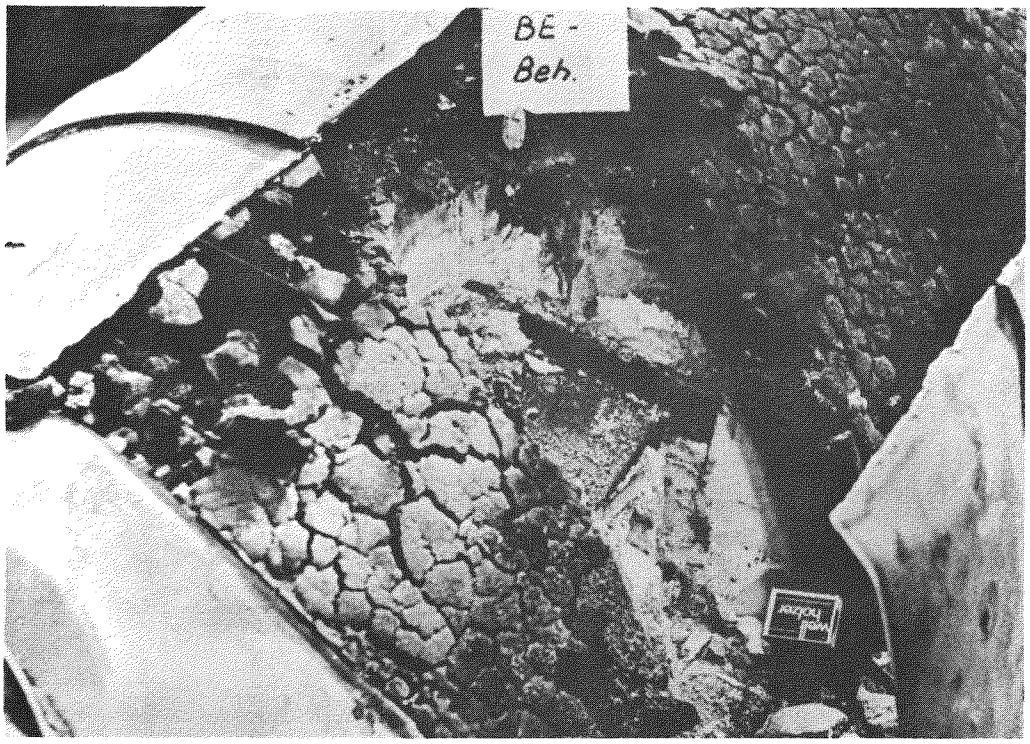


Fig.5. Packaging after test having used a fire-resistant phenolic foam

Shipping Experience

About 25 overpacks have been manufactured in the meantime and are in use for different purposes.

Shipments of solid and liquid Plutonium waste between the ALKEM plant and the Karlsruhe Research Center are performed by Transnuklear on a routine basis today. There haven't been any problems during loading, transport and unloading.

Though originally designed for the shipment of Plutonium waste, the overpack has proven to be an economical packaging system for a variety of none or low heat generating radioactive materials such as fresh fuel elements for HTGR's, UO_2 pellets and others.

ITALIAN TESTING STATION FOLLOWING IAEA REGULATIONS CF6: A FISSILE CLASS II TYPE B PACKAGING

G. F. ELETTI

G. FORASASSI

ABSTRACT

This paper summarizes the main features for testing containers for transport of radioactive materials, which have been set up in Italy at Scalbatraio Testing Center of Pisa University in cooperation with CNEN.

Results are then reported of research and testing series which have brought to finalize the design of a low cost type B, fissile class II container.

These results show that, for a particular version of the container, accelerations, with respect to 9 m drop test, and relevant temperatures, with respect to thermal test, have been reasonably limited.

1. TESTING STATION

1.1. Drop Test Tower

The tower (fig. 1) which can test containers up to 2000 kg of weight is made of the following components:

- tubular frame 15 m high
- target made of highly resistant reinforced

- concrete covered by high resilient steel plate to which target for punch test can be connected
- lifting and release devices
- guided-falling device for impact tests on absorbent materials, structures and parts of containers
- instrumentation for measure of accelerations and displacement.

1.2. Thermal test station

The thermal test station (fig. 2) which uses the open fire method is made of the following components:

- concrete support for combustion pool
- combustion pool exchangeable
- side firewalls of steel to protect container from wind
- fuel drain system to assure quick fire extinction
- instrumentation for temperatures measurements during test.

A part from the preceding test station and following the IAEA "FURNACE TEST", the University of Pisa can use different indu-

strial furnaces electrically operated or with fuel which are located not far from the Scalbatraio Center.

1.3. Stations for spray, hydraulic, tightness and shielding test

The following equipments are operable:

1) Spray test fixture (IAEA)

2) Container for packagings up to 3 m long (fig. 3) which can be pressurized in the range - 0.5 ÷ 6.5 ata. With this equipment water tightness can be tested up to 15 m water height.

3) Equipment for the "Bubble test" following ISO for packaging cylindrical up to 600 mm diameter.

4) Equipment for shielding measurements based also on X and gammagraph.

2. RESEARCH AND DEVELOPMENT

The preceding equipments have been utilized to support the design of a particular packaging and to assure its positive response to IAEA standard test series.

2.1. The CF6 Container

The components of this container (fig. 4), which is made of simple and cheap materials, are:

1) Commercial drum with modified closure

2) Absorbent and insulating medium made of a conglomerate concrete-vermiculite which composition has been determined through several tests to obtain desired mechanical and thermal performances.

3) Internal vessel made of carbon (in the normal version) steel water-tight.

In its last version the CF6 characteristics are:

Inside volume	9 dm^3
Total volume	54 dm
Containment weight	36 kg
Total weight	100 kg
Ratio weights $\frac{\text{containment}}{\text{total}}$	$\frac{1}{2.7}$

2.2. Impact tests

Several tests from 9 m height have been directed to localized impact in:

a) top and bottom

b) Edge (particular for water tightness)with axis inclined to assure that vertical baricentric passes through impact area.

c) generatrix.

Fig. 5, shows accelerograms measured during preceding tests.

Due to lack of space we report simply the main results of tests:

a) No relevant damage has been ever observed on internal vessel.

b) External drums although have been heavily deformed did not to release conglomerate.

c) Residual insulating material never lowered below 40% original thickness.

d) Maximum values of decelerations ranged about 300 ÷ 400 g.

2.3. Thermal tests

After the impact tests, some containers have been tested in the open fire station for 30 min. and then submerged in a water - head of 0.9 m. The temperatures behaviour inside the container has been measured with a series of thermocouples (up to 12) placed in several relevant points within the

different layers of the container.

Particularly the temperature of gasket-point has been measured.

The results of early tests showed that the conglomerate when normally dried and aged do not insulate sufficiently to lower the hot-peak temperatures below 600°C.

These temperatures call for metal gaskets and to not allow lead shieldings when requested.

Nevertheless further studies brought to the conclusion that it was possible to limit considerably the inside temperatures by increasing the free water in the conglomerate.

This possibility it may be very interesting due to the fact that can give more freedom for the contents and easier choose of gasket and shielding materials.

After a series of tests, it has been determined the optimum composition from the view-point of mechanical and thermal characteristics.
Special plastic paints and brazed laminated steel sheets guarantees the requested free water in the conglomerate.

The results of fig. 6 show that prototypes, in this manner designed, reached, for the internal vessel closure, temperatures of 180 ÷ 190°C after about 15' the end of test. In some other tests, perhaps for different ambient conditions, the same temperature did not exceed 100°C.

One prototype of last version of the CF6 container, after impact tests, has been subjected to the "Furnace test" following the conditions, of IAEA[3].

For this test it has been utilized an electrical furnace "SFEAT" of 3 m^3 volume which has been brought to 830°C for 6 h before the actual test.

The diagram of pag. 7 shows that while the outside temperature of container reached that of the furnace at the end of test, the internal vessel gasket temperatures did not exceed 100°C.

Briefly after the thermal tests it has been observed:

1) Max temperature measured on the external surface of container, during the fire tests ranged between 910 and 1060°C and it was variable during single tests.

2) Average thermal fluxes absorbed by the container both during fire tests and furnace test ranged about 18. 000 Ca/m^2h.

3) The total results of internal temperatures obtained for the two types of tests were reasonably similar.

4) Thermal insulation of CF6 container is mainly guaranteed by the free-water in the conglomerate.

3. CONCLUSIONS

It can be said that the CF6 container met the requirements of Section VII of IAEA[1] after the series of tests therein described.

Max values of decelerations measured on the internal vessel (during the 9 m free drop test) ranged around 350 ÷ 400 g.

During the thermal tests, referring to the CF6 version with controlled free water in the conglomerate, the internal ves-

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NOTE:

Permitted contents from Certificate of CNEN for the CF6 Packaging.

a) U powders up to 5% w/o:

max weight	18 kg
max number of packages	45
transport index	1.4

sel gasket temperatures did never exceed 200°C thus permitting use of silicone gaskets.

Those results justify a further development of this type of economical container using a larger commercial drum of 200 liters (50 gallons).

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b) U powders up to 93% w/o:

- for content up to 10 kg:
  - max number of packages 6
  - transport index 8

  for content up to 18 kg (max):
  - max number of packages 2
  - transport index 25

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5) CITTI P., FORASASSI G., GUERRINI B. — Trasporto del materiale radioattivo: attrezzatura di collaudo ed esperienze eseguite su prototipi del contenitore CF6. Istituto Impianti Nucleari dell'Università di Pisa RL 113(72) Tip. Editrice Pisana 1972.

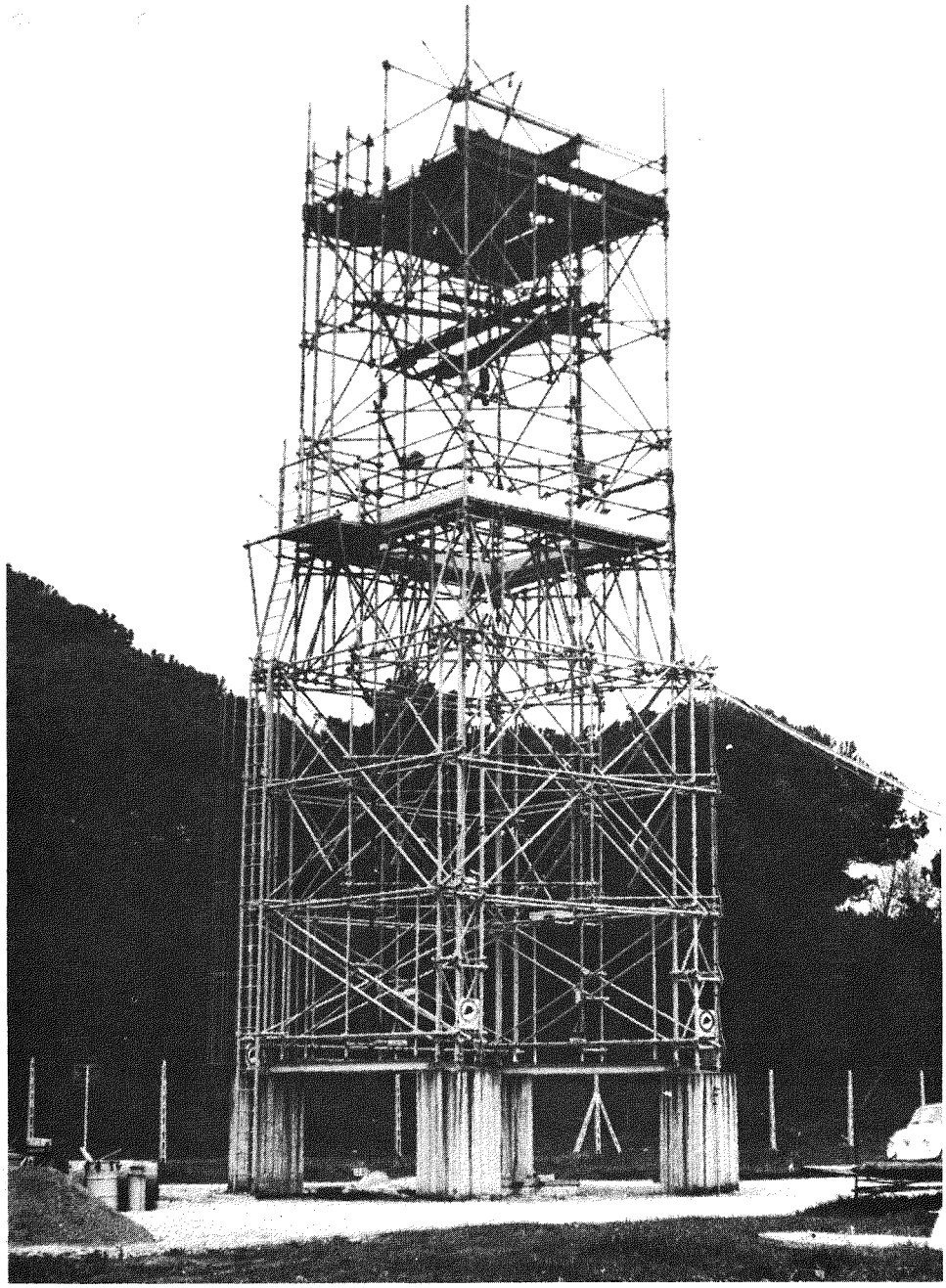


Fig. N. 1 - Drop test tower

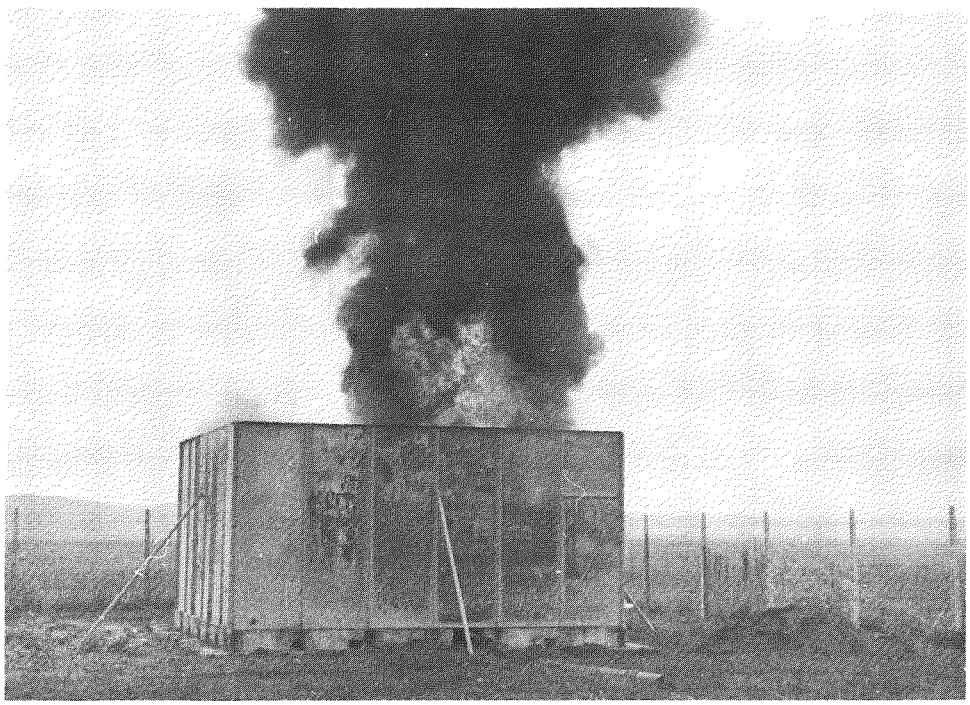


Fig. N. 2 - Fire test station

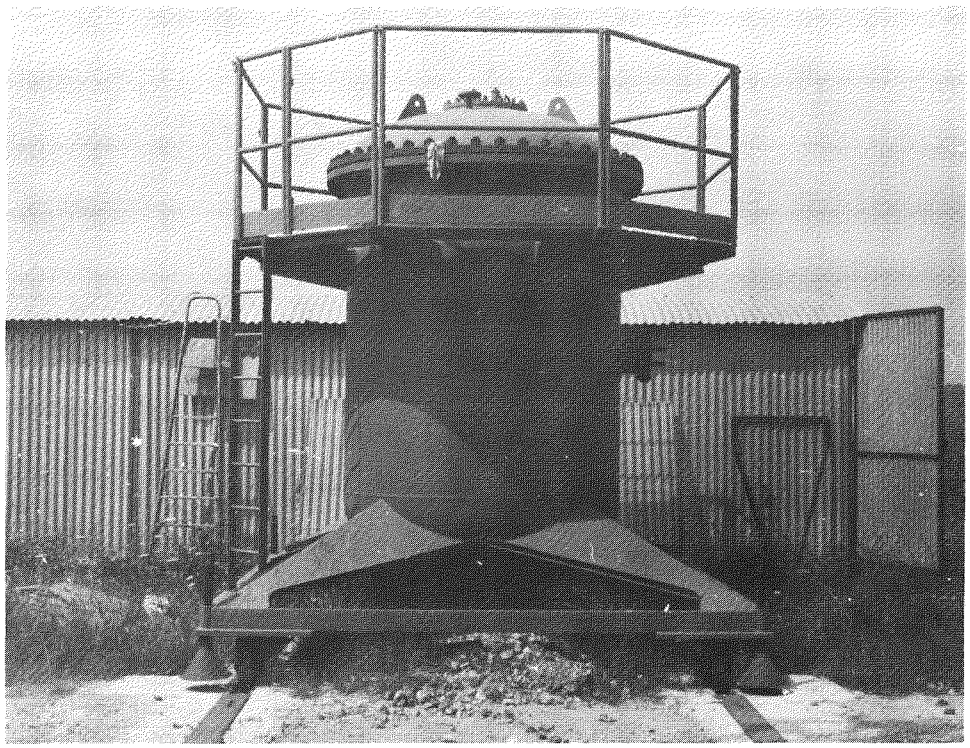


Fig. N. 3 - Hydraulic & tightness test container

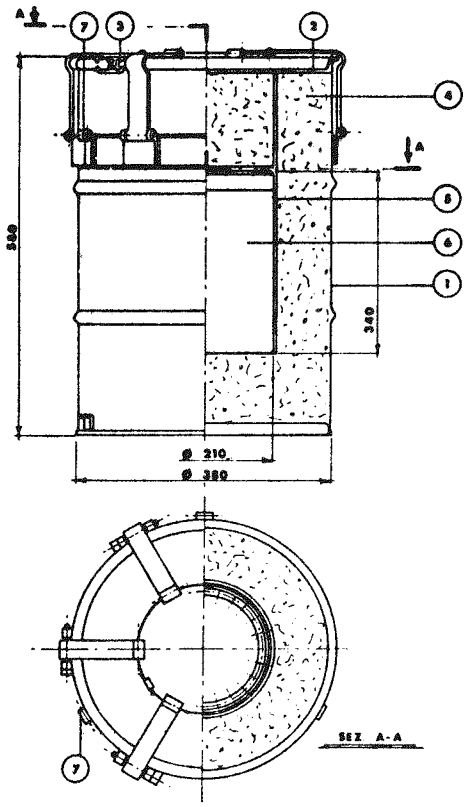


Fig. N. 4 - CF6 Packaging:

1) external container
2) cover
3) closure system
4) conglomerate
5) laminated sheet
6) internal vessel
7) lifting hooks

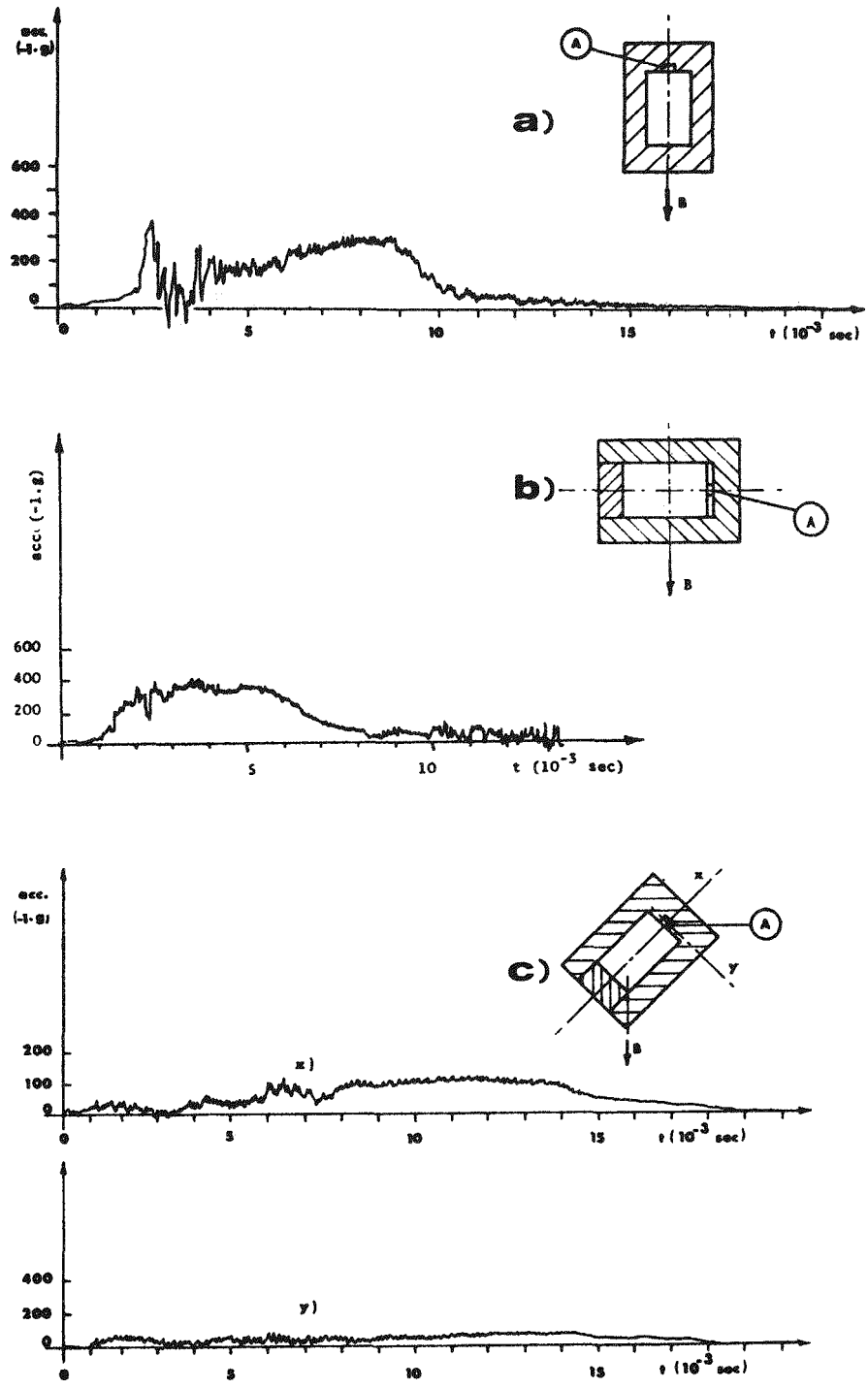


Fig. N. 5 - Accelerations recorded for three types of drop tests where:

A) Position of accelerometers

B) Direction of drop

a) bottom

b) generatrix

c) edge

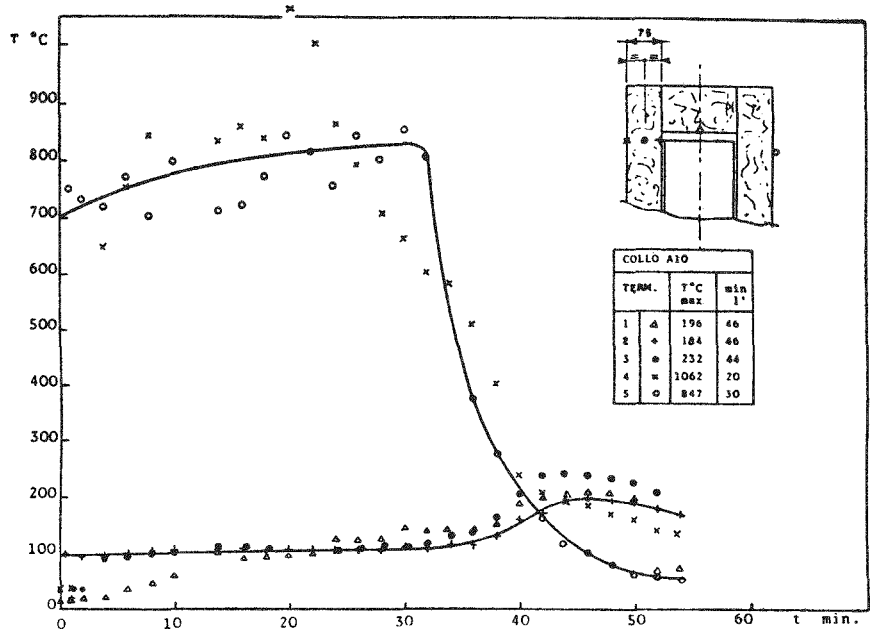


Fig. N. 6 - Temperatures recorded on a CF6 prototype during fire test

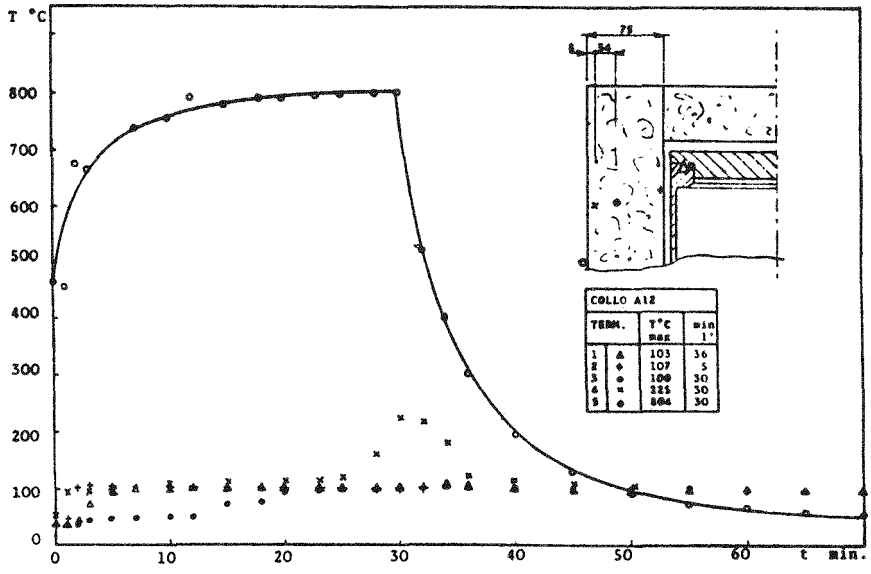


Fig. N. 7 - Temperatures recorded on a CF6 prototype during a furnace test.

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EXPERIMENTAL STUDY OF THE STRESS-STRAIN PROPERTIES OF CASK MATERIALS UNDER SPECIFIED IMPACT CONDITIONS

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ABSTRACT

A study was undertaken to experimentally determine the plastic dynamic compressive stress-strain characteristics of mild steel and stainless steel for strains up to 0.5 inches per inch. The specimen was placed on a load cell and impacted with a free falling weight. The force-time data was obtained from the specimens using a special data acquisition system. A digital computer program was used to calculate the engineering stress and strain for all data points. When the dynamic stress-strain properties are compared with static and quasi-static properties, both materials exhibit a significant increase in strength.

INTRODUCTION

An investigation was undertaken to determine the stress-strain properties of mild steel and stainless steel subjected to gross plastic compressive strains at strain rates comparable to those experienced in shipping containers for radioactive materials under accident conditions. The design criteria for structural integrity of these containers is governed by regulations,[1,2] covering both normal conditions and hypothetical accident conditions of transport. The hypothetical accident condition of interest here specifies that the shipping package or container be capable of withstanding a free drop of 30 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position in which maximum damage is expected.

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*Oak Ridge National Laboratory is operated by Union Carbide Corporation under contract with the United Stated Atomic Energy Commission.

Analyses of containers have generally resulted in ultraconservative design in that most of these analyses were based on material properties determined by static tests. However, many materials have impact properties that vary significantly from their static properties. Generally, the stress for a given strain will be higher for high strain rates that occur under impact conditions. Therefore, the determination of stress-strain properties under impact conditions is of importance. While developed for a specific purpose, the information resulting from this investigation should be of general interest and is applicable where dynamic compressive stress-strain relationships are needed.

TEST PROGRAM AND APPARATUS

The general test philosophy of impacting cylindrical specimens with a free falling weight had been used successfully in prior work.[3] It was felt that this approach offered the most promising means of obtaining the desired data within the limits of engineering accuracy at a minimum cost.

The impact tests were conducted at the Oak Ridge National Laboratory Drop Tower shown in Fig. 1. This structural steel tower is 40 ft high and its drop base is comprised of a surface of armor plate steel embedded in steel-reinforced concrete. Vertical guide rails were attached to the tower to guide the impact hammer during its free fall. The basic structure of this hammer weighs 100 lb. This basic hammer weight can be increased in 21-lb increments by the addition of steel blocks to the hammer, and a maximum weight of approximately 500 lb can be obtained.

Force was measured with a load cell which was an 8-in.-long cylinder with an outside diameter of 6.0 in. and an inside diameter of 4.5 in. It was fabricated of AISC 4340 steel heat treated for maximum toughness. The load cell has a design load capacity of 3000 kips. Bonded resistance strain gages were secured to the periphery of the load cell with epoxy resin and wired into a full Wheatstone bridge arrangement as illustrated in Fig. 2. This arrangement afforded a 5.2 amplification of the output signal, as compared with the output of a single strain gage.

A 12-volt automotive battery was used as a power source for the Wheatstone bridge, which was connected to a bridge balancing unit. The output

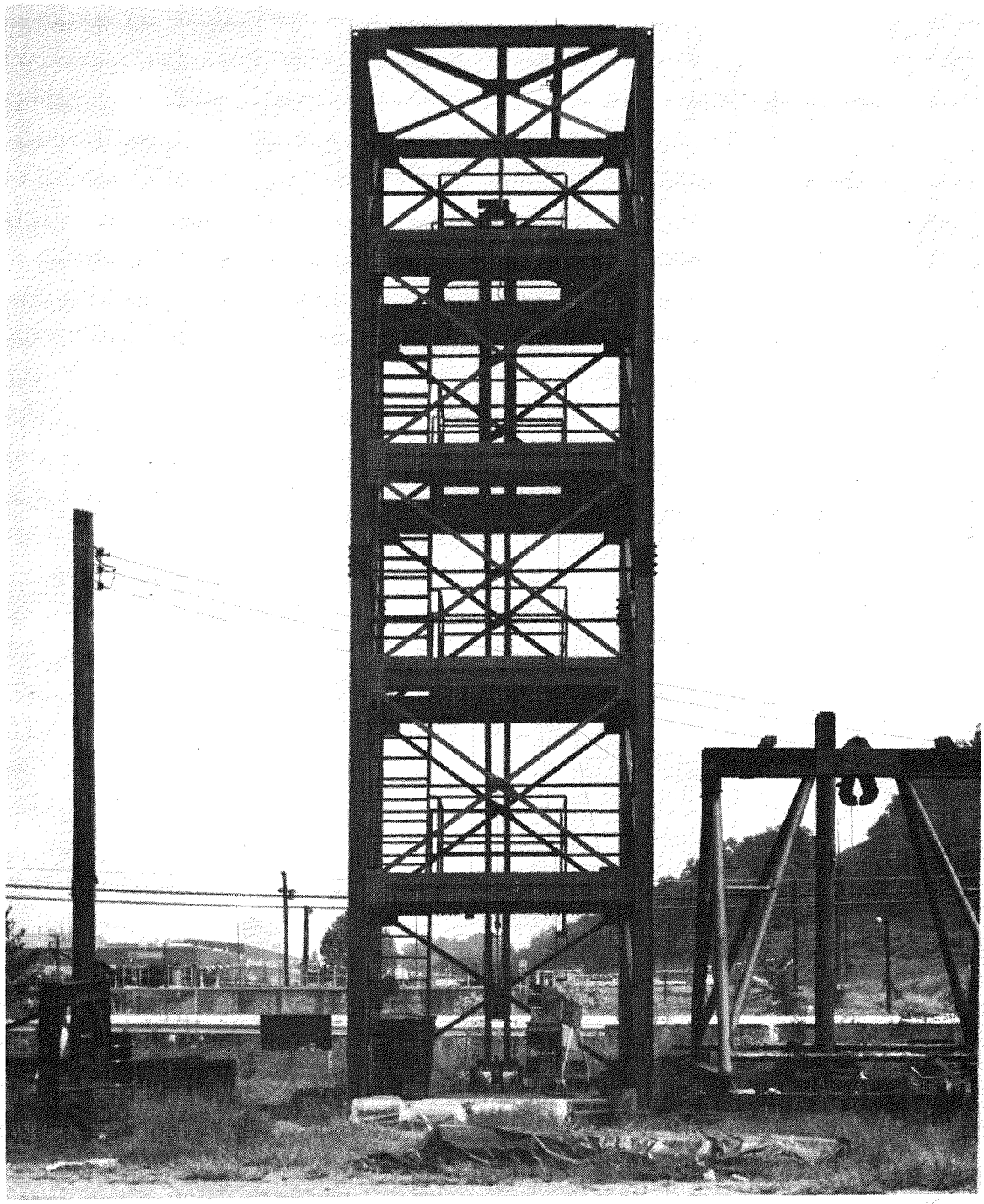


Fig. 1. Oak Ridge National Laboratory Drop Test Tower.

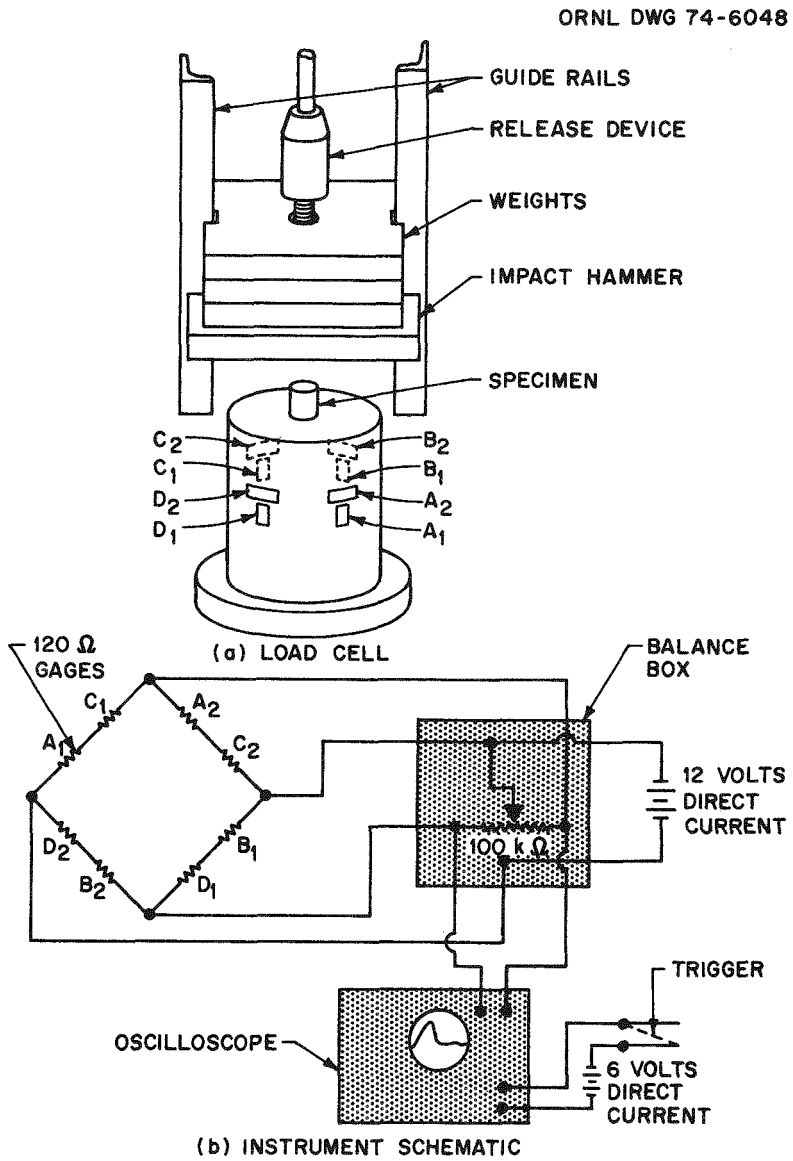


Fig. 2. Typical Load Cell and Schematic Diagram of Instrumentation Used to Measure Force-Time Relationships During the Impact Tests.

of the bridge was fed to one channel of a dual-beam Tektronix Type 565 oscilloscope with a Type 3A3 differential amplifier.

The load cell, amplifier, and oscilloscope package were calibrated up to the design capacity of a conventional 120,000-lb compression testing machine. The fact that track friction was negligible had been demonstrated in previous work.[4]

Right circular cylindrical test specimens with diameters of 1 in. by 1 in. long were machined so that the ends were parallel to each other and perpendicular to the center line of each cylinder within 0.001 in. in 4 in. Material for the specimens was AISC 1020 steel hot rolled and annealed and Type 304L stainless steel hot rolled and annealed.

PERFORMANCE OF IMPACT TESTS

The specimens were impact tested, and the force-time data for each impact were measured and photographically recorded with the equipment previously described. The specimens were impact tested with different weights dropped from different elevations. The weights varied from 100 to 280 lb, and the drop heights from 15 to 30 ft. Specimens after impact testing are shown in Fig. 3.

The procedures used to conduct the impact tests for each of the specimens were as follows. The test specimen diameter and length were measured to the nearest thousandth of an inch and recorded. The specimen was placed on the load cell and visually centered with the scribed alignment rings on the top surface of the cell. The desired sweep rate of 0.5 msec/div and a force scale of 36,000 lb/div were set on the oscilloscope. The camera was set for time exposure with the scale illumination off and the shutter open. The weight was hoisted to the desired height and released. The free-falling weight contacted the trace trigger just prior to contacting the specimen, initiating the single-sweep trace. The falling weight then contacted the specimen and transmitted force to the load cell. As the specimen deflected and load was applied to the cell, the previously balanced bridge became unbalanced and effected a change in the scope voltage that deflected the oscilloscope trace by an amount proportional to the load. A typical force-time photograph of the oscilloscope trace is shown in Fig. 4.

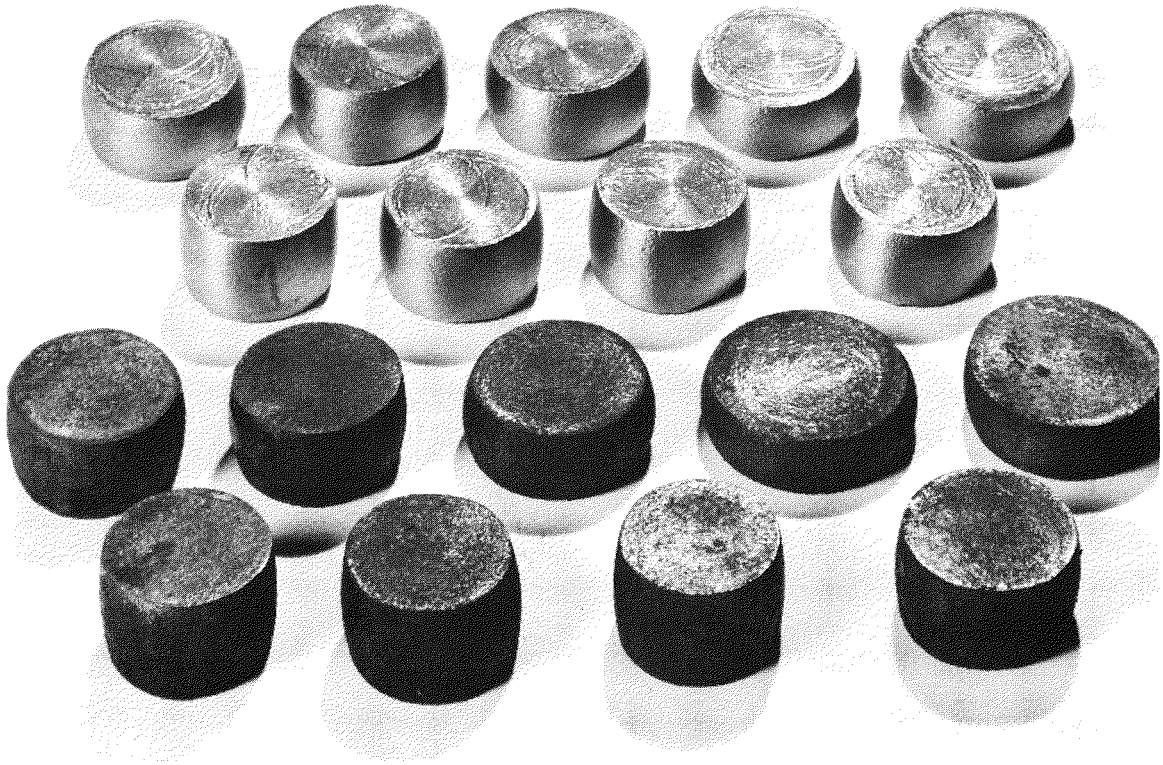


Fig. 3. Typical Specimens After Impact Testing.

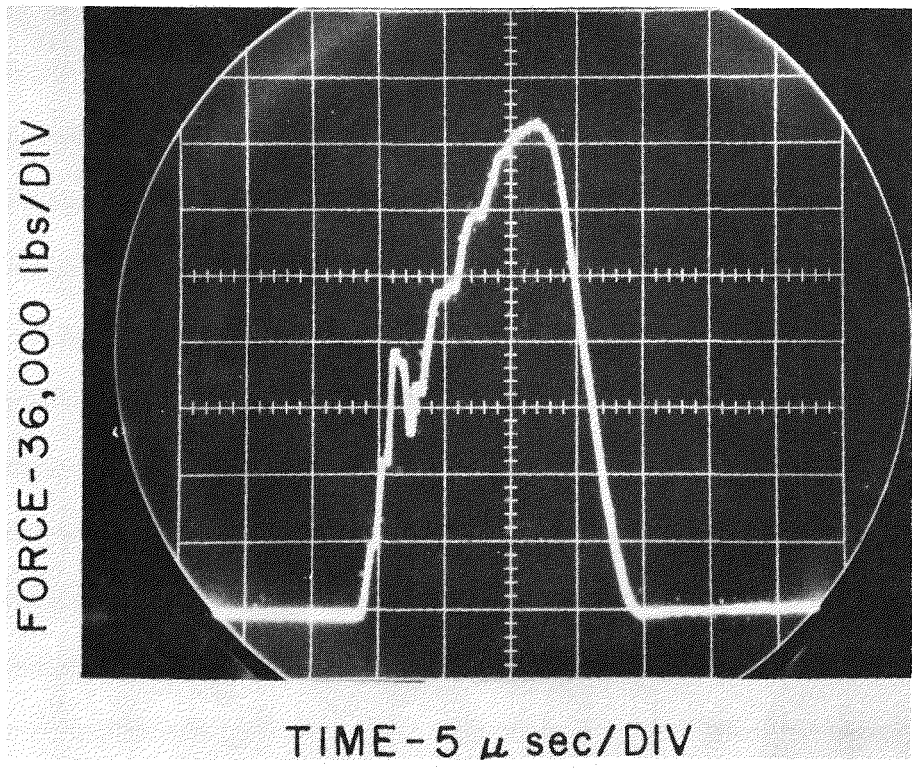


Fig. 4. Typical Force-Time Photograph.

DATA TRANSFORMATION AND EVALUATION

For each of the specimens, the force and time were scaled from the photographs of the oscilloscope traces. Since the mass of the free falling weight was much greater than the mass of the specimen and the strains in the specimen were essentially plastic, the force recorded from the load cell could be used to calculate the acceleration of the top surface of the specimen from Newton's second law.

A computer program was written to numerically integrate the discrete force-time data and obtain velocity and displacement as functions of time.

It was observed that the volume of the specimen and the variation of the cross-section along the length remained approximately constant after impact. These approximations along with the force, velocity, and displacement data were used in the computer program to calculate engineering stress and strain for the discrete data points taken for the specimens.

Figures 5 and 6 are computer plots of the stress-strain data for all data points.

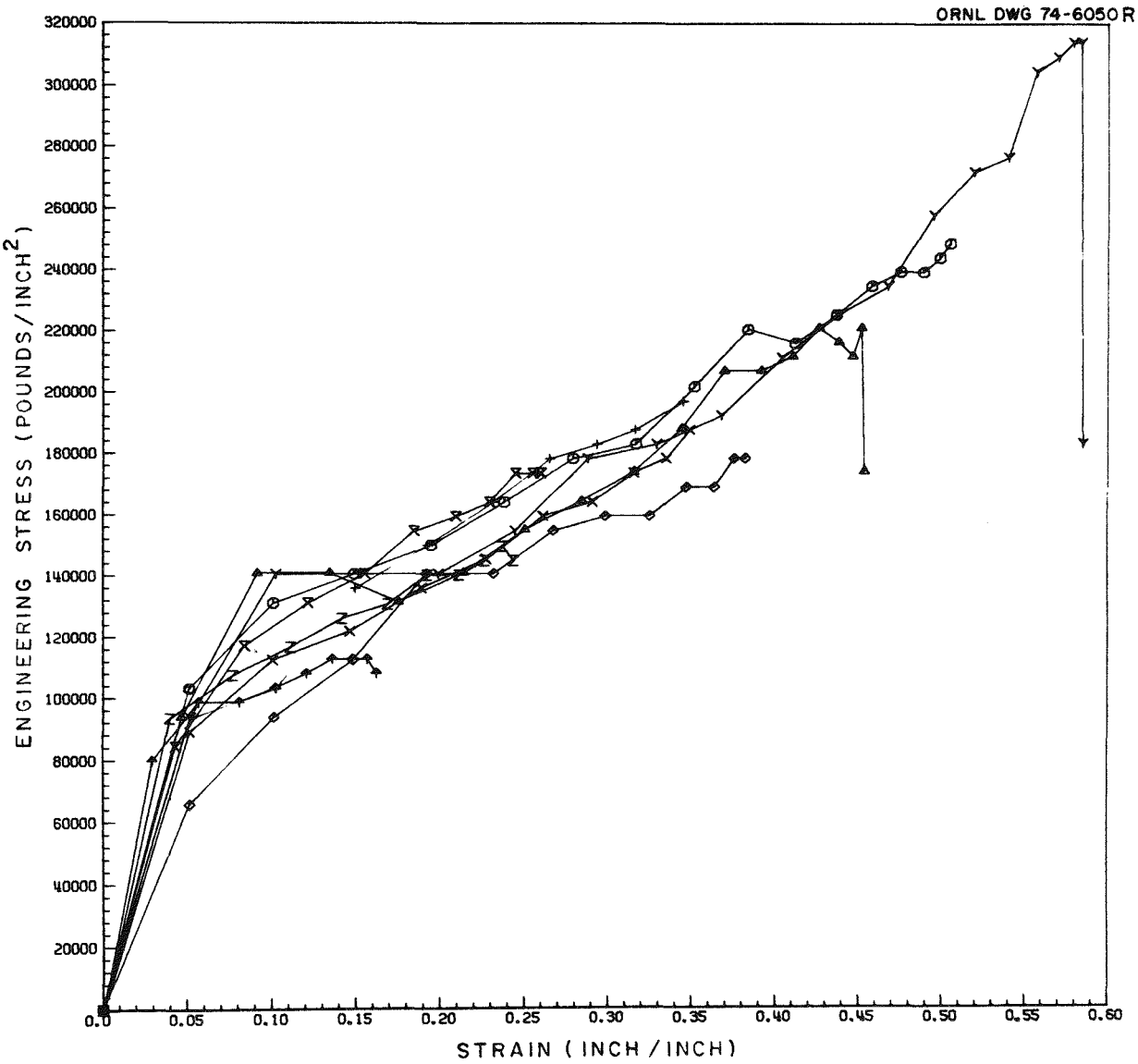


Fig. 5. Dynamic Compressive Stress-Strain Data for AISC 1020 Hot Rolled Steel.

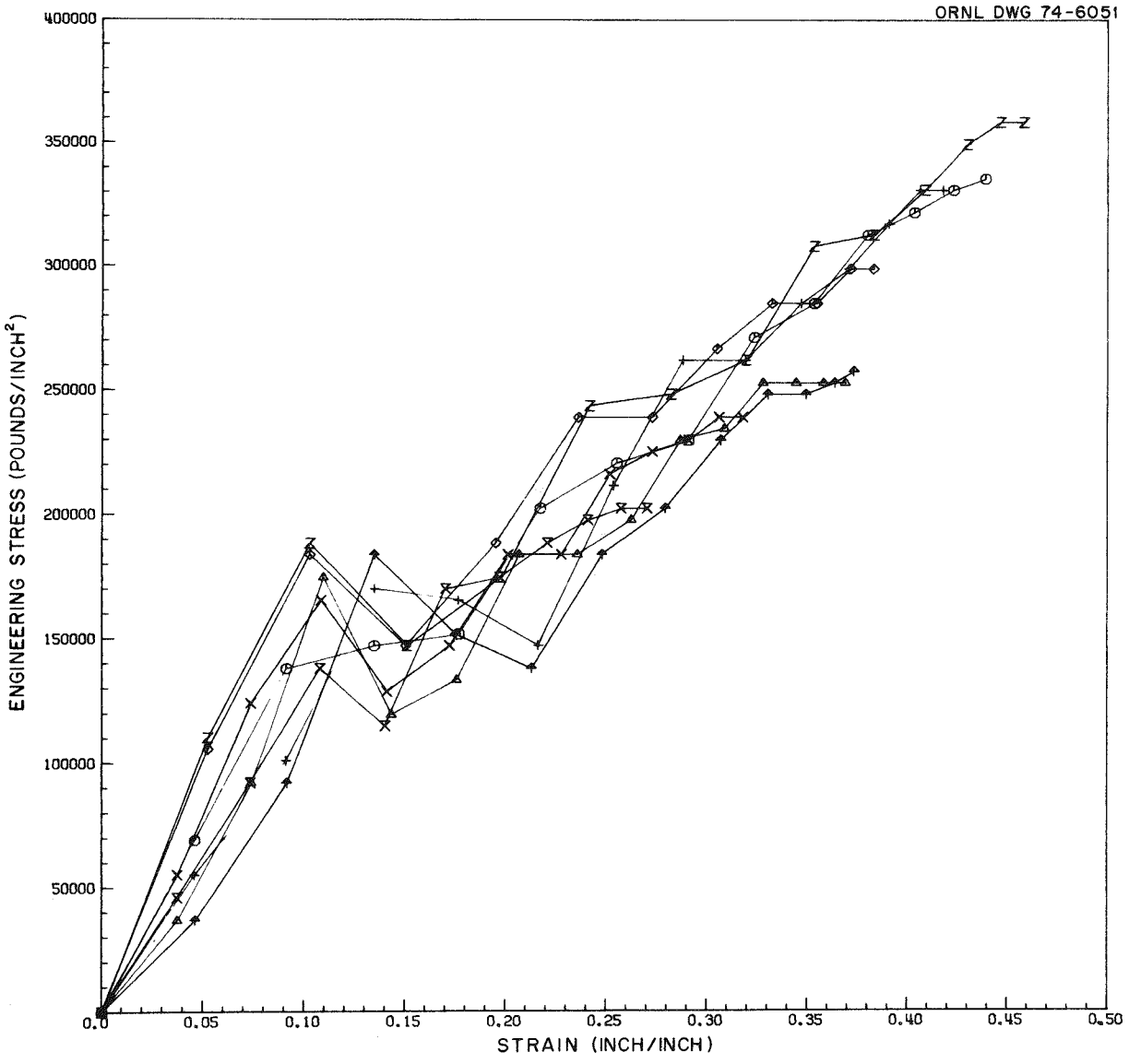


Fig. 6. Dynamic Compressive Stress-Strain Data for 304L Hot Rolled and Annealed Stainless Steel.

The following equations were derived using a least-squares method and describe the experimentally measured stress as a function of strain.

For mild steel, the polynomial equation

$$\sigma = 1.27798558 \times 10^3 + (4.07864936 \times 10^6)\epsilon - (7.81107560 \times 10^7)\epsilon^2 + (9.63696088 \times 10^8)\epsilon^3 - (7.49671720 \times 10^9)\epsilon^4 + (3.54881228 \times 10^{10})\epsilon^5 - (8.91654632 \times 10^{10})\epsilon^6 + (4.08811848 \times 10^{10})\epsilon^7 + (3.99127412 \times 10^{11})\epsilon^8 - (1.12951700 \times 10^{12})\epsilon^9 + (1.27302446 \times 10^{12})\epsilon^{10} - (5.43858552 \times 10^{11})\epsilon^{11} \quad (1)$$

applicable to computer solutions and the straight line,

$$\sigma = 73{,}000 + 345{,}000\ \epsilon \quad (2)$$

applicable to hand calculations were written. Likewise, for stainless steel,

$$\sigma = (-3.28041320 \times 10^2) - (4.89858432 \times 10^6)\epsilon + (4.85760448 \times 10^8)\epsilon^2 - (1.59814612 \times 10^{10})\epsilon^3 + (2.94739248 \times 10^{11})\epsilon^4 - (3.32476216 \times 10^{12})\epsilon^5 + (2.38366924 \times 10^{13})\epsilon^6 - (1.10540172 \times 10^{14})\epsilon^7 + (3.30126304 \times 10^{14})\epsilon^8 - (6.12699608 \times 10^{14})\epsilon^9 + (6.42770848 \times 10^{14})\epsilon^{10} - (2.91202204 \times 10^{14})\epsilon^{11} \quad (3)$$

and the straight line,

$$\sigma = 50{,}300 + 642{,}000\ \epsilon\ . \quad (4)$$

Curves of these equations are plotted in Fig. 7.

CONCLUSIONS

The results of the impact tests show that stainless steel exhibits dynamic compressive strength significantly greater than mild steel. However, when the dynamic compressive stress-strain properties are compared with static and quasi-static properties, both materials exhibit a significant increase in strength. The equations, developed empirically,

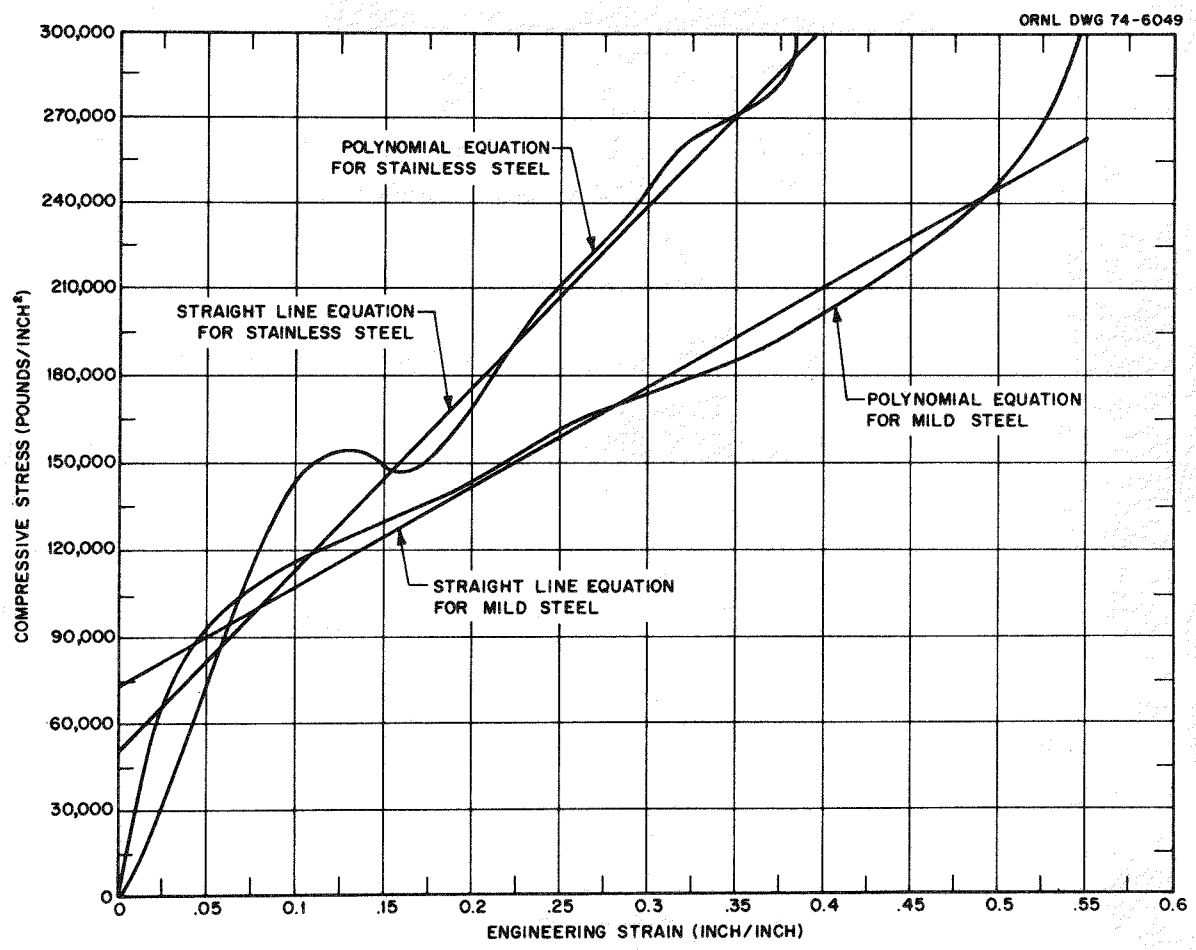


Fig. 7. Equations 1, 2, 3, and 4.

adequately represent the dynamic stress-strain relationship for stainless steel and mild steel within the limits of the impact data. Therefore, these equations and the curves plotted (shown in Fig. 7) can be used to estimate the dynamic compressive deflections and stresses of members in shipping casks that are being evaluated for compliance with the regulations governing the 30-ft free fall accident condition.

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DROP AND FIRE TESTS OF HALF-SCALE MODEL OF TN-9 SHIPPING CONTAINER

G. P. Wachtell

ABSTRACT

The tests were done under a contract with Transnuclear, Inc., in support of the safety analysis report of the TN-8/9 shipping container. These containers were licensed by the USAEC in May 1974. The model, supplied by Transnucleaire, S. A., Paris, France, survived the drop and fire tests without losing leak tightness of the cavity. The lead temperature remained well below its melting temperature. Computed thermal performance for both the model and full-scale shipping container showed that the lead temperature remains at least 60°F below the melting temperature.

For a half-scale model of a 30-minute full-scale fire test, the heat flux should be double that of the full-scale test, and the test duration should be 7.5 minutes. The actual test duration was 10-$\frac{1}{3}$ minutes, to compensate for a less than doubled heat flux. For the computed model thermal performance, a doubled flux was assumed for 10.35 minutes, compensating approximately for the absence of payload self-heating in the model.

DESCRIPTION OF MODEL

The model, supplied by Transnucleaire, S. A., Paris, France, was a cylinder 9.4 ft long, including detachable shock-absorber end pieces, and had 58 longitudinal copper fins 6.64 ft long, 0.57 ft tall and 0.00656 ft thick, attached to a cylindrical shell of 0.0328 ft thick steel. Including fins, the model OD was 2.54 ft. The steel shell housed a lead lining, separated from it by a 0.0164 inch layer of an insulating cement having a thermal conductivity of about 1 Btu/ft-hr-°F, and about 1/3 as much after water, driven off by heat, escapes through vent holes after fusible plugs melt. The lead had an OD of 1.3 ft and surrounded an octagonal steel-lined cavity having sides tangent to a 0.78 ft circle. Shock absorbers, which were steel shells filled with balsa wood, were bolted to the cask model at each end. The cask body is barbell shaped

and consists primarily of lead. The cask ends are balsa filled steel shells, which surround the lead filled cylindrical portion, and are fitted with trunnions for lifting and base plates for setting down in the horizontal position.

The space between the fins was filled with a neutron shield resin to about half the fin height. The actual full scale fins are cut into strips by radial cuts. The strips are each twisted about a radial axis to separate them, improve convective heat transfer, and help anchor the resin. Under fire conditions, radiative heat transfer is dominant, so the difference in fin shape is significant only during cool-down after the fire, when the model is conservative. A cylindrical dummy payload weighing 703 pounds was present in the drop tests, as well as in the subsequent fire test. The total model weight, including dummy payload, was about 9500 lbs.

DROP TESTS

The drop tests were performed on an 18 inch thick concrete slab with a high tensile steel plate, 3 in. thick x 5 ft x 20 ft., weighing 122 pounds per square foot placed on top and held down by I-beams placed across each end of the plate and hooked to fixtures embedded in the concrete.

Two high-speed motion pictures of each drop test were taken. The drop tests were performed in the following order:

1. Thirty foot drop onto corner with lead electrically preheated to about 175°F.
2. Thirty foot drop onto the same end, with axis vertical.
3. Forty inch end drop onto a three inch diameter flat-ended pin twelve inches long, measured from the top of a 5 inch tall conical base.
4. Forty inch drop onto pin with model upside down with axis horizontal.
5. Thirty foot drop onto a pair of trunnions, with model axis horizontal.

In the first two tests, the impact occurred on the end on which the model contained an inner cover for inserting the payload. The other end was used in the third test. Peak-reading accelerometers yielded the follow-

ing decelerations: (Two accelerometers were used in tests 2, 3, and 5.)

Test No.	1	2	3	4	5
Deceleration (g's)	160	180 180	60 20	20	180 150

Previous studies[1] have shown the validity of impact model testing for lead-lined containers. Figures 1-5 show the external damage due to the tests.

In Figure 1, the partial collapse of the corner of the shock absorber may be seen.

In Figure 2, the surface of the steel plate shows the circular outline of the shock absorber, as well as the flattened end of the shock absorber itself, after the second test. Figure 5, taken after the fifth test, shows another view of the flattened end. Shock absorber thickness was reduced to 3.5 inches.

In Figure 3, the hole made in the shock absorber in the third test is shown. Punch penetration was not more than 6-3/4 inches. In later examination after the fire test the shock absorber was removed. The inner shock absorber surface was found to have been indented and slightly cracked. This may be partly due to the fact that shims consisting of square washers around each bolt had been provided to allow space for thermocouples that had been installed for the fire test. It had been considered desirable to install these thermocouples before the drop tests so that the shock absorber need not be disturbed between the drop tests and the fire test. Without the shims, the inner face of the shock absorber would have been supported by the cask end. It was also found that the cask end itself was slightly indented. The cask end had two small openings, covered by discs with O-ring seals. One was in the center. The other, fitted with the gas pressurizing connection, was off center. The indentation was immediately adjacent to the off-center opening, so that the plane of the disc was rotated slightly as a result of the indentation.

Figure 4 shows the hole made in the model in the fourth test. After the fire test, when the shock absorbers and inner cover were removed, visual inspection of the cask cavity showed no apparent distortion of the cavity.

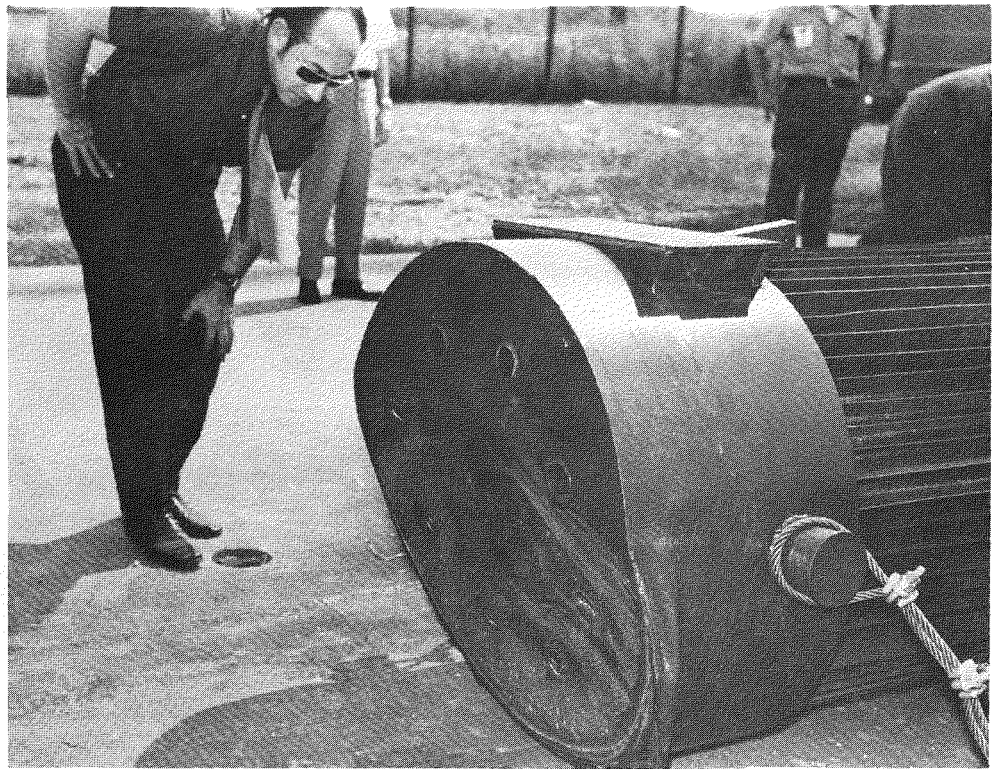


Fig. 1. Thirty Foot Corner Drop, Showing Partial Collapse of Corner of Shock Absorber.



Fig. 2. Thirty Foot End Drop, Showing Flattened End of Shock Absorber.

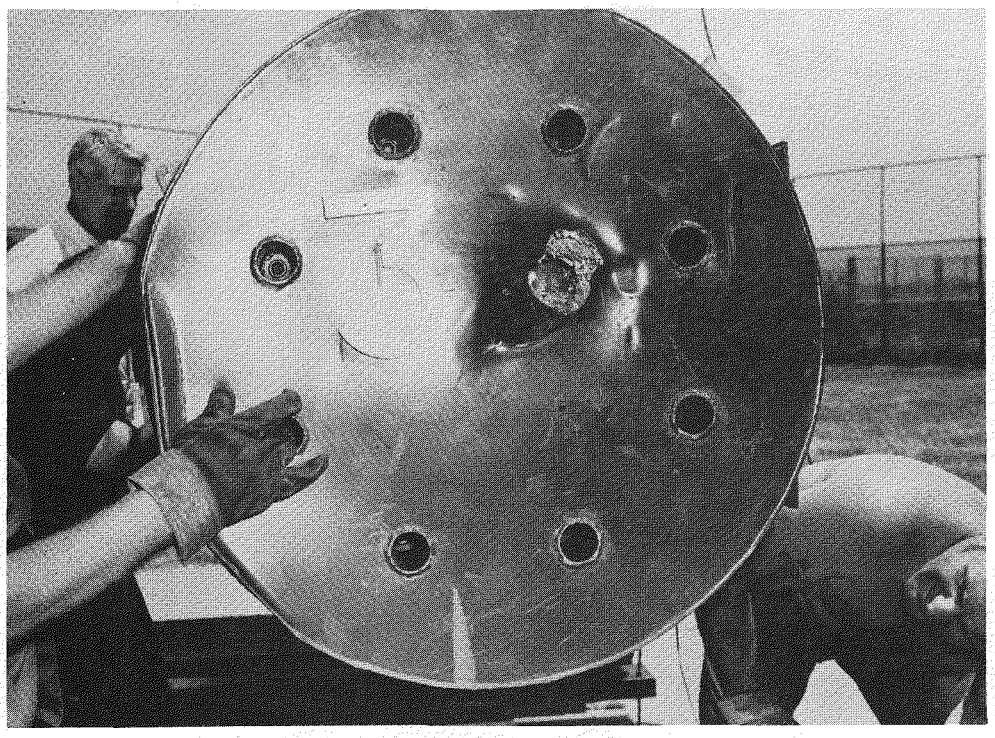


Fig. 3. Forty Inch End Drop Onto Three Inch Diameter Pin.



Fig. 4. Forty Inch Drop Onto Pin With Model Upside Down.

Figure 5 shows one of the trunnions after the fifth test. Both trunnions had the same general appearance.

Freon leakage tests made after each drop test showed no evidence of leakage. The presence of a fairly strong breeze created some doubt that a leak would have been detected, although an attempt was made to shield the freon detector from the wind. After the fire test, the model was returned to The Franklin Institute for inspection. Leak tests were conducted indoors. The leak tests were of three kinds:

1. The cavity was found to hold a pressure of 10 psig for 64 hours, with a loss of only 2 psi. The loss of 2 psi is accounted for by the drop in temperature of the gas in the cavity after its initial compression, and does not indicate a leak.

2. "Leak-Tec" was used at all the possible leakage sites. In this method, small bubbles are an indication of a gas leak. No bubbles were seen.

3. The freon test did not disclose any leaks.

FIRE TEST

The fire basin was an 18-foot square filled with 2 ft of water and 4 inches of number two fuel oil. The initial fuel surface was 2 inches below the top of the basin wall. Five gallons of gasoline were added just before ignition to insure rapid spread of the flame.

The test stand (Figure 6) consisted of a pair of pedestals 30 inches above the initial fuel surface. A frame supported ten thermocouples for measuring fire temperature. Two were on the model axis. The other eight were in the vertical midplane normal to the axis.

The fire test (Figure 7) was to model a full-scale 30-minute test in a fire with T = 1475°F, emissivity 0.9, and cask surface absorptivity 0.8. A half-scale model exposed to a double-flux source for 7.5 minutes would yield a conservative test[2,3], since reradiated flux for a given surface temperature is not doubled. Since the actual test fire did not produce a doubled flux, the test duration was increased so that the time integral of incoming flux had the same value as in a double-flux test.



Fig. 5. Thirty Foot Drop Onto Pair of Trunnions, With Model Axis Horizontal.

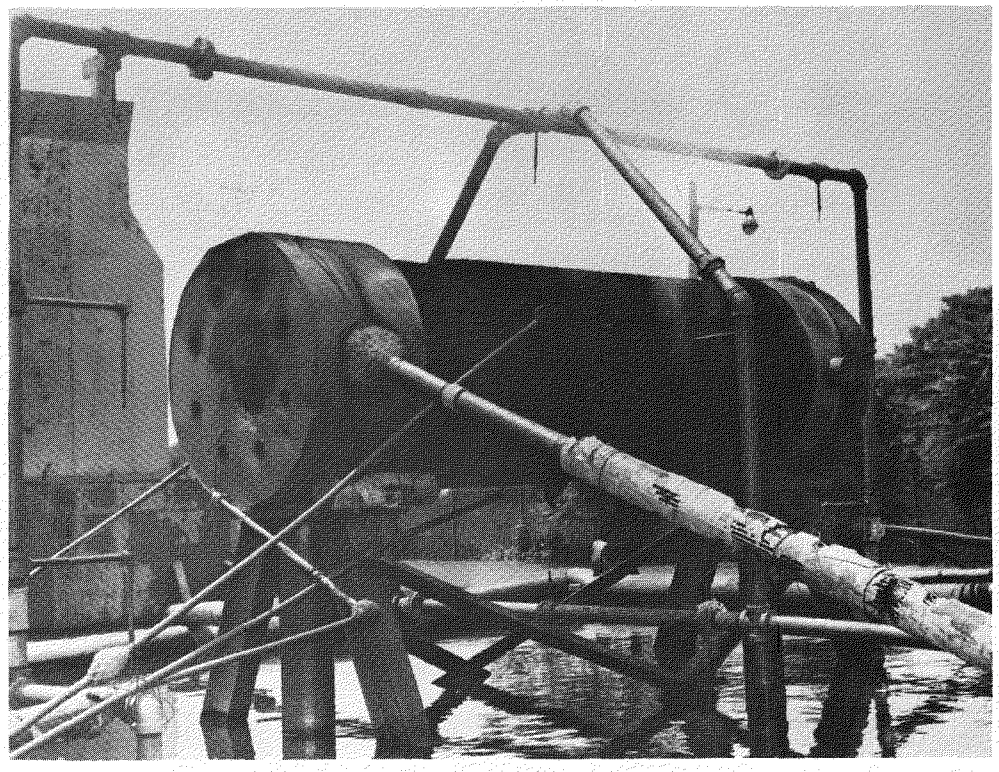


Fig. 6. Model On Test Stand Following Fire Test. Note Frame Supporting Fire Thermocouples Approximately One Foot From Model.

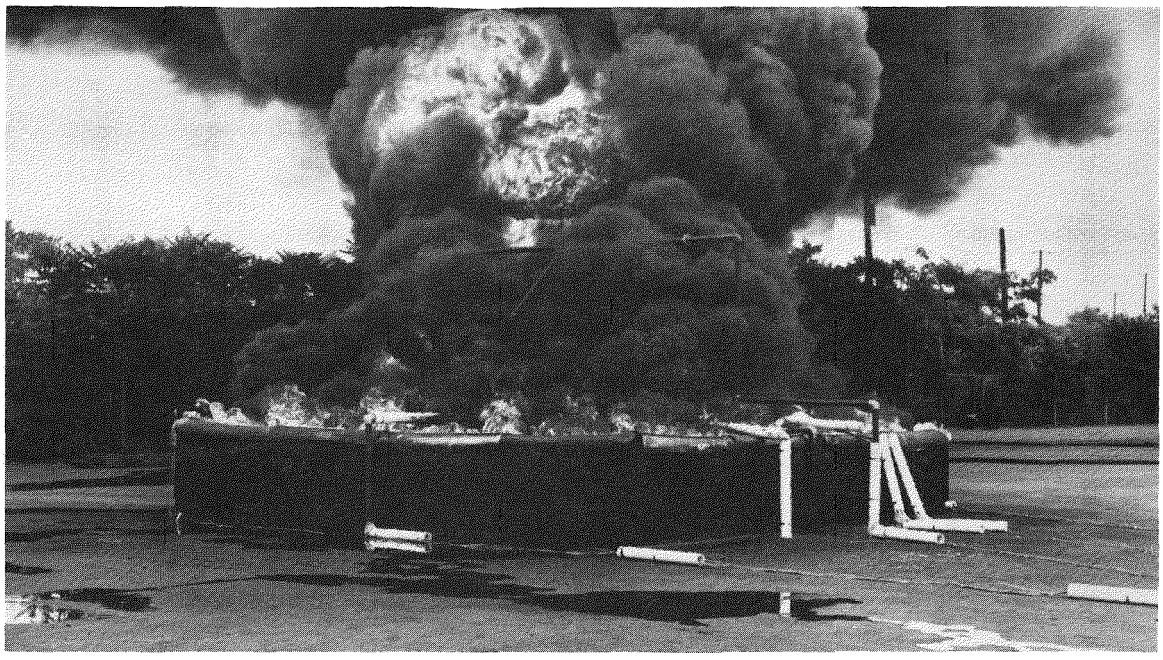


Fig. 7. Fire Test In Progress.

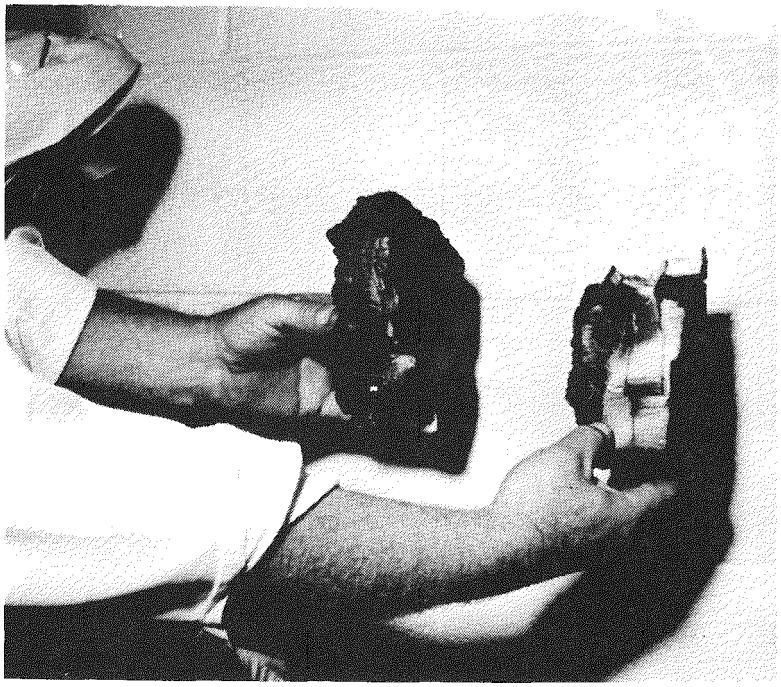


Fig. 8. Charred Balsa Wood Removed From Shock Absorber After Fire Test. Incomplete Charring Indicates Protection of End of Model.

The model test did not include internal heating of the payload.

Expressing temperatures in °F, we have for the equilibrium temperature T_{tc} of a thermocouple in a fire of temperature T_f:

$$0.9\ (T_f + 460)^4 = (T_{tc} + 460)^4$$

We assume that the desired fire duration is such that the total heat delivered is the same as in the ideal double-flux case. That is, t should be inversely proportional to $(T_{tc} + 460)^4$ and should be 7.5 minutes in the double-flux case.

In the double-flux case,

$$(T_f + 460)^4 = 2\ (1475 + 460)^4$$

leading to $T_{tc} = 1781$ °F. Thus,

$$t = 7.5\ \left(\frac{1781 + 460}{T_{tc} + 460}\right)^4 \text{ minutes}$$

This led to Table 1 for making a rapid decision for test determination. During the test, there was little time to average the thermocouple readings. The decision to call for termination of the test was based on the impression at the time that the readings appeared to have some average value. Table 1 was essential in making this decision.

The precise duration of the fire test is, of course, a matter of definition because it took a few seconds for the fire to spread when it was ignited, and a few seconds to extinguish it. Taking the timing uncertainties into account (for example, disregarding most of the initial build-up of the fire) the total test duration was 10 1/3 minutes.

Two minutes after the fuel oil fire was extinguished, the resin was observed to be burning, and the model itself was then doused with water.

The fire thermocouple readings were interpreted by calculating a mean radiation value for each thermocouple. That is, for a given thermocouple with T_i representing reading number i in °F, its mean temperature was found as

Table 1. Model Test Duration

Average Thermocouple Indication (°F)	Test Duration (min.)
> 1780	7.5
1745 - 1780	8.0
1710 - 1745	8.5
1680 - 1710	9.0
1650 - 1680	9.5
1625 - 1650	10.0
1600 - 1625	10.5
1575 - 1600	11.0
1555 - 1575	11.5
1535 - 1555	12.00

$$T = \left[\frac{1}{n} \sum_{i=1}^{n} (T_i + 460)^4\right]^{1/4} - 460$$

where n is the number of readings for that thermocouple during the fire test. This constant value of T would yield the same total heat input as the actual time-varying temperature.

One fire thermocouple failed. The other nine thermocouples yielded radiation mean temperatures between 926 and 1981°F, indicating substantial dependence of temperature on position. The radiation mean for the four thermocouples around the middle of the model was 1579°F, corresponding to a desired test duration of 10.9 minutes, or about 0.5 minutes longer than the actual test duration. However, the temperature above the model was high enough (1737°F) so that the region containing two thermocouples indicating temperature in the resin was exposed just a little longer than necessary. These resin thermocouple readings are therefore valid indications of performance. The temperature opposite the end of the model on which the pin drop test was performed was 1981°F, higher than needed for the actual test duration.

The two resin thermocouples, located on the interface between resin and steel shell, were unfortunately the only thermocouples attached to the model that did not fail. One indicated a temperature that gradually rose to 800°F at about t = 8.3 minutes, and then fell slowly to 700°F by the end of the test. The other indicated a temperature that rose gradually to 486°F shortly before the end of the test, falling to 480°F by the end of the test.

The higher peak reading is about 375°F lower than the calculated temperature (See Figure 10) of the steel when the incoming flux is reduced to zero. We conclude that the maximum temperature in the lead was far below the melting temperature of lead. Tests done by Transnuclear on a full scale slice supplied temperature data lost in the test reported here. Their measurements verified that temperatures were acceptable in the components of the shipping container.

After the fire test, the shock absorber was removed from the cask end on which the pin drop test was performed. The slight crack in the inner face of the shock absorber, mentioned in the description of Figure 3, was found. Hot gases may have been able to penetrate through this crack into the space between the cask end and the shock absorber. Nevertheless, there were the following indications that the temperature on the cask end remained well below the melting temperature of lead (621°F):

1. "Tempilstik" markings (which change color at specified temperatures) indicated that the maximum temperature was between 400 and 450°F near the axis on the end of the model shipping container, and between 450 and 500°F near the edge of the inner surface of the shock absorber.
2. Strips of masking tape had been used to hold the shims in place on the end of the model shipping container before the shock absorber was attached. Most of the tape survived without being scorched.
3. The inner cover on the other end of the model was removed. Butyl rubber O-rings were visually inspected and judged to be in excellent condition.
4. The shock absorber was opened. The balsa wood was found to be severely but incompletely charred (Figure 8), indicating successful protection of the end of the model by endothermic reaction.

Calculations were made of the thermal performance of the full scale shipping container and of the model. To compensate for neglect of axial heat conduction, two rates of self-heating by the payload were assumed for the full-scale shipping container: An average value of 6919 Btu/hr ft and a "peak" value of 10,892 Btu/hr ft.

Heat was assumed to be lost, after 0.5 hrs when the incoming radiant heat flux was reduced to zero, through a film coefficient $h = 1$ Btu/ft^{2}hr°F to an ambient temperature of 100.4°F. (Transnuclear's safety analysis report shows satisfactory calculated temperatures with an ambient temperature of 130°F.) Three values of insulation thickness, δ_{ins}, were used: 6, 10 (nominal value), and 14 mm. For the model, the ambient temperature was taken to be 90°F, and the duration of the heat flux 0.1725 hrs (10.35 min.), but the flux was taken as double that for the full-scale shipping container. The calculation for the model was therefore more severe than the actual test.

Table 2. Calculated Maximum Lead Temperatures

Cement Thickness (mm)	Q_{self} (BTU/hr) per ft of cask§	Maximum Lead Temperature (°F)	Time (hrs.)
10	Peak	530	1.3
10	Average	498	1.2
6	Peak	561	1.1
6	Average	544*	0.6
14	Peak	521	1.6
14	Average	478	1.4
5 (Model)	None	509*	0.325

* Lead outer surface

§ Average = 6919, Peak = 10,892

For purposes of computation, cylindrical geometry was assumed. Thirteen space nodes were used: Node 1, exposed part of fin; Nodes 2-5, fin covered by resin; Nodes 6-8, steel shell; Nodes 9-12, lead; and Node 13, payload. The computational time step for the full-scale shipping container was 0.00025 hrs, and ¼ as large for the model.

Figure 9 shows computed temperatures versus time for the full-scale shipping container with cement thickness 10 mm and average self-heating of payload, at four selected nodes. Figure 10 shows the computed thermal performance of the model for the same nodes.

In all cases, heat continued to flow into the lead after the incoming flux was reduced to zero, because the steel temperature did not immediately fall below the lead outer node temperature. Because of this, as well as payload self-heating in the full-scale shipping container, the computed maximum lead temperature occurred after the incoming flux was reduced to zero.

The calculated maximum lead temperatures are summarized in Table 2. In all cases, the calculated maximum lead temperature is seen to be well below 621°F, so that there is no risk of lead melting.

SUMMARY AND CONCLUSIONS

A double-flux fire test of a half-scale model is conservative, because reradiation is not doubled, so the net absorbed flux is more than doubled. The test that was performed differed from a true double-flux test in the following ways:

1. Initial temperature and internal heating of the payload were not simulated.
2. The test duration was longer than the 7.5 minutes that would be needed in a double-flux test, to compensate for the fact that the actual test fire provided less than a doubled flux.

Two thermocouples on the model performed satisfactorily. These were located on the interface between resin and steel, and read several hundred degrees lower than the temperature calculated for the steel in a double-flux experiment. This can be explained as due to endothermic decomposi-

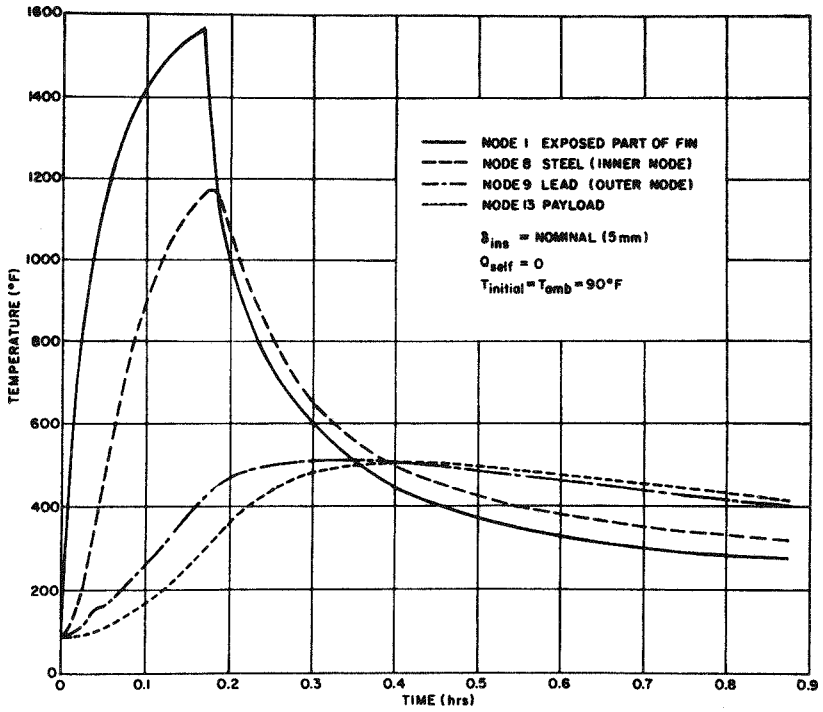


Fig. 9. Calculated Full-Scale Thermal Behavior in 30-Minute 1475°F Fire, With Fire Emissivity 0.9 and Surface Absorptivity 0.8, for Average Payload Self-Heating and Nominal Insulation Thickness.

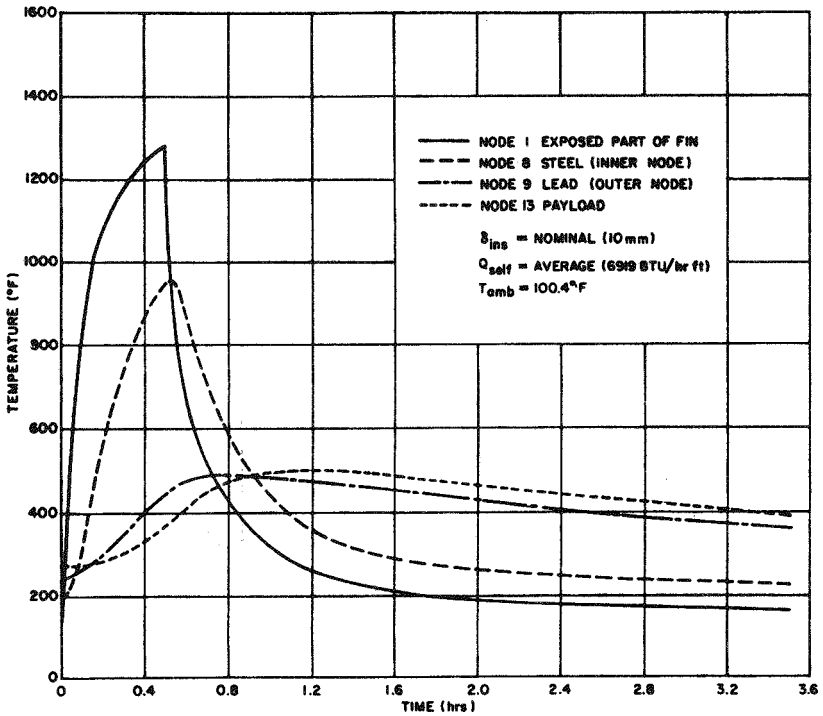


Fig. 10. Calculated Half-Scale Model Thermal Behavior in Double-Flux Fire of 10.35-Minute Duration. This is Longer Than the 7.5-Minute Duration Needed for Conservative Modeling, and More Severe than the Actual Model Test.

tion and evolution of gases by the resin, and also by the fact that the fire temperature was lower than that for a doubled flux.

All indications showed that the lead temperature remained well below the melting point throughout the model. The cavity remained leak-tight. Although some melting of lead should not be expected to damage the shipping container, the test did not determine whether melting would have done any harm.

Calculations of the thermal performance of the model in a double-flux experiment and of the corresponding full-scale shipping container in the correct experiment, show that the maximum temperature of the lead is at least 60°F below the melting point in both cases, even with the most unfavorable choices of cement thickness and self-heating in the full-scale shipping container calculation.

Early dousing of the model was necessary because the resin continued to burn after the fire was extinguished. We have been informed, however, that the resin composition has been modified so that it would no longer continue to burn.

ACKNOWLEDGEMENTS

Through a subcontract with the Naval Air Development Center, (NADC), Warminster, Pennsylvania, facilities of the Naval Air Engineering Center (NAEC), located at the U. S. Naval Base, Philadelphia, Pennsylvania, were used for the drop tests. The NADC staff at the Naval Base provided excellent coordination of various groups, including the photographers and the NAEC crew of riggers.

The electrically operated release hook was kindly loaned by Eastern Rotorcraft Corp., Doylestown, Penna.

The fire test was performed at the Naval Damage Control Training Center, United States Naval Base, Philadelphia, Pa. The Training Center permitted us to use a fire basin normally used in teaching the art of extinguishing fuel fires. They also supplied the fuel, ignited the fire, and extinguished it quickly upon our signal. Their cooperation and permission to use their facilities are appreciated.

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COMPARISON OF ANALYSES FOR SHIPPING CONTAINER LID REMOVAL

Louis R. Fuka

ABSTRACT

Recently, there has been extensive concern about radioactive material shipping container lids remaining on the containers after a 9.14 m (30 foot) drop test. This is a problem that is much easier to counter in the design stage of the shipping container than to analyze after the container is constructed. However, there are many shipping containers in use that were designed and constructed before the engineering state of the art had advanced to where effective measures were taken in design to counter the lid removal problem of the containers. Different analyses dealing with lid integrity have been suspect without some corroboration by testing the actual container. An excessive amount of money would be used by either testing or scrapping each type of container which was designed without adequate attention to lid removal.

It is often possible to check the safety of existing shipping container lids by using approximate, relatively simple analysis methods mentioned in this paper. If safety is not assured by these simpler analytical techniques, it is possible to then use more sophisticated analytical techniques, testing and/or modification of the original shipping container design. The important point is that the design of shipping containers in use should be periodically checked by applying current analytical techniques to insure shipping container integrity.

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AN EXPERIMENTAL STUDY ON PUNCTURE RESISTANCE OF SPENT FUEL SHIPPING CASK BY DROP IMPACT TESTS

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A. Onodera

ABSTRACT

The work reported here was performed to develop the experimental equation which satisfies the criteria proposed by the Japan Atomic Energy Commission (J.A.E.C.) pertaining to the puncture resistance of the outer shell of lead shielded cask used for shipping radioactive materials.

A series of puncture test was conducted on the scale models of flat plate consisting of 3-layered members of frame work which represent the fundamental structure of prismatic cask, and also on the scale models of cylindrical cask. This series of test provided about forty test data on the puncture energy. Such affecting parameters as shell thickness, punch diameter, radius of punch edge etc. were determined by the non-linear regression method, and the experimental equation was obtained (Equation 6).

The equation thus obtained was compared with those obtained elsewhere[1] (Eq. (7) — Eq. (9)) and the equation[2] (12) recommended by the Advisory Material for the IAEA Regulations on the Safe Transport of Radioactive Materials. It was found that the equation (Eq. (6)) gives less shell thickness than the that recommended by IAEA (Eq. (12)).

INTRODUCTION

In 1968 the Japan Atomic Energy Commission proposed safety assessment criteria for fissile materials shipping containers. The criteria are principally based on the IAEA Regulations on the Safe Transport of Radioactive Materials (1967) and specify normal and accident conditions which the containers should meet. To further specify test conditions and to obtain experimental equations in compliance with criteria, the Japan Society of Mechanical Engineers performed a series of drop test[3]~[6] since 1968 under the auspices of Science and Technology Agency.

The work presented here was a part of the series of drop test, and was made particularly on the puncture resistance of spent fuel shipping cask when dropped.

The prototype cask was assumed to be of cylindrical type of 80 tons. The static puncture test was carried out on the cylindrical outer shell in the laboratory. The open-door drop tests were performed on the scale models of 1/4 to 1/2.
These tests were conducted to establish the equation on puncture resistance of the outer shell.

EXPERIMENTAL PROGRAM AND APPARATUS

As shown in Fig.1, the prototype cask was assumed to be of a cylindrical type of 80 tons consisted of stainless steel inner shell, lead shielding and low carbon steel outer shell. The scale models as shown in Table 1 were used for testing. For the flat plate models, the same structure as the cylindrical cask models was employed except for the thickness of outer shells. In the flat plate models, it was arranged so that the thickness could be changed in a range between 3 to 8 mm.

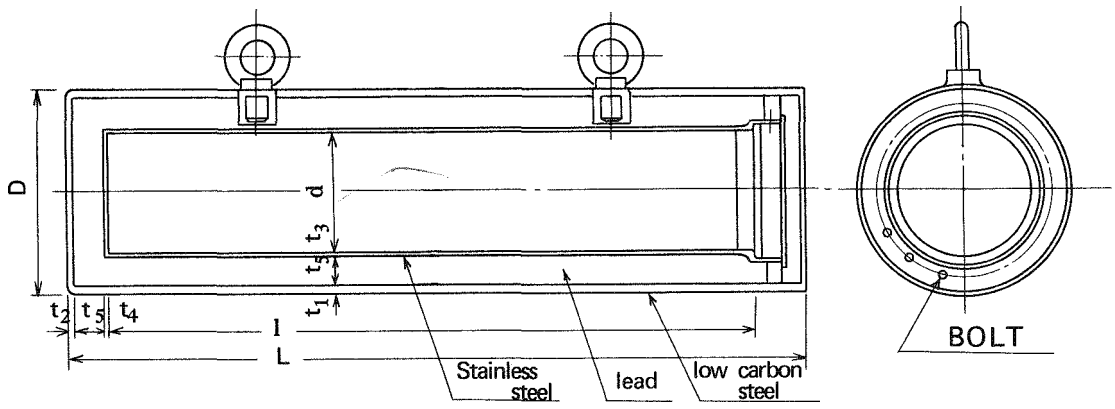


Fig. 1. Geometry of model cask

Table 1. Dimension of prototype cask and scale models

	Model No. 1	Model No. 2	Model No. 3	Model No. 4	Model No. 5	Prototype cask
scale	1/8.5	1/8	1/4	1/2.5	1/2	1
Weight (ton)	0.12	0.15	1.25	5	10	80
D (mm)	188	203	406	650	812	1,624
d (mm)	118	125	250	400	500	1,000
L (mm)	610	717.5	1,438	2,290	2,864	5,312
l (mm)	544	631	1,250	2,000	2,500	5,000
t_1(mm)	4.46	7	14	22	28	56
t_2(mm)	4.46	7	14	22	28	56
t_3(mm)	1.4	2	4	7	8	16
t_4(mm)	1.4	6	12	19	24	16
t_5(mm)	29.3	30	60	90	120	240

The test program consisted of the following three phases:

Phase 1 — Static puncture test: To clarify the influence of the shape of punch, punch diameter, etc. on the puncture energy, No.3 model was punched statically by a 400 ton Amsler Universal Testing Machine.

Phase 2 — Basic static and dynamic test: To clarify the influence of the punching speed on the puncture energy, Models Nos. 1 and 2 were punched statically and dynamically by a 100 ton Amsler Universal Testing Machine and an Impact Testing Machine.

Phase 3 — Free drop puncture test: To clarify the relationship between the thickness of cask outer shell and puncture energy, and the dimensional effect of cask model, Models Nos. 3 through 5 were dropped onto the target from various heights by means of 10 ton Dropping Tower. The target consisted of reinforced concrete block weighting 210 ton covered with two sheets of steel plate, 70 and 50 mm in thickness respectively. The punch was fixed on the steel plate.

RESULT AND CONSIDERATION

The major experimental results of phases 1,2 and 3 are shown in Appendix 1. We found:

1) Effect of the shape of punch on the puncture energy.

In this study, the influence of punch diameter, radius of the punch edge and shape of the top end of punch on the puncture energy was investigated through a static puncture test. The experimental results are illustrated in Fig.2. In Fig.2, when the punch diameter is over 40 mmϕ, there exists a linear relationship between the puncture energy and punch diameter on the log-log graph, if the radius of punch edge is constant. For the punch diameter from 27 mmϕ to 21 mmϕ, the influence of radius of the punch edge becomes smaller. For the punch diameter of 21 mmϕ, no difference is observed in the puncture energy. The puncture energy increases when smaller diameter punches are used. This is due to the fact that the average punch stress exceeds the average yield stress of punch material, resulting in the compressive deformation of punch in the axial direction, and increased the punch diameter.

The range where the sharpness of shape of the punch edge is noticeably effective on the puncture energy is

$$Pm \leq Y$$

Where

$$Pm = \frac{P}{S} = \frac{0.8\pi dt\ \sigma_B}{\frac{\pi}{4} d^2}$$

This can be arranged as follows:

$$t \leq \frac{d\ Y}{3.2\ \sigma_B} \qquad \cdots\cdots (1)$$

Where

p : Puncture force (kg)

s : Cross-sectional area of punch (mm^2)

d : Punch diameter (mm)

t : Outer shell thickness (mm)

σ_B : Ultimate tensile strength of outer shell (kg/mm^2)

Pm : Average stress of punch (=p/s) (kg/mm^2)

Y : Average yield stress of punch (kg/mm^2)

$$\left(= \frac{\text{Ultimate tensile strength + Yield strength}}{2} \right)$$

It is assumed that the possibility of puncture increases as the edge of punch becomes sharper below certain thickness of outer shell regardless of the weight of cask. Also at a region where the thickness of outer shell exceeds a certain value, the puncture energy increases as punch diameter increases.

Static puncture tests were conducted to find the degree of safety when the cask drops at a certain slanted angle against the horizontal. Punches having the top end angles of 60° and 90° as well as semi-spherical end punches were employed for these tests. It was found that in the case of a punch having a top end angle of 60°, the puncture energy becomes 25% lower as compared with that of the standard shape punch of the same diameter; in the cases of a punch having a top end angle of 90° or of semi-spherical end punch, the puncture energy becomes 10% and 35% larger respectively as compared with the prescribed shape punch of the same diameter, resulting in lower puncture possibilities.

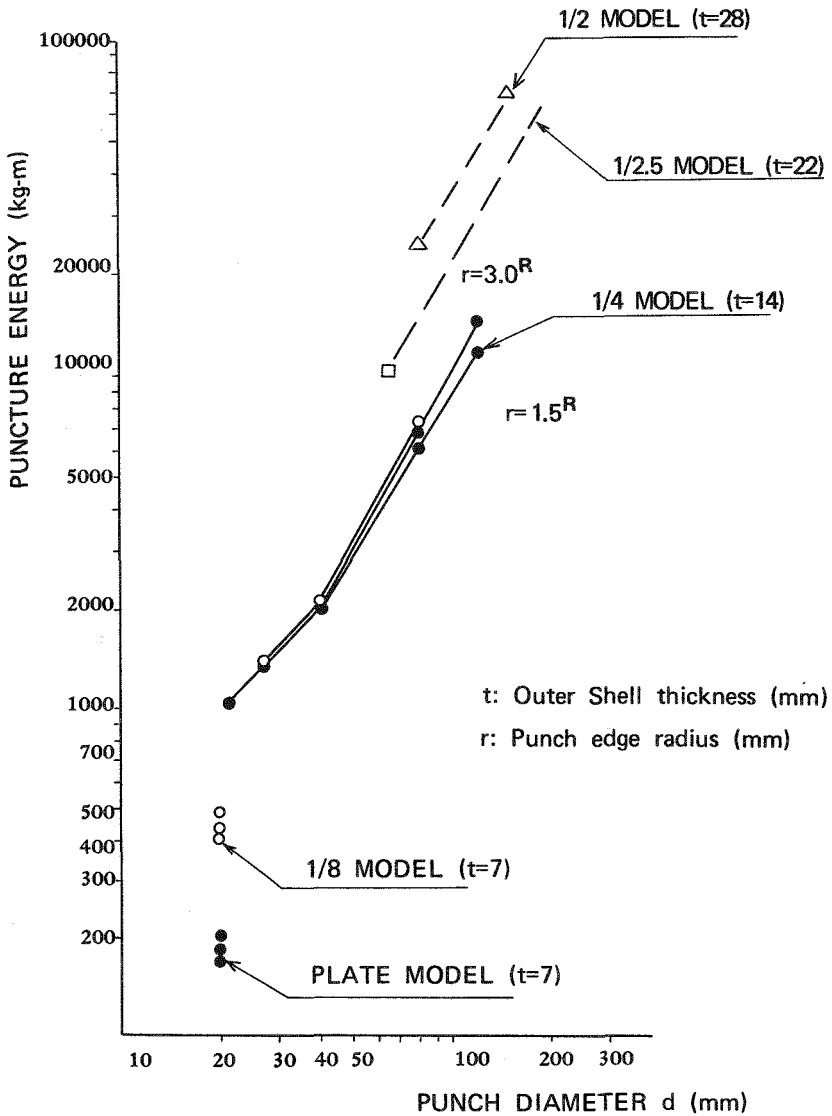


Fig. 2. The effect of punch diameter on puncture energy

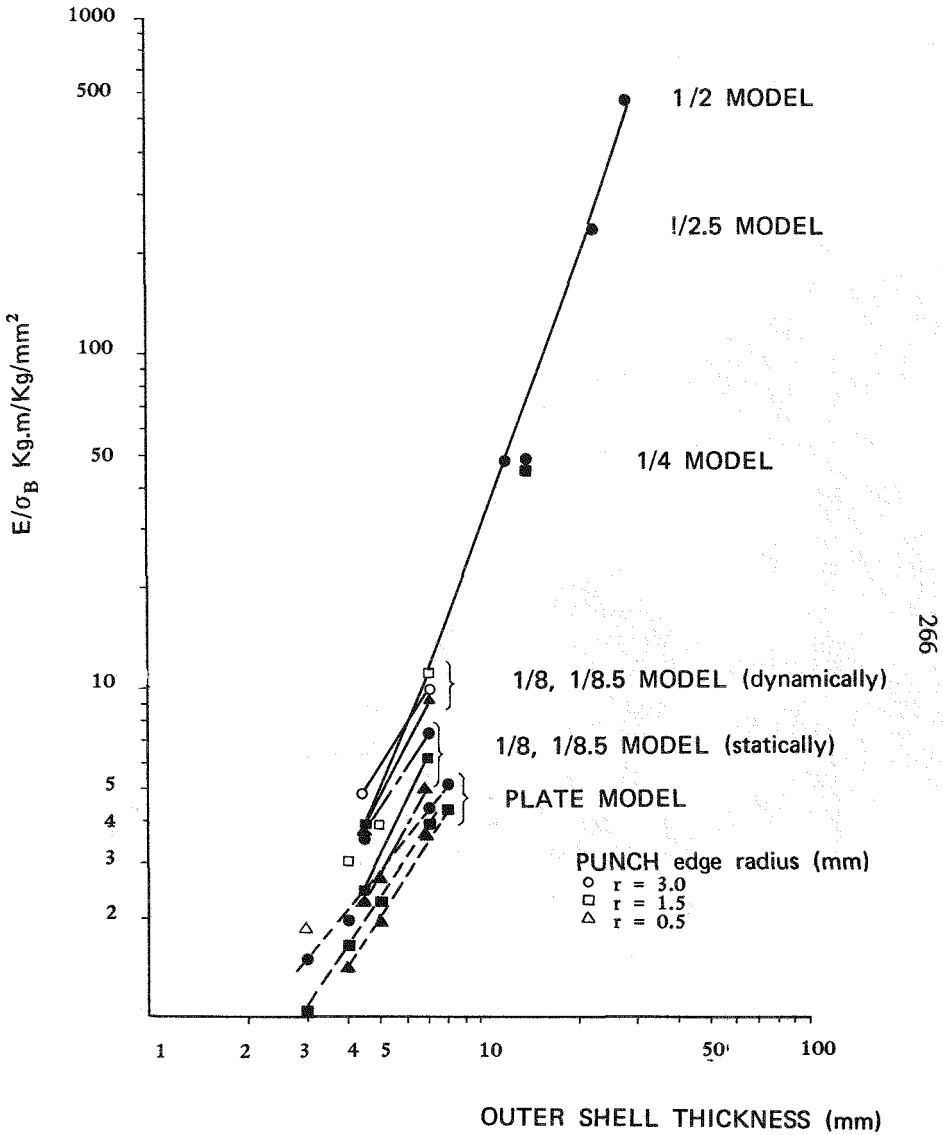


Fig. 3. The effect of outer shell thickness on puncture energy

This indicates that when the cask drops horizontally onto the flat end punch, puncture of outer shell will most possibly take place.

2) Effect of the cask shape on the puncture energy

As shown in Fig.3, there exists a linear relationship on a log-log graph between the thickness of cask outer shell and the puncture energy, with the radius of the punch edge being constant. Even if the thickness of outer shell is same, the puncture energy is different between the cylindrical cask model and the flat plate model. This indicates that the cylindrical model represents a lower puncture possibility. Another experiment was made by employing the cask scale models of 1/8.5 to 1/2 with the punch diameters scaled in the same ratios.
As is clear from Fig.3, each result is plotted on the same straight line, and similar relationships are established in the puncture phenomena of cask models.

3) Relationship between the static puncture energy and the dynamic puncture energy

To clarify the relationship between the static and dynamic puncture energy, Models No.1 and 2 and the flat plate model were punched statically and dynamically. The results of this experiment is shown in Fig.4. From Fig.4, it is noticed that there exisits a linear relationship between the static puncture energy (Estatic: kgm) and the dynamic puncture energy (E dynamic: kgm). The following equation is obtained by employing the method of least squares:

$$E_{dynamic} = 1.63\ E_{static} + 5.19 \quad \cdots\cdots (2)$$

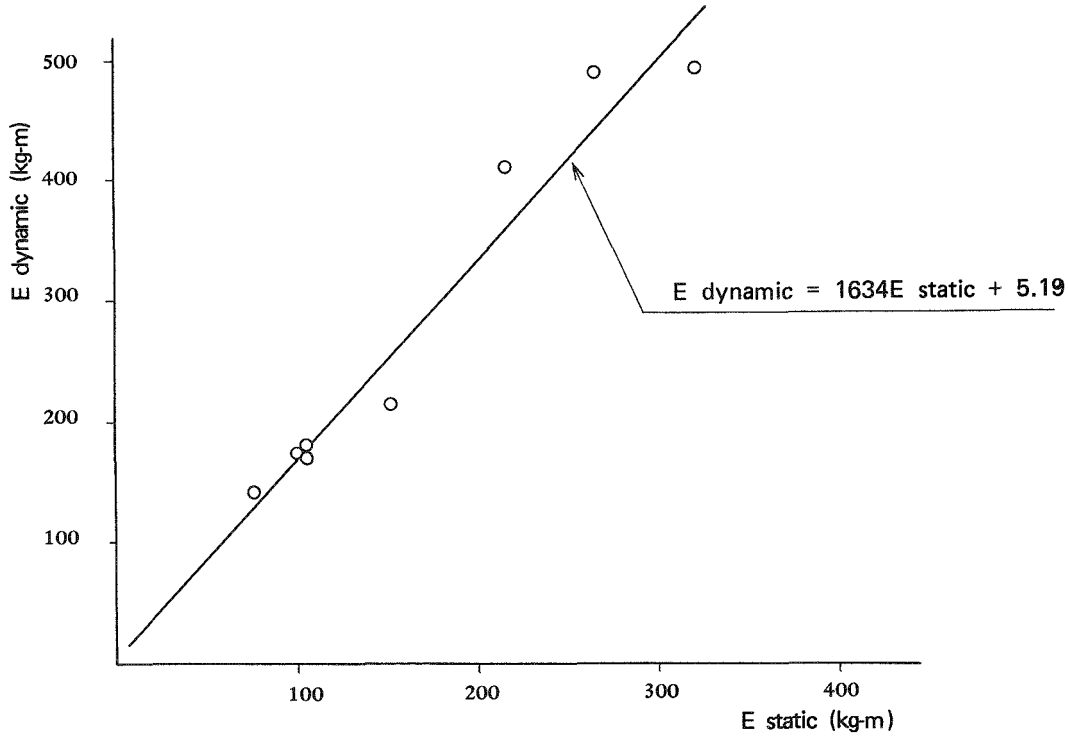


Fig. 4 The corelation between static puncture energy and dynamic puncture energy.

Since the constant term is within the experimental tolerance, it can be neglected. Thus the following equation is derived:

$$\frac{E\ \text{dynamic}}{E\ \text{static}} = 1.65 \qquad \cdots\cdots (2)'$$

4) *Derivation of the experimental equation*

As shown in Fig.2, when the thickness of cask outer shell is fixed in 1/4 model, the puncture energy increases with a certain ratio together with the increase in the radius of the punch edge (r mm), along with the increase in punch diameter(d mmϕ). As shown in Fig.3, when the punch diameter is fixed, the puncture energy decreases together with the increase of the radius of punch edge, along with the increase in the thickness of cask outer shell (t mm). Based on these data, the puncture energy (E kgm) can be expressed as follows:

$$E = At^{\delta+\epsilon r} d^{\zeta+\eta r} \quad (\epsilon<0, \eta>0) \qquad \cdots\cdots (3)$$

Where

$\delta, \epsilon, \zeta, \eta$: constants

When comparing the cases of flat plate model and cylindrical model having the same outer shell thickness, the puncture energy is larger in the case of cylindrical model.

By expressing the effect of the curvature of cask outer shell (D mmϕ) on the puncture energy as a ratio between the cask outer shell thickness and the cask diameter (t/D) and a ratio between the punch diameter and the cask diameter (d/D), and by expressing the effect of the shape of punch as a ratio between the radius of the punch edge and the punch diameter (r/d), Equation (3) can be expressed as follows:

$$E = \left(\alpha + \beta\frac{t}{D} + \gamma\frac{d}{D} + \theta\frac{r}{d}\right) t^{\delta+\epsilon r} d^{\zeta+\eta r} \quad (\epsilon<o, \eta>o) \cdots\cdots (4)$$

Where

$\alpha, \beta, \gamma, \theta, \delta, \epsilon, \zeta, \eta$: constants

When taking into consideration of the effect of the materials for cask outer shell (σ_B kg/mm^2) by the results shown in Ref. (1), Equation (4) can be expressed as follows:

$$E/\sigma_B = \left(\alpha+\beta\frac{t}{D}+\gamma\frac{d}{D}+\theta\frac{r}{d}\right) t^{\delta+\epsilon r} d^{\zeta+\eta r} \qquad (\epsilon<o, \eta>o) \qquad \text{..... (5)}$$

The results of static puncture tests as converted into dynamic data by using Equation (2) and the results of dynamic puncture tests are shown in Appendix 1. The unknown quantities, namely 8 constants in the equation can be calculated as follows through the non-linear regression method with the data shown in Appendix 1:

$$E/\sigma_B = \left(0.0030 + 0.047\frac{t}{D} + 0.002\frac{d}{D} + 0.006\frac{r}{d}\right) t^{1.585-0.11r} d^{1.465+0.077r} \qquad \text{..... (6)}$$

5) Comparison and evaluation of the experimental equation

A comparison between the experimental equation and the equations cited in other references (1), (7) is of interest. The equation in Ref. (1) is quoted here after converting into metric system:

In the case of prismatic cask:

$$E/\sigma_B = 2.4 \times 10^{-3}\ d^{1.6}\ t^{1.4} \qquad \text{..... (7)}$$

or

$$E/\sigma_B = 1.3 \times 10^{-1}\left(\frac{d}{12.5}\right)^{1.6} t^{1.4} \qquad \text{..... (8)}$$

In the case of cylindrical cask:

$$E/\sigma_B = 10^{-3}\ [\ (10.5d - 0.0554D)\ t^2 - 11.8t^3\ ] \qquad \text{..... (9)}$$

The calculation results by Equation (6) and equations (8) through (9) are shown in Appendix 1 and 2. Appendix 1 shows the experimental results of the present work, while Appendix 2 shows the experimental results by Ref. (1) and (7). In Appendix 2, when Equation (6) was used,

the radius of the punch edge was assumed to be 0.1 mm for calculation. Table 2 illustrates the average errors in calculated values and experimental values. From Table 2 it is clear that Equation (6) relatively well represents not only the experimental results of the present work but also those of Ref. (1) and (7).

Table 2 Error distribution calculated by various equations

(%)

Type / Number	Results		Results in Ref. 1,7		Total
	Plate	Cylinder	Plate	Cylinder	
	n=1~17	n=18~43	n=44~81	n=82~95	n=1~95
$\frac{1}{b-a+1}\sum_{n=a}^{b}\left\lvert\frac{E_6-E_{dyn}}{E_{dyn}}\right\rvert$	4.2	12.3	8.1	22.3	10.6
$\frac{1}{b-a+1}\sum_{n=a}^{b}\left\lvert\frac{E_8-E_{dyn}}{E_{dyn}}\right\rvert$	32.9	54.7	8.4	13.4	24.0
$\frac{1}{b-a+1}\sum_{n=a}^{b}\left\lvert\frac{E_9-E_{dyn}}{E_{dyn}}\right\rvert$	—	40.2	—	1.3	26.6

The energy of impact on the cask when dropped can be expressed as follows, taking the hypothetical accident conditions by the AEC into consideration:

$$E = 1000\ WH \qquad \text{..... (10)}$$

Where

H: Height from which the cask is dropped

(=1.012m)

W: Weight of the cask (ton)

By rearranging Equations (6) and (8) by substituting Equation (10) and "d = 152.4 mmϕ".

$$W/\sigma_B=10^{-3}\left(0.0030+0.047\frac{t}{D}+0.006\frac{r}{152.4}+0.002\frac{152.4}{D}\right)t^{1.585-0.11r}152.4^{1.465+0.07r} \qquad \text{..... (11)}$$

$$W/\sigma_B=0.0070t^{1.4} \quad \text{or} \quad t = 34.6\left(\frac{W}{\sigma_B}\right)^{0.71} \qquad \text{..... (12)}$$

For better presentation of Equation (11), the outer diameter of prototype cask in Table 1 was introduced, and calculation was made by taking the radius of punch edge as a parameter. The results are shown in Fig. 5.

From Fig. 5 it is clear that, possibility of puncture becomes higher as the end of punch becomes sharper when the outer shell thickness is up to about 40 mm; when the outer shell thickness exceeds this level, possibility of puncture becomes higher as the radius of punch edge becomes larger. It is considered that when the thickness of cask outer shell becomes larger than the value shown in Equation (1), the compression speed of punch, in the case of static puncture, is low, resulting in even deformation over the total punch and an increase in punch diameter. Thus possibility of puncture becomes low. However, in the case of dynamic puncture, only the top end of punch tends to expand toward the radial direction in the early stage of impact. Thus those having a large radius of punch edge become the same shape as that of small punches having smaller radius of punch edge in the early stage of impact, which invites a higher possibility of puncture. It is considered that those having a smaller radius of punch edge have a lower possibility of puncture as they are changed into a trumpet shape.

From Fig. 5 it is clear that the thickness of outer shell of the prototype cask should be larger than those indicated by thick solid lines. Fig. 6 shows the results of calculation made in the similar way as for Fig. 5 by using Equation (11) with the cask outer diameter as a parameter. This figure shows the minimum thickness requirements versus the weight of prismatic and cylindrical casks. Equation (12) is also illustrated for reference.

From Fig. 6 it should be noted that Equation (11) as obtained in the present work signifies that the outer shell thickness can be made smaller than those specified by Equation (12) under the normal weight of the casks. However, it also signifies that when the casks are very large or small, the thickness should be made larger than the calculated values by Equation (12). Since calculation by Equation (11) is complicated, use of Fig.6 is recommended for quick calculation of the necessary thickness of outer shells of prismatic and cylindrical casks.

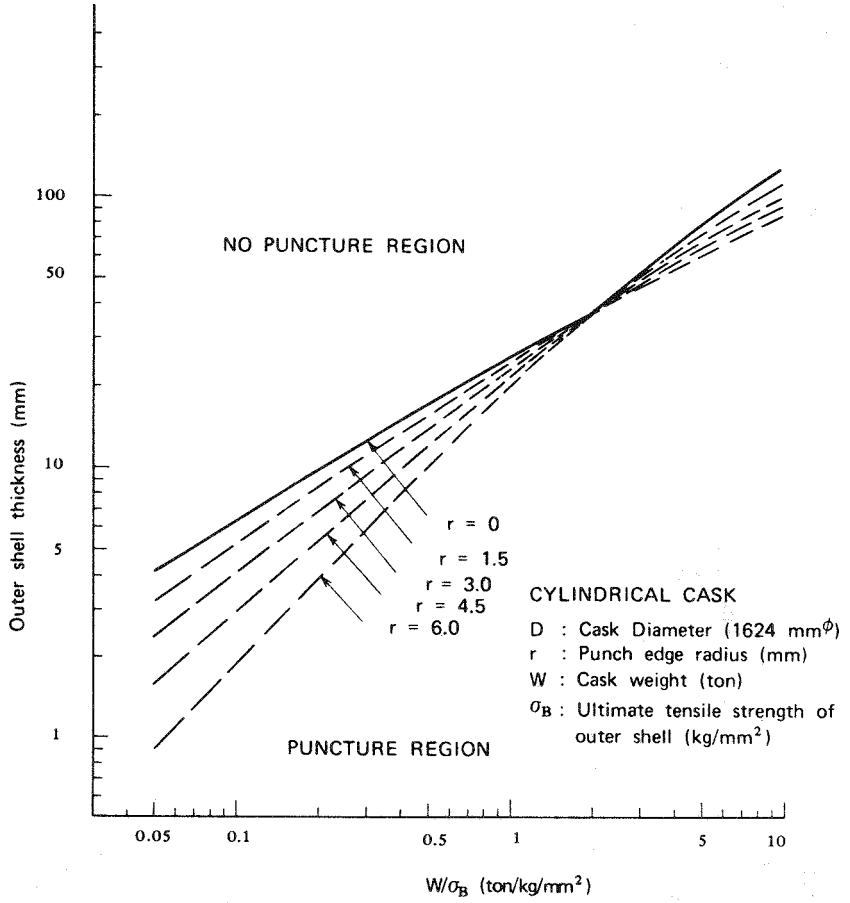


Fig. 5 The effect of punch edge radius on puncture energy.

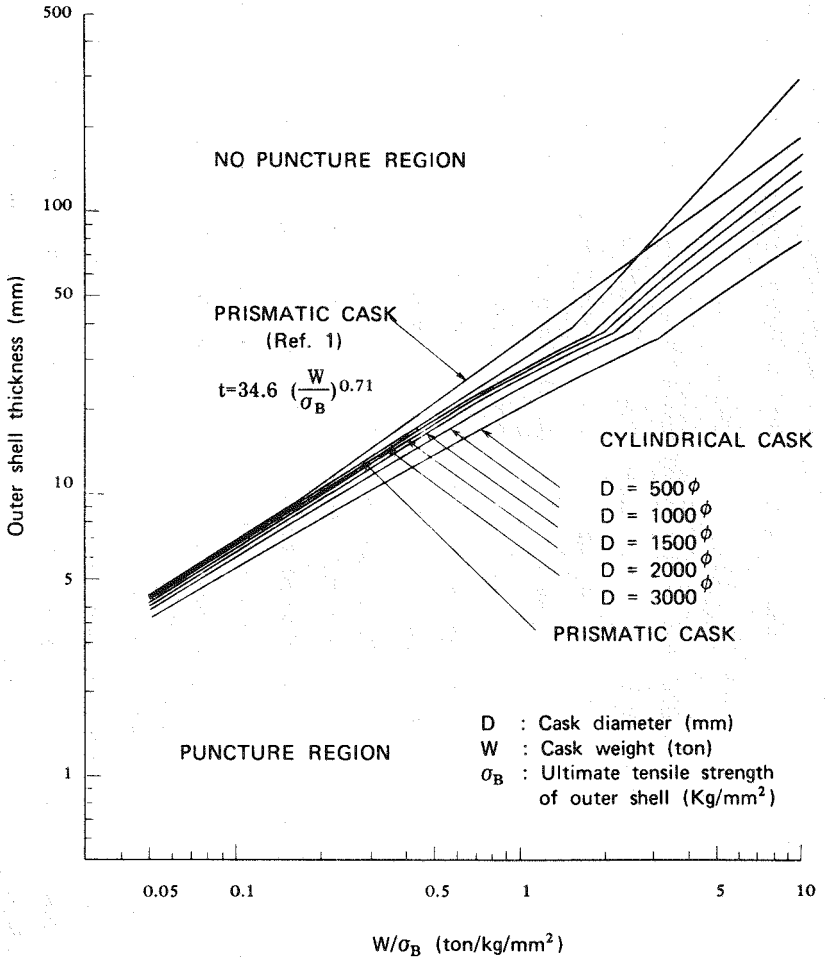


Fig. 6. Outer shell thickness versus weight of cask to cause puncture.

CONCLUSION

1. Equation (6) is an experimental equation to give the puncture energy on the outer shell of lead shielded cask. When the hypothetical accident conditons are incorporated into Equation (6), the thickness necessary for the cask outer shell becomes as shown in Fig. 6. It is recommended to use Fig.6 to determine the outer shell thickness of prismatic and cylindrical lead shielded casks.

2. The effect of the radius of the punch edge on the puncture energy should not be neglected. Possibility of puncture becomes higher as the punch edge becomes sharper when the cask outer shell thickness is below about 40 mm. When the thickness becomes larger, possibility of puncture becomes larger as the radius of punch edge becomes larger.

3. If the geometric conditions are kept constant, it can be considered that nearly a similar relationship can be established between the outer shell thickness and the puncture energy of the cylindrical cask.

4. To puncture the outer shell, more energy is required for cylindrical cask than for prismatic cask of the same weight and shell thickness.

5. Under the hypothetical conditions, the cask outer shell has the highest possibility of puncture when the center of gravity of cask and the center of punch are aligned, and the cask is dropped in the horizontal attitude.

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APPENDIX 1 The results of puncture tests.

No.	D mm	d mm	r mm	t mm	σ_B Kg/mm²	Estatic Kg-m	Edy-namic Kg-m	E' Kg-m	E_6 Kg-m	E_6-Edyn / Edyn x100%	E_8 Kg-m	E_8-Edyn / Edyn x100%	E_9 Kg-m	E_9-Edyn / Edyn x100%
1	〃	20.3	0.5	3	47.7		90		74.85	-16.83	61.53	-31.63	–	–
2	〃	〃	1.5	〃	〃	50		86.89	90.17	3.77	〃	-29.19	–	–
3	〃	〃	3.0	〃	〃	71.4		121.86	117.8	- 3.33	〃	-49.51	–	–
4	〃	20.3	0.5	4	47.7	67.7		115.81	116.08	0.23	92.04	-20.52	–	–
5	〃	〃	1.5	〃	〃	78.3		133.13	135.18	1.54	〃	-30.86	–	–
6	〃	〃	〃	〃	〃		140		〃	- 3.44	〃	-34.26	–	–
7	〃	〃	3.0	〃	〃	93.9		158.62	167.81	5.79	〃	-41.97	–	–
8	〃	20.3	0.5	5	47.7	93.1		157.32	163.16	3.71	125.79	-20.04	–	–
9	〃	〃	1.5	〃	〃	10.7		180.03	185.05	2.79	〃	-30.13	–	–
10	〃	〃	〃	〃	〃		182		〃	1.68	〃	-30.88	–	–
11	〃	〃	3.0	〃	〃	12.5		209.44	220.81	5.43	〃	-39.94	–	–
12	〃	〃	0.5	7	〃	171		284.61	272.61	- 4.22	201.48	-29.21	–	–
13	〃	〃	1.5	〃	〃	185		307.48	297.13	- 3.37	〃	-34.47	–	–
14	〃	〃	3.0	〃	〃	202		335.26	334.01	- 0.37	〃	-39.10	–	–
15	〃	20.3	0.5	8	〃	206		341.8	334.2	- 2.22	242.9	-28.94	–	–
16	〃	〃	1 5	〃	〃	206		341.8	358.56	4.91	〃	-28.94	–	–
17	〃	〃	3.0	〃	〃	246		407.16	393.64	- 3.32	〃	-40.34	–	–
18	188	20.3	0.5	4.46	44.3	101		170.23	179.63	5.52	99.55	-41.52	132.27	-22.30
19	〃	〃	0.5	〃	〃		173		〃	3.83	〃	-42.46	〃	-25.54
20	〃	〃	1.5	〃	〃	108		181.66	201.66	10.98	〃	-45.2	〃	-27.19
21	〃	〃	〃	〃	〃		168		〃	20.0	〃	-40.74	〃	-21.27
22	〃	〃	3.0	〃	〃	155		258.46	238.13	- 7.87	〃	-61.48	〃	-48.82
23	〃	〃	〃	〃	〃		215		〃	10.76	〃	-53.7	〃	-38.48
24	203	20.3	0.5	7	44.3	217		359.77	396.18	10.12	187.12	-47.99	258.97	-28.02
25	〃	〃	〃	〃	〃		408		〃	- 2.9	〃	-54.14	〃	-36.53
26	〃	〃	1.5	〃	〃	268		443.11	419.1	- 5.42	〃	-57.77	〃	-41.56
27	〃	〃	〃	〃	〃		486		〃	-13.77	〃	-61.5	〃	-46.71
28	〃	〃	3.0	〃	〃	324		534.61	453.56	-15.16	〃	-65.0	〃	-51.56
29	〃	〃	〃	〃	〃		489		〃	- 7.25	〃	-61.73	〃	-47.04
30	406	21.2	1.5	14	44.3	1042		1707.84	1166.53	-31.7	529.31	-69.01	496.18	-70.95
31	〃	〃	3.0	〃	〃	1056		1730.72	1120.12	-35.28	〃	-69.42	〃	-71.33
32	406	27.1	1.5	14	44.3	1406		2302.62	1698.87	-26.22	784.01	-65.95	841.00	-63.48
33	〃	〃	3.0	〃	〃	1408		2305.89	1652.51	-28.34	〃	-66.0	〃	-63.53
34	406	40.6	3.0	12	44.3		2120		2521.95	18.96	1206.39	-43.09	1672.67	-21.10
35	406	40.6	1.5	14	44.3	2017		3001.01	3189.73	6.29	1496.97	-50.12	2071.78	-30.96
36	〃	〃	3.0	〃	〃	2150		3518.33	3188.08	- 9.39	〃	-57.45	〃	-41.11
37	406	81.2	1.5	〃	〃	6080		9140.03	9684.54	- 2.57	4537.95	-54.35	5773.26	-41.92
38	〃		3.0	〃	〃	6743		11023.39	10268.78	- 6.85	〃	-58.83	〃	-47.63
39	〃	〃	〃	12	〃		8750		9816.87	12.19	〃	-48.14	4392.13	-49.80
40	406	121.2	1.5	14	〃	11495		18788.25	18912.34	0.66	8681.72	-53.80	9420.04	-49.86
41	〃	〃	3.0	〃	〃	14533		23752.41	20861.31	-12.17	〃	-63.45	〃	-60.34
42	650	65.0	3.0	22	43.5		10300		17685.47	13.45	5876.65	-42.95	8145.58	-20.92
43	812	81.2	3.0	28	47.4		24500		24866.29	1.5	12813.75	-47.7	17734.08	-27.62

APPENDIX 2 The results of drop tests in reference (1), (7).

No.	D mm	d mm	r mm	t mm	σ_B Kg/mm²	Edynamic Kg- m	E_6 Kg-m	$\frac{E_6 - Edyn}{Edyn} \times 100\%$	E_8 Kg-m	$\frac{E_8 - Edyn}{Edyn} \times 100\%$	E_9 Kg-m	$\frac{E_9 - Edyn}{Edyn} \times 100\%$
44	〃	12.7	0.1	1.98	59.85	19.18	22.82	18.98	20.39	6.31	–	–
45	〃	〃	〃	3.05	32.68	23.13	24.58	6.27	20.38	-11.89	–	–
46	〃	〃	〃	0.97	30.25	3.88	3.76	- 3.10	3.8	- 2.06	–	–
47	〃	〃	〃	1.93	65.98	21.82	24.15	10.68	21.68	- 0.64	–	–
48	〃	〃	〃	2.72	38.24	20.05	24.05	19.95	20.34	1.25	–	–
49	〃	〃	〃	0.79	30.73	3.21	2.76	-14.02	2.89	- 9.97	–	–
50	〃	〃	〃	1.6	31.96	8.87	8.72	- 1.69	8.08	- 8.91	–	–
51	〃	〃	〃	1.88	61.35	20.23	21.53	6.41	19.41	- 4.05	–	–
52	〃	〃	〃	1.98	46.55	20.78	17.75	-14.58	15.86	-23.68	–	–
53	〃	15.24	0.1	1.98	59.85	28.98	29.78	2.76	27.3	- 5.80	–	–
54	〃	〃	〃	0.79	60.87	7.33	7.14	- 2.59	7.67	4.64	–	–
55	〃	〃	〃	3.05	32.68	29.12	32.07	10.13	27.29	- 6.28	–	–
56	〃	〃	〃	0.97	30.25	5.10	4.9	- 3.92	5.08	- 0.39	–	–
57	〃	〃	〃	0.69	66.53	6.8	6.3	- 7.35	6.93	1.91	–	–
58	〃	〃	〃	2.72	38.24	28.85	31.29	8.46	27.15	- 5.89	–	–
59	〃	〃	〃	0.79	30.73	3.99	3.6	- 9.77	3.87	- 3.01	–	–
60	〃	〃	〃	1.6	31.96	11.19	11.38	1.70	10.82	- 3.31	–	–
61	〃	7.91	0.1	1.26	29.46	2.97	2.76	- 7.07	2.49	-16.16	–	–
62	〃	〃	〃	1.83	33.61	5.43	5.68	4.60	4.8	-11.60	–	–
63	〃	〃	〃	2.97	33.12	12.25	12.01	- 1.96	9.34	-23.76	–	–
64	〃	9.53	0.1	1.26	29.46	3.82	3.61	- 5.50	3.35	-12.30		
65	〃	〃	〃	1.83	33.61	7.12	7.44	4.49	6.46	- 9.27		-
66	〃	〃	〃	2.97	33.12	15.47	15.72	1.62	12.56	-18.81	–	–
67	〃	11.09	0.1	1.83	33.61	8.63	9.27	7.42	8.24	- 4.52	–	–
68	〃	11.43	0.1	1.83	33.61	9.37	9.69	3.42	8.65	- 7.68	–	–
69	〃	12.7	0.1	1.3	29.46	6.00	5.76	- 4.00	5.53	- 7.83	–	–
70	〃	〃	〃	1.51	32.12	8.14	8.25	1.12	7.72	- 5.16	–	–
71	〃	〃	〃	1.83	33.61	10.85	11.35	4.15	10.24	- 5.62	–	–
72	〃	〃	〃	2.79	42.26	21.56	27.67	28.34	23.3	8.07	–	–
73	〃	〃	〃	3.01	33.12	24.09	24.32	0.95	20.22	-16.06	–	–
74	〃	〃	〃	3.45	43.59	29.84	39.85	33.55	32.34	8.38	–	–
75	〃	14.02	0.1	1.83	33.61	12.73	13.06	2.59	11.99	- 5.81	–	–
76	〃	15.88	0.1	1.26	29.46	9.02	7.61	-15.63	7.59	-15.85	–	–
77	〃	〃	〃	1.83	33.61	16.20	15.65	- 3.40	14.63	- 9.69		
78	〃	12.7	0.1	0.79	59.2	5.62	5.28	- 6.05	5.54	- 1.42	-	–
79	〃	〃	〃	1.27	〃	9.53	11.22	17.73	10.82	13.54	–	–
80	〃	〃	〃	1.59	〃	15.27	15.94	4.39	14.8	- 3.08	–	–
81	〃	〃	〃	1.88	〃	22.43	20.79	- 7.31	18.74	-16.45	–	–
82	106.43	10.16	0.1	1.91	36.07	10.09	12.46	23.49	8.17	19.03	10.30	2.08
83	118.11	〃	〃	〃	〃	9.98	12.16	21.84	〃	18.14	10.21	2.30
84	132.08	〃	〃	〃	〃	9.93	11.86	19.44	〃	17.72	10.11	1.81
85	156.46	〃	〃	〃	〃	9.81	11.48	17.02	〃	16.72	9.93	1.22
86	106.43	12.7	〃	〃	〃	13.95	17.45	25.09	11.68	-16.27	13.81	- 1.00
87	118.36	〃	〃	〃	〃	13.82	16.99	22.93	〃	-15.48	13.72	- 0.72
88	130.81	〃	〃	〃	〃	13.78	16.61	20.54	〃	-15.24	13.63	- 1.09
89	156.72	〃	〃	〃	〃	13.58	16.01	17.89	〃	-13.99	13.44	- 1.03
90	168.91	〃	〃	〃	〃	13.49	15.8	17.12	〃	-13.42	13.35	- 1.04
91	182.37	〃	〃	〃	〃	13.41	15.7	17.08	〃	-12.90	13.25	- 1.19
92	118.11	15.24	0.1	1.91	36.07	17.04	22.42	31.57	15.63	- 8.27	17.23	1.12
93	132.59	〃	〃	〃	〃	16.93	21.82	28.88	〃	- 7.68	17.12	1.12
94	168.91	〃	〃	〃	〃	16.69	20.77	24.45	〃	- 6.35	16.86	1.02
95	175.36	〃	〃	〃	〃	16.61	20.63	24.20	〃	- 5.90	16.81	1.20

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INVESTIGATION OF DOT SPECIFICATION 2R CONTAINERS TO MEET SPECIAL FORM CERTIFICATION

R. W. Andrae

ABSTRACT

A test program was conducted to investigate the qualifications of 2R, pipe nipple-cap type containers as a special form encapsulation. The testing included a 30 foot (9.14m) drop, impact with a 3 pound (1.36 kg) weight, heating to 1475°F (1075 K) and quenching in water. The results indicate that the use of the 2R, pipe nipple-cap container as a special form encapsulation should probably be restricted to the shipment of solids and, then, only under the conditions that soluble solids are adequately protected by an inner container.

INTRODUCTION

Radioactive materials can be packaged for shipment by common carrier much more conveniently when the material meets special form criteria as defined by the DOT Specification. Special form can be attained either by having the material itself in massive solid form, or by virtue of its encapsulation. This report is concerned with the latter, encapsulation; and, in particular with pipe nipple-cap type containment meeting the DOT specification 2R.

It has been our judgment at LASL that the use of the 2R, pipe nipple-cap container as a special form encapsulation should be restricted to the shipment of insoluble solids and not liquids or gases because of the uncertainty of the leak tightness of the threaded joints. Even such limited usage, however, must soon be supported by appropriate documentation affirming that the special form criteria have been met. Because the DOT specifications only stipulate the performance specifications for qualification and not the design details of a container to provide compliance, the continued use of 2R containers as a special form encapsulation is dependent upon adequate test data showing their re-

sponse to conditions of interest.

DISCUSSION

As a practical expediency, the test specimens in this study did not contain any radioactive substance. The determination as to whether or not test-incurred damage meant the possible release of radioactivity in an actual case was a matter of interpretation of the test data. Obviously, the loss of a cap or the fracture of the container walls would represent very serious damage, but this would be easily detected by a physical examination of the tested article. Just a loosening of the cap, on the otherhand, was a more subtle form of damage that could be serious and required further evaluation. Three such measurements, each involving a different medium, were used for this purpose at various points in the test program. These were (1) leak rates for helium, (2) leak rates for air and (3) the amount of water leaked in during a 24 hour period. The helium leak test is by far the most sensitive techniques used.

SPECIMENS

The test specimens were made from off-the-shelf, nominal 4 inch sch 40 (101.6 mm) pipe nipples and caps. A single nipple was cut in half to form 2 test specimens. The cut end of each specimen was welded shut with a flat plate of the same material. The caps were all 304 stainless steel and the body was either 304 stainless steel or black iron. Twelve specimens of each material combination were used making a total of 24 specimens. The all stainless steel specimens were numbered 1 - 12 and, those with the black iron bodies 21 - 32.

APPARATUS

The special tests prescribed in the DOT specification Par. 173.398 required the following test set-ups: (1) a tower to drop the specimen through a distance of 30 ft. (9.14 m) onto an essentially unyielding surface; (2) a device for impacting the specimen with a one inch (25.4 mm) cylindrical steel rod weighing 3 lbs (1.36 kg) from

a height of 40 in. (1.02 m); and (3) a furnace for heating the specimen to 1475°F (1075 K) together with a way of immediately quenching it in water. Fig. 1 shows the lower portion of the drop tower, and the way in which the specimen was suspended from the release mechanism. The essentially unyielding surface consisted of thick steel plates resting on a concrete slab. The attitude of the specimen at impact was determined by its attitude at the instant of release because no rotation occurred during its descent (Fig. 2). The percussion test fixture is shown in Fig. 3. The specimen rested on a ½ inch (12.7 mm) lead plate (4.2 Vickers hardness). The furnace (Fig. 4) had a clam shell arrangement whereby the heated specimen could be droped directly into a tank of water.

TESTING

The basic test philosophy was not only to find out the effects of the prescribed sequence of tests on the given container configuration, but also to gain knowledge that might be useful in the event a better container is required. A summary of the test program used is shown in Fig. 5. The decision on which containers were to be tested was made as the program progressed.

The initial assembly was done in as uniform a manner as possible in order to have a common starting point for each container. The threads were luted with a high temperature, anti-sieze, sealing compound* and the cap was installed by torquing it to 200 ft. lbs (271.2 J). No attempt was made to clean-up the threads which were especially rough in some of the caps. As will be shown in Table I, in spite of the care taken during assembly, the leak tightness of the containers was far from uniform probably due to the lack of uniformity at the thread interfaces. It was first thought that a low initial leak rate was an important factor in determining a container's ultimate leak rate at the conclusion of the testing sequence. It will be shown later, however, that the phenomenon encountered during the quench

*Fel-Pro-C5A

following the heating to 1475°F (1075 K) effectively nullified any gain realized from a low initial leak rate.

The helium leak test was made by internally pressurizing the container to 30 psi (0.207 MPa) and reading the leak rate directly or, in cases where the leak rate was higher than the scale on the detector, by calculating the leak rate as the product of the pressure drop (atmos.) and the container volume (cc) divided by the time (sec.). The latter approach for finding leak rates was also used for the air measurement which began with an internal pressurization of 50 psi (0.345 MPa). All the leak rates shown in the tables have been normalized to the condition of helium at 30 psi (0.207 MPa).

A physical examination of the dropped specimens showed a slight flat spot on the cap (Fig. 6) and a hand-loose cap on all the containers with stainless steel bodies. Although the cap could be turned through a small angle, it could not be removed easily indicating an out-of-roundness condition had occurred. Table I shows the effects of the drop on both the leak rates and the amount of water taken on during a 24 hour immersion. The leak rates were higher after the test as indicated by the air check, but they still do not represent a high enough value for water leakage. A possible clue to account for the better performance of the containers with black iron bodies is that the combination of stainless steel and black iron assembled more smoothly during the torquing operation.

The percussion test caused little or no damage. No water immersion check was made, but the leak rates shown in Table II are essentially unchanged as a result of this test. The flat, circular end of the steel rod impacted the container on the body just below the cap.

It took approximately 30 minutes to first bring the furnace to 1475°F (1075 K) and then hold at this temperature for 10 minutes as required by the criteria. After this period, the clam shell door of the furnace was opened and the specimen released into the can of water. This quench phase proved to be a very disasterous experience for these containers. Apparently the body having a larger exposed area but a smaller mass than the cap cooled more rapidly thereby creating a gap

to open in the threaded joint for water to pour in. Ultimately, of course, when the entire container reached the same temperature, this gap would close again. To support this theory, 4 specimens were air cooled after heating before they were immersed in water. The results of these tests are shown in Table III. Experiments II, III and IV were attempts to keep the body from becoming cooler than the cap during quench, but their results are inconclusive indicating that this condition was not attained for the entire quenching period.

SUMMARY AND CONCLUSIONS

The special tests prescribed in Par. 173.398 of the DOT Specification were strictly adhered to in testing a particular 2R, pipe nipple-cap container configuration to find out whether or not this container met the special form encapsulation criteria. On the basis of the test data the following conclusion was reached:

> A 2R, pipe nipple-cap container made with a stainless steel cap and having either a stainless steel or black iron body can be considered a special form encapsulation providing it is used for the shipment of solids only, and, further, that if these solids are water soluble, they are adequately protected by an inner container. It should be noted that the problem of criticality can be avoided if the shipment is confined to a single container whose inner diameter is less than 5 in. (127 mm).

Table 1. Effect of Drop Test

Specimen Number	Leak Rate Before Test		Leak Rate After Test	Water Leaked over 24 Hrs.	
	g/cc		g/cc	g	%
	He	Air	Air		
1	1.1	0.87	270.	23.4	1.8
3	5.2×10^{-6}	0.50	46.	27.3	2.1
4	0.61	-	90.	11.7	0.9
7	0.19	-	300.	-	-
8	0.066	0.12	640.	19.5	1.5
10	7.6	0.26	914.	24.7	1.9
11	0.19	0.19	61.	6.5	0.5
12	8.0	7.3	256.	6.5	0.5
21	$<1.8 \times 10^{-10}$	0.0	7.1	2.6	0.2
22	4.0	0.0	7.3	0.0	0.0
23	0.034	0.12	142.	0.0	0.0
24	0.053	-	60.	2.6	0.2
25	0.059	0.0	21.	0.0	0.0
26	0.059	0.0	7.3	0.0	0.0
29	0.68	0.12	11.	0.0	0.0
31	$<1.9 \times 10^{-10}$	0.0	4.4	0.0	0.0

Table 2. Effect of Percussion Test

Specimen Number	Leak Rates Before Test		Leak Rates After Test
	g/cc		g/cc
	He	Air	Air
1		270.	250.
3		46.	1600.
4		40.	36.
5	$<1.8\times10^{-10}$	0.42	1.5
6	$<2.1\times10^{-10}$	0.0	0.0
7		20.	4.4
8		640.	800.
10		914.	1100.
11		61.	200.
12		256.	400.
21		7.1	11.
22		7.3	17.
23		142.	80.
24		6.7	4.7
25		21.	74.
26		7.3	2.8
27	0.057	0.027	0.0
29		11.	20.
31		4.4	6.3
32	0.11	0.11	0.029

Table 3. Effect of Heat-Quench Test

Specimen Number	Leak Rate Before Test		Water Leaked over 24 Hrs.		Remarks
	g/cc		g	%	
	He	Air			
2	$<1.6\times10^{-10}$			36.	Untested
9	$<1.6\times10^{-10}$			29.	
28	$<2.0\times10^{-10}$			72.	
30	$<1.9\times10^{-10}$			75.	
5		1.5		33.	Percussion only
6		0.0		36.	
27		0.0		75.	
32		0.03		70.	
Experiment I					
11		200		10.	Air cooled
12		400		10.5	
26		2.3		1.8	
31		6.3		0.8	
Experiment II					
8		C.R.*		50.	Re-tightened cap asbestos on body
10		C.R.		59.	
25		C.R.		49.	
29		C.R.		31.	
Experiment III					
1		C.R.		64.	Asbestos 30 sec. delay
3		4.0		36.	
21		C.R.		28.	
22		300.		40.	
Experiment IV					
4		180.		47.	Asbestos 90 sec. delay
7		C.R.		36.	
23		80.		54.	
24		C.R.		37.	

* Cap Re-tightened



Fig. 1 Lower Portion of Drop Tower

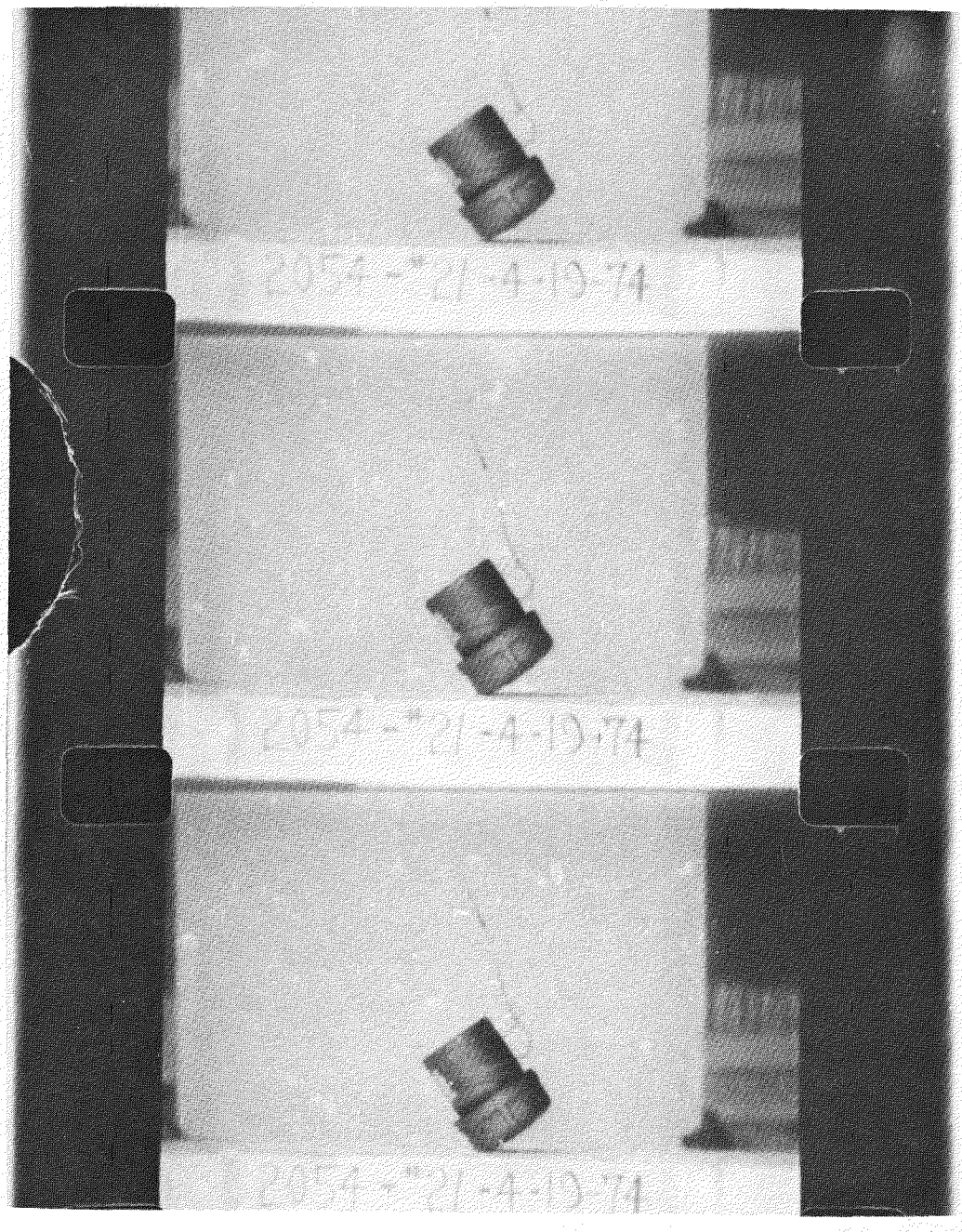


Fig. 2 Dropped-Specimen Impacting Surface

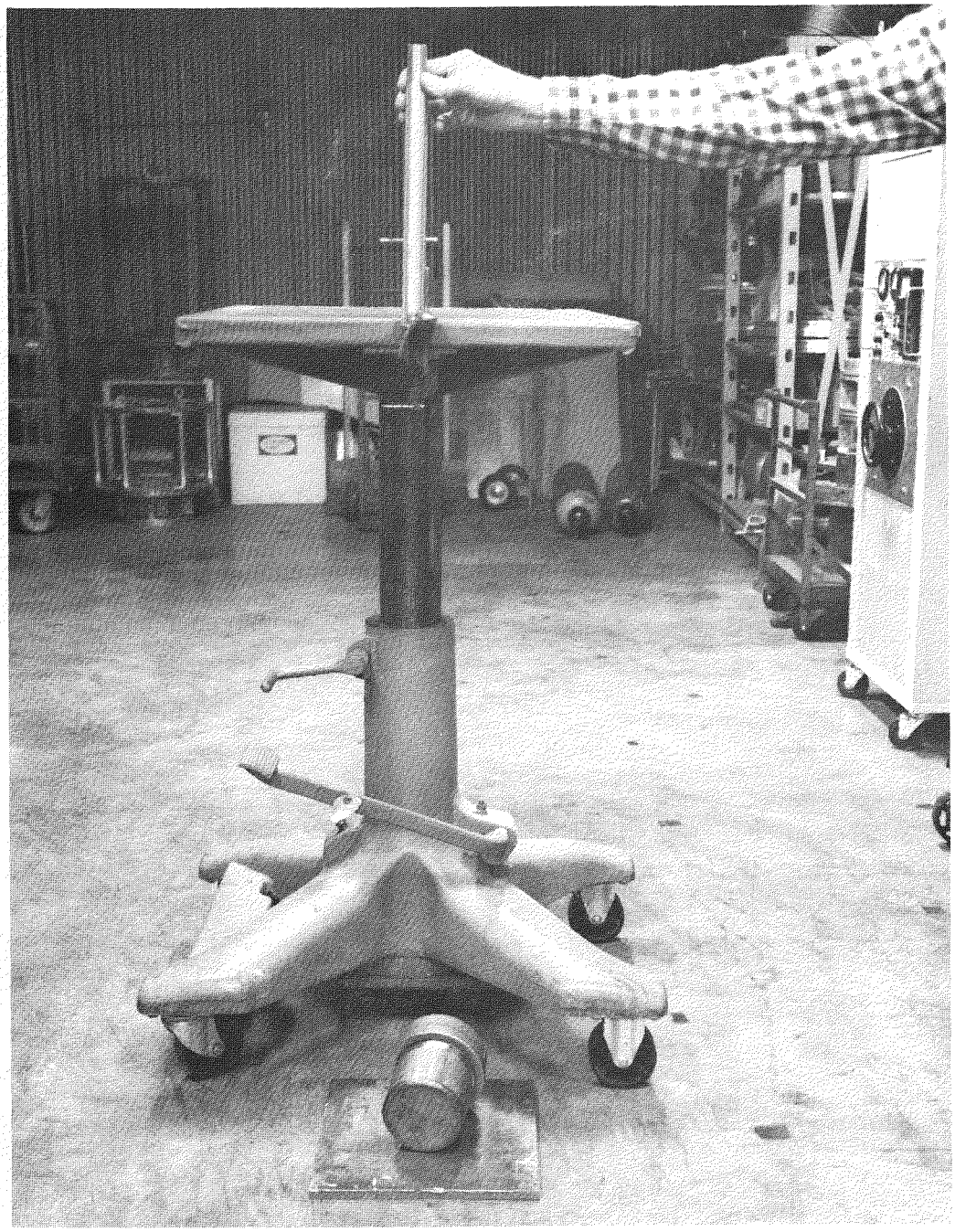


Fig. 3 Percussion Test Set-Up



Fig. 4 Heat-Quench Test Set-Up

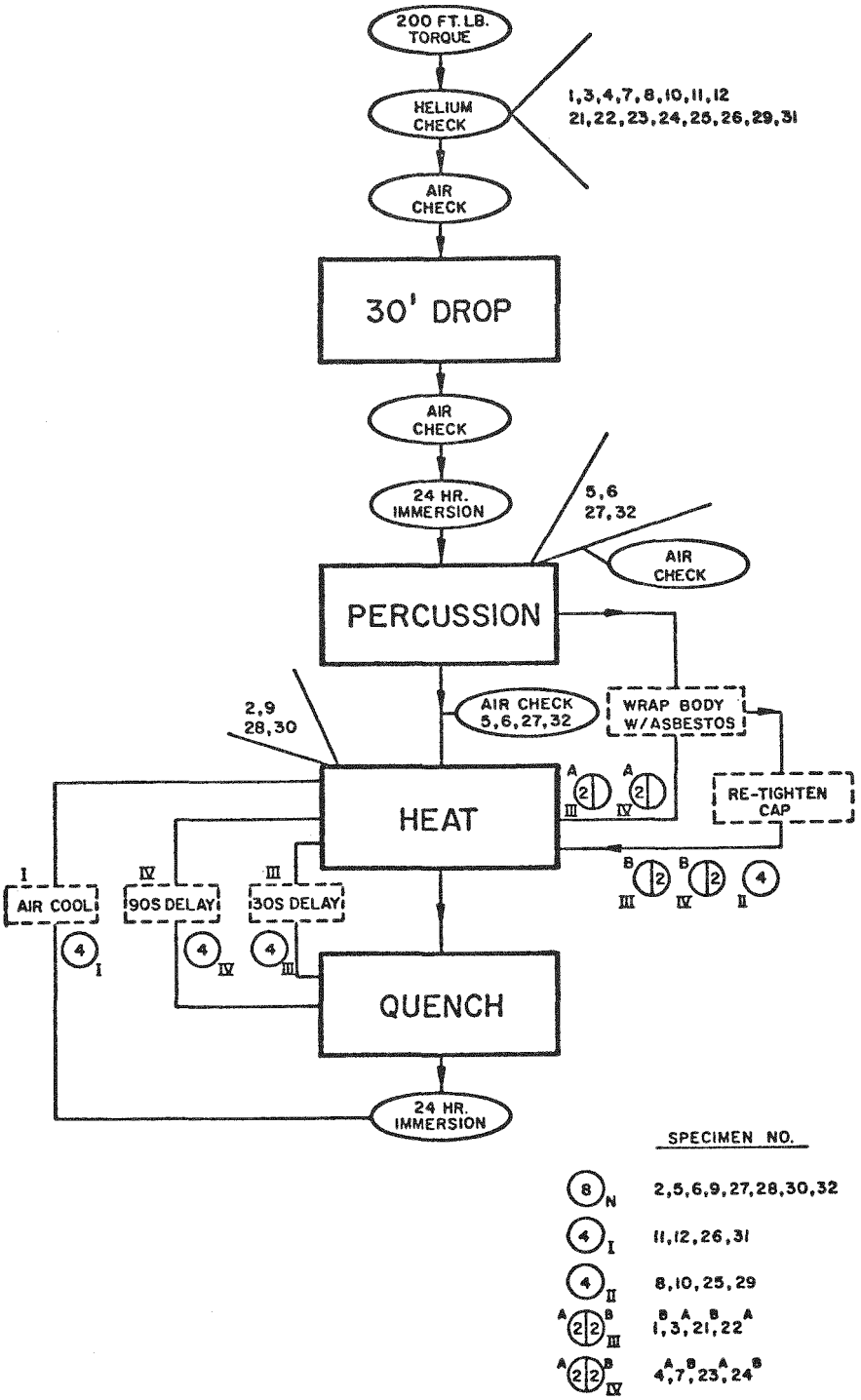


Fig. 5 Test Program Flow Diagram



Fig. 6 Dropped-Specimen Damage

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IMPACT ANALYSIS OF METAL CANS DAMAGED DURING A RADIOACTIVE MATERIALS SHIPMENT

J. M. Taylor

ABSTRACT

The impact properties of metal cans used to package radioactive materials was determined. The study was done because metal cans of the same type were damaged during a radioactive shipment. The impact properties of the metal cans were used to estimate the impact force that was applied to the damaged metal cans. The study revealed that the damaged metal cans were subjected to considerable impact force inside a DOT-6M shipping container. Drop tests, using a metal pipe from a DOT-6M and loaded in a similar fashion as the shipment, required 1000 drops (dropping height - 3") to cause extensive damage to the cans similar to the damage incurred during shipment. Vermiculite was evaluated as a protective packaging material during drop tests. It proved to be inadequate due to its high compressibility.

INTRODUCTION

This study was undertaken to a) estimate the magnitude of impact force that caused damage to several metal cans during a radioactive material shipment; b) determine the impact properties of metal cans of the same type used for packaging; and c) evaluate vermiculite as a protective packaging material.

The study was initiated because several metal cans containing UO_2 powder were damaged during shipment to Piketon, Ohio. Some of the more severely damaged cans are shown in Figure 1. The metal cans were packaged inside an approved radioactive shipping container (110 gallon, DOT-6M) which was not damaged during shipment. Metal cans of two sizes (3-1/2"D x 3-1/2"H, 4-1/4"D x 5-9/16"H) were stacked inside the pipe of the container and were free to move laterally. Since the shipping container was not damaged, it was suspected that the cans were damaged in transit, due to the bouncing of the cans. In order to confirm this, several columns of cans were impact-tested in a cyclic drop tester.

Deformation similar to the deformation the metal cans received during shipment, was achieved. Vermiculite was evaluated as a protective packaging material, using the cyclic drop tester.

EXPERIMENTAL PROCEDURE

The metal cans tested, Figure 2, are ones commonly used at Hanford for a variety of functions. The cans are made from tin-coated steel plate 0.011-0.014" thick. The body of the can is rolled to shape and the side seam is welded. The bottom is mechanically attached to the body by a crimping technique and the lids are made by a drawing process.

The metal cans were impact-tested using a drop tester, Figure 3, and a cyclic drop tester, Figure 4. The plastic deformation of the metal cans as a function of impact force was determined using the drop tester. The impact force on a single can was varied by dropping the crosshead from different heights and by varying the weight of the crosshead. The drop tester is capable of delivering 120 ft-lbs of energy. Metal cans stacked two high and offset, and four high of different sizes, were also plastically deformed using the drop tester (Figures 5, 6)

The cyclic drop tester was used to simulate impact shock that metal cans experience during shipment. The pipe shown in Figure 4 is one that was taken from a DOT-6M container. The pipe can be raised and dropped through different vertical distances. The number of drops per time can also be varied. To simulate the conditions under which metal cans containing UO_2 were severely damaged during shipment to Piketon, Ohio, four quart cans and six pint cans were filled with lead shot and placed inside the pipe. The cans were stacked inside the pipe as shown in Figure 7, which as far as could be determined, was the way they were stacked for shipment.

The gross weights of the indivual cans are given in Table 1. Empty cans and rags were used to fill the void space above the filled cans inside the pipe. No additional packing material was added to the pipe and the cans were free to move laterally the same as during actual shipment. The pipe was lifted a distance of 1 inch and allowed to free fall for 1000

drops. The frequency was approximately 1 cycle/second. The pipe was also dropped 3 inches for 1000 drops with and without packing material surrounding the cans. The number of drops was chosen arbitrarily since there was no way to determine the number of impacts the container received during transport. The actual shipment traveled around 4000 miles, so the 1000 drops corresponded on the average to an impact every 4 miles.

Table 1. Weight of UO_2 Powder and Metal Cans Packaged for Shipment in a 110 Gallon DOT-6M Shipping Container.

Can Numbers	Gross Weight - lbs.
1	4.11
2	1.39
3	14.37
4	2.34
5	1.23
6	1.21
7	14.38
8	1.80
9	16.40
10	0.4

RESULTS AND DISCUSSION

The reduction in height as a function of a repeated constant impact force (4.5 ft-lb) is seen in Figure 8. The relationship is linear for both size cans. To reduce the height of the cans by 50%, 37 impacts and 105 impacts of 4.5 ft-lb were required for the 3-1/2"D x 3/1/2"H and the 4-1/4"D x 5-9/16"H cans respectively. The cans deformed by crinkling and folding.

The deformation behavior of the 3-1/2"D cans under variable impact load, was non-linear as seen in Figure 8. The deformation behavior beyond 12% deformation for the 4-1/4"D can could not be determined, due to the limitations of the drop tester. Possibly, if the 4-1/4"D can could have been subjected to higher impact loads, it might have deformed in a non-linear fashion also. A single impact force of 120 ft-lbs reduced the height of the 3-1/2"D can by 43%, while a total impact force of 144 ft-lbs (4.5 ft-lbs x 32 impacts) was required to reduce the height of the 3-1/2"D can by 43% under repeated impacts of 4.5 ft-lbs. Differences in the

deformation characteristics under a single impact force as compared to a repeated impact of a constant force, is thought to be due to some sort of strain rate sensitivity. However, the impact force vs. deformation curve is linear up to 10% deformation regardless of whether the impact force is applied in one impact or several.

The total deformation, when the cans were offset, was greater due to the impact force acting on a smaller cross sectional area (14% less). The reduction in height as a function of impact was a linear function for the 4-1/2"D x 5-9/16"H can (Figure 8).

The impact properties of the column of four cans was much different than the single or double can impact properties. The reduction in column height as a function of a repeated constant impact force is shown in Figure 9. The percent deformation or column height decreases with increasing number of impacts. It is interesting to note that the smaller cans positioned between the larger cans underwent no deformation during the test. The decrease in height was due totally to the deformation of the 4-1/4"D cans. From the curve (Figure 9) it is seen that after about 20 impacts the column height does not change significantly. Impacting the larger can with the smaller can caused the lid to become depressed. As the number of impacts increased, the denting increased until the side wall of the can began to roll inward. It is speculated that as the sidewall was strained by being rolled inward, work-hardening occurred and deformation stopped. This type of deformation is different from the buckling or crinkling of the cans side wall by direct impact over the entire surface of lid area. It is not entirely clear why the smaller cans do not start to deform when the larger cans cease to deform. One possible explanation would be that the larger cans still deform, but they are elastically strained rather than plastically deformed. It was observed that during the drop test of the column of four cans that greater springback occurred as the number of impacts increased. It can be assumed that if the impact force was increased, deformation would start again.

It was intended that the data obtained from the drop tests could be used to estimate the impact energy delivered to the cans during shipment.

It became apparent, however, that every position of impact that a column of cans could assume in a shipping tube, could not be tested, using the drop tester.

The cyclic drop tester more closely matched shipping conditions and dropping the tube a known distance a known number of times, provided known impact data and corresponding measurable deformation. The damage sustained to 10 cans after dropping the tube a distance of 1 inch for 1000 cycles, is seen in Figure 10. The actual damage was slight; the volume change measured, along with the impact forces for the cans damaged, is given in Table 2.

Table 2. Deformation of Metal Cans Dropped 1000 Times in the Cyclic Tester a Distance of One-inch per Drop

Can Position No.	Can Size	% Change in Volume	Single Impact Force ft-lbs	Total Impact Energy ft-lbs
1 (bottom)	4-1/4"D	2.8	4.46	4,458
3	4-1/4"D	3.2	3.15	3,147
4	3-1/2"D	6.0	2.95	2,952
5	3-1/2"D	5.0	2.85	2,849
7	4-1/4"D	2.0	1.55	1,549

The most damaged can of the 4-1/4"D size was can Number 3. This can was positioned between two smaller cans so the top as well as the bottom of the can was deformed. Can number 7 was also positioned between two smaller cans but the impact force on the can was less. The bottom can was a large can which received the highest impact force. The only deformation visible was two dents on the top of the side wall and a depressed lid. The bottom was not damaged. It is felt that the reason the bottom can received less deformation than can 3 was because only the top was deformed. The lid area of can 1 stiffened up and resisted further deformation while can 3 continued to deform at least on the bottom. The smaller cans (4,5) had become offset during testing. The lid and bottom of can 5 was damaged while only the lid and sidewall of can 4 was damaged. Based on the impact curves shown in Figure 8, the cans should have deformed much more. However, it appears that the deformation characteristics of the cans inside the cyclic tester conform more to the deformation properties shown in Figure 9.

It can be assumed, based on Figure 9, that the total deformation took place after relatively few impacts. After the cans had become work hardened to the point deformation ceased, the cans proceeded to bounce or elastically absorb the impact force until the test was completed.

A second cyclic drop test consisted of dropping the tube 3-inches for 1000 drops. The test resulted in extensive damage to the cans. A few of the more severely damaged cans are shown in Figure 11. The bottom cans (1,2) were compressed to the point that they were nested in one another. Also, the smaller cans 4,5, and 6) had become mashed together. The damage sustained by the cans during the second test was more extensive than what actually occurred during shipment. The estimated volume change, along with the impact force is given in Table 3.

Table 3. Change in Volume as a Function of Impact Force For Cans Dropped 1000 Times in the Cyclic Tester a Distance of 3 inches Per Drop

Can Position No.	Can Size	% Change in Volume	Single Impact Force ft-lbs	Total Impact Energy ft-lbs
1 (bottom)	4-1/4"D	76	13.38	13,375
2	3-1/2"D	80	13.03	13,033
3	4-1/4"D	25	9.44	9,440
4	3-1/2"D	60	8.86	8,855
5	3-1/2"D	55	8.55	8,548
6	3-1/2"D	45	8.25	8,245
7	4-1/2"D	10	4.65	4,650
8	3-1/2"D	1 - 2	4.20	4,200
9	4-1/4"D	2	0.1	100
10	3-1/2"D	--		

The impact forces were severe enough that can 1 and can 2 were deformed to the volume of their contents (lead shot). From Table 3 it is seen that impact forces of 13 ft-lbs per impact is sufficient to cause extensive damage to the bottom cans. It is not known how many impacts were necessary to completely compress the cans.

From the cyclic drop tests some estimate can be made of the impact forces that were applied during shipment. The volume of the bottom can was reduced by 66% after shipment. This was the maximum amount the can could be compressed due to the volume of UO_2 it contained. To cause this

type of damage, it can be assumed that the bottom can saw forces between 4 ft-lbs/impact to 13 ft-lbs/impact. This doesn't rule out the possibility that damage could have been sustained in a few very severe bounces of the container. But, it would appear more likely that damage would occur after several moderate bounces. However, a 13 ft-lb impact on the bottom can required the shipping container or tube to bounce or drop a distance of 3 inches; this would be considered by some to be a severe bounce. Based on the tests performed, it can be speculated that the shipping container was bounced or dropped at least 3 inches several times. This indicates that the shipping container was shipped under fairly severe conditions, realizing that the total package weighed over 500 lbs.

One additional cyclic drop test was done with the cans in the same stacking sequence as before, but surrounded with vermiculite. About 1-1/2" of vermiculite separated each can inside the pipe. The vermiculite and the cans completely filled the shipping pipe so that empty cans or rags were not needed to fill the void space. After the pipe was dropped a distance of 3 inches for 1000 drops, the vermiculite became highly compressed. Damage to the cans was considerable (Figure 12). However, the overall damage to the cans was less with the vermiculite than without. Vermiculite proved to be an inadequate packing material under the circumstances of the test. Vermiculite is a very compressible material, having considerable void space. Thus, there is a little spring back or elasticity. When the material is fully compressed it loses its cushioning properties and the ability to absorb energy so that the impact force can be transmitted through it. As seen in Figure 13, the vermiculite was compacted to such a degree that it formed a hard heel on the bottom of the can.

This study has shown that the packaged contents inside approved specification containers can be subjected to fairly severe impact forces and requires careful packaging methods to be used.

Probably all of the damage that occurred to the cans during shipment could have been avoided if the following steps had been taken:

- The heaviest cans placed at the bottom of the pipe.
- Cans of one size taped together to prevent lateral movement.

- Suitable packing material (rubber disks) separating the large column of cans from the smaller column of cans.
- A suitable rubber disk or other shock absorbing material placed in the bottom of the shipping tube.

This study has also shown that research needs to be done to determine impact properties and limitations of common protective packaging materials.

ACKNOWLEDGMENT

The author expresses his appreciation to R. T. Lansiedel for his able assistance in performing the experimental work and for his inovative talent in designing the cyclic drop tester.

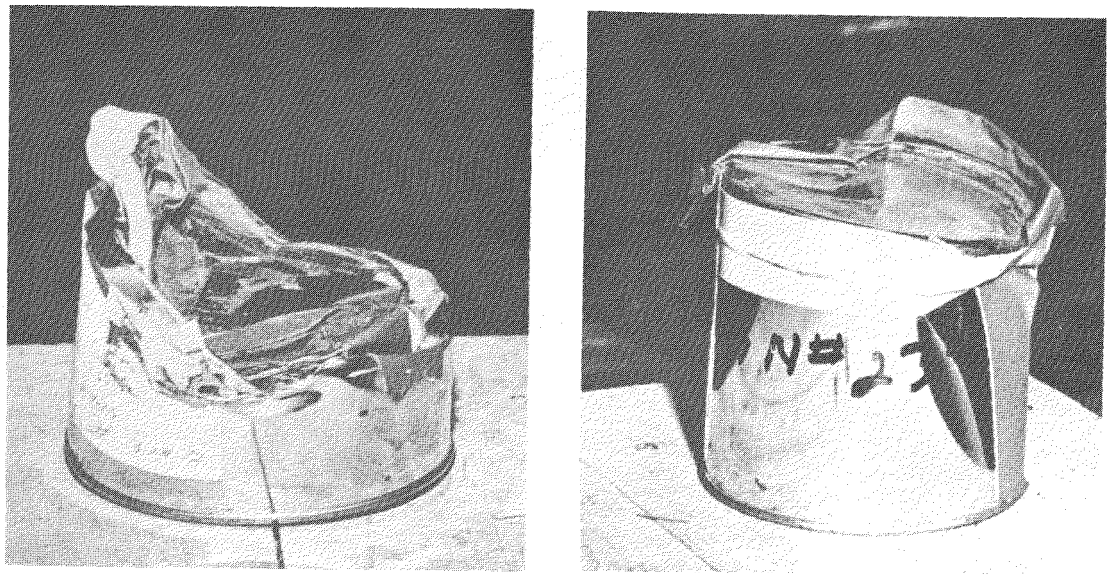


Fig. 1. Metal Cans Damaged in a DOT-6M Container During a Radioactive Materials Shipment.

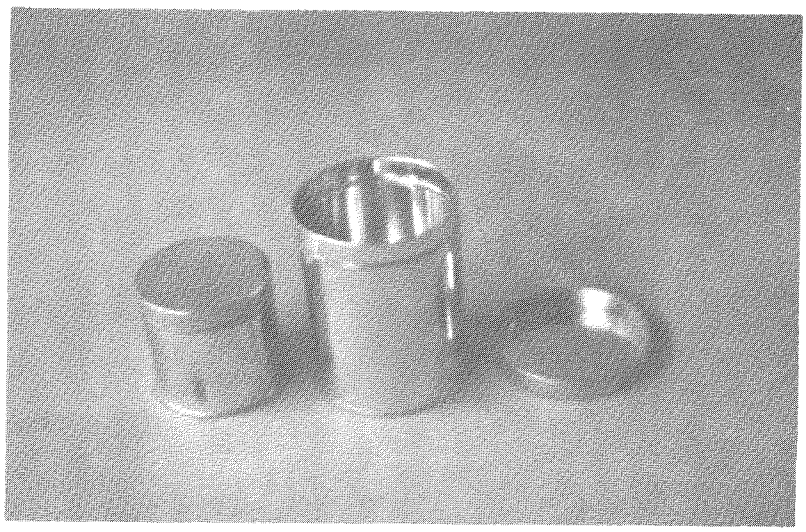


Fig. 2. Type of Metal Cans Used in Packaging Radioactive Material.

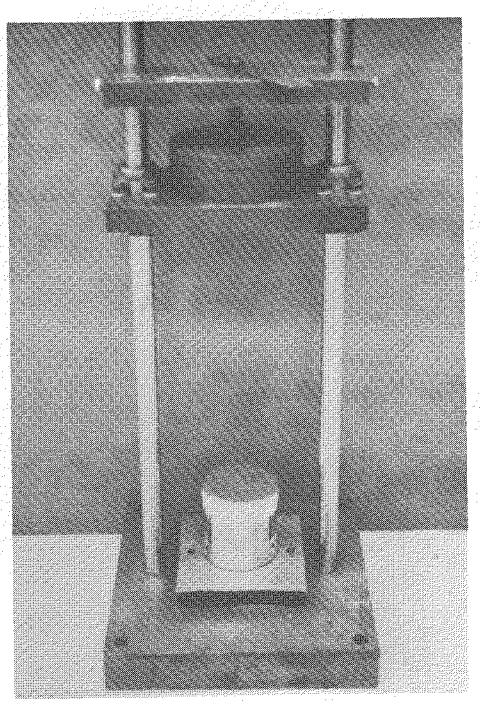


Fig. 3. Variable Height - Weight Drop Tester.

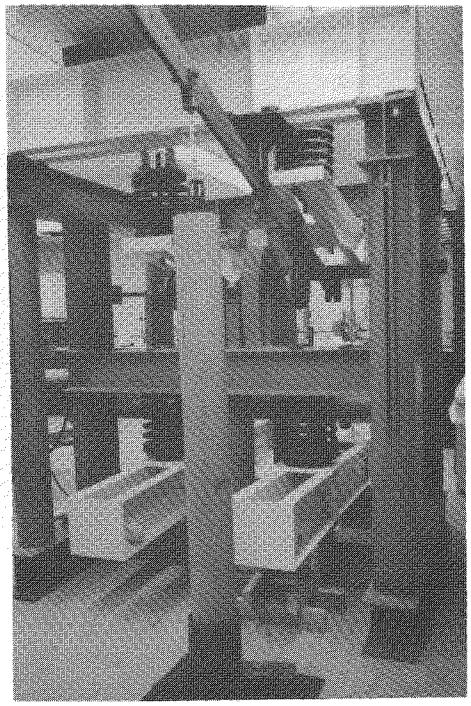


Fig. 4. Cyclic Drop Tester.

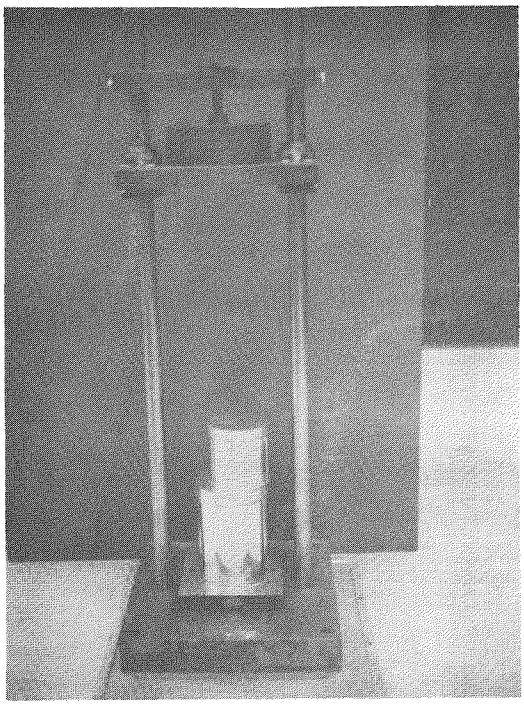


Fig. 5. Typical Position of Metal Cans Tested in Offset Position.

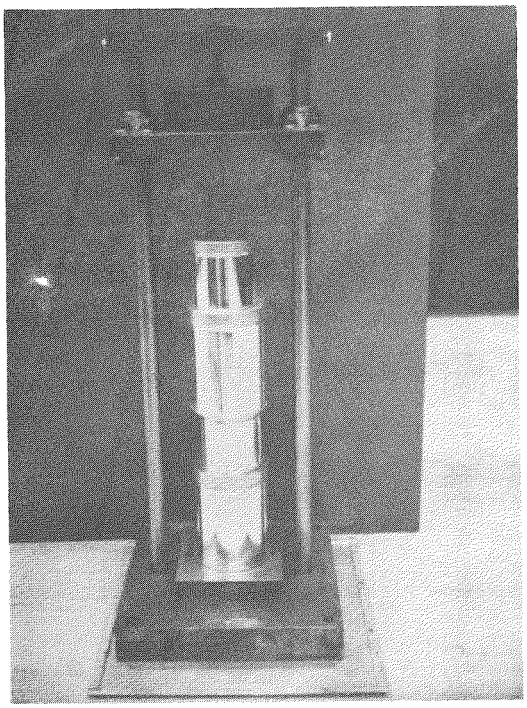


Fig. 6. Stacking Sequence of Different Size Cans Tested in Drop Tester.

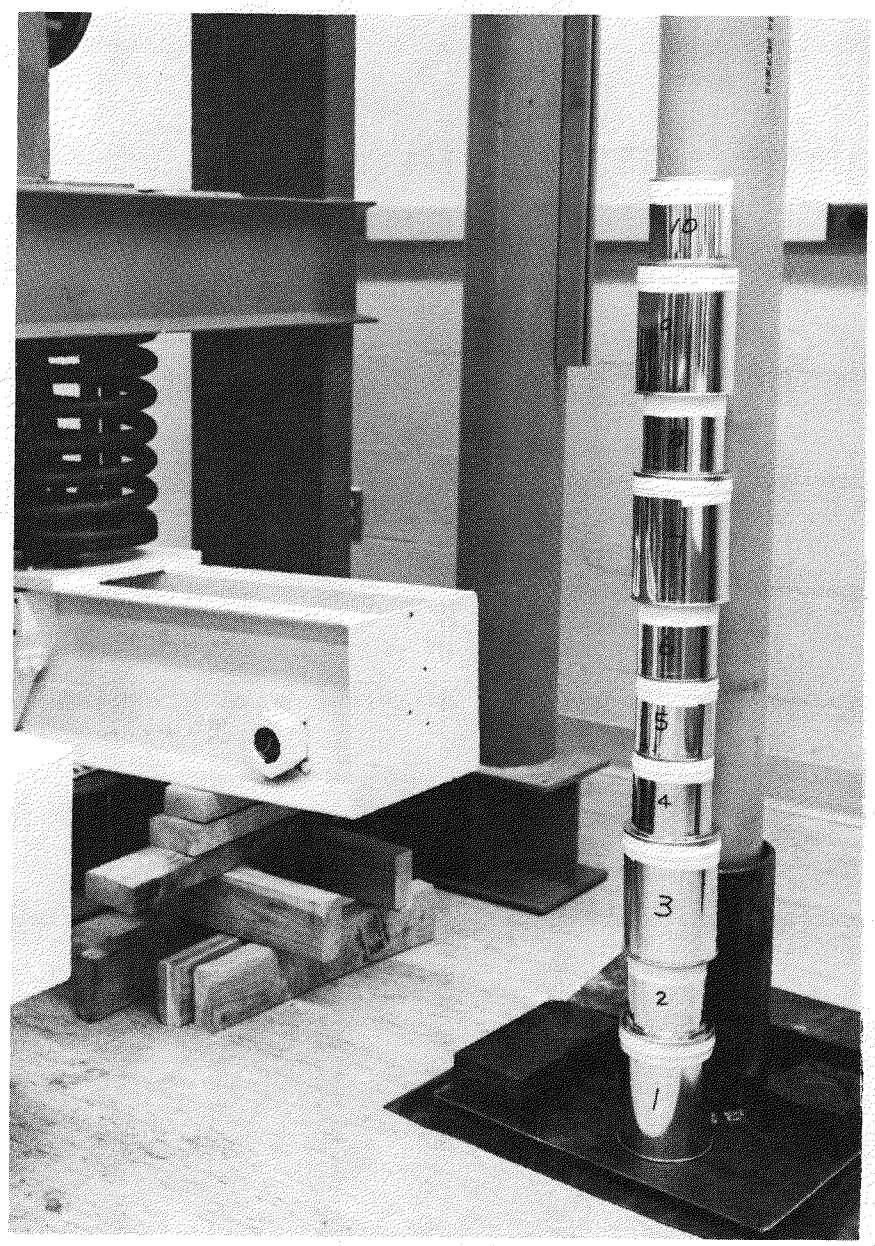


Fig. 7. Stacking Sequence of Metal Cans Loaded in a DOT-6M Shipping Tube.

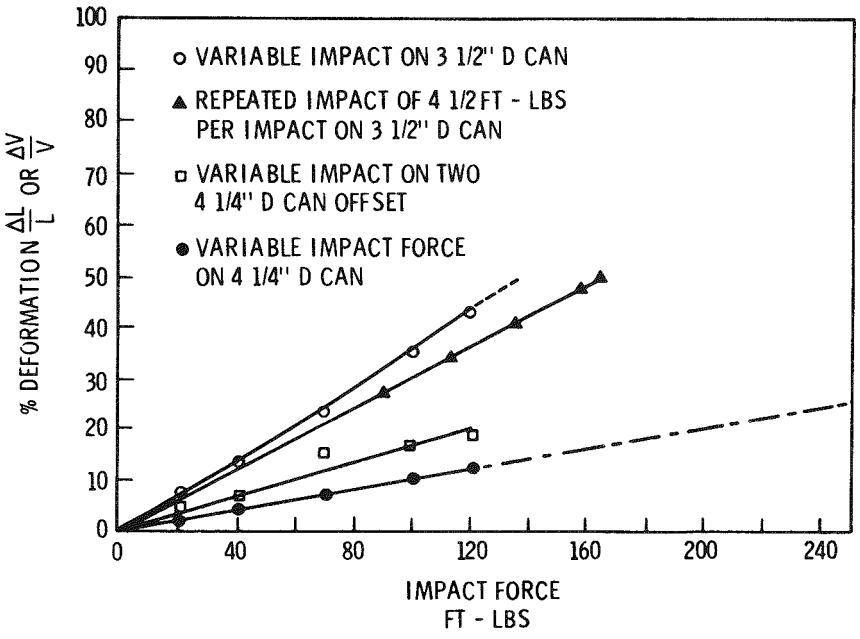


Fig. 8. Deformation Properties of Metal Cans as a Function of Constant and Variable Impact Forces.

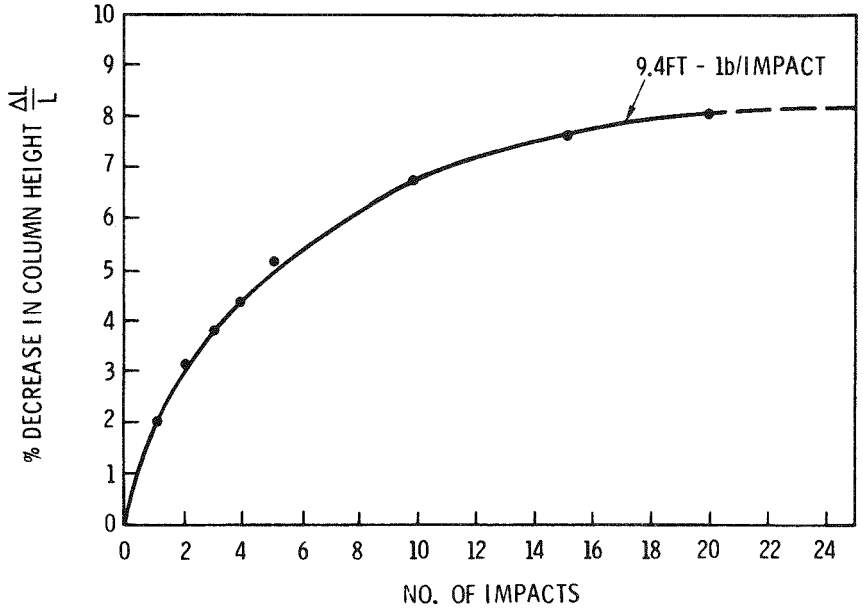


Fig. 9. Deformation of a Column of 4 Cans as a Function of Repeated Constant Impact Force.

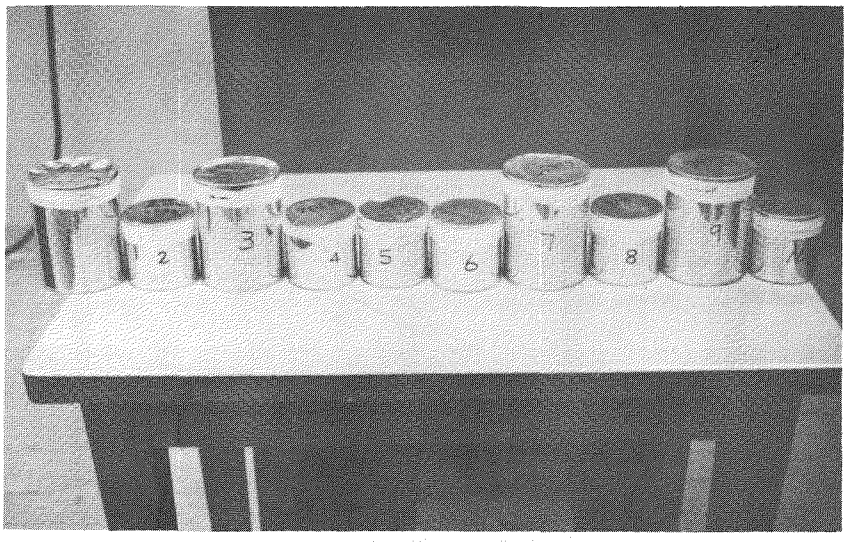


Fig. 10. Extent of Damage to Cans Dropped 1000 Times in Cyclic Tester a Distance of One Inch Per Drop.

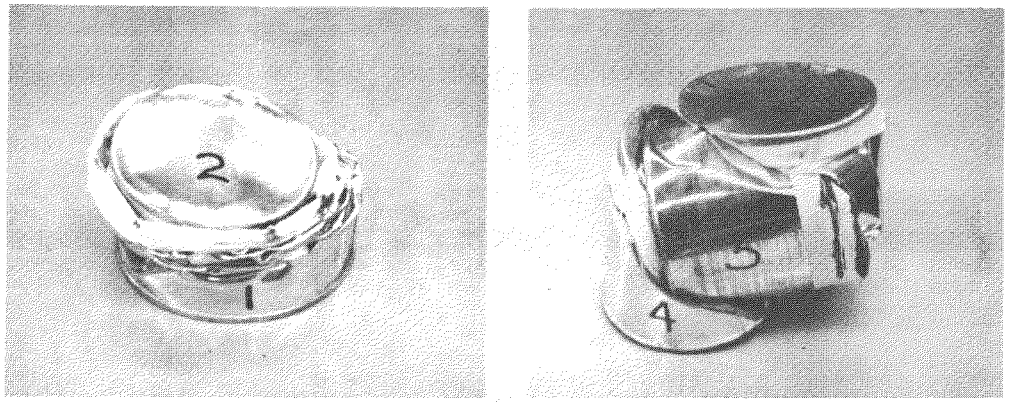


Fig. 11. Extent of Damage to Metal Cans Dropped 1000 Times in Cyclic Tester a Distance of Three Inches Per Drop.

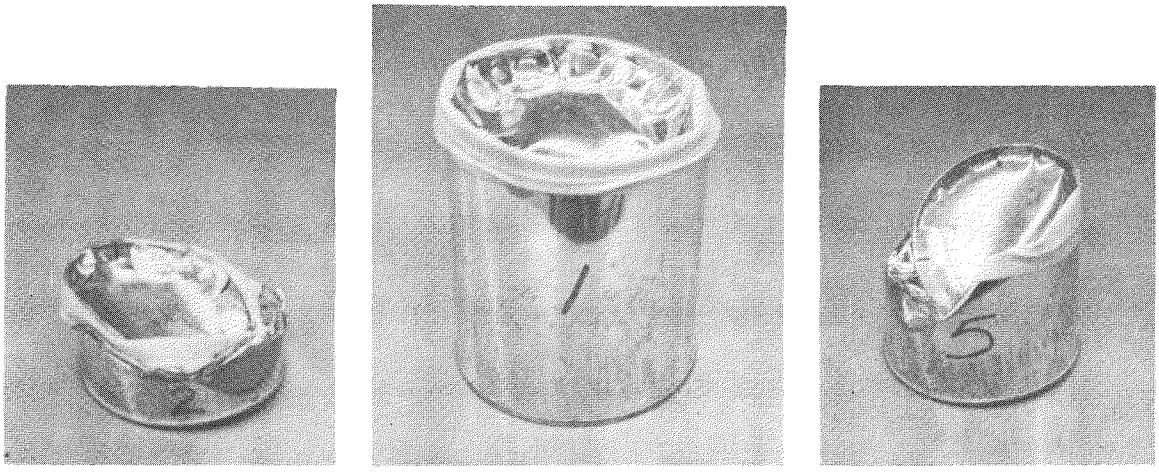


Fig. 12. Extent of Damage to Metal Cans Surrounded by Vermiculite, Dropped 1000 Times in Cyclic Tester a Distance of Three Inches Per Drop.

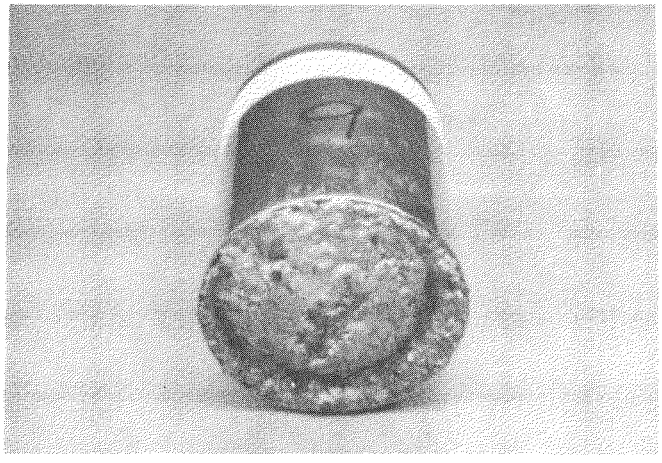


Fig. 13. Compressed Vermiculite Adherring to Bottom of Can After Removing From Cyclic Drop Tester. Dropped 1000 Times, Three Inches Per Drop.

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ENERGY ABSORPTION CHARACTERISTICS OF STAINLESS STEEL BOLTS UNDER SHIPPING CASK IMPACT CONDITIONS

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ABSTRACT

An impact testing program was conducted at Oak Ridge National Laboratory to determine the energy absorption characteristics of stainless steel bolts when subjected to impact load conditions typical of those which would result from the 30-ft free drop hypothetical accident conditions for shipping casks. Stainless steel bolt specimens were impacted by dropping a free-falling weight from a fixture which transmitted a tension load to the bolt. Force-time relationships occurring during the impact event were recorded. These data were numerically converted into deformation-time and energy-time relationships. Curves of absorbed energy for various bolt geometries were prepared to illustrate their behavior. Improvements in the energy absorption capabilities by reduction in the shank diameter of the bolts were investigated resulting in the determination of an optimum ratio of shank to tensile area.

INTRODUCTION

The shipment of radioactive material from one location to another requires the use of shielded containers which are subject to regulations[1,2] governing their structural integrity, shielding and heat dissipating capabilities. In addition to stipulating structural standards for packaging, these regulations require that the effectiveness of the shipping container be maintained within specified limits during and after certain prescribed hypothetical accidents.

Some type of bolted closure is included in the design of almost all shipping casks. To show that a shipping cask complies with the regulations, the adequacy of these bolted closures to fulfill their design function

*Oak Ridge National Laboratory is operated by Union Carbide Corporation under contract with the United States Atomic Energy Commission.

throughout the conditions of the hypothetical accidents must be demonstrated. This requires that the behavior of these bolts under the dynamic loading conditions they would experience when subjected to the accident condition be known. The dynamic yield strength, the fracture elongation, and the energy absorption characteristics are typical of the data required to determine the adequacy of these bolts. The investigation reported herein was undertaken to develop these data for stainless steel bolts to be used in securing shielding closures, with special emphasis placed on the energy absorption characteristics of these bolts when subjected to loadings typical of those expected in the hypothetical 30 foot free drop of a shipping container.

The program consisted of three parts: (1) determination of the optimum stress area ratio that will result in maximum energy absorption, (2) a study of energy absorption characteristics as a function of bolt geometry, and (3) a study of the effect of impact velocity on the dynamic behavior of these bolts.

ANALYTICAL BASIS OF EXPERIMENTAL PROGRAM

The basic concept of the experimental program involved dropping a known weight, in the form of a drop hammer, from a predetermined height onto a fixture in which the bolt to be tested is mounted and recording the force with respect to time experienced by the bolt throughout the duration of the resulting impact. This force-time relationship can be converted to an acceleration-time relationship by using Newton's Second Law of Motion. As part of the analytical basis of this program, two assumptions were made. It was assumed that the time of energy transfer between the falling drop hammer and the movable part of the test fixture is short when compared with the yielding time of the specimen and that the falling drop hammer and movable part of the fixture move together as a single mass immediately after their collision. Based on these assumptions, the acceleration of the drop hammer and movable portion of the fixture moving as a single mass at time t is given by

$$A(t) = \frac{1}{m_1 + m_2} F(t), \qquad (1)$$

where

m_1 = mass of drop hammer,

m_2 = mass of movable portion of fixture, and

$F(t)$ = instantaneous force transmitted to the specimen at time t.

The acceleration-time relationship can be converted to a velocity-time relationship by using the equation

$$V(t) = V_o - \int_{\tau=0}^{\tau=t} A(\tau)\, d\tau \,, \tag{2}$$

where

$V(t)$ = velocity of the combined masses moving as a single mass at time t and

V_o = initial velocity of the combined masses moving as a single mass at time = 0.

The initial velocity of the combined masses moving as a single mass can be determined by observing that the velocity of the combined masses must be zero at the time when unloading begins. The initial velocity of the combined masses is then determined from the equation

$$V_o = \int_{\tau=0}^{\tau=t_1} A(\tau)\, d\tau \,, \tag{3}$$

where t_1 = time at which the force begins to rapidly decay to zero. The relationship expressed in Equation 3 is necessary because energy is not conserved and the momentum of the system may or may not be conserved.

The displacement-time relationship can be obtained by using the equation

$$X(t) = \int_{\tau=0}^{\tau=t} V(\tau)\, d\tau = \int_{\tau=0}^{\tau=t} V_o\, d\tau - \int_{\tau=0}^{\tau=t} \int_{\tau=0}^{\tau=t} A(\tau)\, d\tau\, d\tau \,, \tag{4}$$

where $X(t)$ - deformation of the specimen at time = t. When both the force and the displacement as functions of time are known, the amount of energy absorbed by the speciment can be determined by using the equa

$$U = \int_{0}^{X} F \, dX \, , \tag{5}$$

where

U = energy absorbed for a specified amount of deformation,

X = deformation of specimen, and

F = applied force.

At the end of the impact duration, the amount of energy absorbed by the specimen can be compared with the kinetic energy of the falling mass at time equal zero. The impulse of the applied force integrated over the impact duration can be compared with the total change in momentum of the system to evaluate the validity of the original assumptions.

EXPERIMENTAL EQUIPMENT AND INSTRUMENTATION

The impact tests were conducted at the Oak Ridge National Laboratory (ORNL) Drop Tower. This structural steel tower is 40 feet high, and its base consists of an armor plate embedded in steel-reinforced concrete. Two 3-by-3-inch angles are attached to the tower vertically to guide the drop hammer during its free fall. The basic structure of the drop hammer weights 99 pounds. Weights in the form of steel blocks can be added to the hammer in 21-pound increments to obtain a maximum weight of approximately 500 pounds.

The bolt specimen to be tested is secured between the fixed and movable flanges of the impact test fixture illustrated in Figure 1. The movable portion of this test fixture has a total weight of 36 pounds and is comprised of the anvil, three pins, and the movable lower flange. When the drop hammer strikes the anvil, the load is transmitted downward through the three pins to the lower movable flange. The specimen, which is loaded by the movable flange at the bottom of the fixture, is secured to the upper fixed flange by a tension load cell. Force measurements are made either by using both a tension load cell and a compression load cell or by using only a compression load cell, depending on the size of the bolt being tested.

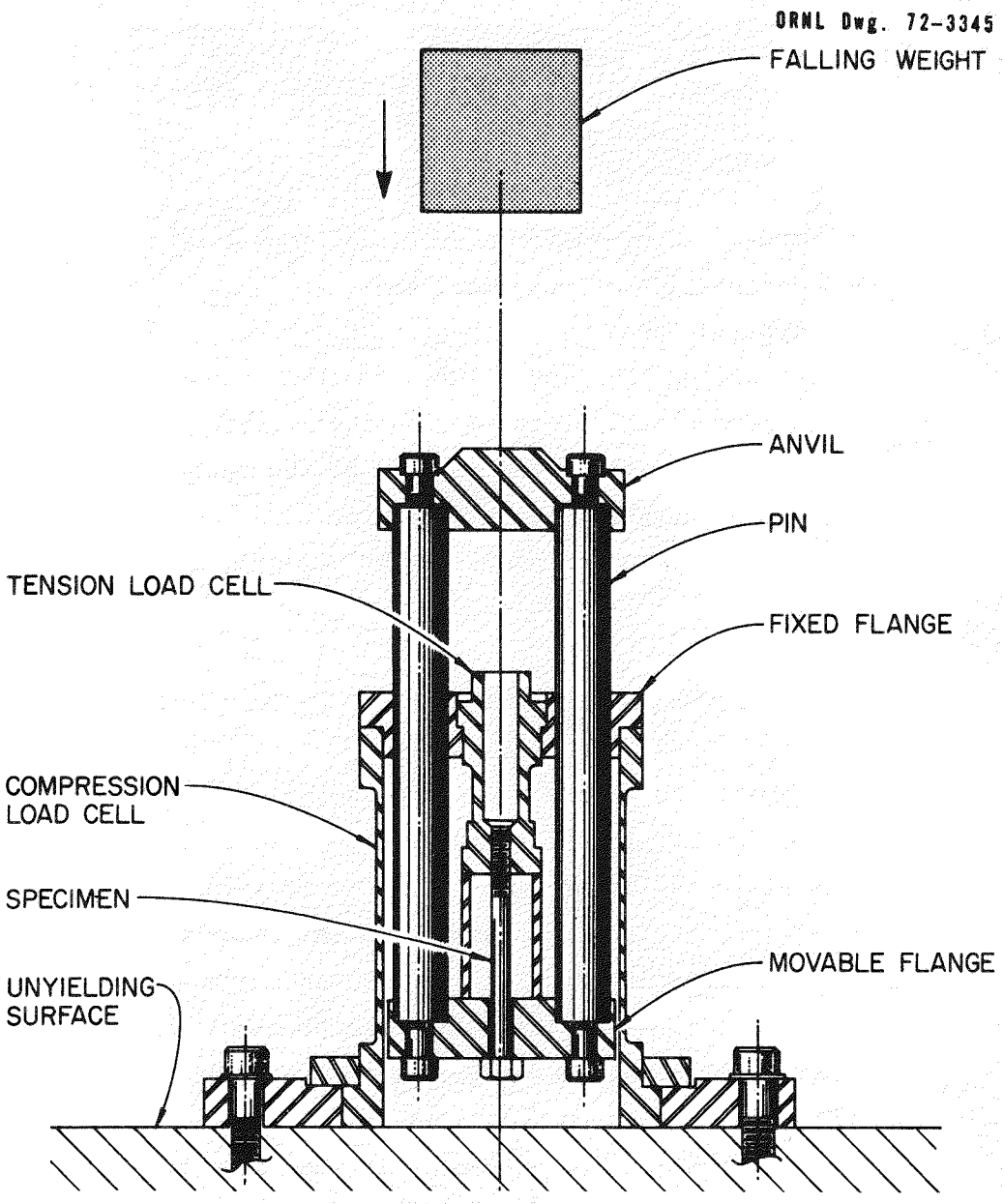


Fig. 1. Bolt Impact Fixture.

Five load cells were designed, fabricated and instrumented for this experimental program. Each load cell was equipped with bonded resistance strain gages wired into a full Wheatstone bridge circuit. The output of each bridge was fed to one channel of a dual-beam oscilloscope with a dual trace differential amplifier. The ORNL Drop Tower, the drop hammer, weights, specimen fixture, and the measuring and recording equipment used in the impact tests are shown in Figure 2.



Fig. 2. Equipment Used in Bolt Impact Tests.

TEST SPECIMENS

The selection of specimens to be tested was directed toward the development of impact data for commercially available stainless steel bolts with a potential for use in shipping cask applications. The specimens tested were coarse threaded bolts procured in accordance with the requirements of Military Standard MS-35307[3] which specifies the dimensions and the minimum tensile strength of the bolts and limits the material of manufacture to 300-series stainless steel. In addition to meeting the requirements of this standard, the manufacturer of the test

specimens was requested to certify the material, the method of manufacture, and the method of thread forming used. The selection of the size of the specimens to be tested was based on typical shipping cask applications and the load capacity of the ORNL Drop Tower. Bolts of varying lengths with diameters of 3/8, 1/2, 5/8, 3/4 and 1 inch were selected.

The specimens procured in accordance with the military standard were both hot-forged bolts and cold-headed bolts with rolled threads. The hot-forged specimens included both strain hardened bolts with ground threads and as-forged bolts with machine-cut threads. For convenience, the differences were categorized as Type 1, Type 2, and Type 3. The strain-hardened hot-forged bolts with ground threads were designated as Type 1, the hot-forged bolts with machine-cut threads were designated as Type 2, and the cold-headed bolts with rolled threads were designated as Type 3.

The studies of the optimum stress area ratio made during this investigation were conducted on three series of specimens. Two of these were Type-1 bolts with free lengths of 1.50 and 2.75 inches, and the third was Type-3 bolts with free lengths of 2.50 inches. The term "free length" is used here to describe the length of the bolt measured from the undersurface of the head to the inside surface of the engaged nut. The specimens for the optimum stress area ratio studies were prepared by reducing the shank of the bolts by turning to give a predetermined value of the ratio of the shank area to the tensile stress area. As used in this investigation, the stress area ratio is the ratio of the shank area to the tensile stress area of the bolt. The tensile stress area is the cross-sectional area of the threaded portion of the bolt determined from the mean of the pitch and root diameters of the threads. This shank-to-tensile-stress-area ratio was varied from 0.95 to approximately 1.35. The ratio of 1.35 is that for a bolt with a full-diameter shank.

IMPACT TESTS

Impact tests were conducted on a total of 295 stainless steel bolts. Each bolt to be tested was secured in the impact test fixture, and the desired weight to be dropped and drop height were selected. The load

cell bridge circuits were balanced, the proper sweep rate and voltage amplification were set on the oscilloscope, and the mechanical trigger was set a small distance above the top of the anvil of the bolt impact test fixture.

When all of the equipment was adjusted and the proper instrument settings were made, the drop hammer was raised to the selected drop height and released. When the free falling hammer closed the contacts on the mechanical trigger, the single sweep of the oscilloscope was initiated. The impact of the drop hammer on the anvil of the bolt test fixture transmitted a force through the specimen to the load cells producing an output voltage proportional to the applied force. This voltage, which changed rapidly with time, was displayed on the screen of the oscilloscope and recorded by the scope camera as a time exposure.

A typical impact force-time photograph made during the tests are shown in Figure 3. The upper trace shown in this figure is the signal from the tension load cell. The impact force is the product of the displayed force signal and the load cell calibration constant. The lower trace shown in Figure 3 is the signal from the compression load cell.

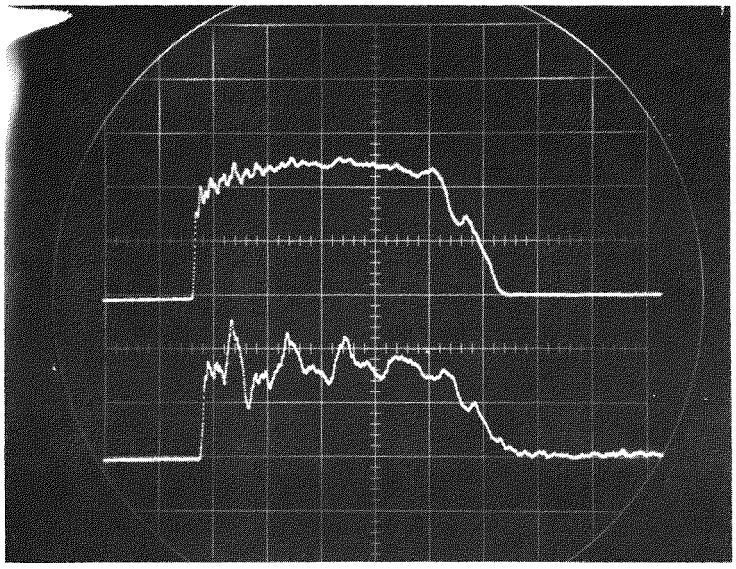


Fig. 3. Force-Time Photograph Taken During Impact Test.

EVALUATION OF TEST RESULTS

Impact tests directed toward determining the optimum stress area ratio for the different bolt sizes and material conditions were made prior to the tests directed toward studying the energy absorption characteristics of these specimens. The optimum stress area ratio determined from the results of the first tests was used as the basis for the reduced-shank bolts tested in the energy absorption study. Impact tests were also conducted to determine the effect of impact velocity on the dynamic behavior of bolts used in shipping cask applications. The assumptions and techniques that formed the analytical basis of this experimental investigation were assessed.

Optimum Stress Area Ratio

The first impact tests of bolt specimens were conducted to determine the optimum ratio of the shank area to the tensile stress (threaded) area for bolts with a nominal diameter of 5/8 inch. Eighty-five specimens consisting of three series of bolts with stress area ratios ranging from 0.95 to 1.35 were tested for this purpose. The three series of specimens included Type-1 bolts with a free length of 1.5 inches, Type-1 bolts with a free length of 2.75 inches, and Type-3 bolts with a free length of 2.5 inches. The results of these tests were plotted in terms of the absorbed energy and fracture elongation as a function of the stress area ratio for the three series of specimens tested.

The absorbed energy at the fracture elongation of the specimens as a function of the stress area ratio is plotted in Figure 4. All of the points on the curves except four represent fractured specimens. The dashed vertical lines indicate the approximate stress area ratios at which the mode of failure changed. Data points to the left of these lines denote shank failures and data points to the right denote thread failures. Several data points were obtained for each stress area ratio of specimens that did not fracture, and these points would lie below the curves for all three series of bolts.

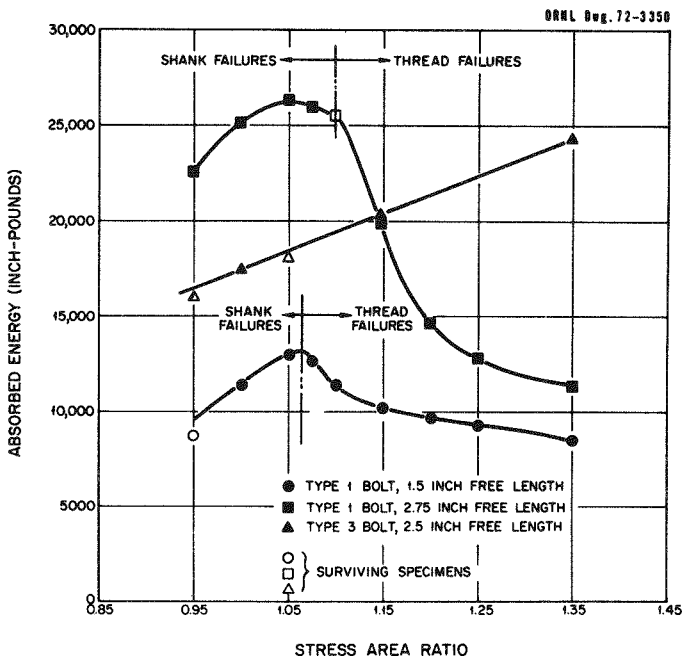


Fig. 4. Absorbed Energy as a Function of Stress Area Ratio for Three Test Specimens.

Both series of the Type-1 bolts exhibited an optimum stress area ratio. Although tests of more specimens would be required to establish the ideal optimum area ratio, the data indicate that a conservative ratio for the material conditions and free lengths tested would be 1.05. A ratio slightly less than the optimum should be selected to assure that failure will occur in the shank of the bolt.

The behavior of the Type-3 bolts with rolled threads and a free length of 2.5 inches was very different from that of the Type-1 hot-forged steel bolts. The Type-3 bolts with a stress area ratio of 1.35 (full-diameter shank) experienced shank failure. A reduction in the shank area of this specimen type resulted in a proportional reduction in the amount of energy absorbed. The amount of energy absorbed appears to be a linear function of the shank-to-tensile-stress-area ratio for the Type-3 bolts tested.

It should be noted that the energy absorption curves for the Type-1 stainless steel bolts illustrated in Figure 4 are similar in shape to those reported by Love and Pringle[4] for carbon steel bolts. The increase

in the amount of energy absorbed by the Type-1 hot-forged stainless steel bolts with their shanks reduced to the optimum ratio over that absorbed by similar bolts with full-diameter shanks was approximately 50% for the bolts with a free length of 1.5 inches and about 130% for the bolts with a free length of 2.75 inches. The same reduction in the shank area of the Type-3 cold-headed bolts with rolled threads resulted in a decrease in the energy absorbed of about 25%.

Energy Absorption Characteristics Under Impact Loading

Impact tests were made on a total of 195 stainless steel bolts with both full-diameter shanks and shanks reduced to the previously determined optimum stress area ratio to investigate their energy absorption characteristics. The data resulting from these tests were analyzed by using the analytic techniques discussed.

The dynamic energy absorption characteristics of 5/8- and 3/4 inch-diameter Type-1 stainless steel bolts with both full and reduced shanks are illustrated in Figure 5. The values, plotted on the abscissa, in each plot are ratios of the free length of the specimen to the nominal diameter.

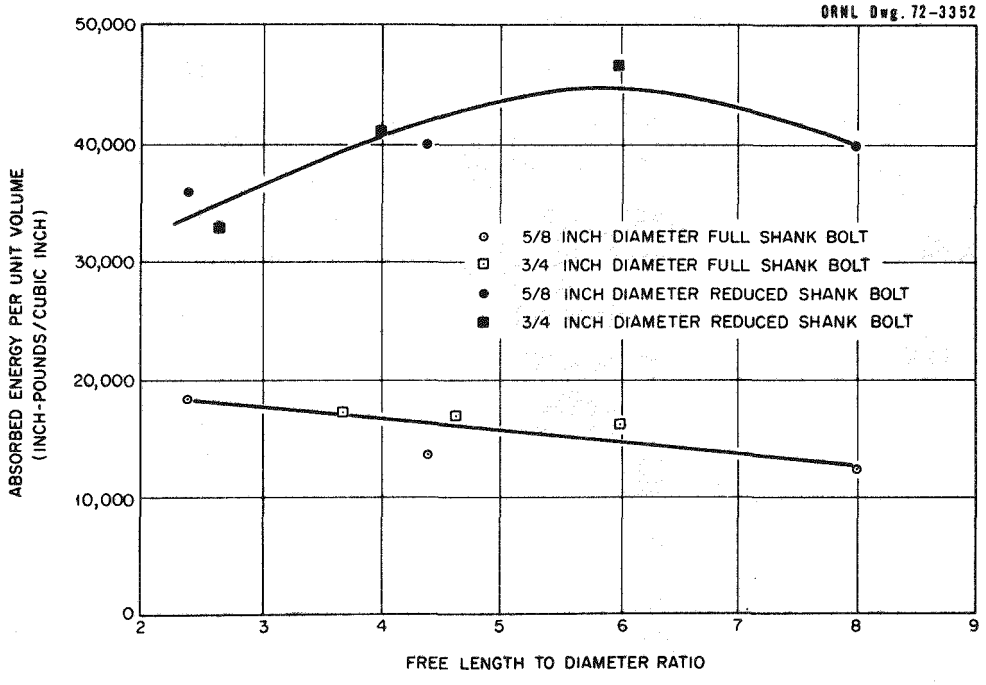


Fig. 5. Absorbed Energy per Unit Volume as a Function of Free-Length-To-Diameter Ratio.

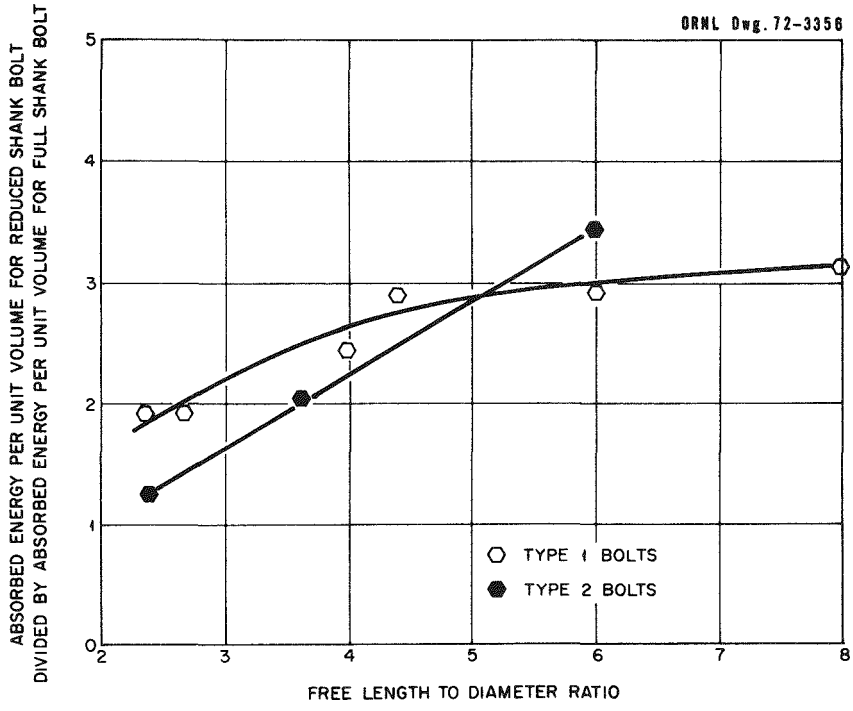


Fig. 6. Ratio of Impact Energy Absorbed by Reduced Shank Bolts to Energy Absorbed by Full Shank Bolts.

The absorbed energy per unit volume is expressed in inch-pounds per cubic inch, where the volume was determined from the nominal or reduced-shank diameter and the free length of the bolts tested.

Perhaps the value of shank area reductions is best illustrated in Figure 6, which is a plot of the ratio of the impact energy absorbed per unit volume by the reduced-shank bolts to that absorbed ty the full-shank bolts as a function of their free-length-to-diameter ratio for the Type-1 and Type-2 bolts tested. It is clear that the energy absorption capability of the bolts is greatly increased by reducing the shank. For free length to diameter ratios greater than six the ratio of the energy absorbed by reduced-shank bolts to that absorbed by full-shank bolts appears to be essentially constant at a value of 3 for the Type-1 bolts. An insufficient number of Type-2 specimens was impact tested to determine the magnitude of the increase in energy absorption resulting from shank reduction, but it is obvious that the value approaches and may exceed that for Type-1 bolts. The similarity between the energy absorption and fracture elongation characteristics of full-shank Type-2 bolts with a diameter of 1 inch is illustrated in Figure 7.

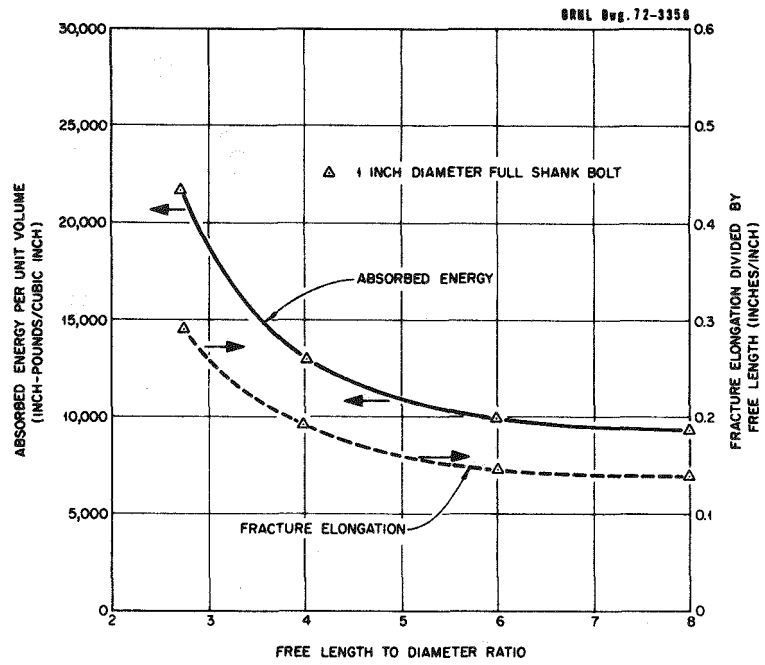


Fig. 7. Absorbed Energy and Fracture Elongation as a Function of Free-Length-To-Diameter Ratio.

The energy absorption characteristics of full-shank Type-3 bolts with diameters of 3/8, 1/2, and 5/8 inch were investigated. The amount of energy absorbed by these full-shank bolts relative to that absorbed by the full-shank Type-1 and Type-2 bolts should be noted. The Type-3 bolts with full-diameter shanks absorbed almost 75% more energy per unit volume than the full-shank Type-1 bolts and perhaps 150% more than the full-shank Type-2 bolts with a free-length-to-diameter ratio of 4. All failures occurring in these Type-3 bolts were shank failures. For each failed specimen, four to six other specimens of the same type were impact tested and absorbed less energy than the fractured specimen and were elongated proportionately less.

Typical Type-3 specimens with full-diameter shanks of 3/8, 1/2, and 5/8 inch that were tested by both static and dynamic methods are shown in Figure 8. From left to right in this photograph are two 3/8-inch-diameter bolts, the first of which was tested statically and the second dynamically. The third and fourth specimens are 1/2-inch-diameter bolts tested statically and dynamically, respectively, and the fifth and sixth specimens are 5/8-inch-diameter bolts tested statically and dynamically, respectively.

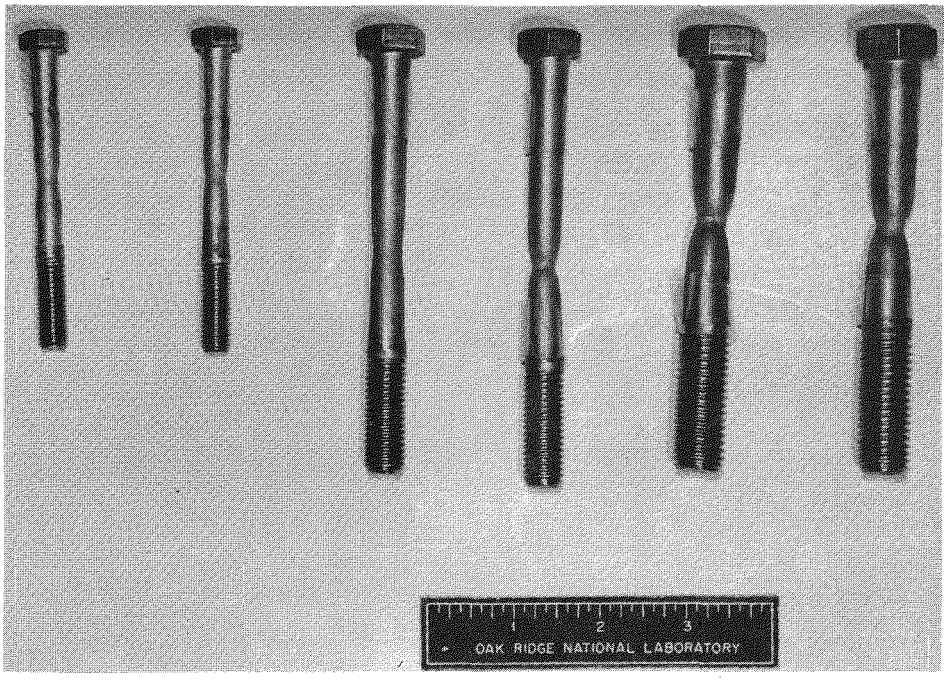


Fig. 8. Typical Type 3 Specimens Tested by Static and Dynamic Methods.

EFFECT OF IMPACT VELOCITY

Many shipping casks are designed to employ energy absorption devices to reduce the effective impact velocity experienced by the cask under accident conditions. It is therefore desirable to know the effect of impact velocity on the dynamic behavior of bolts used in shipping cask applications. For this purpose, impact tests were conducted on 15 Type-3 bolts with a nominal diameter of 5/8 inch and a free length of 2.75 inches. In these tests, the impact velocity was varied over a range of 20 to 43 ft/sec. This was the range of the terminal velocity of the crop hammer just prior to impact with the anvil of the bolt test fixture. The effective impact velocity experienced by the bolt specimens ranged from 18 to 31.5 ft/sec. This difference in values was attributed to the energy transfer between the drop hammer and the movable portion of the test fixture.

The amount of variation in the energy absorbed by the test specimens is illustrated in Figure 9. As may be seen in this plot, there is

essentially no variation in the energy absorption capability of this bolt type and size over the range of impact velocities applicable to shipping casks subjected to the prescribed accident conditions.

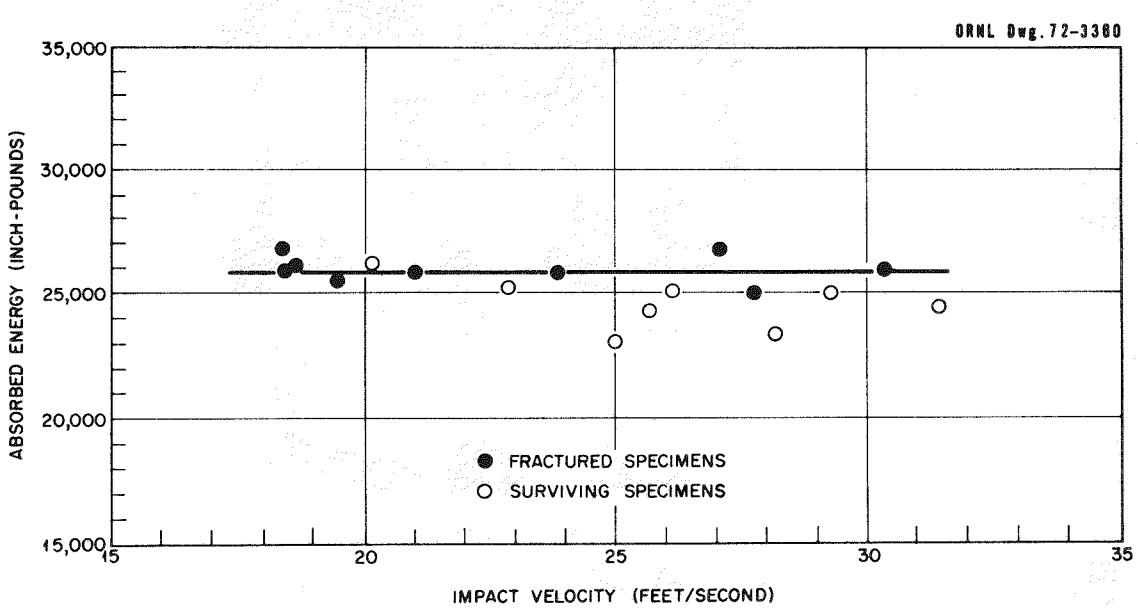


Fig. 9. Absorbed Energy as a Function of Impact Velocity for Type-3 Bolts.

EVALUATION OF ANALYTICAL TECHNIQUES

Certain assumptions were made in developing the analytical basis for this experimental program. The assumption that the drop hammer and movable portion of the bolt test fixture move as a single mass after impact was valid as long as the mass of the drop hammer was greater than twice the mass of the movable portion of the fixture. The calculated elongation compared favorably with the final measure elongation of the specimens tested, the major portion agreeing within 20%. A comparison of the values for impulse and change in momentum of the drop weight indicates that they were within 15% for most surviving specimens. For surviving specimens the absorbed energy represented approximately 70 to 80% of the impact energy.

CONCLUSIONS

The dynamic behavior of the stainless steel bolts impact tested in this experimental program can be characterized by such data as the optimum stress area ratio, the energy absorbed and fracture elongation as a function of the bolt geometry and material conditions, the dynamic yield force, and the effects of impact velocity. These data were developed for the bolts tested and are presented in USAEC Report, ORNL-TM-1312, Vol. 12, in a form that will allow designers and analysts of shipping casks to observe the effects of different material conditions, reductions in shank diameter, varying length-to-diameter ratios, and the impact velocity on the dynamic behavior of these bolts. The dynamic behavior of stainless steel bolts was shown to be highly dependent upon their method of manufacture and final material condition.

This optimum stress area ratio of Type-1 bolts with free-length-to-diameter ratios in the range of 2.4 to 4.4 was found to be approximately 1.05. This is a conservative value in that it is slightly less than the apparent optimum but was selected to result in shank failure. No optimum stress area value was determined for the Type-2 bolts tested. The optimum stress area ratio for the Type-3 bolts tested was found to correspond to the full-diameter shank condition of these bolts.

The energy absorption characteristics exhibited by the bolts tested showed the importance of the free-length-to-diameter ratio with respect to the dynamic behavior of both Type-1 and Type-2 bolts. These bolts with full-diameter shanks exhibited an energy absorption capacity in the range of 10,000 to 20,000 inch-pounds per cubic inch. This capacity decreased with increasing length-to-diameter ratios. The Type-3 bolts with full-diameter shanks exhibited an energy absorption capacity in excess of 30,000 inch-pounds per cubic inch for free-length-to-diameter ratios ranging from 2 to 6. Reducing the shank diameter to an approximate optimum diameter greatly enhanced the energy absorption capacity of the Type-1 and Type-2 bolts. For these bolts, the increase in absorbed energy per unit volume was a function of the free-length-to-diameter ratio. This increase in absorbed energy ranged from 1.3 to approximately 3 for length-to-diameter ratios ranging from 2.5 to 8.

The dynamic properties of Type-3 bolts with full-diameter shanks were found to be insensitive to impact velocity over the range of velocities used in this investigation. It appears that this is also true for Type-1 and Type-2 bolts with reduced-diameter shanks. The effect of impact velocity on the behavior of Type-1 and Type-2 bolts with full-diameter shanks is probably a function of the exposed thread length of these bolts.

REFERENCES

1. United States Atomic Energy Commission Manual, Volume 0000, General Administration, Part 0500, Health and Safety, Chapter 0529, Safety Standards for the Packaging of Radioactive or Fissile Materials, August 22, 1966.

2. Code of Federal Regulations, Title 10, Part 71, Transport of Licensed Materials, also see Federal Register, Vol. 31, pp. 9941-9949, July 1966.

3. Military Standard MS-35307, "Screw, Cap, Hexagon Head (Finished Hexagon Bolt), Steel, Corrosion Resisting, Passivated, UNC-2A," U. S. Government Printing Office, May 1969.

4. J. Love, Jr., and O. A. Pringle, "The Influence of Shank Area on the Tensile Impact Strength of Bolts," Transactions of the ASME, Vol. 78, pp. 1489-1495, October 1956.

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CURRENT RESEARCH AND DEVELOPMENT ON SPENT FUEL SHIPPING CASKS IN JAPAN

S. Aoki

ABSTRACT

Nowadays in Japan more than 25 nuclear power stations in operation or under construction. The first reprocessing plant will start to operate in this year. In near future a lot of spent fuels shall be transported between these nuclear power stations and the reprocessing plant. Providing for such situation, the Japanese National Standards for Safe Transport of Fissile Fuels enacted in 1968 are now being reexamined referring to the 1973 revised edition of IAEA Regulations for the Safe Transport of Radioactive Materials. On the other hand, the studies necessary to design the spent fuel shipping casks have been carried out by the Japan Society of Mechanical Engineers, Fire Research Institute, Ship Research Institute, Mitsubishi Atomic Power Industries,Inc. and also by Hitachi Ship Building Engineering Co. Ltd. Some of the results from those experiments have been already reported and the others are presented in this symposium.

In this paper the result obtained from " An Experimental Study on Thermal Performance of Spent Fuel Shipping Cask " conducted in 1973 is mentioned. The purpose of this test is to examine the effect of the test duration on the reduced scale of cask model. The temperature distributions and the temperature change behaviors during 5, 7, 10 and 30 minutes tests are compared and discussed. The tests shall be carried out also in 1974 using larger scale models.

INTRODUCTION

In Japan, 10 nuclear power stations (BWR 6, PWR 3, GCR 1) are now in operation or critical. The spent fuels from some of those stations were transported by sea to Windscale U.K. for reprocessing. On the other hand, the Japanese reprocessing plant at Tokai is expected to become in operation in late 1974. Some of the spent fuels from MIHAMA (PWR) and FUKUSHIMA (BWR) nuclear power stations shall be transported to the reprocessing plant for the test runs in near future. In the case of Japan all nuclear power stations are located on seashore and therefore it seems to be most convenient to transport by ship from the piers of stations to the nearest harbor to the reprocessing plant (Hitachi Harbor) and then on road between the plant and the harbor. Fig.1 shows the location of nuclear power stations and the Tokai reprocessing plant in Japan. Considering these situation, various preparations for transportation of spent fuels are now being made concretely.

In this paper, I would like to mention about the works for improving the Japanese National Standards for safe transport of spent fuels, design criteria for the transport ship and also about research and development works on shipping casks which have been carried out in these past years.

REVISION OF JAPANESE NATIONAL REGULATIONS FOR SAFE TRANSPORT OF RADIOACTIVE MATERIALS AND DESIGN CRITERIA OF TRANSPORT SHIP CARRYING PACKAGES

Just after IAEA recommended the 1973 revised edition of IAEA regulations for the safe transport of radioactive materials, the working party for looking over the 1968 Japanese national standards was established within the frame of the Atomic Energy Bureau in the Agency of Science and Technology. After the works have been finished on the end of 1973, the Atomic Energy Committee of Japan has begun to revise the old standards and reg-

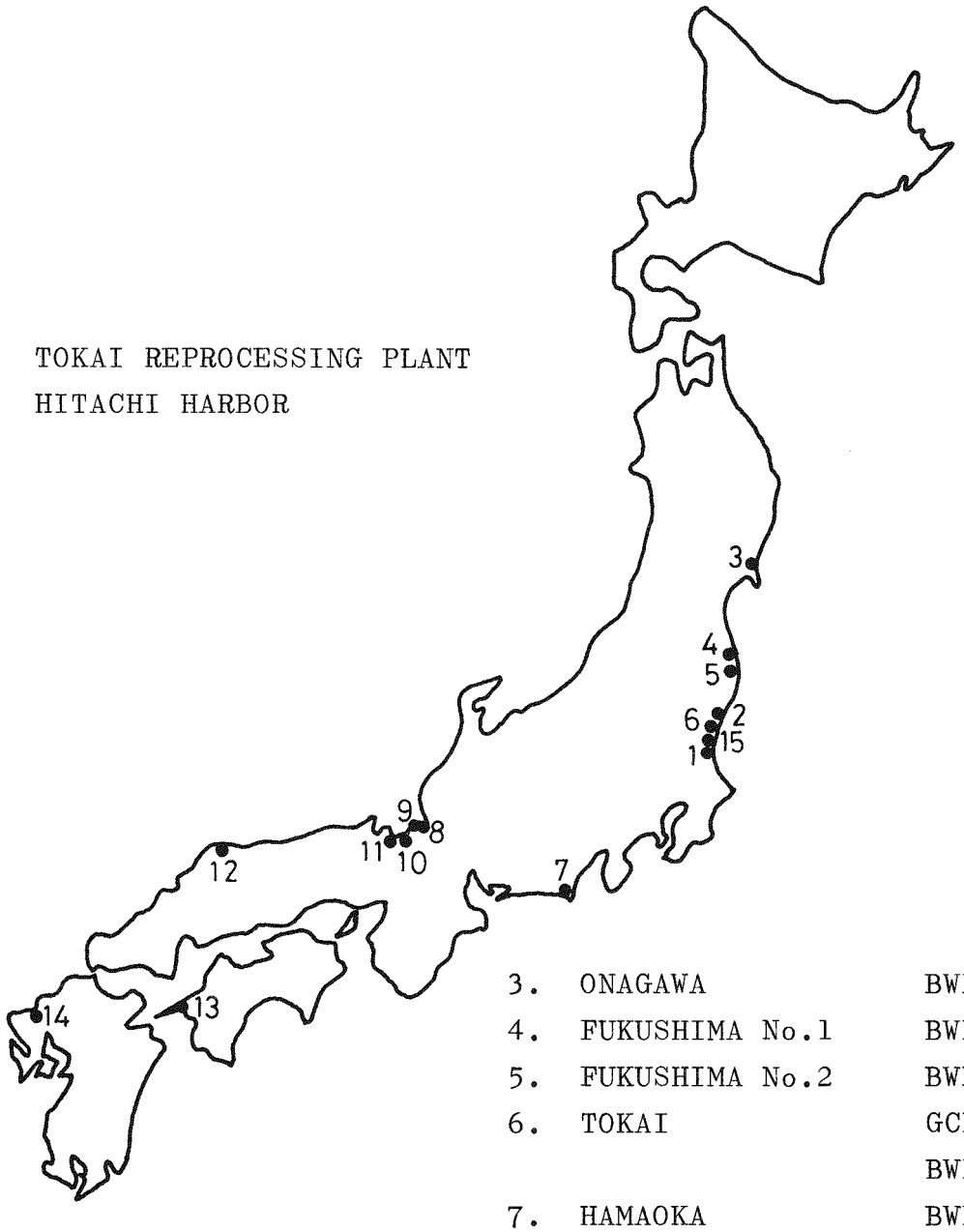


Fig. 1. Locations of Reprocessing Plant, Harbor and Nuclear Power Stations in Japan

3.	ONAGAWA	BWR	1
4.	FUKUSHIMA No.1	BWR	6
5.	FUKUSHIMA No.2	BWR	1
6.	TOKAI	GCR	1
		BWR	1
7.	HAMAOKA	BWR	1
8.	TSURUGA	BWR	1
9.	MIHAMA	PWR	3
10.	OHI	PWR	2
11.	TAKAHAMA	PWR	2
12.	SHIMANE	BWR	1
13.	IKATA	PWR	1
14.	GENKAI	PWR	1
(15.	JAERI	BWR	1)

ulations concerning the safe transport of radioactive materials. Especially in regard to the maximum permissible levels of non-fixed radioactve contamination and the activity limits for loss of radioactive contents at normal transport condition and also at accident condition, it is very important to make the standard conform to the allowable activity limits in environment which has been prescribed in the related national regulations for radioactive materials in Japan.

The other serious problem which has been discussed is the leakage of radioactive contents from spent fuel shipping casks which are dropped into the sea at harbor during loading and unloading and has sunk onto the deep sea bottom in collision accidents in the near region around the Japan Islands. The counterplans are now being investigated. For instance, one of them is the modification of shipping cask in order to prevent the increase of pressure difference between the ambient and the inside of cask by means of adopting a pressure balancing valve and the other one is the establishment of escort and accident provision for the transportation on water. Considering such situations the Ministry of Transport is studying the criteria of structure design for transport ship carrying packages of spent fuels, in order to make minimum the probability of sinking and also the radiation hazards at normal and accident conditions.

THE EXPERIMENTAL STUDY ON SPENT FUEL SHIPPING CASK

In order to obtain the basic data for shipping cask design, some experimental studies have been performed since 1967.

In 1967 the Japan Society of Mechanical Engineers (JSME) established a research subcommittee for design of spent fuel shipping cask to conduct a series of tests supported by the Atomic Energy Bureau. Those tests has been continued up to 1970 by JSME. Since 1971 Hitachi Ship Building & Engineering Co. Lt

has carried out two years and then again JSME has conducted the thermal test in 1973 and perhaps will continue it in 1974 too. The shielding test has been also done before and after the thermal tests by the Ship Research Institute. Also the Fire Research Institute has carried out the thermal tests of container using a furnace since several yearsago and some of them have been already reported in the past International Symposiums. A drop test of shipping container for PWR type new fuel assembly has been conducted by Mitsubishi Atomic Power Industries, Inc.

The tests which have been carried out in Japan up to date are as follows:

I. By the Japan Society of Mechanical Engineers;

1) AN EXPERIMENTAL STUDY ON THE HEAT REMOVAL FROM SPENT FUEL SHIPPING CASK (1967) [1,2]
2) AN EXPERIMENTAL STUDY ON THE DROP IMPACT OF SPENT FUEL SHIPPING CASK (I) (1968) [3,4]
3) AN EXPERIMENTAL STUDY ON THE DROP IMPACT OF SPENT FUEL SHIPPING CASK (II) (1969) [4]
4) AN EXPERIMENTAL STUDY ON THE DROP IMPACT OF SPENT FUEL SHIPPING CASK (III) (1970) [5]
5) AN EXPERIMENTAL STUDY ON THE THERMAL PERFORMANCE OF SPENT FUEL SHIPPING CASK (I) (1973) (to be mentioned in this paper)

II. By the Fire Research Institute;

1) STUDY ON FIRE RESISTIVITY OF RADIOACTIVE ISOTOPE CONTAINER (1965) [6]
2) TEST AND THEORETICAL ANALYSIS OF FIRE RESISTIVITY OF BETA-RAY THICKNESS GUAGE CONTAINERS (1968) [7]
3) STUDY ON FIRE RESISTIVITY FOR IRRADIATED FUEL SHIPPING CASKS (1971) [8]
4) STUDY ON FIRE RESISTIVITY FOR SPENT FUEL SHIPPING CASK (1973) (to be presented in this symposium)

III. By the Ship Research Institute;

1) GAMMA RAY SHIELDING EFFECT OF A LEAD SHIELD FOR THE RADIOACTIVE MATERIALS SHIPPING CASK AND ITS INTEGRITY INSPECTION

METHOD (1973) (to be presented in this symposium)

IV. By Hitachi Ship Building & Engineering Co. Ltd.;

1) AN EXPERIMENTAL STUDY ON PUNCTURE RESISTANCE OF SPENT FUEL SHIPPING CASK (1971) (to be presented in the symposium)

2) AN EXPERIMENTAL AND ANALYTICAL EVALUATION OF IMPACT LIMITERS (1972) (to be presented in the symposium)

V. By Mitsubishi Atomic Power Industries,Inc.;

1) DROP TESTS OF SHIPPING CONTAINERS FOR PWR TYPE NEW FUELS (1971) [9]

2) OVER-THE-ROAD TEST OF SHIPPING CONTAINER FOR PWR TYPE NEW FUEL (1971) [4]

AN EXPERIMENTAL STUDY ON THE THERMAL PERFORMANCE OF SPENT FUEL SHIPPING CASK

The purpose of thermal test was to measure the transient temperature distribution using the scale model under open fire condition and also to obtain the data concerning the scale effect at thermal test.

Alike the case of the experimental studies on drop impact of spent fuel shipping cask from 1968 to 1970, the prototype cask was postulated to be a horizontal cylinder type consisted of an inner shell (stainless steel),a lead-filled shielding and an outer shell (low carbon steel), having the weight of 80 t. At 1973's test the scale models were of 1.25 t in weight,which corresponded to a quater scale model of the prototype. Fig.2 and Table 1 show the dimension of scale model. The temperatures were measured with thermocouples at 46 points on and in the specimens. The ambient temperatures around the test specimen were also measured at 16 positions. These data were recorded every moments. Fig.3 shows the positions at which the temperature of specimen were measured.

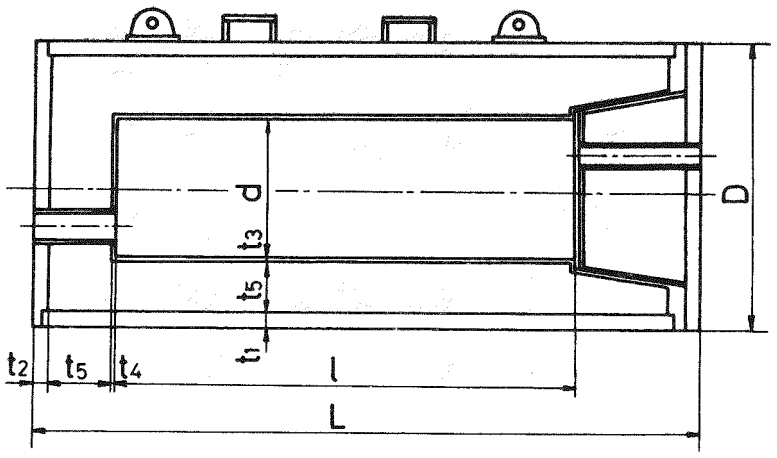
In order to investigate the effect of the test duration on lead-filled shielding and to ascertain the relation between the test duration and the reduced scale, 5, 7, 10 and 30 minute

were selected as the test duration for two scale models, one of which was used only for 30 minutes test. After the 10 minutes test and 30 minutes test were finished, the deformations of the test specimens were observed and the change of weight caused by the melting and expansion of lead was measured. The resolidification behaviors were examined by means of cutting the test specimens lengthways and in round slices. Fig. 4 shows the comparison of temperature behaviors on the outer surface at 5, 7, 10 and 30 minutes tests. Fig.5 shows also the comparison of temperature behaviors on the inner surface at every tests. Fig.6 shows the temperature changes at the outer and inner surface and also at the center of lead-filled shielding during 30 minutes test. The effectiveness of lead-filled shielding were inspected by means of Gamma-ray before and after the thermal test and its results were compared with the results by observation.

In 1974 the 1/2 scale model (10 t) and the 1/2.5 scale model (about 5 t) shall be tested. Using these results, we would like to select finally the optimum test duration for the thermal test with scale model. The details of the thermal tests shall be presented in next symposium after all tests shall have been finished.

ACKNOWLEDGEMENTS

The author wishes to express his gratitutde to the Atomic Energy Bureau for the financial support. He also wishes to express his thanks to the members of the Research Subcommittee for Thermal Test of Spent Fuel Shipping Cask in the Japan Society of Mechanical Engineers, who take parts in the test project,especially to the Fire Research Institute and also Kimura Kakoki Co. Ltd., who carried out the experiments.



	Scale	Weight(t)	D(mm)	d(mm)	L(mm)	l(mm)	t_1(mm)	t_2(mm)	t_3(mm)	t_4(mm)	t_5(mm)
MODEL-I	1/4	1.25	406$^{\phi}$	250$^{\phi}$	1438	1250	14	14	4	12	60
MODEL-II	1/4	1.25	406$^{\phi}$	250$^{\phi}$	1438	1250	14	14	4	12	60

Fig. 2. Table 1. Dimension of Scale Model

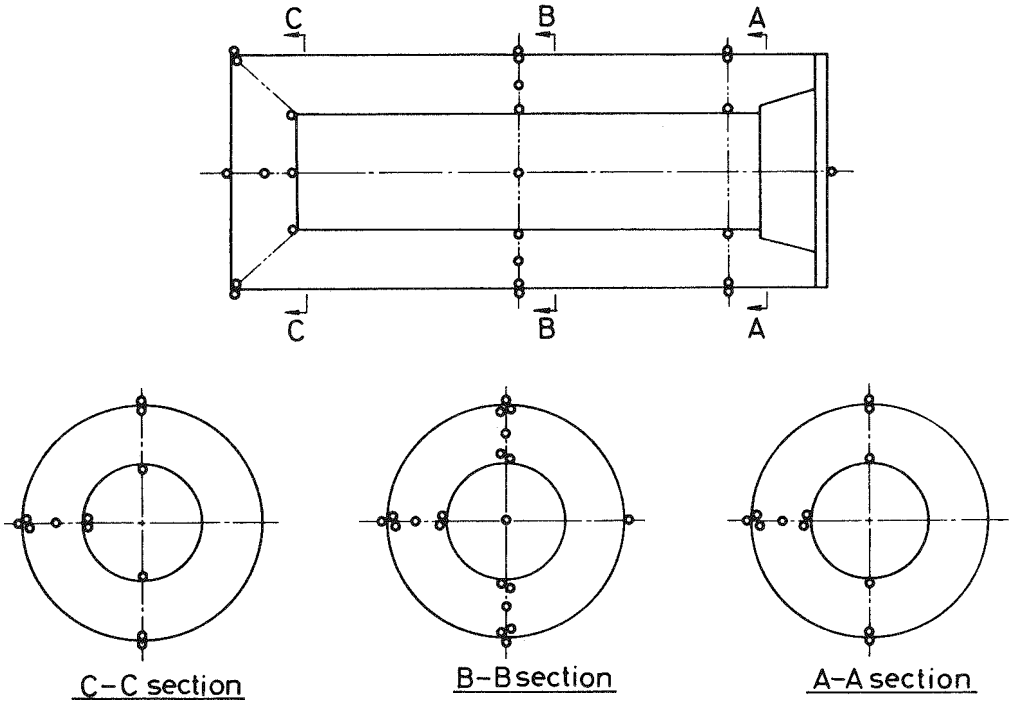


Fig. 3. Points of Temperature Measurement

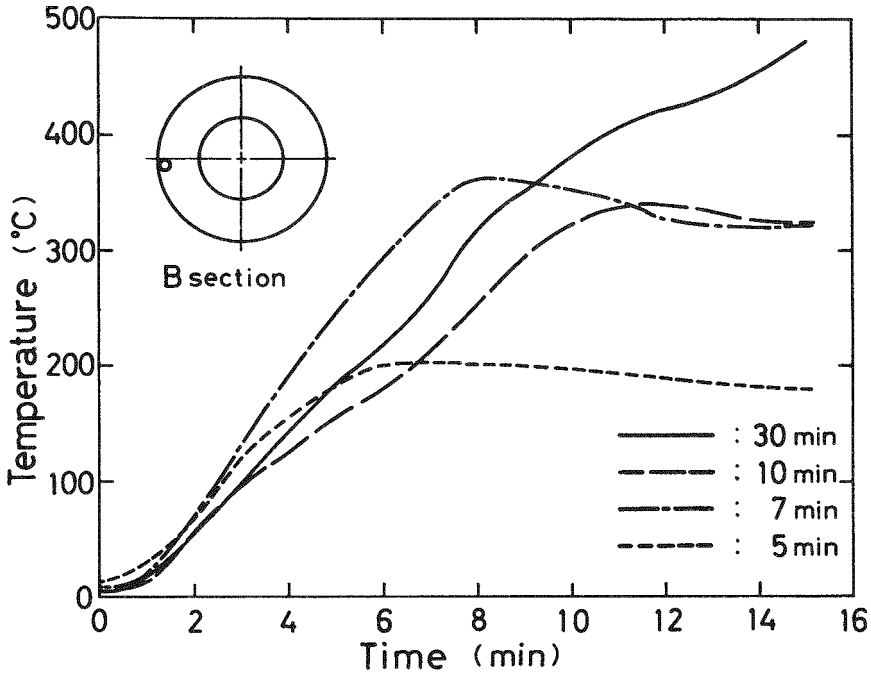


Fig. 4. Comparison between Temperature Changes of the Outer Surface during 5,7,10 and 30 min Tests

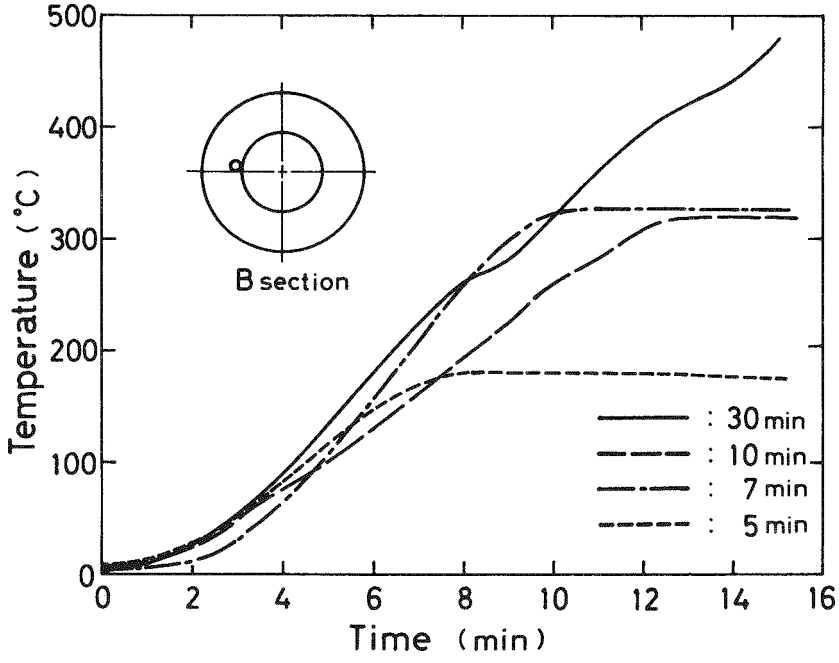


Fig. 5. Comparison between Temperature Changes of the Inner Surface during 5,7,10 and 30 min Tests

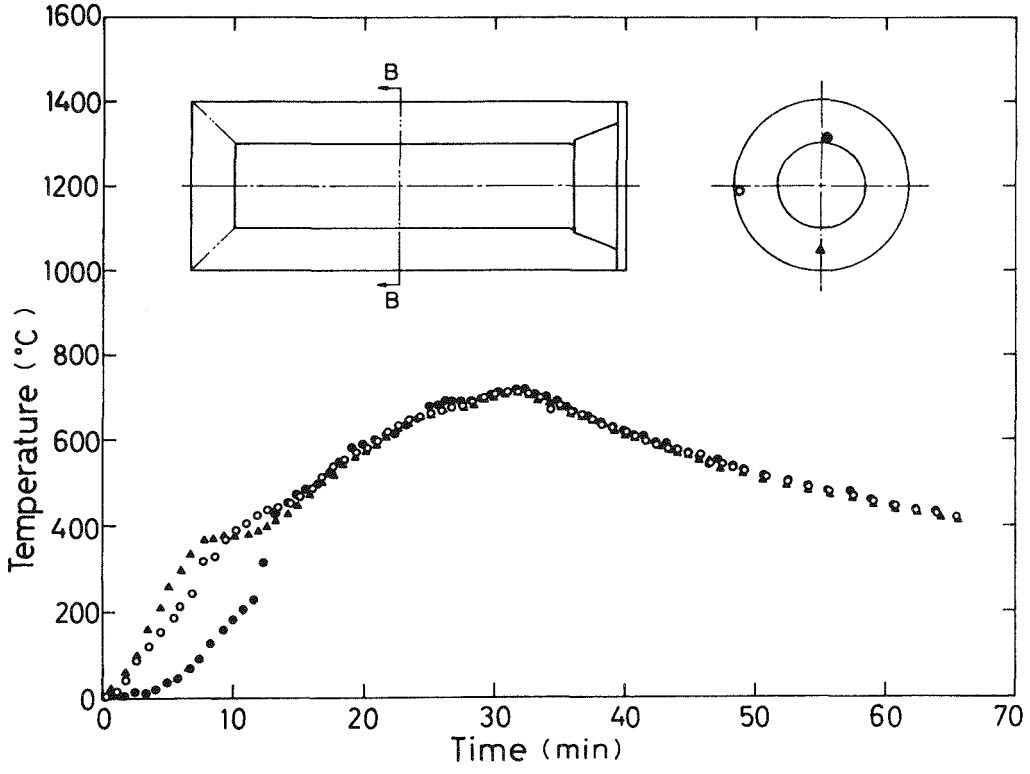


Fig. 6. Temperature Change Behavior at 30 min Test

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A TRANSPORTATION RESEARCH AND DEVELOPMENT PROGRAM

by
James A. Sisler

ABSTRACT

This paper explores the background and need for Research and Development (R&D) in the area of packaging and transportation of radioactive materials. Current and future R&D programs and related data gathering functions are identified in general terms and briefly discussed. Details and current results of most of these programs can be found elsewhere in these proceedings.

Starting in 1972, the AEC's Division of Waste Management and Transportation initiated a program in nuclear materials packaging and transportation R&D. Specific ongoing programs are described along with funding and manpower resources devoted to these programs. In addition, identified R&D programs of other AEC divisions are briefly described to provide a better perspective on the total AEC transport R&D.

INTRODUCTION & BACKGROUND

Many people will ask, "Why do we need research and development in an area that has had such a perfect safety record?" Others will state, "Although there have been no injuries so far in nuclear materials transportation, this is only freakish good luck and you must prove to me absolutely that there will never be an accident that will endanger any segment of the public, or I will never accept your statements that such transport is safe." The first group of people are hard to convince that any funds should be spent on R&D. The second is a growing group that will probably never be satisfied at any level of spending because no endeavor undertaken by man is ever absolutely safe. It is apparent, however, that studies must be undertaken to identify risk, and new technology developed to assure that the consequences of the inevitable accident are minimized to acceptable levels.

Defining an "acceptable level of risk" is one of our key problems. Obviously, man is willing to accept some risk because risk of death or injury is a constant fact of life. How can we agree on what is acceptable? Man's greatest fear is the fear of the unknown. As far as the public is concerned, the nuclear business is largely an unknown, therefore, something to be feared. We must then, it seems, find some way to educate the public to understand enough of the complexities of nuclear safety and hazards to agree on an acceptable risk. The only alternative I can see is to find a public spokesman that most will believe, and educate him. In these days of public distrust, and popularity of establishment critics, I doubt such a figure can be found. Perhaps some middle ground can be realized if enough of us who understand our problems will undertake the responsibility of answering criticism, and speaking eloquently on subjects of public concern. In any event, we in the Division of Waste Management and Transportation (WMT) believe that one of our most difficult responsibilities is finding ways to explain to the concerned

public what it is we are trying to achieve, and what are the real risks involved. Research for the advancement of transport technology will be of no value if the public is too frightened to accept its application. If we fail, I believe mankind will be the poorer for it, because it will then be unable to fully utilize all the enormous benefits which can be had by taming that fearsome, yet gentle giant, called "nuclear technology."

The Transport Safety Record

What is the record in nuclear transportation during the nuclear era, which started in the mid 1940's? Let's look briefly at the records compiled by AEC's Division of Operational Safety starting in 1949. First, we will need to define the six classifications of accidents or incidents that can be experienced in transportation of nuclear materials. These are:

Class I, Radiation Release - The vehicle has been involved in an accident or package damage is suspected. The shipment is delayed or stopped. No radioactive material is released and there is actually no loss of integrity to the package.

Class II, Radiation Release - The package integrity is breached. However, there is no release of radioactive materials.

Class III, Radiation Release - Radioactive material is released from the package but is confined to the vehicle.

Class IV, Radiation Release - Radioactive material is released to the ground or trafficway with no runoff or aerial dispersal.

Class V, Radiation Release - Radioactive materials is released, resulting in aerial dispersal.

Class VI, Radiation Release - Radioactive material is released and enters a watercourse, either directly or after spilling to the ground or trafficway.

The following three tables provide a summary of the distribution of the 182 accidents or incidents recorded.

Table 1 - Incident Experience By Year

Year	No. of Incidents	Year	No. of Incidents	Year	No. of Incidents
1949	1	1957	6	1965	18
1950	1	1958	5	1966	6
1951	1	1959	15	1967	6
1952	2	1960	12	1968	11
1953	0	1961	9	1969	13
1954	1	1962	14	1970	6
1955	2	1963	9	1971	4
1956	5	1964	15	1972	12
				1973	8

Table 2 - Incident Experience By Class Of Radiation Release
Number of Incidents

Class	1971-1973	1968-1970[a]	1965-1967[a]	1963-1964	1962	1957-1961	1949-1956
I	5	15	18	12	9	29	6
II	0	5	6	1	1	7	2
III	12[a]	4	4	6	2	7	0
IV	5	5	4	5	2	3	3
V	1	1	0	0	0	1	1
VI	0	0	0	0	0	0	1
TOTAL	23	30	32	24	14	47	13

[a]This number includes contaminated packages which may have been contaminated by some mechanism other than release.

Table 3 - Incident Experience By Type - Irrespective of Transport Mode (Unless Specified)

Type	1971 1973	1968 1970	1965 1967	1963 1964	1962	1957 1961	1949 1956
Handling	10	15	6	2	1	15	0
Impact (Collision - Truck)	1	5	13	10	4	10	7
Impact (Collision - Air)	0	1	0	0	0	0	1
Impact (Collision - Rail)	1	0	1	0	1	2	0
Fire	2	1	1	0	2	5	3
Vehicle or Equip. Failure	2	0	0	1	0	5	0
Leakage of Containers	2	5	7	5	3	4	0
Derailment	4	1	1	1	2	3	2
Tiedown of Bracing Failure	0	0	0	4	1	2	0
Aerial Dispersal	1	0	0	0	0	1	0
Packaging Failure	0	2	3	1	0	0	0

Examination of the preceding tables will reveal several significant facts. First, although the annual number of shipments of nuclear material has grown from essentially zero in the 1940's to approximately one million a year by 1973, there has been no corresponding increase in incidents or accidents. Second, only Class V or Class VI incidents have much potential for injury to the general public. You will note that only four Class V and one Class VI incidents have occurred in 25 years. The consequences of these incidents were very limited and no injuries resulted. Such a record is not completely dependent on good luck. It was the result of constant attention to safety. As the number of shipments has grown, the safety effort in terms of new technology development, better safety standards, and funding has increased.

Shipment Trends

As mentioned earlier, shipments of nuclear materials have increased drastically in recent years. If you will examine some of the other papers in these proceedings, you will discover that all projections point to further drastic growth in shipment numbers. I will not repeat these numbers here. It is important to note, in order to keep your perspective,

that there are over 30 billion shipments each year of other hazardous materials. That is about one shipment of nuclear materials for every 30,000 shipments of other hazardous materials. The Department of Transportation estimates that there are about 45,000 carriers and 11,000 shippers of hazardous materials, at 82,000 locations, who handle two billion tons of hazardous materials each year. Only a tiny fraction of that amount consists of radioactive materials. If you consider that about 90-95% of all nuclear shipments are radiopharmaceuticals, the remaining total, which includes all of the potentially dangerous shipments, is still a very small number. I am not including a detailed list of nuclear material shipments because such a list does not exist to my knowledge. The radiopharmaceutical business is rather competitive and accurate shipment figures are difficult to obtain. If anyone has these numbers, I would like to know about it.

Why A Research and Development Program

In essence then, the five primary reasons why we have an R&D program are as follows:

1. To understand and define in some logical manner the hazards in transport of nuclear materials.
2. To understand the transport environment and keep ahead of a rapidly expanding business by constantly improving our standards to meet requirements.
3. To improve technology to reduce risks to as low as practicable, and to lower costs. Accidents and excessive safety are both expensive.
4. To protect the environment from any adverse effects of nuclear transport.
5. To satisfy public demand for clear and concise information which will permit them to determine whether they are willing to accept the small but finite risk involved in nuclear materials transport, and to provide assurance that this risk is as low as practicable.

Early Nuclear Transport R&D

Prior to 1972 and the formation of WMT, all AEC R&D in nuclear packaging and transport was the responsibility of the particular AEC Program Division concerned with making the shipments. This is still true to some degree. As an example, the Reactor Research and Development Division is involved in R&D in LMFBR fuel shipments, Military Application has responsibility for weapons, Naval Reactors looks after Navy reactor fuels, etc. WMT is concerned with transport R&D programs that concern more than one division or AEC as a whole.

Prior to 1972, many millions of dollars were spent by AEC Program Divisions on packaging technology, transport safety, and other related studies. Some of this work was, and still is, classified. However, most of the technology development has now been reported in the unclassified literature. That technology which is still classified is related to military programs.

CURRENT WMT TRANSPORT R&D PROGRAM

Objective

The ultimate objective of this program is the development of new technology and a comprehensive body of standards encompassing the needs for packaging, materials handling equipment, procedures, and systems related to the safe transport of radioactive materials. We will examine critically the adequacy of existing standards to assure safe transport without excessive and expensive conservatism which is solely compensation for deficiencies in technical knowledge. We will seek ways to integrate AEC developed standards and systems into the commercial transportation industry, and last but not least, we will seek new ways to overcome the knowledge barrier that is now so prominent between nuclear materials transport safety technology on one hand, and the concerned public and segments of the transport industry on the other. To state all of this in simple terms, we are trying to determine the best way to safely ship nuclear materials between two points in a manner the public and transport industry will accept.

Current Programs

The following WMT programs are current for fiscal year (FY) 1974. Although descriptions for each program are brief, more detail is available on most programs elsewhere in these proceedings.

Packaging Specification Development

1. Contractor - Oak Ridge National Laboratory (ORNL)
2. Starting Date - Late FY 1973
3. Target for Completion - Continuing
4. Funding - FY 1974 - $100,000 -- FY 1975 - $100,000 (estimated)
5. Summary - This program will evaluate currently used AEC package designs having wide acceptance and usage. From these designs a family of standardized package specifications will be developed suitable for incorporation into the DOT and IAEA regulations. Technological improvements will be incorporated in the new standards as needed.

Development of Packaging Technology for Plutonium Nitrate

1. Contractor - Battelle Pacific Northwest Laboratories (PNL)
2. Starting Date - Late FY 1973
3. Target for Completion - FY 1976
4. Funding - FY 1974 - $50,000 -- FY 1975 - $100,000 (estimated)
5. Summary - This program is studying the feasibility of reducing plutonium nitrate solutions to a solid by dehydration techniques such as vacuum drying. This should minimize the evolution of gas during transport, thereby simplifying package design, and permitting larger quantities of plutonium per package while still permitting the receiving facility to easily convert back to a liquid of any acid concentration.

Physical Parameters of Transportation Accidents

1. Contractor - Sandia Laboratories, Albuquerque (SLA)
2. Starting Date - Late FY 1972
3. Target for Completion - FY 1975
4. Funding - FY 1973 - $130,000 -- FY 1974 - $50,000 -- FY 1975 - $25,000 (estimated)
5. Summary - This AEC/DOT jointly funded study is determining the physical parameters affecting cargo packages under a variety of severe transportation accident conditions. The study is examining the type of physical forces involved in transport accidents which cause package damage and identifying, if possible, test criteria to duplicate the accident damage. A comparison will then be made between: (1) forces resulting from the regulatory accident damage test; and (2) forces resulting from transport accidents. The study must determine if the two types of forces are, in fact, comparable. If they are not, then the differences will be identified with recommendations being made for modification of the regulatory test standards.

Radiological Risk Associated with the Transportation of Radioactive Material

1. Contractor - PNL
2. Starting Date - FY 1973
3. Target for Completion - FY 1976
4. Funding - FY 1973 - $30,000 -- FY 1974 - $43,000 -- FY 1975 - $120,000 (estimated)
5. Summary - This study has developed a risk computational model for risk evaluation in the transport of nuclear materials. After constructing fault trees, defining accident environment (frequency vs. severity), determining package behavior to the accident environment, and ranking risk of package releases; an assessment will be made of the level of risk associated with various transport modes, types of material, material form, and specific transport routes. Because of the gross lack of definitive data in many areas affecting the study, this first risk assessment may require further refinement. One major value of the study will be in identifying the level of accuracy possible with current data.

Testing of Large Obsolete Casks

1. Contractor - ORNL
2. Starting Date - Late FY 1973
3. Target for Completion - FY 1976 or 1977
4. Funding - FY 1974 - $100,000 -- FY 1975 - $100,000 (estimated)
5. Summary - Large shipping casks are very expensive to design, analyze, and fabricate. Very little actual accident-damage test data exists. To avoid expensive over-design and still meet all safety criteria

more precision is needed in our safety analysis techniques. This program will utilize obsolete large casks for a comparative study between predicted cask response, and actual test results obtained by subjecting these casks to the regulatory package damage tests. Knowledge gained at this point will be used to improve analytical techniques.

Commercial Radioactive Waste Fixation Program

1. Contractor - PNL
2. Starting Date - FY 1973
3. Target for Completion - Continuing
4. Funding - FY 1973 - $897,000* -- FY 1974 - $2,400,000* -- FY 1975 - $2,665,000* (estimated)
5. Summary - One phase of the Commercial WFP Program (WFP-4) is a safety and systems evaluation of interim storage, conditioning, transportation, retrievable surface storage, emplacement in disposal scheme, and final disposal of high-level radioactive waste. This work will include a risk analysis of the transportation steps in the system with emphasis on transport of glass and calcine waste forms. This program will utilize methodology developed under the PNL risk assessment program.

*These are program totals and do not reflect the small portion devoted to transportation risk analysis.

Specification 7A Package Certification to Meet DOT Requirements

1. Contractor - Monsanto Research Corporation, Mound Laboratory (MRC)
2. Starting Date - FY 1974
3. Target for Completion - FY 1975
4. Funding - FY 1974 - $10,000 -- FY 1975 - $22,000 (estimated)
5. Summary - The DOT proposed a regulation in HM-111, Notice 73-7, to require: "Each shipper of a Spec. 7A packaging must maintain on file for at least one year after the latest shipment, and be prepared to provide the Department, a complete certification and supporting safety analysis demonstrating that the construction methods, packaging design, and materials of construction are in compliance with the specification." Other Type A packagings will no longer be authorized. This program is intended to provide such DOT certifications for all AEC Spec. 7A packagings. This will avoid the duplication of effort involved in each contractor doing his own certifications.

Impact Analysis of a Spent Fuel Cask

1. Contractor - SLA
2. Starting Date - FY 1974
3. Target for Completion - FY 1974
4. Funding - FY 1974 - $25,000

5. Summary - Members of the public, certain common carrier groups, and others have repeatedly questioned the ability of radioactive material shipping containers to retain their contents following an extreme transportation accident. Therefore, this study will perform an impact analysis involving a railroad train moving 80 mph and a common spent fuel cask which is mounted on a truck parked at a grade crossing.

Note: The following ongoing WMT programs are directed toward the integration of AEC-developed packaging systems and transport standards into the commercial transport system including the necessary public acceptance of this integration. Current efforts are directed toward development of needed management tools required to accomplish such a program including a statistical transport data program, and transport-related educational aids.

Development of Statistical Transport Data Program

1. Contractor - ORNL
2. Starting Date - FY 1974
3. Target for Completion - FY 1974
4. Funding - FY 1974 - $20,000
5. Summary - This study will develop a statistical transportation data bank on the AEC/contractor transportation program. This data bank will provide a basis for determining the nature (traffic pattern, number of shipments, shipping costs, etc.) of AEC/contractor freight shipments of radioactive materials and other commodities which are transported by commercial for-hire transportation companies. The data bank also will provide periodic shipment data summaries and a history of AEC/contractor's transportation shipment experience.

Transporting Radioactive Cargoes Exhibit

1. Contractor - Oak Ridge Associated Universities (ORAU)
2. Starting Date - Late FY 1972
3. Target for Completion - Continuing
4. Funding - FY 1973 - $10,000 -- FY 1974 - $14,000 -- FY 1975 - $15,000 (estimated)
5. Summary - The exhibit is designed to be used primarily at convention sites. Its purpose is increased awareness of the present and potential requirements for the transport of radioactive materials. It emphasizes the AEC's continuing concern for the safety of every citizen and the protection of our environment. It provides WMT with a means for identification of areas of public concern in the transport of nuclear materials.

Transportation of Radioactive Materials Film

1. Contractor - SLA
2. Starting Date - FY 1973
3. Target for Completion - FY 1974

4. Funding - FY 1973 - $30,000 -- FY 1974 - $15,000

5. Summary - This program will develop and produce a 16mm color-sound educational motion picture which: (1) is understandable to the average layman; (2) will explain the complex safety procedures involved in design, development, and approval of packaging for radioactive materials; (3) will provide enough explanation about the nature of radioactive materials to dispel the "mushroom cloud" syndrome and explain the reasons for different levels of shielding; (4) will explain how and why radioactive materials are shipped by all modes of commercial transport worldwide; and (5) will explain why the shipment of radioactive materials will, of necessity, increase in the future.

OTHER AEC TRANSPORT R&D PROGRAMS

Nuclear Weapon Safe Secure Trailers

1. Contractor - SLA

2. Starting Date - FY 1970

3. Target for Completion - Design and fabrication work is complete. Further improvements are undergoing evaluation.

4. Funding - FY 1972 - $2,693,000 (Accumulated total thru FY 1972) -- FY 1973 - $612,000 -- FY 1974 - $795,000 -- FY 1975 - $256,000 (estimated)

5. Summary - This is a program to develop, manufacture hardware, and provide improvements in a safe secure highway transport system for nuclear weapons.

Nuclear Weapon Accident Resistant Container (ARC)

1. Contractor - SLA

2. Starting Date - FY 1972

3. Target for Completion - Open

4. Funding - FY 1972 - $798,000 -- FY 1973 - $1,216,000 -- FY 1974 - $1,060,000 -- FY 1975 - $3,074,000 (estimated)

5. Summary - This is a program to develop an accident resistant container suitable for air transport of nuclear weapons. The ARC is designed to survive aircraft accidents.

Nuclear Weapon Helicopter Accident Resistant Containers (HARC)

1. Contractor - SLA

2. Starting Date - FY 1974

3. Target for Completion - Open

4. Funding - FY 1974 - $500,000 -- FY 1975 - $840,000 (estimated)

5. Summary - This is a program to develop an accident resistant container suitable for helicopter transport of nuclear weapons.

The HARC is designed to survive helicopter accidents.

ATMX Rail Cars

1. Contractor - SLA
2. Starting Date - FY 1972
3. Target for Completion - Open
4. Funding - FY 1972 - $54,000 -- FY 1973 - $12,000 -- FY 1974 - $95,000 -- FY 1975 - $840,000 (estimated)
5. Summary - This is a program to improve safety and security in the rail shipment of nuclear weapons.

SECOM (Security Communications System)

1. Contractor - Scientific Radio Systems Inc. (Operator is AL)
2. Starting Date - FY 1972
3. Target for Completion - Continuing
4. Funding - FY 1972 - $925,000 -- FY 1973 - $208,000 -- FY 1974 - $13,000 -- FY 1975 - $5,000 (Modification cost is estimated to be $288,000)
5. Summary - The SECOM is a network of five base stations, each hard wired to AL for remote operation. Provides 48 state coverage for both HF voice and digital communications in the weapons program.

Special Nuclear Material (SNM) Vehicle Concept Development

1. Contractor - SLA
2. Starting Date - FY 1973
3. Target for Completion - Open
4. Funding - FY 1973 - $43,500 -- FY 1974 - $100,000 -- FY 1975 - $150,000 (estimated)
5. Summary - This program will develop a concept for an unclassified secure motor vehicle which may be used to transport safeguards material over the highway. The concept includes penetration resistant vehicle walls, and immobilization systems. Funding for construction of a prototype is included in the FY 1975 funds.

Communications Systems for SNM Special Vehicles

1. Contractor - SLA
2. Starting Date - FY 1973
3. Target for Completion - Open
4. Funding - FY 1973 - $41,000 -- FY 1974 - $210,000 -- FY 1975 - $550,000 (estimated)
5. Summary - This experimental program is examining the feasibility of various two-way radio communications equipment for the purpose of developing a constant communications capability for transport vehicles

carrying safeguards material anywhere in the contiguous 48 states.

Spent Fuel Shipping Study

1. Contractor - ORNL and Hanford Engineering Development Laboratory (HEDL)
2. Starting Date - FY 1972
3. Target for Completion - FY 1975
4. Funding - FY 1973 - $75,000 -- FY 1974 - $75,000 -- FY 1975 - $25,000 (estimated)
5. Summary - The objective of this study is to examine alternative approaches to shipping nuclear power fuels and wastes in such a manner as to minimize radiation exposures to workers and the public. Information developed in the study will be helpful in establishing procedures and standards for container R&D programs for advanced reactors.

   The projected nuclear fuel requirements in the U.S. for the Year 2000 will be utilized as the base case for comparing the alternative approaches. Among the assumptions selected for examination are: the establishment of small and large nuclear parks, shipping by barge when water routing is possible, and removal of some fission products prior to shipping.

LMFBR Fuel Shipping Task of ORNL Aqueous Reprocessing Program

1. Contractor - ORNL
2. Starting Date - FY 1966
3. Target for Completion - Open
4. Funding - FY 1972 - $400,000 -- FY 1973 - $225,000 -- FY 1974 - $50,000 -- FY 1975 - $50,000 (estimated)
5. Summary - This task performed parametric trade studies of spent fuel shipping concepts for optimum LMFBR fuel cycle. Experimental work was performed to support a sodium filled cask concept. Reports were issued covering cask seals, heat transfer, structural and shock absorber testing. A conceptual design for an LMFBR shipping cask was completed.

FFTF/Demonstration Plant Cask Design

1. Contractor - Aeroject Manufacturing Company
2. Starting Date - FY 1971
3. Target for Completion - FY 1974
4. Funding - FY 1973 - $170,000 -- FY 1974 - $250,000 (Reflects redesign cost due to change in DEMO LMFBR fuel assembly length) -- FY 1975 - Residual licensing support only (less than one man year)
5. Summary - Provides design, Safety Analysis Report, procurement, and technical interaction for cask regulatory approval. First SAR submittal scheduled for March 1974.

Cask procurement, based on competitive bidding will aim for cask delivery in the December 1978 time period.

RRD Fuel Shipping Program

1. Contractor - Savannah River Operations Office
2. Starting Date - FY 1969
3. Target for Completion - Continuing Program
4. Funding - Annual Level of Effort Funding: $100,000 Operational Services, $200,000 Technical Support
5. Summary - Provides technical support and operational services in packaging and transport of RRD fissile and radioactive fuels, materials, and experiments.

Individual tasks in support of this program are as follows:

1. Develop and test "special form" inner containers for use in transport of irradiated fuel test pins from the HFEF facility inside existing shipping casks servicing that facility.
2. Develop detailed designs for a series of standard shipping containers for irradiated experiments.
3. Assistance in transportation management when requested by RRD contractors in meeting their responsibilities for proper packaging and transportation of fissile or irradiated fuels.
4. Performance of RRD off-site fuel shipments when requested and assigned by RRD.
5. Prepare RDT Standards for packaging and transportation of RRD programmatic fissile and radioactive material:

   E12-1 Container for Transporting Unirradiated Fuel Pins and Assemblies

   E12-5 Use and Handling of Radioactive Materials Shipping Containers

   E12-6 Design of Ancillary Features of Radioactive Materials Shipping Containers

   E12-7 Procedures for Inspection and Preventive Maintenance of Radioactive Materials Shipping Containers

   E13-14 Procedure for Fragility Rating of Unirradiated Fuel Pins and Assemblies with Respect to Transport

   F8-9 Design Basis for Resistance to Shock and Vibration in Truck Transport

   F8-10 Transportation Management of R/A Materials

   F8-11 Tie-Down for Transport of R/A Material Containers by Truck

   F8-12 Tie-Down for Transport of R/A Material Containers by Rail

RDT Standards Program at ORNL

1. Contractor - ORNL
2. Starting Date - FY 1971
3. Target for Completion - Open
4. Funding - The two listed tasks are a small part of a $1,500,000 standards program.
5. Summary - The following two tasks in transportation standards are a small part of a large ORNL program to administer all and develop some RDT standards. Of approximately 300 standards under development or completed, 11 deal with transportation standards. These 11 standards include the two below and nine listed under the RRD Fuel Shipping Program.

   E12-4 Shielded Shipping Cask for Reactor Fuel Elements (Approved June 1973)

   F8-7 Transportation of Critical Components and Equipment (For other than radioactive material a draft standard has been prepared. Disposition is to be determined.)

Assessment of Radiation Exposures in Aircraft

1. Contractor - FY 1973 - Northwestern University
   FY 1974 - Harvard University
   FY 1975 - To be selected
2. Starting Date - FY 1973
3. Target for Completion - FY 1975
4. Funding - FY 1974 - $4,000 -- FY 1975 - $10,000 (estimated)
5. Summary - Each year about 800,000 packages containing radioactive material are transported in the United States, most of them by passenger carrying aircraft. The objective of this study is to assess the total annual radiation doses received by the passengers and crew and the maximum doses received by individual passengers or crew members.

Assessment of Radiation Exposures to Transport Workers

1. Contractor - States of New Jersey, Oregon, Minnesota, and City of New York
2. Starting Date - FY 1974
3. Target for Completion - Open
4. Funding - FY 1974 - $10,000 -- FY 1975 - $25,000 (estimated)
5. Summary - There has been little or no regular and systematic surveillance or enforcement activity and no information collected on a regular basis as to the adequacy of packages relative to design standards and labeling requirements, the state of their physical upkeep, or the level of radiation from packages and resultant exposures of transport workers. These studies constitute a pilot program with several states to furnish surveillance information that

will become part of the basis for improved transport packaging requirements.

Drop Test Engineering Analysis

1. Contractor - Los Alamos Scientific Laboratory (LASL)
2. Starting Date - FY 1973
3. Target for Completion - Open
4. Funding - FY 1974 - $30,000 -- FY 1975 - $45,000 (estimated)
5. Summary - The purpose of this program is to develop improved methods of analyzing the structural response of shipping containers under impact loading conditions. The results will enable the REG staff to more accurately determine the extent to which a package is damaged by the accident test conditions and will increase the efficiency of the safety assessment for increasingly larger and more complex shipping containers.

Structural Analysis of Shipping Containers

1. Contractor - Naval Ordnance Laboratory (NOL)
2. Starting Date - FY 1973
3. Target for Completion - Open
4. Funding - FY 1974 - $30,000 -- FY 1975 - $40,000 (estimated)
5. Summary - This program provides the Regulatory staff with expert structural and mechanical engineering consultants to assist with special problems that may arise during the course of shipping container safety reviews. The program also provides the REG staff with access to advanced and specialized computer codes which can be used to solve complex structural engineering problems or to independently assess the integrity and safety of shipping containers.

Computer Assistance

1. Contractor - ORNL
2. Starting Date - FY 1972
3. Target for Completion - Open
4. Funding - FY 1974 - $50,000 -- FY 1975 - $70,000 (estimated)
5. Summary - This program provides expert assistance in developing and modifying computer codes that are used to assess the thermal, shielding, and criticality aspects of specific designs of shipping containers with respect to compliance with the standards and adequacy with respect to safety.

PROPOSED NEW R&D FOR FY 1975

WMT Programs

The WMT programs listed below are proposed for initiation in FY 1975.

Development of Safer Drum Lid Closures

1. Contractor - SLA
2. Starting Date - FY 1975
3. Target for Completion - FY 1975
4. Funding - FY 1975 - $80,000 (estimated)
5. Summary - The standard DOT specification drum lid closures used in many radioactive materials packaging designs have proven in numerous tests to be the weakest point in the design. This program will be an effort to improve transport safety by development of a new lid design that will be at least as strong as the bottom of the drum, and less susceptible to operator error than current locking ring designs.

Development of a System for the Collection, Storage, and Retrieval of Transportation Accident Environmental Data

1. Contractor - SLA
2. Starting Date - FY 1975
3. Target for Completion - Continuing
4. Funding - FY 1975 - $85,000 (estimated)
5. Summary - R&D studies now being funded are developing large quantities of transport accident environmental data. Other ongoing studies such as risk analysis are finding that there is still a serious shortage of good quality data. To avoid loss of valuable R&D resources once a program is complete, to locate new data sources, and to identify needs for new data collection, a system will be developed to collect and store transportation accident environmental data in a form which is retrievable for future studies.

Training Film - Safe Packaging of Radioactive Materials

1. Contractor - SLA
2. Starting Date - FY 1975
3. Target for Completion - FY 1975-76
4. Funding - FY 1975 - $50,000 (estimated)
5. Summary - AEC has an unexcelled safety record in the transport of radioactive materials over the past 25 years. However, AEC/REG feels that with a constantly increasing volume of shipments, the threat to public safety from operator error in package assembly will become a serious problem. There seems to be general agreement that operator error can be a problem and should not be ignored. New package approval procedures, documentation, quality assurance requirements, and transport regulatory requirements have all appeared in recent months. All of the new requirements will necessitate some training of packaging personnel. New employees are also a constant training problem. It is, therefore, evident that a high quality package assembly training film is needed. This program will develop the script and produce a 16mm color-sound motion picture suitable for such training.

Other Proposed Programs

At the time this paper was written, I was able to identify two other R&D programs proposed by the AEC Directorates of Licensing and Regulatory Standards. These are listed below. There will probably be a few additional programs initiated by other AEC divisions, but these were not yet developed to the point where it was feasible to report them.

Shielding Requirements and ALAP

1. Contractor - To be selected
2. Starting Date - FY 1975
3. Target for Completion - Open
4. Funding - FY 1975 - $15,000 (estimated)
5. Summary - Application of the as-low-as-practicable principle to shipments of radioactive materials needs to be assessed in terms of costs vs. benefits. This study will analyze alternative methods for reducing the allowable radiation levels on the outside of packages, including additional shielding, larger containers, and smaller allowable activity of contents.

Packaging Standards and Specifications

1. Contractor - To be selected
2. Starting Date - FY 1975
3. Target for Completion - Open
4. Funding - FY 1975 - $50,000 (estimated)
5. Summary - This subprogram will assess current packaging standards and specifications for achieving as-low-as-practicable levels of radiation and radiation exposure in transport.

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TRENDS AND PROJECTED SHIPMENTS IN THE NUCLEAR FUEL CYCLE INDUSTRY TO THE YEAR 2000*

Royes Salmon, J. O. Blomeke, and J. P. Nichols

ABSTRACT

Projections of the production, shipment, and accumulation of nuclear fuel, plutonium, and radioactive waste from 1974 to 2000 are presented. The nuclear power reactor complex is assumed to consist of LWRs using enriched uranium and plutonium recycle fueling, HTGRs, and LMFBRs. Installed nuclear electric capacity is taken as 102, 500, and 1200 GW(e) at the ends of calendar years 1980, 1990, and 2000, respectively, with an LMFBR capacity of 200 GW(e) at the end of year 2000.

INTRODUCTION

This paper presents some projections for the nuclear fuel industry with emphasis on shipments of fuel, plutonium, and various types of wastes. Amounts of radioactive wastes accumulated at repositories also are shown. Earlier projections, based on the AEC's "most likely" forecast of the growth of nuclear power as shown in WASH-1139(72),[1] were reported in ORNL-TM-3965 (February 1974).[2] In this paper, these projections are updated in accordance with more recent AEC forecasts. The forecast we used was the AEC Office of Planning and Analysis intermediate case B, which is the higher of their two intermediate cases.[3] Table 1 shows the projected growth of nuclear power for case B. Generating capacity is 102 GW(e) by the end of 1980, 500 GW(e) by the end of 1990, and 1200 GW(e) by the end of 2000. Also shown are the estimated capacities of LWRs, HTGRs, and LMFBRs at each of these time points. Plutonium recycle in LWRs was assumed to take place from 1977 to 2000. HTGR growth shows a fairly successful market penetration of about 123 GW(e) by year 2000. LMFBR growth starts with commercial introduction in 1988 and reaches 200 GW(e) installed capacity at year 2000.

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

Table 1. Projected Growth of Nuclear Power, GW(e)[a]

	LWR	HTGR	LMFBR	Total
1980	101	1		102
1990	449	47	4	500
2000	877	123	200	1200

[a]Based on AEC Office of Planning and Analysis Intermediate Case B.

Reactor mass balance data for the LWR and HTGR were taken from ref. 2. For the LMFBR, we used the early oxide mass balance developed by Hanford Engineering Development Laboratory for a recent cost-benefit study. The breeding ratio is 1.15.

We inserted these reactor growth rates and reactor mass balance data into the KWIKPLAN code to develop the annual fresh fuel fabrication and spent fuel reprocessing rates for each reactor type for each year from 1970 to 2000. Table 2 shows the quantities fabricated in the years 1980, 1990, and 2000. Table 3 shows the quantities of spent fuel reprocessed in each of these years. Next, using the ORIGEN code,[4] we calculated the radioactivity in the spent fuels at various decay times. This formed the basis for estimates of the radioactivity in various types of waste and of the amounts of radioactivity accumulated at waste repositories over extended periods of time.

The remainder of this paper is in two parts. The first presents a summary of the projected shipments of fresh fuel, spent fuel, and plutonium. The second covers shipments and accumulations of the various types of waste.

Table 2. Fuel Fabrication Loads[a]

	LWR-U	LWR-Pu	HTGR	LMFBR	Total
1980	3,950	260	30		4,240
1990	12,500	1,500	520	210	14,730
2000	22,100	1,780	1,030	4,620	29,530

[a]Metric tons of heavy metal per year.

Table 3. Fuel Reprocessing Loads[a]

	LWR-U	LWR-Pu	HTGR	LMFBR	Total
1980	1,400	220	2		1,622
1990	7,000	1,340	190	20	8,550
2000	17,500	1,400	710	1,850	21,460

[a]Metric tons of heavy metal per year.

FUEL AND PLUTONIUM SHIPMENTS

Table 4 shows projected shipments of fresh fuel for each of the reactor types. The number of shipments required per year was based on estimated vehicle loads of 5.0, 1.0, and 2.8 metric tons of heavy metal for the LWR, HTGR, and LMFBR respectively. The average number of vehicles in transit at a given time was based on the estimated transit times shown for shipment from the fabrication plant to the reactor.

Table 5 shows the same type of projections for spent fuel shipments. Estimates were made of both truck and rail shipments for LWR fuel, with the fraction shipped by rail gradually increasing because of its lower cost and greater capacity. The quantity of radioactivity per shipment was based on cooling times of 150, 365, and 90 days for the LWR, HTGR, and LMFBR, respectively.

Projected shipments of plutonium are shown in Table 6. Loads per vehicle were based on truck capacities of 0.1 metric ton of plutonium when shipped as nitrate solution and 0.5 metric ton of plutonium when shipped as the oxide. It was assumed that shipment as the oxide will account for most of the shipments by 1980 and all of the shipments in 1990 and thereafter.

In making these projections it was assumed that spent fuel discharged from reactors is shipped off-site to a central reprocessing plant. If the industry develops around nuclear power centers with on-site fabrication and reprocessing facilities, this would reduce the off-site shipments accordingly.

SHIPMENT AND ACCUMULATION OF WASTES

To estimate the future magnitude of waste management operations, fuel cycle process flowsheets were constructed for each reactor, based on present or near-future technology. Ten types of radioactive wastes were defined on the basis of their characteristics and of their packaging, shipping, and disposal requirements. For each of these types, annual rates of generation, annual number of shipments, and total amount accumulated were calculated through the year 2000.

Table 4. Fresh Fuel Shipments

	1980	1990	2000
LWR			
Metric tons/year	4200	14,000	24,000
Load per vehicle, metric tons	5.5	5.5	5.5
Shipments per year	760	2,550	4,360
Time in transit, days	2.8	2.8	2.8
Vehicles in transit	6	20	33
HTGR			
Metric tons/year	30	520	1,030
Load per vehicle, metric tons	1	1	1
Shipments per year	30	520	1,030
Time in transit, days	4.0	3.5	3.0
Vehicles in transit	0.3	5	8
LMFBR			
Metric tons/year		210	4,620
Load per vehicle, metric tons		2.8	2.8
Shipments per year		75	1,650
Time in transit, days		3.5	3.0
Vehicles in transit		1	14

Table 5. Spent Fuel Shipments

	1980		1990		2000	
	Truck	Rail	Truck	Rail	Truck	Rail
LWR						
Metric tons/year	600	1020	2440	5900	4000	14,900
Load per vehicle, metric tons	0.44	3	0.44	3.5	0.44	4[a]
Shipments per year	1360	340	5540	1680	9090	3720
Time in transit, days	2.5	9	2.5	8	2.5	7
Vehicles in transit	9	8	38	37	62	71
HTGR						
Metric tons/year		2		190		710
Load per vehicle, metric tons		0.6		0.6		0.6[a]
Shipments per year		3		320		1180
Time in transit, days		8		8		7
Vehicles in transit		0.1		7		23
LMFBR						
Metric tons/year				20		1850
Load per vehicle, metric tons				2		2[a]
Shipments per year				10		930
Time in transit, days				8		7
Vehicles in transit				0.2		18

[a]Megacuries per shipment: LWR = 16, HTGR = 3, LMFBR = 20.

Table 6. Projected Shipments of Plutonium

	1980	1990	2000
Metric tons fissile Pu/year	13	70	260
Load per vehicle, metric tons Pu[a]	0.4	0.5	0.5
Shipments per year	32	140	520
Time in transit, days	2	2	2
Vehicles in transit	0.2	0.8	2.8

[a]Truck shipment, based on 0.1 ton Pu nitrate or 0.5 ton Pu oxide.

High-Level Wastes

High-level wastes produced during the reprocessing of irradiated fuels contain virtually all of the nonvolatile fission products, several tenths of a percent of the uranium and plutonium in the spent fuel, and all the other actinides formed by transmutation of the uranium and plutonium during irradiation. Regulations require that the aqueous high-level wastes be solidified within 5 years after generation and shipped to a federal repository within 10 years after generation.

Although solidification of high-level wastes has not as yet been practiced industrially, indications are that they may be packaged in steel canisters about 1 ft diam by 10 ft long for shipment to a repository. Studies of conceptually designed shipping systems show that up to 12 such canisters, each containing 6.3 ft^3 of 10-year-old waste, could be shipped by rail in a 100-ton cask.[5,6]

Table 7 shows the rate of generation of high-level waste, the annual rate of shipment 10 years after generation, and the number of shipments per year. The quantity of radioactivity per 75 ft^3 shipment is about 10 megacuries. Also shown are the total volume and radioactivity accumulated at the repository.

Cladding Wastes

Cladding wastes consist of solid fragments of Zircaloy and stainless-steel cladding and structural components of the fuel assemblies that remain after the fuel cores have been dissolved. In addition to induced activity, the cladding waste was assumed to contain 0.05% of the actinides and 0.05% of the fission products in the spent fuel. It was estimated that after compaction, the final volume would be 2.1 ft^3 per metric ton of LWR fuel and 8.7 ft^3 per metric ton of LMFBR fuel. The compacted cladding is packaged in canisters 10 in. diam by 10 ft long. Each canister contains 3.5 ft^3 of waste. After storage of 1 year at the reprocessing plant, the waste is shipped by rail to a repository. Each rail cask carries 27 canisters of LWR cladding (95 ft^3) or 36 canisters of LMFBR cladding (127 ft^3).

Table 7. Solidified High-Level Waste Shipments

	1980	1990	2000
Rates of Generation and Shipment			
Rate of generation, ft^3/year	3200	17,900	47,700
Rate of shipment, ft^3/year	0	3200	17,900
Ft^3/shipment		75	75
Shipments/year		43	240
Megacuries/shipment		10	10
Total Accumulation at Repository			
Volume accumulated, ft^3	0	14,100	113,000
Megacuries accumulated	0	1770	14,100

Table 8. Shipments and Accumulation of Cladding Wastes

	1980	1990	2000
Rate of Production and Shipment			
Volume shipped, ft^3/year	2,900	15,600	50,300
Volume per shipment, ft^3	95	95	102
Shipments/year	31	164	495
Megacuries/shipment	0.3	0.3	0.5
Total Accumulation at Repository			
Volume accumulated, ft^3	11,400	97,100	415,000
Megacuries accumulated	28	190	820
Thermal power accumulated, MW	0.3	2.1	7.7

Table 8 shows the rates of production and shipment in cubic feet per year, the number of shipments per year, and the quantity of radioactivity per shipment. Also shown is the total quantity accumulated at the repository. The total accumulated by the end of year 2000 is 415,000 ft^3, with a radioactivity of about 820 megacuries and a heat release of 7.7 megawatts.

Noble Gases

Table 9 shows the projected rate of shipment and accumulation of noble gas fission products. It was assumed that krypton and xenon are packaged in 50-liter cylinders at 2200 psig, stored for 1 year at the reprocessing plant, and shipped to a repository by rail. Each rail shipment consists of six gas cylinders in a water-filled cask.[7]

Iodine

Rates of shipment and accumulation of fission-product iodine are shown in Table 10. It was assumed that essentially all of the iodine was recovered, converted to potassium iodide, and stored for 1 year before shipping to a repository. The number of shipments per year was estimated based on the assumptions that each reprocessing plant would make one shipment per year and that there would be approximately 3, 6, and 8 reprocessing plants operating in the years 1980, 1990, and 2000. Total radioactivity accumulated is quite small in comparison with other wastes.

Tritium

Each 1000-MW(e) light water reactor was assumed to produce each year 20,000 gal of tritiated water containing 5 μCi/ml. After storage at the power station for 1 year, the water is shipped to a disposal site in 4000-gal tank trucks.

Fission-product tritium is recovered at LMFBR power stations and from spent fuel at reprocessing plants. It was assumed that this tritium

Table 9. Shipment and Accumulation of Noble Gas Fission Products

	1980	1990	2000
Rate of Generation and Shipment			
Generation, megacuries/year	13	76	220
Shipment, cylinders/year	82	470	1,300
Number of shipments/year	14	78	217
Megacuries/shipment	1	1	1
Total Accumulation at Repository			
Cylinders accumulated	300	2,800	11,800
Megacuries accumulated	44	350	1,400
Thermal power, MW	0.1	0.6	2.3

Table 10. Shipment and Accumulation of Fission Product Iodine

	1980	1990	2000
Rate of Production and Shipment			
Iodine production, kg/year	350	2,030	5,770
Production, curies/year	48	276	780
Shipments/year[a]	3	6	8
Curies/shipment[a]	16	46	98
Total Accumulation at Repository			
Iodine accumulated, kg	1,270	12,000	52,000
Curies accumulated	170	1,630	7,000
Thermal power, W	0.1	1.1	4.6

[a]Estimated, based on one shipment per year per reprocessing plant.

is recovered in water at a concentration of 200 Ci/liter. This is then immobilized as calcium hydroxide and is shipped to a repository by truck. A truckload consists of 10 ft^3 of calcium hydroxide or about 0.03 MCi.

Table 11 shows the rates of production, shipment, and accumulation of tritium on the basis described.

Alpha Solid Wastes

Volumes of alpha waste produced were based on the following assumptions: (1) production of 10,000, 20,000, and 4000 ft^3 per ton of plutonium processed in fuel preparation, fabrication, and reprocessing plants, respectively, (2) average Pu content of 0.25 g/ft^3 before compaction, (3) compaction by factors of 3 to 7.5 during 1976-1987, and by a factor of 10 thereafter. Shipment to the repository was assumed to be by rail, 1000 ft^3 per carload, 1 year after generation. Table 12 shows the estimated volumes shipped and accumulated. Uranium wastes from the HTGR fuel cycle are based on assumptions similar to those given for plutonium-bearing wastes.

Alpha-Beta-Gamma Wastes

It was estimated that 10,000 ft^3 of alpha-beta-gamma wastes are generated per metric ton of plutonium or HTGR uranium recovered from spent fuels at reprocessing plants. The average Pu or uranium concentration was assumed to be 0.025 g/ft^3, and it was assumed that 0.025% of the fission products are present. Compaction factors were assumed to be the same as for alpha wastes. Shipment to the repository was assumed to be by motor freight, 1 year after generation, each shipment consisting of 75 ft^3 of waste in a lightly shielded cask. Shipments and accumulations are shown in Table 13.

Beta-Gamma Wastes

Production of beta-gamma wastes from fuel preparation, fabrication, and reprocessing was estimated at 40,000, 80,000, and 16,000 ft^3 per

Table 11. Shipment and Accumulation of Tritium

	1980	1990	2000
Tritiated Water from Reactors			
Production, 10^6 gal/year	1.6	8.1	16.8
Production, megacuries/year	0.03	0.14	0.3
Shipments/year	400	2020	4200
Accumulated volume, 10^6 gal	8	56	190
Fission-Product Tritium from Reprocessing			
Production, ft^3/year[a]	280	1600	4500
Production, megacuries/year	0.8	4.7	13
Shipments/year	56	312	880
Accumulated volume, ft^3[a]	2000	19,000	80,000
Accumulated megacuries	3	23	88

[a]Volume given as calcium hydroxide, 50% of theoretical density.

Table 12. Shipment and Accumulation of Alpha Solid Wastes

	1980	1990	2000
Plutonium Alpha Wastes			
Production, 10^3 ft^3/year	110	230	1140
Production, megacuries/year	1	7	33
Shipments/year	110	230	1140
Accumulated volume, 10^3 ft^3	320	2400	8400
Accumulated megacuries	2.4	34	170
Uranium Alpha Wastes			
Production, 10^3 ft^3/year	1	13	78
Production, curies/year	30	1400	8100
Shipments/year	1	13	78
Accumulated volume, 10^3 ft^3	2.5	51	530
Accumulated curies	93	5200	78,800

Table 13. Shipment and Accumulation of Alpha-Beta-Gamma Wastes

	1980	1990	2000
Production, 10^3 ft^3/year	45	88	340
Production, megacuries/year	0.4	2.4	7
Shipments/year	600	1200	4500
Accumulated volume, 10^3 ft^3	220	980	3000
Accumulated megacuries	1.1	7	26

metric ton of plutonium or HTGR uranium. Estimated radioactivity was 1 millicurie per cubic foot. Production of beta-gamma wastes from reactors was based on 4000 ft^3/year from each 1000 MW(e) of LWR or HTGR capacity and 2000 ft^3/year from each 1000 MW(e) LMFBR. The reactor wastes were assumed to contain 1 curie per cubic foot. These wastes were assumed to be shipped by motor freight to surface burial grounds. Each shipment consists of 475 ft^3 in 55-gal drums, 64 drums per load. It was estimated that 1 acre of land is needed for each 50,000 ft^3 of waste. Table 14 shows the production, accumulation, and land area used.

Trends in Fuel and Waste Shipping

For what it may be worth, here are the trends we see:

- More stringent recovery of the radioisotopes from the effluents of all types of nuclear installations.
- More emphasis on shipping various types of waste to federal repositories, rather than to burial grounds.
- Continued development of improved shipping concepts for various types of waste (and, of course, for fresh and spent fuels).
- Shipment of a greater proportion of spent fuel by rail rather than by truck, because of the greater ratio of fuel load to cask weight.
- Shipment of plutonium in the solid oxide form rather than as nitrate solution.

Table 14. Shipment and Accumulation of Beta-Gamma Wastes

	1980	1990	2000
Production, 10^6 ft^3/year	1.3	8.7	39
Production, megacuries/year	0.1	0.5	1
Shipments/year	2700	18,300	82,200
Accumulated volume, 10^6 ft^3	4.2	52	270
Accumulated megacuries	0.3	1.6	5
Total burial area used, acres	84	1050	5300

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1984 FACT OR FICTION? OR THE TRANSPORTATION REQUIREMENTS OF THE NUCLEAR REPROCESSING INDUSTRY

C. W. SMITH R. H. JONES

The abstract and/or paper is printed in Volume 3.

SPENT FUEL CASKS - WORLDWIDE STATUS

J. D. ROLLINS D. M. COLLIER

The abstract and/or paper is printed in Volume 3.

WHAT IS A CASK LIKE YOU DOING IN A PLACE LIKE THIS? DISCUSSION OF USAEC REQUIREMENTS FOR IN-PLANT CASK DROP ANALYSIS AND PROTECTION

R. H. JONES
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V. K. VIRA

The abstract and/or paper is printed in Volume 3.

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"SPECIALIZED TRANSPORTATION FOR THE NUCLEAR AGE"

Bert Jody, Jr.

During the Spring of 1973 a legislative workshop, sponsored by the Southern Interstate Nuclear Board in cooperation with the Oak Ridge Operations Office of the Atomic Energy Commission and its operating contractors was held to discuss the topic - "Radioactive Shipments Through Our States - The Weakest Link?"

Many of you here today may have attended that meeting and are now wondering whether you will be subjected to a review. That is not my intention. I came away from that meeting with the conclusion that transportation did indeed represent the weakest link in public protection from radiation incidents and with some serious questions of carrier involvement of my own.

I will now confine my remarks to the motor carrier industry for which I'm sure you will be grateful, and I do agree that at least in part, transportation does represent the weakest link. Why shouldn't it? Historically, transportation has always been the weakest link because of industrial priority systems. The scenario has read something like this: Plant design, fabrication, execution, manufacture, sales, and then way down stage -- "Will someone please find somebody to haul this stuff?" "You mean we have a traffic department?" "What was that traffic guy's name?" Then in desperation somebody grabs Joe Woodenspoke, the local box and refuse hauler, and another transport system is born! Sometimes it's successful. The oddity in the nuclear field seems to be that a good deal of thought has been given toward transportation of

materials prior to the actual time of movement.

All of this has presumably led neatly into my topic, "Specialized Transportation for the Nuclear Age". Whether it has or not, I would like to discuss for a few minutes three areas which are vital to any specialized system.

First, transport equipment. Transport equipment for the nuclear carrier must take on the aspect of dedication to the principal of cargo safety above all other criteria since nuclear cargos have a twin consistency. They are of extreme value or hazard and usually co-exist.

Until very recently, chief reliance for safety in transit has been placed in the containment vessel itself. The exceptional background and continuing revision of containment vessel safety must serve as a basis for the carrier industry to meet that standard of excellence within its own confines. The movement of all types of nuclear materials over the highway system of the United States will, by 1980, number in the multi-thousands per year where a much lower load level is the rule today. The emphasis on transport safety must surely shift to some degree to include responsible specialized design of nuclear transport by a developing nuclear carrier industry. Some thoughts and suggestions in this area must include the need for highly reliable light weight power equipment in order to meet delivery and security requirements while providing optimum tonnage for cost consideration. We are all too painfully aware of the problems being encountered in spent fuel delivery to forget that. In the past, use of ultra light weight sleeper cab equipment and matching trailers has caused higher breakdown costs and resultant delivery problems. New construction

materials must be found to insure a better balance between reliability and weight. Component parts need to be thought out well in advance of production, and a continuing evaluation by the equipment manufacturer and the carrier are needed to insure reliability and the eventual evolution of the nuclear tractor specification.

Retro-fitting of standard production model trailers has been a usual and necessary function to date for the movement of nuclear materials. This has resulted in a high visibility factor which in my opinion can and should be altered in view of increasing public clamor over the movement of nuclears. A marriage of the containment vessel and the trailer to provide a specification vehicle under D.O.T. codes, as has been the case in the tank truck industry, is a possibility. The value of such a system of vehicles is readily apparent. More uniform equipment handling systems, higher trailer utility, and certainly a lower public profile could result.

A welter of optional equipment is available on the market today which should be assessed in the light of specialization. Automatic transmissions in heavy tractors can lead to better engine performance with less driver fatigue. New federal braking requirements will add to the safety factor. Engine fail safe systems can point out potential trouble before costly road failures occur. We must have better, high performance tires to avoid the potential blow out problems, which could negate all the special equipment mentioned to date. No consideration of transport equipment would be complete without some reference to the new regulations for security under 10 CFR 73. Whether the nuclear carrier considers the use of retro-fitted armored cars or takes a hard look at

the Sandia Laboratories design concept for the movement of Special Nuclear Materials, a new era has arrived for the carrier when he must talk of bullet proof glass, armor plate, locking axles, exploding drive-lines, intruder resistant cargo containers, mobile telephone equipped units and armed security forces. This is specialization now, not tomorrow, and the responsible carrier must get in front of these changes rather than be dragged reluctantly behind.

My second area concerns tie down systems and handling procedures. At present, tie down systems range from chains and binders to massive turnbuckle assemblies and permanent or semi-permanent bolted units attached to truck or trailer beds. While there is nothing wrong with any of these systems in its proper place, some need for uniformity would seem to be required. It would be in order to establish at this time a group consisting of cognizant nuclear industry people, concerned nuclear carriers, and representatives of the Department of Transportation to study, revise, and implement more specific tie down criteria which could then eliminate the present touch system of what tie down can or may be used to handle which cylinder, cask, overpack, or other shipping container.

Eventually, clear tie down criteria being established, the specific arrangement, location, load strength, and required D.O.T. approval, should be the concern and responsibility of the carrier moving the load and not the shipper. Looking toward future improvements, tie downs could take the form of a cradle and hinged top much like the puritan punishment stock where cylinders are being moved. This would give a higher immobility to the load and provide protection against overturn

damage like the roll bar on a stock car. These cradle assemblies, made from light weight formed steel, will not add much to the total shipment weight and these are already in a formative stage by our Company. Another departure from standard tie down methods could be the use of two rails mounted parallel over the main beams of the trailer deck which would allow overpacks and other shipping containers of many types, equipped with matching rail lugs, to be moved along these rails and locked in place at points consistent with weight distribution. Here again, the weight gain is minimal and has the advantage of mixed loading capability coupled with the use of ground handling equipment where large overhead cranes are not available.

Another concept is the use of a unit which in appearance resembles a low profile tank trailer. The top half of the cylinder can be laid back in sections to allow loading of 30 inch cylinders in tandem. As each compartment is closed, it provides the required separate overpack integrated into the whole. This idea would require the use of light weight power and under carriage to offset the additional weight of the overpack. The value of such a system lies in the increased rigid strength of the towed unit, its increased safety factor and the low public visibility of which I spoke earlier. From the carrier's standpoint, this vehicle creates less wind drag and increases fuel mileage and since the industry has not yet provided us with nuclear powered tractors, that is a plus.

For those concerned with the movement of radioactive waste in upright casks, it is possible to construct a trailer with a large vertical steel tube or box incorporating the upper and lower impact

skirts. Designed properly, such a unit would eliminate the need for sheerblocking and other conventional tie downs such as solid arms, turnbuckles, or cable attachments. In this case, the weight reduction could mean payload increases which would cut down the total trips per year required to move such waste. Fewer trips mean less exposure to accident and that I presume, makes everyone happy.

Each of these innovations can be used in conjunction with presently acceptable materials handling equipment. In the area of handling procedures, the carrier could provide any number of handling arrangements depending upon the desire of shippers. Rail mounted overhead power hoists, which operate up and down the trailer deck to move small overpacks from the trailer to ground handling equipment, have been in existence for many years in other industries. Hydraulic or electric equipment for loading drums or small cylinders into trailer mounted overpacks are already in use. It is possible to provide a flat bed chassis, whose entire loading deck can be removed by sliding it out of the side rails onto a platform. The unit then has an empty load deck rerun and locked and is ready for departure. The carrier is not waiting or collecting demurrage time and plant handling personnel can unload the lading as their time schedule permits. This concept can be adapted to ship and aircraft loading procedures as well. The possibilities for better handling procedures are endless but require close cooperation between individual shipping systems and carriers. Some sense of industry wide needs and goals in this area would have to be realized before carriers can hope to reach a point of equipment stability. Again, standardized handling procedures, materials handling

equipment and carrier equipment must all be designed to work together for safe and efficient loading or unloading of cargo.

Handling procedure must of necessity include the proper shipping documents. This headache would be less painful if faster and more universal communications equipment were available. The use of telecopiers or transceivers for freight information between the carrier, shipper, and consignee on every shipment could remove most of the delays, missent freight, and anxiety over the whereabouts of someone's expensive property.

The third, and most important area in my view is that of driver and terminal personnel. Present driver hiring practices do not always provide the best possible personnel for specialized transportation. Very often the carrier cannot assign the driver whom he feels is best qualified by attitude and skill to handle the job. Seniority systems and other negotiated items can cause problems for management. The time may have arrived for us to consider the salaried driver-technician who can work and advance outside the restraints of hourly wages or mileage payments. Transport management must develop highly skilled drivers, motivated to do the job, through periodic training programs developed with the shipper. In this way, the carrier can have some assurance that he will be able to hold these skilled people against the enticement of other carriers.

In the final analysis, the skilled driver is the key to safe, efficient delivery. Without his interest, all of the massive effort toward safety in containers, special equipment and advanced handling techniques may come to nothing.

The same must be said for terminal personnel. Dispatchers, clerks, and service people must be made aware of the products to be handled and the need for exact pickup and delivery schedules. Outside of prescribed regulations by the States and the Federal Government, the carrier should in his own interest, make available to operating and terminal personnel comprehensive procedural manuals for response to emergency situations, freight handling, dispatch procedures, cleaning, repair, and any other company data pertinent to maintaining an excellent service relationship with the shipper and the public. I do not refer to an impressive shelf document that looks good, but to one which is lean, dogeared, and in constant use and revision.

In Conclusion.

The nuclear industry has a unique opportunity to develop its own carrier industry. There are at present, a number of healthy motor carrier systems in the United States serving the various needs of other industries. There are general commodity carriers, tank truck carriers, size and weight carriers, household goods movers, and several more. There should be another category, the nuclear carrier. Trying to superimpose already existent systems over specialized nuclear concerns will serve only to bring old problems and cures to a new industry. Where a carrier derives his livelihood there will his interest be. It would be my hope that a nuclear carrier industry can become fact through the combined efforts of the nuclear industry, those carriers whose serious interests lead them in that direction, and the governmental agencies responsible for granting the required operating rights. It has been said that a truly good transaction is one in which everyone is satisfied. A specialized nuclear carrier industry would appear to fit that description.

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THE DEVELOPMENT OF HIGH STRENGTH, LOW WEIGHT, SEMI TRAILERS FOR TRANSPORTATION OF NUCLEAR FUEL

William C. Benson

ABSTRACT

Semi Trailers for transportation of Nuclear Fuel Casks impose new considerations on the vehicle designers. The safety standards, the need to maximize pay loads of a high value commodity, the always loaded duty cycle, the load concentration, vertical loading/unloading mode, personnel protection and air circulation are all unusal criteria to be considered in developing Semi Trailers. Through the development of a series of vehicles, for a variety of casks, legal and permit size, the author has accumulated experience and knowledge that is of extreme importance to anyone undertaking such a development. This paper is devoted to discussing, analyzing, and optimizing vehicle designs in this increasingly important, energy industry.

GENERAL DISCUSSION

We have heard, and will hear more during this seminar, about the problems associated with transportation of nuclear fuel, other radioactive material, and the projected growth in shipments of such material.

The attention of Government is now focused on the energy shortage. Nuclear power plants are one of the few solutions.

The area of my expertise involves the highway transportation of these loads. The area of your interest, at least many of you, is the efficient and economical highway transportation of these loads.

Radio active material is a high value commodity and, as such, the shipping cycle time is important to your economics. The shielding of radio active material is so dense that "Pay loads" which can be transported without permits is in the range of 1500 to 3000 pounds, and has a value of $50,000 to $100,000.

The nuclear industry has been attempting to obtain relief for several years under the rules and regulation of the A.E.C. and various state highway department laws. Many of you participated in the Southern Governor's Conference in 1969 where a major effort was made to get yearly, unrestricted travel permit authority from the various states for gross combination weights approaching 89,000 pounds. We were not successful. Individual states could not agree on a universally acceptable tractor-trailer configuration.

Many of you are also familiar with House Bill H.R. 11870 which has been rather inactive since introduction in 1969. This bill would permit gross combination weights approaching 80,000# on the interstate highway system. This bill has received a major push from the Nixon Administration since early this year, but the Nixon's Administration's ability to push is questionable.

The bill needs pushing for a variety of reasons. Its good points are as follows:

A. It would be deflationary as it would tend to lower the cost of motor freight due to improving productivity of trucks

and drivers.

B. It apparently would not harm the interstate highway system.

C. It would have a major impact on decreasing transportation costs of radio active material.

D. It would generate many new jobs in the truck/semi trailer industry.

Until we are able to get some relief through law, we must continue to work within present regulations (73,280# G.C.W.) and extremely small pay-loads of radioactive materials.

CUSTOMER NEEDS

Speaking from years of experience, it is somewhat fool hardy to approach any technical development without first defining the customer needs.

You share one common need with all shippers, equipment manufacturers and truckers. That need is to maximize pay loads.

Beyond that requirement, the needs of your industry become very uncommon, for a variety of reasons as follows:

A. The radio active nature of your "Payload" requires an extremely heavy shipping container.

B. Your "Payload" is extremely low, 1500# to 3000# without permits and 4500# by restricted travel permits.

C. The value of your commodity is unusally high at approximately $30/lb.

D. Based on $1.00/mile transportation costs, 100 miles of movement adds 6½¢/lb. to your commodity. Based on an empty cask return, the 100 mile round trip adds 13¢/lb.

E. The heavy weight of your payload and container requires a long semi trailer (for bridge formula). The compact size of the pay load imposes severe stresses in a trailer frame.

F. From an engineers standpoint, your semi trailers are always loaded. (The empty cask is a load).

G. Air circulation is necessary around the cask, but a personnel barrier is needed.

H. Access to the casks is usually from above.

I. Safety demands the highest quality of design and manufacture.

TECHNICAL CONSIDERATIONS

GENERAL

During the past five years, I have been living particularly close to those interested in highway transportation of nuclear fuel in the United States.

The first year or so was spent concepting truck semi trailer combinations to transport casks weighing 56,000 to 60,000# within the then anticipated acceptance of gross combination weights of 85,000 to 89,000#.

When this increase did not occur, we were asked progressively, for legal (non permit) combinations, to haul casks having the following

weights:

A. 46,000#
B. 52,000#
C. 54,000#
D. 48,000#

These figures become serious to the semi trailer designer who is on the short end of the stick for reasons as follows:

A. The customer has completed a long lead time and expensive cask design and has it in some stage of licensing.
B. Truck tractor development is also a long and expensive process, so basically, standard lightweight tractors weighing about 14,000# are used.
C. The state load laws permit gross combination weights in the area of 73,280# for non permit operations.
D. The special semi trailer designer gets what is left in weight and time to provide a semi trailer to haul these casks.

What is left for the semi trailer designer by the four casks above, assuming load distribution can be perfect and no allowance made for snow, mud, or ice accumulation is as follows:

A. 13,280#
B. 7,280#
C. 5,280#
D. 11,280#

The above weights are considered maximum and are not just for a normal trailer. Included in those weights must be cask cradles, cask tie downs, personnel barriers, opening roof and hopefully, a tolerance of some type for the semi trailer weight, snow, mud, or ice accumulation. There

is a saving grace. So far, actual cask weights have been less than the customer's estimated maximum. Unfortunately, this is not known until the casks and trailers are completed.

Then comes risk considerations. Your industry demands and must be proud of a near perfect safety record. The semi trailer designer often ends up with a calendar time schedule that precludes an extensive endurance test program.

Our industry, while not as strictly regulated as yours, is not free of government restrictions. Effective Jan. 1, 1975, new safety standards of the Department of Transportation imposes anti-skid controls on semi trailers built after that date. Aside from cost, the weight penalty is 205# for a two axle semi trailer.

Our response has been technically aggressive. Given time and money, we feel almost anything is possible. To date, we have been successful, but the 5,280# unit is still in a preliminary study phase.

Specifics

We have been through numerous studies and four specific development programs to the present time. I feel we, and our customers, have learned a great deal from these programs of a quite surprising nature.

Certainly, through this work, we have been forced to update our knowledge of rule-of-thumb standards that have been developed and used throughout the semi trailer industry over the years.

Some of these rule-of-thumb, practical experience practices which we have found Not Applicable to your needs are as follows:

A. Static proof loading to 150% rated capacity assures structural integrity.

B. Vertical acceleration loads do not exceed the area of 1.4 to 1.5G.

C. Stress reversal does not occur in normal operation.

D. Fatigue characteristics of structural steel can be generalized at 40% to 50% of tensile stress limit.

E. Gusseting of weld joints recovers parent metal strength.

F. Dynamics of semi trailer frames and spring suspensions do not vary substantially.

G. The T-1 class of steel (100,000 PSI yield) used in recent years for light weight trailers is a panacea to trailer designers.

They are Not Applicable because your needs are not typical of the trucking industry for four reasons as follows:

A. Your need to reduce weight to remain legal.

B. Your constant load 100% of the time. No empty return.

C. The always concentrated nature of the load. It is never a distributed load.

D. The relatively high speed movement.

These four unusual needs raise a banner which says "Metal Fatigue-

Be Carefull."

Now I would like briefly to review the projects I have been involved with for transporting Nuclear Fuel, beginning with the General Atomic Hallum Cask (46000#), through the NL Industries Cask (48000#), to the NFS/NAC Cask (52000#), to a proposed cask (54000#).

General Atomic of San Diego (Figure 1)

Our first customer for a fuel cask trailer was General Atomic, for transporting the 46000# Hallum Cask and Cradle. The Cradle attachment devices were to withstand acceleration loads of 10G FWD, 5G Side and 2G UP.

This required a quite un-sophisticated semi trailer, actually a standard T-1 Steel frame platform. We modified the "ST'D" by installing aluminum disc wheels, an aluminum tread plate floor, the tie down devices, and achieved a comfortable weight of 9800# without enclosure. As I recall, the fuel load for that Cask was approximately 150# and personnel barriers were not a requirement.

NL Industries (Figures 2,3, & 4)

From a technical standpoint, the NL Industries requirement began to get tighter. They contracted with us for semi trailers with fenders and cat walks to weigh 8000# or less, to haul a 48000# Cask. NL Industries then added their own cradles and personnel barrier.

The task began to get more severe, but by using fiberglass reinforced plastic fenders, aluminum wheels, disc brakes, tubeless tires, aluminum

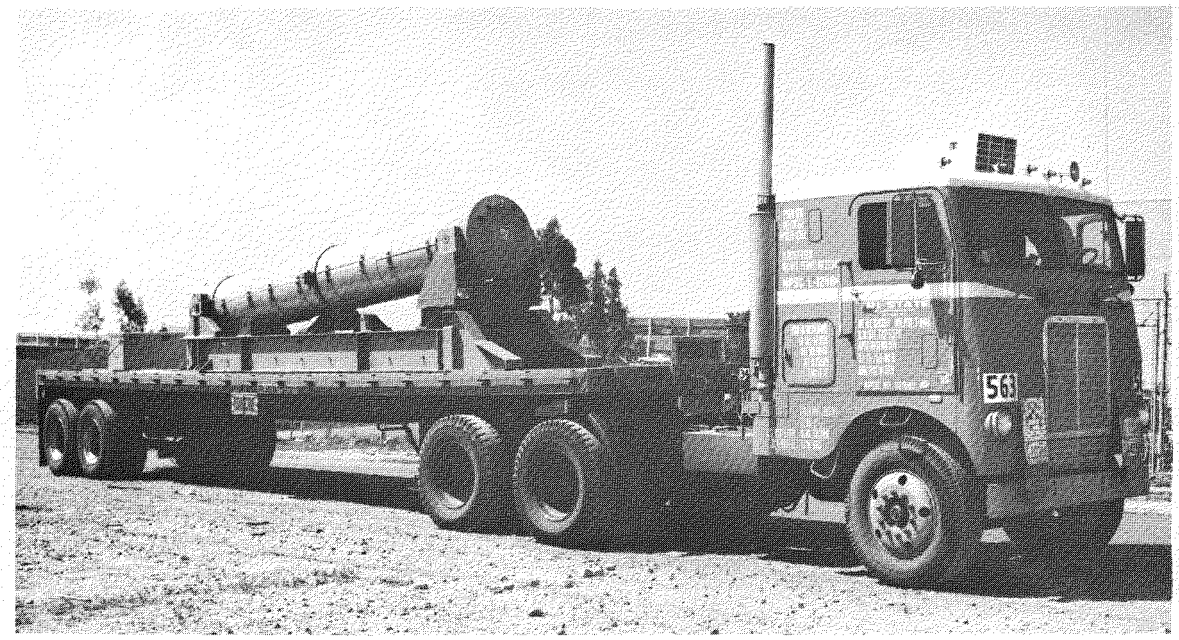


Fig.1. General Atomics Semi Trailer with Cask.

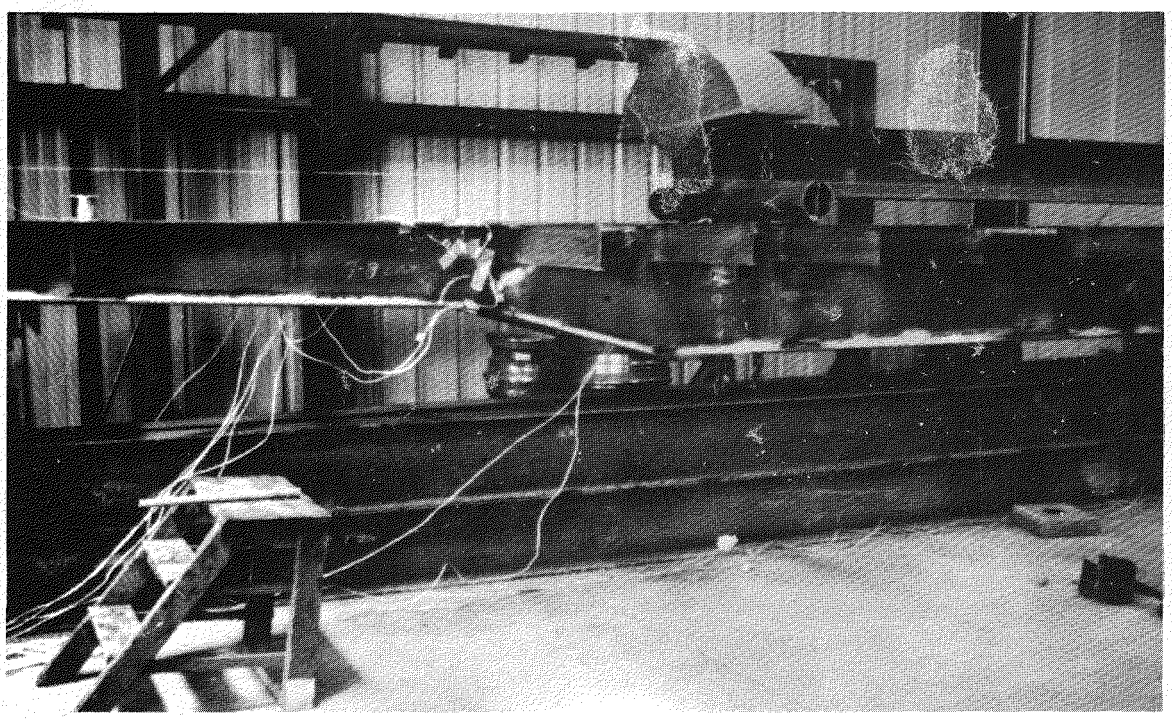


Fig.2. Strain Gage Testing - NL Industries Semi Trailer Frame



Fig.3. Transport Configuration - NL Industries Semi Trailer

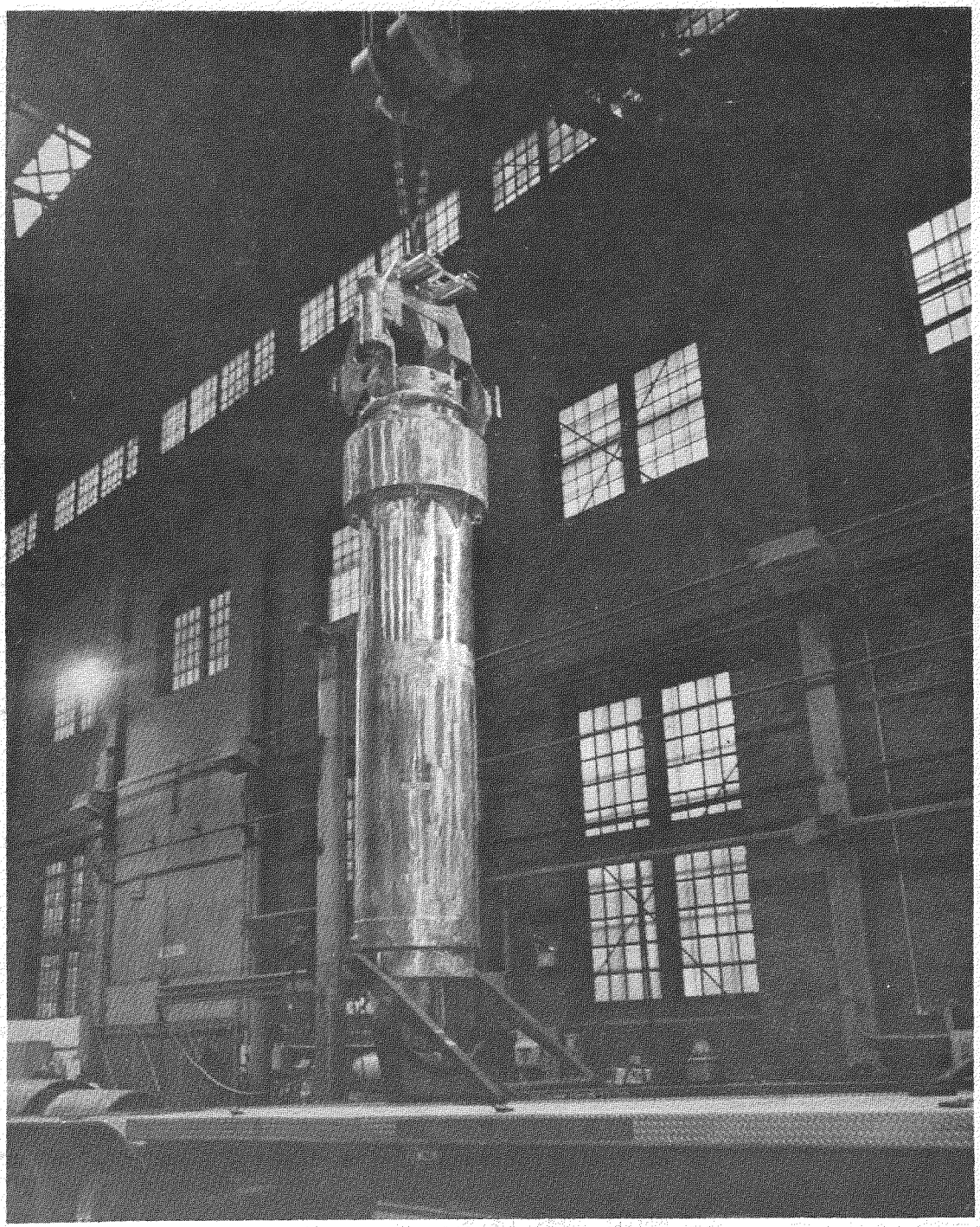


Fig.4. Loading Configuration - NL Industries Semi Trailer

tread plate cat walks, we were able to use conventional fabricated T-I steel beams. We ended up with a 7680# weight. When NL added their cradles and personnel barrier, the weight became 8580#.

Quality requirements were also tightened to include certified welders, non-destructive testing of all structural welds and static proof loading to 2"G" or 96000# (Figure 2).

Nuclear Fuel Services (Figures 5 & 6)

NFS (actually ahead of NL calendar wise) had a more severe problem. They wanted to handle a cask weighing possibly 52000# and wanted a trailer complete with cradles and personnel barrier to weigh 7300#.

To attain that weight required a departure from conventional design as used on prior vehicles.

We ran through several approaches to solving this requirement including the following:

A. A high strength aluminum wide flange beam frame of the 7075T6 class having a tensile strength in the 83,000 PSI range. After consultation with structural engineers in the Aircraft industry, we rejected this due to poor weldability and an endurance limit of only 23,000 PSI. They did suggest we look at Titanium. We did and liked what we saw. More about that later.

B. A monocogue (Van type) construction of aluminum which, though offering basic high strength to weight capability,

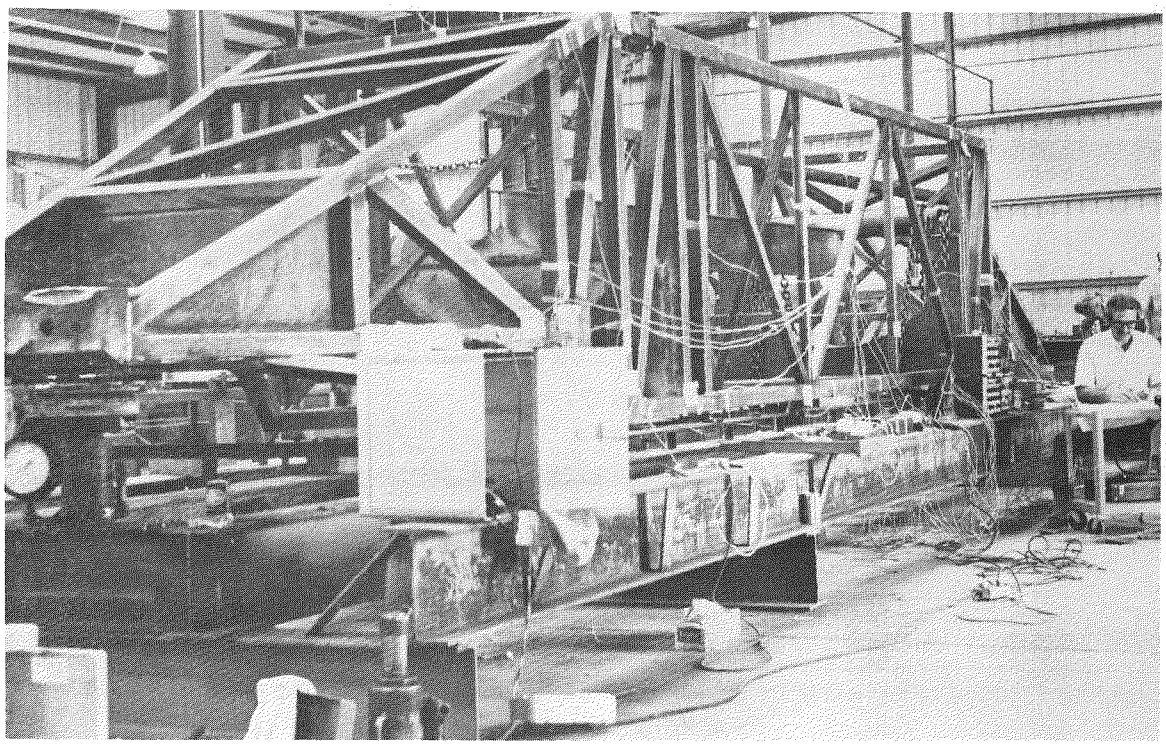


Fig.5. Strain Gage Testing - Nuclear Fuel Services Semi Trailer Frame

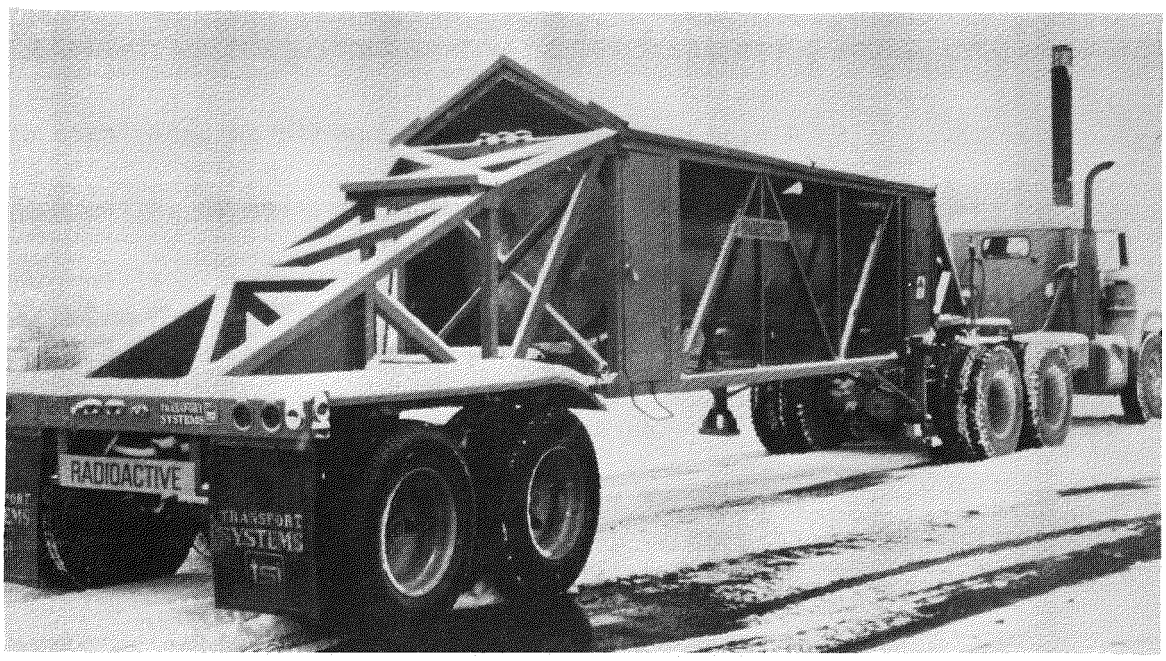


Fig.6. Transport Configuration - Nuclear Fuel Services Semi Trailer Frame

began to look less favorable when other needs were analyzed. The needs which began to erode the basic advantage were open top loading, side door access, air flow cooling, and load concentration.

C. A structural tubing truss of 50,000 PSI yield steel which appeared to fit our required strength to weight ratio and a design with which we were familiar from livestock and grain trailer experience. This design was selected, static proof tested to 2"G", used aluminum wheels, tubeless tires, single leaf springs, aluminum fence and roof and aluminum support legs. The finished weight of the unit was 6900#. We were pleased with that result but were aware that state-of-the-art advance such as this required some evolution to a final design based on user experience.

This design did not prove to be an exeption. We began to get some fatigue cracks, in high stress areas, in or adjacent to welds at 10,000 to 17,000 miles.

Fortunately, the NFS Cask was lighter than target so we were able to add some weight in the form of added gussetts as we chased fatigue cracks around structural weld joints.

At that time, we began an educational process, intensive in nature, at considerable expense, to learn more about several things as follows:

A. The dynamic strains and stresses of a concentrated load on

truss type and beam type semi trailers.

B. The safe fatigue limits on various structural materials from grade 50 steel through grade 100 (T-1) steel and the affect of welding on these materials.

C. Optimization of joint gusset designs.

D. Ride characteristics of some light weight trailer suspensions.

We had some difficulty in finding valid fatigue information on structural steels although data on alloy steels, aluminum and titanium is available from aerospace research.

While the search for fatigue data was made, one NFS cask trailer was instrumented with strain gages and accelerometers. Road tests were initiated over a variety of roads and courses.

Then we began to get surprises, both from the road test data and structural steel fatigue data. The surprises, in order, from the road tests, were as follows:

A. Peak vertical acceleration (gravitational plus dynamic) measured at the C.G. was -2.3"G".

B. The maximum dynamic stress was 47,800 PSI (T) in a material having a 50,000 PSI minimum yield.

C. Strain reversals occurred in the trailer structure. Dynamic acceleration at the C.G. exceeded gravity by a margin of up to ±0.3g.

D. Some of our gusseting resulted in unsymetrical loading across the square structural tubing.

We also received some information from road tests that we expected such as maximum accelerometer and stress levels occur during high speed crossing of railroad tracks.

The surprises from the fatigue data (mostly studies by U.S. Steel) was later substantiated through an independent consulting group.

These findings, at least in our case, were appreciably different from rule-of-thumb information in many text books and engineering manuals that "conservative estimate of fatigue strength is 40-50% of tensile strength." Based on that rational, we could be using a "conservative" fatigue stress in the area of 25,000 to 32,000 PSI, for 50,000 yield material and 50,000 to 62,500 PSI on 100,000 yield material.

What we found, as reliable data, is tabulated in Table I for ASTM A 501(A36), A 500-Grade B (50,000 yield), and ASTM A 514(100,000 yield). The Table is based on three assumptions as follows:

A. Safety factor against fatigue failure= 1.42

B. Stress ratio= 0 Minumum

C. Number of stress cycles of 600,000 and 2,000,000.

Table 1 - Allowable Fatigue Stress Level in Welded Material

Grade of Steel	Allowable Stress @ 600,000 Cycles	Allowable Stress @ 2,000,000 Cycles
36,000 Yield	16,900 PSI	13,400 PSI
50,000 Yield	19,700 PSI	16,200 PSI
100,000 Yield	24,900 PSI	19,000 PSI

The biggest surprise comes from the relatively small gain in fatigue stress by using 100,000 yield material over lower strength steels.

Going back to the data obtained during the dynamic road test, and projecting approximately one million miles as the life of the trailer, a load spectrum for fatigue analysis was compiled based on 500,000 miles of expressway route, 500,000 miles of paved secondary route (smooth to bumpy asphalt), 50,000 miles of very rough patched asphalt-concrete, and an arbitary series of harder shocks for an additional 50,000 miles.

Based on this load spectrum, a cumulative fatigue damage analysis was performed on the critical structural members of the semi trailer indicating that approximately 520,000 cycles at the higher levels will occur during the 1,100,000 miles.

With these new inputs, a re-design of the semi trailer was initiated to soften the ride characteristics and to increase the working area of all tension members to meet the allowable stress levels indicated in Table 1 for 50,000 PSI yield steel and 600,000 cycles. This structural re-design increased the weight of the basic semi trailer from 6900# to approximately 7520# which was acceptable to all concerned. In actual fact, due to the NFS Cask weighing less than expected, other

operational refinements were made increasing the weight to 8000#.

Nuclear Assurance Corporation

This corporation negotiated license rights to the NFS Cask and contracted for four semi trailers quite similar in configuration, which have just entered service at this time.

XYZ Corporation

A corporation, desiring to be un-named at this time, has come to us now with a possible requirement for a heavier cask and a lighter semi trailer (perhaps weighing as little as 5500#). Our preliminary estimate indicates a reliable unit can be built in that weight range by extensive use of Titanium material.

Most of us earthlings know little about this material although it is used extensively in aerospace and high temperature industrial pressure vessels. Table 2 is a quick summary of properties of 100,000 PSI yield steel, 7075 T6 aluminum, and 5AL-2.5 SN Titanium.

Table 2 - Comparative Properties - Structural Materials

Property	Grade 100 Steel	7075-T6 Alum.	5AL-2.5 SN Titanium
Density #/cu. in.	0.283	0.101	0.161
Mod. Elasticity	29.7×10^6	10.4×10^6	16×10^6
Tensile Strength	125,000PSI	83,000	115,000PSI
Yield Strength	100,000PSI	73,000	110,000PSI
Elongation	18	11	10-20
Hardness	280Brinell	150Brinell	311Brinell
Fatigue Strength	29,600PSI @ 2,000,000 cy	23,000PSI Limit	95,000PSI @ 10,000,000 cy
Weldability	Good	Fair	Good
Available Form	Plate	Plate-Extrusns.	Plate-Extrusns.

QUALITY CONSIDERATION

A reliable semi trailer is a combination of material, design, test, and quality control.

The quality control program for a semi trailer to haul Nuclear Fuel has been established by working with all our customers. It is as follows:

A. Certification of structural materials.

B. Qualification of welders.

C. 100% Non-destructive testing of welding

D. Static test loading to 200% of rated load.

E. Close field inspection for first 50,000 miles.

For extremely light weight,Truss type, Tubular Semi Trailers, an additional and quite revolutionary structural monitoring system was conceived by Mr. Kenneth Dufrane of Nuclear Fuel Services. This system has been reduced to practice on the NFS and Nuclear Assurance Corporation units.

The system is simple but assures an early warning of impending structural problems. Basically, it consists of the following steps:

A. Welding the tubing subjected to high tensile stresses to pressure vessel standards.

B. Providing a valve to permit pressurizing the tensile members.

C. Pressurizing to approximately 10 PSI,and add a monitoring device to warn of depressurization if crack develops.

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SPENT NUCLEAR FUEL SHIPMENTS BY ROTARY-WINGED AIRCRAFT*

W. L. McKeown

ABSTRACT

The economic feasibility of utilizing rotary winged aircraft for shipments of radioactive material is investigated. Inherent in this shipment method is the consideration of route control and capability to tolerate accident consequences as an alternative or companion transport safety philosophy to the present philosophy of safety accomplished through the design of shipment packaging.

A case study involving shipments of spent fuel from nuclear power reactors to a fuel reprocessing plant in CY 1978 is considered, the preliminary results of which indicate that the transportation costs of this novel shipment mode are comparable to the present day costs associated with rail shipments. Advantages of this transport method related to accident likelihoods and security are presented and the reliability of a particular helicopter capable of performing this mission, the Boeing Heavy Lift Helicopter, are summarized. These preliminary results indicate a need to perform some fundamental accident consequence analysis to determine if the economic feasibility can be complemented by the required "safety" feasibility.

INTRODUCTION

As the nuclear power industry continues to expand, a proportional increase in the number of shipments of radioactive material will be experienced. Conventional rail or truck transport is planned for the bulk of these shipments; however, in view of the magnitude of radioactive material being shipped[1], in particular spent fuel, it may be prudent to evaluate other than surface modes of shipment, namely air. A similar thought is expressed in the AEC's Environmental Survey of

*The costs contained herein serve only as a foundation for this economic analysis and do not reflect actual production or operational costs. Valid costs will become available following production and actual operational experience.

the Transportation of Radioactive Materials[2], where it is noted that economic and safety studies are underway regarding shipment of irradiated fuel elements by air.

The objective of this paper is twofold. The first objective is to suggest consideration of an alternative or companion to the present philosophy of assuring safety in the transportation of spent fuel. This suggestion basically involves a substitute method for achieving the safety presently provided through the design of shipment packaging. This packaging assures that under normal and accident conditions, radioactive material is unlikely to be released while it also provides protection to both the public and transport related workers from direct radiation. Under this philosophy safety in transport does not depend on special routing.[3] The substitution involves a high degree of route control which separates population groups from the low level direct radiation associated with normal transport (~25 mrem/hr - 3 ft from rail car surface[4]) or from the consequences of a highly unlikely radioactive release which may occur as a result of a transporting vehicle accident. The second objective, which follows from the first is to identify the economic factors specific to one method of transportation amenable to this route control philosophy; namely, rotary winged aircraft or specifically the Boeing Heavy Lift Helicopter (HLH).

At this introductory point, it should be emphasized that the authors have not performed any accident consequence analysis related to HLH shipments. Only after such analyses have been performed would it appear prudent to seriously promote this transport method application. Therefore, this paper should be considered as a preliminary approach to a feasibility determination.

BACKGROUND

Some study in the past has been directed to unconventional modes of transport including fixed wing aircraft, conventional and STOL; dirigibles, and air cushion vehicles[5]. These past evaluations of "advanced" concepts generally did not consider or were not amenable to the route control transport safety philosophy. Furthermore, the

advantages of such transport methods (e.g., speed of shipment, increased payload) were generally offset by disadvantages of cost, transportation vehicle availability (e.g., required development of vehicle), ability to provide needed support facilities, and questions of extent of radiological consequences under accident conditions. Helicopters, although considered in these past evaluations, were found to be impractical because of limitations on payload weight and range. However, with advances in the state-of-the-art, notably the development of aircraft such as the Boeing Heavy Lift Helicopter (HLH), a prototype of which is planned for flight in the summer of 1975, there is a distinct possibility that such limitations may no longer be valid. Transportation of spent fuel by rotary-winged aircraft may provide certain advantages at costs competitive with more conventional means of transport.

SAFETY PHILOSOPHY

In accepting this paper for presentation, the Program Committee suggested a discussion of applicable regulations would be in order. The authors knowledge regarding these regulations and interpretations thereof is reflected by the limited discussion which follows.

Since we are discussing shipments of spent reactor fuel which is defined as a "large quantity" of fissile material, the most pertinent regulations under which the proposed rotary-winged aircraft shipment would fall, involve Part 71 of Title 10 and Part 173 of Title 49 of the Code of Federal Regulations. The former part, specifically paragraphs 71.34 through 71.36 deal with package standards, specifically, evaluations of single packages. It is under these paragraphs that reference is made to the required demonstration of the capabilities of the package to survive both normal transport conditions and hypothetical accident environments as specified. Obviously, it would be fortuitous if the hypothetical accident conditions which include a 30 ft. free drop test, a puncture, thermal, and water immersion test have any relationship to accident forces associated with certain unlikely rotary-winged aircraft accidents. Therefore, one approach to

the regulations would involve a redefinition of the hypothetical accident environment. In this approach, package design options, such as crushable members and impact orientation control systems, could be included to assure "gross" survivability of the package in the event of a transport vehicle accident. A second approach is based on a liberal interpretation of the allowance for alternative safety concepts included in the regulations under paragraph 71.34(c) which states that "Normal conditions of transport and hypothetical accident conditions different from those specified in §71.36 may be approved by the Commission if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment". In this approach, safety would be achieved entirely through a very precise route control emphasizing a remoteness from any population groups. This remoteness can be achieved through the HLH flight control system which will allow the aircraft under even severe flight conditions to follow a specified flight path within one mile. In both approaches emphasis would be placed on the reliability of the HLH, highlighted later in this discussion, which is expected to be superior to that achieved by large helicopters under non-battlefield conditions.

The other regulation of major interest is 49 CFR 173.393 which defines the allowable radiation levels at various locations relative to the package. Although the HLH will be capable of handling casks similar to the casks presently utilized for truck shipments, any savings in cask weight achieved through minor shielding reductions can be translated into additional "commodity" weight which, as will be subsequently discussed, plays an important role in shipment economics.

ECONOMICS

In presenting the economics of rotary-winged aircraft shipments, the approach utilized involves a case study of projected 1978 traffic into the Barnwell Nuclear Fuels Plant from all nuclear power plants within the HLH's range capability. The associated transportation costs are compared against the costs assuming rail shipments, the generally most economical present shipment mode. The assumed traffic appears in

Table 1 which represents a projection presented in the summer of last year to the Association of American Railroads by Mr. Walter Potts, the Manager for Nuclear Transportation at the Barnwell Facility.[6]

Table 1. 1978 Traffic - BNFP

Reactor	Type	Location	No. Ass'y Shipped	Indicated Shipping Method
Summer	P	Parr, S.C.	53	Rail
Robinson	P	Robinson, S.C.	52	Rail
McGuire I & II	P	Cowansford, N.C.	144	Rail
Oconee I, II, & III	P	Seneca, S.C.	183	Truck-Rail
Shearon Harris	P	Bonsal, N.C.	56	Rail
Brunswick I & II	B	Leland, N.C.	280	Rail
Crystal River	P	Red Level, Fla.	60	Rail
Sequoyah I & II	P	Daisy, Tenn.	144	Rail
Wattsbar	P	Watts Bar, Tenn.	64	Rail
Surry I & II	P	Porthsmouth, Va.	106	Barge-Rail
North Anna I & II	P	Mineral, Va.	104	Rail
St. Lucie	P	Stuart, Fla.	73	Barge-Rail
			Total - 510 MT.	

P - pressurized water reactor
B - boiling water reactor

The rough cost estimates were broken down into four categories: capital, insurance, handling, and freight. Freight and capital costs account for the major portion of total transportation cost and are the major items addressed in this paper.

Beginning with the relatively straightforward freight costs, Table 2 shows the breakdown of rail freight cost for each facility as estimated in reference 6. It should be noted that these costs reflect only the amounts paid to the railroads and do not allow for any connecting transportation.

In the other columns of table 2 the "freight" cost for HLH shipments are given as a function of the number of spent fuel elements being shipped/flight. These freight costs are based on the estimated

Table 2. Comparative Freight Costs

Reactor	Rail Cost Est.	Est. Rd. Trip Flt. Time (Hr.)*	HLH Freight Cost 3PWR EQ. FE/FLT	4PWR EQ. FE/FLT	5PWR EQ. FE/FLT
Summer	$16,480	1.5 (150 mi)	$23,700	$17,800	$14,200
Robinson	16,480	1.75 (230 mi)	27,000	27,000	16,300
McGuire I & II	55,900	2.4 (300 mi)	103,000	77,300	61,900
Oconee I, II & III	69,150	2.25 (280 mi)	123,000	92,300	73,800
Shearon Harris	26,676	3.2 (430 mi)	53,400	40,000	52,100
Brunswick I & II	53,350	3.1 (420 mi)	129,500	97,000	77,800
Crystal River	31,764	4.25 (600 mi)	85,000	63,750	51,000
Sequoyah I & II	76,500	3.65 (500 mi)	157,000	117,500	94,200
Watts Bar	34,590	3.7 (510 mi)	70,600	59,100	47,400
Surry I & II	59,150	5.25 (760 mi)	166,000	124,100	99,700
North Anna	60,080	5.35 (775 mi)	166,000	124,000	99,700
St. Lucie	45,234	5.6 (815 mi)	122,000	91,400	73,000
Total	$525,000		$1,226,000	$925,000	$740,000
Annual Required Flight Hours (FH)			(1370 FH)	(1031 FH)	(827 FH)
Annual Required FH/1800 FH Assumed Annual Utilization			0.76	0.58	0.46

* Assume 15 min P/L Discharge/Pickup Time, Cruise Speed 160 mph.

PWR EQ. FE/FLT = Pressurized Water Reactor Equivalent Fuel Elements per Flight

flight hours per trip shown and include the operation and maintenance expenses of the aircraft as estimated in Table 3. The flight crew expense shown is based on a crew of 3--a pilot, co-pilot, and cargo control crewman. The other components of the HLH direct operating expenses involve those costs associated with direct maintenance.

Table 3. Estimate of Hourly "Freight" Costs for HLH

I.	Flying Operation	
	Flight Crew	$ 75
	Fuel, Lubes, Misc	260
		$335
II.	Direct Maintenance	
	Flt. Line & Shop Labor	$ 70
	Airframe Overhaul	50
	Engine Overhaul	90
	Component Overhaul	90
	Replenishment Spares	260
		$560
	Total	$895

The incremental transportation cost which dominates the economics and is subject to the widest variation in parameters involves amortization of capital equipments. In somewhat simplistic terms, the trade-offs in this case study involve a high cost transportation vehicle (the HLH), a limited number of relatively moderate cost shipping casks and the cost of logistical support equipments vs. a larger number of high cost shipping casks and transporters. The aquisition cost of an HLH is estimated to be about $6 million.(7) The cost includes the fly-away cost (i.e., includes engine avionics, and furnishings) and a cost associated with ground support equipment and initial spares stockage including airframe, engines, and avionics spares. Depreciation is based on a 10 year life and assumes an annual aircraft utilization of 1800 hours.

To this cost one must add the amortization costs associated with the shipping casks. At this point the variable of shipping cask cost is introduced, a cost which can be significantly affected by the transport safety philosophy and allowable "normal" radiation levels applicable to the shipment. In particular, if the cask can be designed with emphasis being given to handling, normal transportation loads, and shielding while excluding major consideration of hypothetical accident forces, a simpler, less costly, and somewhat lighter cask should result. If furthermore, the general packaging requirements of 49 CFR 173 (specifically the radiation levels associated with the package) could be relaxed to a very limited extent (i.e., allowance of higher than standard dose rates from the package bottom surface which faces away from the HLH crew), a major savings in weight could be achieved which could be traded off against an increased "commodity" payload (number of spent fuel assemblies/cask). Since detailed cask design was beyond the scope of this paper, all costs associated with HLH shipment were parameterized to reflect a range in the realistic number of "equivalent" pressurized water reactor (PWR) elements which could be shipped. A range of 3 to 5 PWR elements was assumed which has previously been considered in the HLH "freight" cost estimates. For capital charge estimates, cask costs of $250,000 and $500,000 were postulated using reference 8 as a departure point. The higher value is typical of the presently available truck casks. The lower value, although somewhat arbitrary, is presented as an example of minimum cost which might be obtained if (1) a cask designed against accident forces is not required and (2) shielding is optimized for protection of flight and ground handling crews.

A closer look at Table 2, in particular, the estimated round trip flight time column, indicates that unless turnaround times for the casks at the reactor and reprocessing plant are significantly improved, it is this operation which will dictate the required number of casks. A requirement for 4 to 8 casks was therefore assumed.

The annualization of a range of cask costs is shown in Table 4. Four cases were considered reflecting the need for : (1) 4 - $250,000

casks; (2) 8 - $250,000 casks; (3) 4 - $500,000 casks; and (4) 8 - $500,000 casks. A fixed charge rate of 24.6%[9] was assumed to annualize the cask costs. Although in the example analysis the annual utilization of these casks, as shown in Table 2, ranged from 827 to 1370 hours, no credit was taken for this non-optimum usage (e.g., transfer and usage at other reprocessing facilities, although certainly possible, was not considered).

The same fixed charge rate used for the casks was assumed to annualize the HLH capital cost. Although the HLH flyaway cost, exclusive of ground support and spare costs, will be amortized over 10 years, the minimal impact of this difference on the fixed charge rate (less than 2%) did not warrant a more sophisticated evaluation. This annual charge for the HLH has been adjusted by the fraction of flight-hours required for spent fuel shipments divided by the aircraft's assumed annual utilization of 1,800 hours. As indicated in Table 2, the flight hours necessary to deliver the assumed traffic of spent fuel, representing about 510 MT or 1/3 of the Barnwell facility design capacity, ranges from 827 to 1,370 hours. Other commercial uses for the HLH, perhaps at other reprocessing plants, are assumed over the remaining annual utilization time. The parameterization of the annualized HLH capital charge against required flight time (which is a function of the cask capability to handle 3, 4, or 5 equivalent PWR fuel elements) is shown in Table 4 under the heading HLH capital charge.

The comparable capital charge for rail shipments was based on the reprocessing facility need for 18-24 rail casks at a cost of $1 to $1.5 million/cask. An adjustment factor of 0.34 was included in the calculation which represents the fraction of the annual reprocessing facility throughput which was assumed to be transported by the HLH (i.e., 510 MT/year divided by the Barnwell facility design capacity for 1,500 MT/year). Again, as with the HLH shipping casks, a fixed charge rate of 24.6% was assumed. Table 4 indicates the resulting annual charge.

It should be pointed out that the assumed number of casks could be significantly increased if recent suggestions by the Association of American Railroads are implemented. Reference 10 indicates that the

Table 4. Annualized Capital Charges* ($)

HLH Shipment									Rail Shipment			
Number & Cost of Casks									Cask Capital Charge			
									$1,000,000		$1,500,000	
4 @ $250,000			4 @ $500,000 or 8 @ $250,000			8 @ $500,000			Number of Casks Required			
									18	24	18	24
Number of Equivalent PWR Fuel Elements Per Cask												
3	4	5	3	4	5	3	4	5				
$246,000			$492,000			$984,000						
HLH Capital Charge ($6,000,000)												
$1.13M	$0.85M	$0.68M	$1.13M	$0.85M	$0.68M	$1.13M	$0.85M	$0.68M				
Total Annual Capital Charge									Total Annual Capital Charge			
$1.4M	$1.1M	$0.92M	$1.6M	$1.3M	$1.2M	$2.1M	$1.8M	$1.7M	$1.5M	$2.0M	$2.3M	$3.0M

*Based on shipment of 510 MT of spent fuel.

Association recommended that spent fuel be transported by special trains, along specified routes, at maximum speeds of 35 mph with trains stopping for passage of other trains. Conservative estimates indicate that such a policy could double both freight and capital charges[11].

The remaining charges to be evaluated involve handling and insurance. The handling charge is a strong function of the number of casks-trips required to transport the spent fuel. Estimates of this charge utilizing the rail cask (10 equivalent PWR elements per cask) and the HLH cask (3, 4, or 5 equivalent PWR elements per cask) are $210K, $480K, $380K, and $320K, respectively.

For rail shipments, the insurance on the fuel value (0.05% of fuel value - $25,000/PWR FE[12]) is not a major cost factor. If this insurance rate applies to cask value, this component of insurance cost would also be small.

For HLH shipments, the significant insurance cost would not be associated with the fuel or casks but with the HLH. The only conservative benchmark available for the analysis on HLH insurance involved a $350/flight hour rate estimate associated with the aircraft if utilized in a demonstration mode. It would be expected that the commercial rate would be significantly reduced; however even this conservative estimate would involve only a 10-15 percent increase in total annual example case charges.

Table 5 summarizes the costs associated with the example case - the shipment of 510 MT of spent fuel into the Barnwell Facility. Although the uncertainty bands on the total costs are large, the comparative nature of the results suggests that the potential for rotary-winged aircraft shipments of spent fuel should not, at this time, be ruled out purely on economic consideration. In fact, the possibility of an economic advantage to such a shipping method may exist.

Table 5. Summary of Annual Costs

	Rail	HLH
Freight	$525,000	$740,000 - $1,226,000
Capital	$1,500,000 - $3,000,000	$920,000 - $2,100,000
Handling	$210,000	$320,000 - $480,000
Insurance	Small ?	?
TOTALS	$2,235,000 - $3,735,000 + insurance	$1,980,000 - $3,800,000 + insurance

SOME NON-QUANTIFIABLE ADVANTAGES OF ROTARY WINGED AIRCRAFT SHIPMENTS

In considering the possibilities of rotary winged aircraft shipments of radioactive material, certain non-quantifiable advantages can be cited. The five advantages listed in Table 6 pertain not only to safety, through an undefineable reduction in accident likelihoods, but also have economic and safeguard implications.

Table 6. Non-Quantifiable Advantages of Rotary Winged Aircraft Shipments

1. Control over Transporting Crew Training and Expertise
2. Security (Radar Followed, Limited Range and Transit Time)
3. Relatively Insensitive to Strikes; Labor Disturbances
4. Independence from Requirements of Shippers Groups
5. No Dependence on Varying Conditions of Transportation Routes (eg. Reduced Speed Trackage)

The safeguards implications would be most appropriate to shipments of LMFBR or HTGR fuel or in the shipment of plutonium or uranium 233 in amounts, alone or in combination, exceeding 5000 grams. The economic case example considered in this paper did not include LMFBR or HTGR fuel. However, it is expected that HLH shipment of these fuels, because of their acknowledged higher transportation costs relative to LWR fuels,[13] will be equally, if not more, competitive than rail shipments.

IMPORTANT HLH SAFETY AND RELIABILITY FEATURES

Although helicopters operating in regularly scheduled service have experienced a degree of safety high enough to permit the commercial carrying of passengers, the HLH should exceed this safety level by a wide margin. Projections of increased safety are being made analytically pending the accumulation of data from actual HLH flight experience.

Some of the more significant advances that will contribute to helicopter safety and reliability are:

1. Rotor Blades
    a. Fiber glass composite construction with a resultant strength increase and elimination of corrosion.
    b. Built-in fault detectors which give early warning of impending blade problems.
    c. Multiple load paths that assume the load in the event of failures elsewhere.
2. Rotor Heads
    a. Elimination of lubrication and thus the safety problems of dry bearings and incorrect lubrication.
    b. Multiple load paths and built-in fault detectors as in the rotor blades.
3. Flight Controls
    a. The world's first electronic helicopter flight control system which will provide about 55 times the reliability of the control system on its predecessor helicopters.
    b. Elimination of high pressure hydraulic lines with their tendencies to malfunction.
    c. Triply redundant independent electrical control lines.
    d. Improved instrument flight capability through computer assisted flight maneuvering.

e. Electronic load stabilization resulting in decreased load oscillation and other hazardous variables.

4. Power Plants
   a. Three engine safety with any two capable of carrying a full load.
   b. Improved fire detection, containment and extinguishing.
   c. Engine operation during cruise at relatively low percentage of total power available.

As can be easily seen by review of the foregoing technological advances, the HLH will probably be the safest most reliable helicopter flying. The HLH will experience far fewer in-flight malfunctions than current helicopters. However, the judicious selection of flight routes over isolated areas should permit a safe and uneventful landing should a mission affecting malfunction occur.

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9. LMFBR Spent Fuel Transport: Parametric Studies, J. Nichols and A. Irvine, ORNL-TM-3503, February, 1972, p. 58.

10. Nucleonics Week, April 18, 1974, p. 1.

11. Reference 10 refers to a doubling of the cost of rail transport (freight cost) based on $14-15/car mile rate. Using this figure for the example case would increase the freight costs of Table 2 by 10-15%. Capital costs would increase directly with the number of casks required.

12. Personnal communication with W. Potts of Allied Gulf Nuclear Services, May, 1974.

13. Reference 1, p. 6-16.