MASTER

PROCEEDINGS

THIRD INTERNATIONAL SYMPOSIUM
PACKAGING AND TRANSPORTATION
OF RADIOACTIVE MATERIALS

AUGUST 16-20, 1971/RICHLAND, WASHINGTON, U.S.A.

SPONSORED BY • U.S. ATOMIC ENERGY COMMISSION AND THE FOLLOWING HANFORD CONTRACTORS
• ATLANTIC RICHFIELD HANFORD COMPANY • BATTELLE-NORTHWEST • DOUGLAS UNITED NUCLEAR, INC.
• ITT/FEDERAL SUPPORT SERVICES, INC. • J A JONES CONSTRUCTION COMPANY
• VITRO CORPORATION/HANFORD ENGINEERING SERVICES • WADCO CORPORATION

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

PROCEEDINGS OF THE
THIRD INTERNATIONAL SYMPOSIUM ON
PACKAGING AND TRANSPORTATION OF
RADIOACTIVE MATERIALS

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness, of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Richland, Washington
August 16-20, 1971

SYMPOSIUM COMMITTEE

C. L. Brown	Symposium Chairman	BNW*
E. M. Johnston	Co-chairman	WADCO*

Arrangements

D. R. Oden	Secretary	BNW
J. S. Stoakes	Treasurer and Registration	BNW
D. A. Hoover	Program	ARHCO*
T. E. Dabrowski	Program	DUN*
E. F. Curren	Program	ARHCO
A. O. Dodd	Program	USAEC-RL
D. A. Knapp	Transportation and Housing	ITT*
W. A. Bevan	Transportation and Housing	ITT
G. M. Dalen	Public Relations	BNW
G. R. Petersen	Public Relations	BNW
W. S. Kelly	Exhibits	BNW
R. J. Bashor	Proceedings	BNW
A. W. DeMerschman	Meeting Arrangements	WADCO
J. B. Edgar	Meeting Arrangements	WADCO
G. F. Penn	Hospitality	USAEC-RL
R. F. Garrison	Tours	USAEC-RL
E. D. Clayton	Banquet Arrangements	BNW
C. N. Zangar	Special Assignment	Vitro
I. E. Dunn	Special Assignment	JAJ*

Advisory Committee

D. R. Swindle	USAEC-HQ
W. A. Brobst	USAEC-HQ
J. A. Sisler	USAEC-HQ
L. B. Shappert	ORNL

Coordinators for AEC-RL

L. F. Perkins	USAEC-RL
C. F. Goodner	USAEC-RL

- * ARHCO - Atlantic Richfield Hanford Company
BNW - Battelle, Pacific Northwest Laboratories
DUN - Douglas United Nuclear, Inc.
ITT - ITT/Federal Support Services, Inc.
JAJ - J. A. Jones Construction Company
Vitro - Vitro Corporation/Hanford Engineering Services
WADCO - WADCO Corporation

CONTENTS

SESSION 2

U. S. Regulations

<u>Title</u>	<u>Author</u>	<u>Page</u>
How Abnormal is Normal?	J. A. Sisler, Transportation Branch, USAEC	1
Today's Role of DOT in Regulating the Transport of Radioactive Material	A. W. Grella, Department of Transportation, Office of Hazardous Materials, Washington, D. C.	25
Transportation Safeguards	D. L. Crowson	35
Improving Applications for Shipping Container Approvals	R. L. Stevenson, Div. of Materials Licensing, USAEC	36
A Guide to the U. S. Transportation Regulations	A. C. Cornish & H. Simens, Bureau of Radiological Health, Calif. Dept. of Public Health, Berkeley, Calif.; and Bechtel Corp.	45

SESSION 3

International Regulations

The 1970 Review of the IAEA Regulations for the Safe Transport of Radioactive Materials	G. E. Swindell, I.A.E.A., Vienna	52
How Changes in the IAEA Regulations May Affect 10 CFR Part 71	R. F. Barker, Product Standards Branch, USAEC	62
Particular Solution to Meet Regulations Requirements	A. Redon & Y. Sousselier, Commissariat a L'Energie Atomique, France	75
AEC Licensing of the Export of Radioactive Materials and Facilities	C. J. Holloway, Jr. & J. Saltzman, Indemnity & Export Control Branch, USAEC	98

SESSION 4A

Isotopes and Waste Packaging Technology (Part 1)

The Design and Testing of a Lithium Hydroxide Shielded Cask for the Transportation of Fast Neutron-Emitting Heat Sources	R. D. Seagren, Oak Ridge National Laboratory	106
Transportation Safety of Trans-Plutonium Actinides	C. M. Copenhaver, USAEC-ORNL	113
Shielded Neutron Shipping Cask	C. L. Hanson, M. S. Coops, E. D. Arnold, Lawrence Radiation Laboratory	125
Recent Developments in Nondestructive Evaluation Applicable to Encapsulated Isotopes	D. R. Newman, J. C. Crowe, Battelle-Northwest	138

<u>Title</u>	<u>Author</u>	<u>Page</u>
SESSION 4B		
Accident Experience		
The AEC Accident Record and Recent Changes in AEC Manual Chapter 0529	W. C. McCluggage, USAEC, Div. of Operational Safety	166
Single Vehicle Wreck on Interstate Highway - February 1970	D. F. Cronin, United Nuclear Corp.	167
Enrichment Errors in Interplant Shipments	D. L. Dunaway, National Lead Company of Ohio	185
Preventing Shipping Incidents: Two Case Studies	R. C. Becker, Lawrence Radiation Laboratory	188
SESSION 5A		
Isotope and Waste Package Technology (Part 2)		
Shielded Transport Containers for Intermediate and High-Level Radioactive Wastes	W. Bechthold, J. Heil, M. C. Schuchardt, R. Wolf, Karlsruhe Nuclear Research Center	195
"Celotex" Insulated Shipping Containers	E. E. Lewallen, E. I. duPont deNemours, Savannah River Plant	209
New Facility for Packaging Radioactive Liquid Wastes	D. E. Bloomfield, Gilbert Associates	210
The Design of Low Cost Fire Shield for Small and Large Radioisotope Shipping Containers	J. D. W. de Lind van Wijngaarden & R. A. Harrod, Atomic Energy of Canada, Ltd.	227
SESSION 5B		
Shipment of UF ₆		
Enriching Services Transportation	B. M. Robinson, USAEC, Oak Ridge, Tenn.	262
Nuclear and Radiological Safety Aspects in Transporting Uranium Hexafluoride	W. A. Johnson, C. E. Newlon, R. G. Taylor, Union Carbide	272
Engineering and Quality Control Considerations in the Safe and Economical Handling of Uranium Hexafluoride Cylinders	J. W. Arendt, Union Carbide	288
Economics of Transportation - Uranium Hexafluoride	G. R. Brooks, A. T. Freeman, J. A. Lamb W. R. Simpson	302

SESSION 6A

Spent Fuel - Experience and Problems (Part 1)

Indian Experience in Transportation of Irradiated Fuel	N. Srinivasan, Government of India	322
Experience in the Transportation of Radioactive Materials in Italy	C. Faloci, A. Susanna, C.N.E.N., Rome, Italy	336
Neutron Shielding Problem - Its Effect on Spent Fuel Shipping Cask Design	J. D. Rollins, Nuclear Fuels	355
Practical Problems in Spent Fuel Shipping Related Cask Concepts	P. Blum, Transnuclear Group, New York	355

SESSION 6B

Isotope and Waste Packaging Technology (Part 3)

Transportation of Radioactive Waste: Worth Its Salt?	W. A. Brobst, USAEC	356
ATMX-500 Railroad Car Radioactive Waste Container	E. P. McDonald, Monsanto Research Corp.	380
ATMX-600 Railcar - A New Concept in Radioactive Waste Shipments	F. E. Adcock, Dow Chemical	399

SESSION 7A

Spent Fuel Experience and Problems (Part 2)

Effect of Highway Weight Limits on Cost of Spent Fuel Shipments	E. D. North, Nuclear Fuel Services	414
Spent Fuel Transportation - State of the Art	R. W. Peterson, Allied Chemical Nuclear Products	415
Radioactive Transportation Activities for a Fuels Reprocessing Plant	D. D. Wodrich, Atlantic Richfield Hanford Company	437
Spent Fuel Transportation Experience	H. Walchli, Westinghouse Corp.	446

SESSION 7B

Isotope and Waste Packaging Technology (Part 4)

Operating Experience in the Shipment of Radioactive Materials	V. K. Iya, R. G. Deshpande, M. D. Kulkarni, Gov't. of India	447
Development, Design and Construction of a Shipping Package for Tritiated Heavy Water	C. G. Cameron, R. M. Smith, AECL-Pinawa, Manitoba, Canada	462

<u>Title</u>	<u>Author</u>	<u>Page</u>
Processing and Packaging of Solid Wastes from BWR's	H. L. Loy, D. C. Saxena, General Electric Co., San Jose, Calif.	478
Design Considerations for High Volume, IAEA Type B Protective Overpack Shipping Containers in Use with Current Transportations Systems	K. Gablin, Protective Packaging, Inc., Tacoma, Wash.	490

PAPERS FROM SESSIONS 8 THROUGH 16
APPEAR IN VOLUME II

<u>Title</u>	<u>Author</u>	<u>Page</u>
SESSION 8A		
Insurance and Radiation Safety		
Insurance and Indemnity Protection for Nuclear Materials in Transport	J. Saltzman & C. J. Holloway, USAEC, Indemnity and Export Control Branch	503
Third Party Liability Insurance and Government Indemnity Associated with the Transportation of Radioactive Material	L. G. Cummings, Marsh & McLennan, Insurance, New York	516
Processing Radioactive Shipments from the Standpoint of Radiation Safety	C. W. Buckland Jr., Los Alamos Scientific Laboratory	532
Radiation Survey of the Transportation of Radioactive Materials	G. D. Schmidt, U.S. Public Health Service, Rockville, Maryland	552
A Method of Controlling Radiation Exposure of Persons in Aircraft During Transportation of Radioactive Materials	P. A. Lecomte, Belgian Airlines, Brussels, Belgium	565
SESSION 8B		
Spent Fuel Packaging Technology (Part 1)		
Analysis of Packaging Compliance to Transport Regulations	H. Lucas and M. Labrousse, Commissariat L'Energie Atomique Fontenay-Aux-Roses, France	592
Review of Experimental Studies on the Fissile Fuel Shipping Cask in Japan	S. Aoki, Tokyo Institute of Technology	603
Evaluation of Metal Seal Criteria for Spent Fuel Cask Closures	R. E. Latham and R. T. Brown, Gray Tool Co., Houston, Texas	615
SESSION 10A		
Spent Fuel Packaging Technology (Part 2)		
Recent Developments in Lead Shields for Transport Containers in the United Kingdom	R. Smith, Lead Development Association, London, England	634
Shielding Aspects of LWR Spent-Fuel Shipping Casks	B. A. Engholm, Gulf General Atomics, San Diego, Calif.	652
Lead Remelt Experience with 60-Ton Cask	E. C. Lusk, Battelle Columbus Laboratories, Columbus, Ohio	664
Weight Minimization of Casks for Shipment of High Burnup and Recycle LWR Fuels	J. L. Ridihalgh and R. E. Best, Battelle Columbus Laboratories	675
Impact Testing of Chemical Lead, Cask Models, and an Energy Absorbing Device	J. H. Evans, Oak Ridge National Laboratory	676

<u>Title</u>	<u>Author</u>	<u>Page</u>
SESSION 10B		
Administrative Experience		
Administration of the Radioactive Materials Transportation Safety Program	W. A. Pryor, Oak Ridge National Laboratory	689
The Responsibilities of the Consignor and the Consignee	K. R. Schendel, Westinghouse Electric Corp., Pittsburgh, Pa.	708
Development of Documentation of the Transport of Radioactive Materials	J. Fairey, UKAEA - AWRE, Berks, England	719 A
Transportation and Handling of Radioactive Pharmaceuticals and Radioactive Chemicals	S. H. Sanger, Cabs Unlimited, Mountain View, Calif.	720
SESSION 11A		
Spent Fuel Packaging Technology (Part 3)		
An Experimental Study on the Drop Impact of Spent Fuel Shipping Cask	S. Aoki, S. Nakata, S. Shimamura, Y. Kanae, A. Muramatsu, H. Nakazawa, Tokyo Institute of Technology	727
Impact Testing of Cask Fin Specimens	F. C. Davis, Oak Ridge National Laboratory	740
Impact Tests of Models at the Savannah River Plant	J. W. Langhaar, E. I. duPont de Nemours Co., Atomic Energy Division, Wilmington, Del.	754
On the Prediction of Deformation and Deceleration of a Composite Cylindrical Body for the Corner Drop Case	K. Lee, ATCOR, Inc., Elmsford, N. Y.	773
Economics, Compositions and Process Equipment for Radioactive Waste Solidification	L. Rutland, ATCOR	785
SESSION 11B		
Nuclear Safety		
Application of Fissile Material Storage Limits to Criticality Safety Analysis in Transport	J. T. Thomas, Union Carbide Corp., Oak Ridge, Tenn.	792
Criticality Safety Evaluations of Shipping Containers Used by Idaho Nuclear Corporation at NRTS	J. K. Fox and W. G. Morrison, Idaho Nuclear Corp., Idaho Falls, Idaho	803
ATMX Railcar - Nuclear Safety Evaluation	J. D. McCarthy, Dow Chemical Company, Golden, Colo.	822

<u>Title</u>	<u>Author</u>	<u>Page</u>
SESSION 12A		
Spent Fuel Packaging Technology (Part 4)		
Study on Fire Resistivity for Irradiated Fuel Shipping Casks	T. Moriya and H. Shimada, Fire Research Institute Japan, Tokyo, Japan	833
Heat Transfer Calculations for Spent Fuel During Hypothetical Accident Conditions	C. J. Anderson, Nuclear Fuels Services, Wheaton, Maryland	846
A Thermal Test Evaluation Method for Lead & Uranium Shielded Casks	J. C. Glynn and R. H. Odegarden, USAEC, Washington, D.C.	846
Shipping Container Design for Fire Accident Protection, Minimum Restriction to Decay Heat Removal	A. Serkiz and E. C. Lusk, Battelle Columbus Laboratories, Columbus, Ohio	847
SESSION 12B		
Non-Irradiated Fuel Packaging Technology		
Evaluation of Plutonium form on Safety During Transportation	L. M. Knights	860
Drop Tests of Shipping Containers for PWR Type New Fuels	S. Aoki, S. Nakada, S. Shimamura, Y. Kanae, T. Koguchi, and Y. Seki, Tokyo Institute of Technology	861
Shipping Package for Unirradiated Power Reactor Fuel Assemblies	K. E. Kropp, United Nuclear Corp., New Haven, Conn.	873
A Simple Shipping System for Power Grade Plutonium Oxide	R. E. Giebel and R. G. Leeb1, Dow Chemical Co., Golden, Colo.	886
SESSION 13A		
Spent Fuel Packaging Technology (Part 5)		
Tiedown Systems for Casks	E. E. Lewallen, E. I. duPont deNumours, Aiken, South Carolina	886
Compact Metallic Impact Limiters for Shipping Containers	R. J. Burian and E. C. Lusk Battelle Columbus Laboratories	887
Shock and Vibration Measurements During Normal Rail and Truck Transport	J. T. Foley and M. B. Gens, Sandia Laboratories	905
Upgrading of PM-3A Casks	R. A. Scaggs, E. I. duPont deNemours, Aiken, S.C.	934

<u>Title</u>	<u>Author</u>	<u>Page</u>
SESSION 13B		
Needs and Progress in Standardization		
Future Trends in Automated Freight Handling Systems	J. C. Wolff, Director, Drake Chelan--Stewart Dougall, Inc., New York, N.Y.	944 <i>A</i>
Quality Control of Shipping Cask Fabrication	A. Short, Vitro Corp., Richland, Washington	945
SESSION 15		
New Spent Fuel Cask Designs		
Design, Fabrication, and Performance Testing of Model 160 Shipping Cask	C. M. Kershner, Allis-Chalmers, York, Pa.	973
Uranium Cask for Shipment of Fort St. Vrain HTGR Spent Fuel	C. R. Davis, Gulf General Atomic, San Diego, Calif	974
Shipping Casks for Spent Fuel with High Burn-Up	F. Schmiedel and R. Dietrich, Fried, Krupp GmbH, Nuclear Division, Essen, Germany	989
General Description of Expected Size and Capability of 1100 MWe Reactor Spent Fuel Shipping Casks	J. P. Malone, ATCOR, Elmsford, N.Y.	990
SESSION 16		
Advanced Technology		
Contribution to the Design of Type B Packages for High Residual Power Irradiated Fuels Elements	H. Bruel and C. Bochard, Societetbatel SLPI, France	995
An LMFBR Fuel Shipping Cask Concept and Observations Pertaining to Hazard Survival Potential	A. R. Irvine and L. B. Shappert, Oak Ridge National Laboratory	1007
Air Cushion Support for Highway Transport of Heavy Loads	W. H. Trask, Battelle-Northwest	1015
Computer Movies for Simulation of Mechanical Tests	L. H. Baker, B. J. Donham, W. S. Gregory, and E. K. Tucker, LASL, Los Alamos, New Mexico	1028
PAPERS NOT PRESENTED		
Heat Transfer Analysis of a Cesium-137 Shipping Cask	G. R. Bray, J. J. Darby, and C. E. Gruhl	1043
A Study of the Effect of Impact and of the Use of a Toroidal Shell as an Energy Absorber on a Cylindrical Container of Lead and Steel	L. R. Shobe and B. R. Dewey	1052

✓ INSURANCE AND INDEMNITY PROTECTION FOR NUCLEAR MATERIALS IN TRANSPORT

Jerome Saltzman
C. James Holloway, Jr.

ABSTRACT

To provide protection to the general public and promote the growth of the nuclear industry a unique system of liability coverage for nuclear operations including transportation of nuclear materials, has been developed. The principal features of the insurance and indemnity aspects of shipments of radioactive materials are: (a) the formation of nuclear liability insurance and nuclear property insurance pools by the casualty and property insurance companies, and (b) the establishment, through Federal legislation, of a program of government indemnity and underlying financial protection (usually nuclear liability insurance) to cover public liability for nuclear incidents. Liability insurance coverage for transportation is provided by means of two basic policies - the Facility Form and the Supplier's and Transporter's Form. Indemnity coverage is supplied to certain licensees and contractors of the Commission. Nuclear property insurance provides first party coverage for certain nuclear property. Some insurance coverage is available for ocean transportation and proposed legislation would extend indemnity coverage for some aspects of ocean transportation. There are also international conventions covering third-party liability of interest to persons involved in the transport of radioactive materials. While certain problem areas remain in the insurance and indemnification of transportation of radioactive materials, by cooperative efforts, these problems can be overcome.

INTRODUCTION

The rapid increase in the volume and scope of shipments of radioactive materials has been facilitated by a system of insurance and indemnity which recognizes the need for flexibility in providing coverage for an industry utilizing newly developed technologies. The very concepts of coverage by

nuclear liability and property insurance pools supplemented by government indemnity is a reflection of the belief of the nuclear, transportation, and insurance industries and the legislative and executive segments of the Government that novel insurance approaches were needed to assure protection to the general public and to promote the growth of the nuclear industry.

The principal features of the insurance and indemnity aspects of shipments of radioactive materials are: (a) the formation of nuclear liability insurance and nuclear property insurance pools by the casualty and property insurance companies, and (b) the establishment, through Federal legislation, of a program of government indemnity and underlying financial protection (usually nuclear liability insurance) to cover public liability for nuclear incidents.

NUCLEAR INSURANCE POOLS

Four pools were formed in 1956 by the casualty and property insurance industry to provide unprecedented amounts of liability and property insurance capacity identified by them and the nuclear industry as required by the nuclear activities in the private sector of the economy made possible and encouraged by the Atomic Energy Act of 1954. The insurance companies owned by stock holders (the stock insurance companies) formed two pools each with an underwriting capacity of about \$46.5 million. The Nuclear Energy Liability Insurance Association (NELIA) was established to provide liability coverage and the Nuclear Energy Property Insurance Association (NEPIA) was established to provide insurance for the purchaser's own property. The insurance companies owned by their policy holders (the mutual insurance companies) also in effect formed two pools, each with a capacity of \$13.5 million. Nuclear liability insurance was written through Mutual Atomic Energy Liability Underwriters (MAELU) and nuclear property insurance was written by Mutual Atomic Energy Reinsurance Pool (MAERP).^{*} Thus, when the pools were formed in the mid-1950s, the combined underwriting capacity that could be written for any single nuclear liability insurance risk was \$60 million and the combined underwriting capacity that could be written for any single nuclear property risk was \$60 million dollars. This total insurance capacity of \$120 million which could be placed against a single risk to cover

---

^{*}MAERP is actually a single pool for liability and property, which allocates its capacity to the two types of coverage. MAELU is 100% insured by MAERP.

both liability and property losses represented essentially the entire capacity that organizations constituting the world-wide insurance market could make available for this type of risk. Almost one-third of this capacity represented reinsurance in the foreign insurance markets (such as the Lloyds Syndicates).

The underwriting capacities for nuclear liability and property insurance have increased through the years so that today there is \$82 million in nuclear liability insurance available for each risk and \$84 million in nuclear property insurance for a total insurance coverage (liability and property) of \$166 million for each risk. In the past year or two, major undertakings by the industries involved have attempted to pool capacity for potential catastrophic losses in such areas as the operation of jumbo aircraft and supertankers. The combined capacity of the four nuclear pools remains one of the largest amounts of insurance funds amassed for a single risk even when measured against these undertakings.

PRICE-ANDERSON INDEMNITY

While the formation of the pools appeared at the time to provide adequate property coverage for nuclear installations, the capacity developed by the nuclear liability pools, albeit in unprecedented amounts, was still not considered to be sufficient to cover potential, though hardly credible, liability claims from catastrophic nuclear accidents. Congress, through the Joint Committee on Atomic Energy (JCAE), and the executive branch of Government through the Atomic Energy Commission, after some two years of studies and hearings, developed the Price-Anderson system. This system, enacted in 1957 as an amendment to the Atomic Energy Act, was designed to supplement the nuclear liability insurance that the pools were developing so as to best serve the dual goals of protecting the public and fostering private participation in the atomic energy program by assuring that, in the event of a serious accident, funds would be available for the payment of damages.

LICENSEES OF THE COMMISSION

Under the Act, certain licensees are required to furnish proof of financial protection (in the form of nuclear liability insurance or as otherwise provided in the Act) to cover public liability claims up to an amount specified by the AEC. When financial protection is required,

the AEC must provide indemnity protection for up to \$500 million for each nuclear incident over and above the amount of financial protection. In some instances, financial protection is not required and indemnity is provided either from the "first dollar" or in excess of \$250,000. The liability of all persons liable is limited to the combined sum of the financial protection and the Commission's indemnity, up to a total of \$560 million. Under the Act, the AEC requires financial protection of, and indemnifies, all licensees for nuclear reactors and fuel reprocessing plants. Licensees authorized to operate reactors designed with a power level of greater than 100 electrical megawatts are required to provide financial protection equivalent to the maximum capacity available from private sources, which is currently \$82 million. The AEC indemnifies these reactor licensees in excess of this amount up to \$560 million.

The AEC has discretion under the Act to require financial protection of, and to indemnify, nuclear materials licensees. While the Commission has not exercised this discretionary authority to extend indemnity to materials licensees to date (except for fuel in storage prior to reactor operation) it has the matter under continual study.

COMMISSION CONTRACTORS

The second major group to which Price-Anderson protection is extended is the Commission's contractors. Under the Price-Anderson Act, the Commission is authorized to indemnify contractors for the construction of production or utilization facilities (such as nuclear reactors) and such other of its contractors who are engaged in activities for the benefit of the United States involving a risk of public liability for a substantial nuclear incident. The indemnification protection for Commission contractors is in the amount of \$500 million. While the Price-Anderson Act authorizes the Commission to require its contractors to furnish private financial protection such as nuclear liability insurance, the Commission has chosen not to require its contractors to purchase such insurance, though sometimes insurance otherwise applicable is permitted to be maintained (as in the case of a facility in which both Commission and private work is performed).

The Commission has entered into indemnity agreements with all of its prime contractors who are engaged in an activity involving the risk of a substantial nuclear incident.

LIABILITY INSURANCE COVERAGE FOR TRANSPORTATION

The two nuclear liability pools utilize two basic forms of liability insurance policy which provide transportation coverage for their insureds. The first form is the nuclear energy liability policy (facility form) which has been approved by the Commission for use in meeting its requirements for underlying financial protection. This facility form is also sold by the pools to operators of nuclear installations, such as fuel fabricators, who are not subject to the Price-Anderson Act. The second form is used by pools to provide nuclear liability insurance to suppliers and transporters who are not operators of nuclear installations, and is called the nuclear energy liability policy (supplier's and transporter's form).

Facility Form

The facility form covers loss and loss expense resulting from liability because of bodily injury or property damage caused by "the nuclear energy hazard." Under the policy, the "nuclear energy hazard" is the radioactive, toxic, explosive, or other hazardous properties of nuclear material, when such materials are at the facility designated in the policy or, subject to specified limitations, are being transported to or from the facility.

One novel feature of the facility form of policy is that it covers the liability of any person or organization (other than the United States and its agencies) for a covered nuclear incident. (Conventional liability policies, on the other hand ordinarily cover only the liability of the named insured.) Another novel feature is the coverage of the off-site property of persons liable for the incident as if this off-site property had belonged to another. Also unusual is the policy's coverage for transportation.

Under the facility form, when it is certified by the licensee as evidence of financial protection, coverage is provided in the course of transportation (including handling and temporary storage incidental thereto) within the territorial limits of the United States, its territories, Puerto Rico and the Canal Zone. The insured shipments covered by the policy are shipments of source material (uranium and thorium), special nuclear material

(enriched uranium and plutonium), spent fuel or waste* that is being transported to the facility designated in the policy from any location (except an indemnified nuclear facility) or from the facility to any other location. The transportation coverage is provided until the material is removed from the transporting conveyance for any purpose other than the continuation of its transportation. While there are certain exceptions which should be reviewed by anyone transporting nuclear materials under this type of policy, in general it can be stated that where two facility forms of nuclear liability policies apply to a single shipment, the policies are worded to assure that it is the shipper's (rather than the receiver's) policy that will provide the coverage.

The coverage is subject to various exclusions in the policy, particularly the exclusion from coverage of "property damage to nuclear material in the course of transportation to or from the facility including handling or storage incidental thereto." However, in general, the insurance policies cover liability for damage to companion cargo, and to such property as rights-of-way, roads, terminals, ports, bridges and other property used in connection with the transportation of the material, even if the person found to be liable for the nuclear incident is the owner of such property.

Supplier's and Transporter's Form

The transportation coverage provided by the pools under the supplier's and transporter's form is intended among other things to supplement the transportation coverage in the facility form of policy. The supplier's and transporter's form, in addition to providing coverage for liability resulting from transportation of nuclear materials not within the scope of the facility form and not within the scope of Price-Anderson indemnity, also provides liability coverage to a named insured (such as a transportation firm) who might be uncertain as to any other coverage or the limits of coverage for shipments it is handling. Like the facility form of policy, the supplier's and transporter's form is omnibus with respect to transportation coverage, but

---

* The pools do not normally afford nuclear liability coverage under the facility form of policy to licensees for radioactive isotopes (called byproduct material) but coverage for liability from applications of radioisotopes in normal commerce is afforded by the conventional liability insurance market.

otherwise, unlike the facility form, covers only the named insured. This form provides coverage for damage to the property of others, not the off-site property of the named insured. Like the facility form, the supplier's and transporter's form does not usually cover liability arising out of the shipment of commercial radioisotopes; however, liability for such shipments of radioisotopes can be covered through general liability insurance policies.

TRANSPORTATION COVERAGE UNDER PRICE-ANDERSON INDEMNITY

The transportation coverage under the Commission's indemnity agreements with licensees and under its agreements with its contractors is quite similar to the coverage afforded by the facility form of insurance policy. The indemnity agreements cover all of the "radioactive material" in the course of transportation from the indemnified location and the material to the indemnified location (so long as the transportation is not "from" another indemnified location). This coverage applies to material whose transportation to or from the location is not interrupted except for continuation of the transportation or incidental temporary storage. The radioactive material covered by the policy is source, special nuclear, and byproduct material, which is used or to be used in, or produced as a result of the operation of, the indemnified facility. This definition then, could include some byproduct material (radioisotopes) which would not be covered in the facility form or shipper's and transporter's form of liability insurance policy.*

The indemnity agreement excludes coverage for damage to the radioactive material being transported, as does both forms of insurance policy. However, in contrast to the policies, the indemnity agreement also excludes coverage for damage to the container of the material and damage to the vehicle on which the container is being transported. An additional exclusion in the agreement is the workmen's compensation claims of employees of persons indemnified whose duties involve the performance of work on the vehicle transporting the radioactive material.

---

* Often, however, the practical effect of the limitation of coverage of byproduct material to such material "used or to be used in or produced as a result of the operation of the . . . facility" in the indemnity agreement will mean that the covered byproduct material is "spent fuel or waste" and within the coverage of the nuclear facility pool insurance.

NUCLEAR PROPERTY INSURANCE

Since 1959 the standard fire and property insurance policies have contained an exclusion of damage to the insured's property and loss of revenue of the insured caused by nuclear reaction, nuclear radiation, or radioactive contamination regardless of the source of the incident or whether the incident falls within the scope of the Price-Anderson Act. We understand that the insurers consider that property damage caused by a nuclear incident would often be covered by a nuclear liability insurance and that this coverage should be excluded from the conventional fire and property policy (i.e., non-nuclear policy) to avoid duplication of insurance for the same property.

Since, however, certain property (such as the nuclear facility property, the nuclear material involved in the incident itself, and for indemnity coverage also the container of the material and the transporting vehicle) is excluded from both coverage of nuclear liability insurance (and indemnity) and, in effect, conventional property insurance there was a market for coverage of excluded business property and the nuclear insurance pools established forms of property insurance to provide this coverage.

The nuclear property pools, Nuclear Energy Property Insurance Association (NEPIA) and Mutual Atomic Energy Reinsurance Pool (MAERP) currently supply a total of \$84 million per policy to cover damage to the property of an operator of a nuclear facility caused by an accident at the nuclear facility. This standard nuclear energy property insurance policy covers damage to the property (which is delineated geographically and described in the Declarations page of the policy) from radioactive contamination (from onsite sources) and all risks of direct physical loss except as provided in the policy. This policy then provides first-party property insurance at the described location against conventional fire and other property risks as well as the nuclear risks. It does not provide coverage for nuclear risks where the source of the nuclear contamination is outside the location described in the agreement. Nor does this form of nuclear property insurance provide coverage away from the location such as for material in transport.

NEPIA and MAERP do provide nuclear property insurance for nuclear fuel in transport, except while water borne, within the continental United States

(and Canada where the origin or destination of the transportation is in the United States) and the containers, materials, and supplies shipped with the fuel. There is one form of policy to insure the shipper for the property loss and another form to insure the carrier. The policies provide insurance coverage only to the named insured and his legal representatives.

INDEMNITY AND INSURANCE FOR OCEAN TRANSPORTATION

The indemnity protection afforded by the Price-Anderson Act applies (with certain exceptions*) only to incidents arising within the United States. Thus, for example, irradiated fuel from an overseas reactor being sent to the United States for reprocessing at a licensed or contractor-operated reprocessing plant would be indemnified (under the licensee indemnity agreement or contractor agreement) only for the portion of the trip when it is in the course of transportation within the territorial limits of the United States. If an incident were to occur outside of the territorial limits of the United States, the person liable and the public would not receive protection for the damage suffered on the high seas, in a foreign country, or even in the U. S. itself.

Legislation was introduced in the 91st Congress to provide Price-Anderson indemnity coverage for high seas transportation of nuclear materials in American flag vessels but was not acted upon before Congress adjourned. (As of March 31, 1971 consideration was being given to introducing similar legislation in the present Congress). The bill would have extended Price-Anderson coverage to nuclear incidents arising out of the transportation of source, special nuclear, or byproduct material on vessels of U. S. registry and which take place in the course of such transportation outside of the territorial limitations of the United States or any other nation. The person indemnified would have been limited to the licensee or

---

* See 170.d. and 170.l. of the Act. Under subsection 170.d. extraterritorial indemnity may be applicable to cover liability of persons who are contractors or subcontractors of any tier under Commission contract or other project to which 170.d. indemnity has been extended, if the incident involves a facility or device owned by and used by or under contract with the United States.

contractor of the AEC with whom an indemnity agreement is executed and any other person: (a) who may be liable by reason of his activities within the U. S. (such as an American cask designer or manufacturer, or stevedore in a U. S. port); (b) who may be liable in connection with the ownership or operation of the vessel; (c) on board the vessel transporting the nuclear material while such vessel is on the high seas who may be liable (such as a seaman); (d) and any other person who has his principal place of business in the U. S. and who may be liable (such as an employee of an American firm supervising loading at a foreign port). As in the case with domestic indemnity coverage, loss of the transporting vessel, and its equipment, fuel and stores (which we understand, in some cases may be covered by hull insurance) are not covered under the proposed legislation nor are the nuclear materials being transported and the containers used.

Liability for covered ocean transportation would be limited to \$100 million together with underlying insurance providing financial protection up to an aggregate liability of \$115 million. We understand that \$15 million represents the currently available maximum "protection and indemnity" coverage in the marine nuclear field (third-party liability). In view of this, the amount of financial protection to be required under the proposed legislation would be \$15 million, although the Commission would be authorized to establish a greater or lesser amount of financial protection, taking into consideration the amount of liability insurance available from private sources and the cost and terms of insurance. The \$100 million in indemnity for persons indemnified in connection with each nuclear incident would be reduced by the amount that financial protection required exceeds \$15 million so that the aggregate liability shall remain at \$115 million even if more insurance is made available.

INTERNATIONAL CONVENTIONS ON LIABILITY

The Paris Convention of July 29, 1961, on Third Party Liability in the Field of Nuclear Energy, and the Vienna Convention of April 29, 1963, on Civil Liability and Nuclear Damage were drafted to cover the liability of the operator of nuclear installations for damage suffered anywhere, including during the transportation of nuclear material, except in the territory of

nonadhering states. The U.S. has not signed or adhered to any of the conventions. At present, of the international conventions in the field of nuclear liability only the Paris Convention on Third-Party Liability in the Field of Nuclear Energy, has been ratified by a sufficient number of countries to bring it into force.

PROBLEMS RELATED TO INSURANCE AND INDEMNITY COVERAGE FOR NUCLEAR MATERIALS IN TRANSPORTATION

International

As we describe in the preceeding section, the problem of insufficient liability protection for shipments from the United States or to the United States on the high seas should be met if proposed legislation to extend Price-Anderson protection for shipments on U. S. flag vessels is reintroduced in Congress and becomes law.

Domestic

Bridge, Tunnel, Port, and Tollroad Authorities

As was described earlier, the fire and property insurance carriers in 1959 adopted a nuclear exclusion clause in all of their policies which excluded from coverage, first-party damage and loss of revenue caused by the radioactive materials hazard. Nuclear property insurance that does not contain this exclusion was available in the U. S. only for nuclear installations, and under limited conditions, shippers and carriers of nuclear materials. Some bridge, tunnel, port and tollroad authorities were concerned that in the event of an accident that damaged their facilities or caused a loss in revenue of their facilities, they would not be able to recover against the liability pools for damage to an authority's own property without establishing the liability against specific defendants. In view of this concern, they did not consider nuclear liability insurance as an adequate substitute for the first-party coverage which would be provided to them by conventional property insurance without a nuclear exclusion.

In 1966, Congress enacted Public Law 89-645 amending the Price-Anderson Act to provide for prompt and equitable settlement of claims arising out of a nuclear incident, and to provide for waivers of defenses with respect to claims arising out of an "extraordinary nuclear occurrence" as that term

is defined in the Act. In accordance with the 1966 amendments to the Price-Anderson Act, in 1968 the Commission amended Part 140 of its regulations to establish criteria setting forth the basis on which the Commission would make a determination as to whether or not an "extraordinary nuclear occurrence" has taken place and incorporating waivers of certain defenses in the facility form of nuclear energy liability policy and the forms of indemnity agreement. An endorsement was added to all nuclear insurance policies provided as proof of financial protection for indemnity purposes, and the indemnity agreements were correspondingly amended, to require that specific issues or defenses be waived by a defendant in the event of an extraordinary nuclear occurrence. A claimant might still have to proceed with an ordinary legal action involving proof of the nature, cause and amount of damages, but with the mandatory waiver of defenses his burden would be eased by the elimination of such issues as negligence, contributory negligence, assumption of risk, charitable or governmental immunity, and, to a certain extent, statute of limitations.

We believe that these waivers of defenses provisions go far in meeting the concern expressed by the transportation authorities with respect to having to establish negligence under the facility forms of nuclear energy liability insurance, at least in the case of extraordinary nuclear occurrences. However, in an AEC-sponsored industry conference called in 1968 to consider the (at that time) proposed amendment to the Commission's regulations providing for waivers of defenses, the representative of one of the authorities expressed some remaining reservation particularly with respect to obtaining first-party property insurance for those nuclear incidents that did not qualify as extraordinary nuclear occurrences.

Railroads

Although we have had no recent discussions with the railroad industry regarding liability insurance and indemnity protection for the transportation of spent fuel, the Traffic Executive Association, Eastern Railroads, did discuss with us several years ago, concerns which they had at that time. The principal problems were: (a) their uncertainty as to whether their liability under the Federal Employer's Liability Act (FELA) would be covered by government indemnity under the Price-Anderson system and nuclear liability insurance (or on the other hand, excluded as workmen's compensation claims); and (b) their concern that in particular cases they may not know whether a specific shipment of nuclear reactor fuel is covered by the Price-Anderson Act.

With regard to the first of these concerns, the railroad representatives were advised that the FELA liability is covered by the underlying insurance policy and the government indemnity is in excess thereof. We understand, however, that there may be forms of nuclear liability insurance, other than the one approved by AEC as acceptable proof of financial protection under the government indemnity, which do not cover FELA liability.

With regard to the second concern, the AEC established a procedure under which a certification can be issued by the AEC to a carrier, at the time of delivery to the carrier of the spent fuel, which would certify that the spent fuel during specified transportation is covered by government indemnity. Several railroads have availed themselves of this certification procedure and have carried shipments of fuel as contract carriers.

CONCLUSION

Although the insurance industry and the government have endeavored to maintain flexibility and adapt coverage to keep pace with the growing complexity and volume of nuclear material shipments, and much has been accomplished, we can see that some problem areas remain. By the cooperative efforts of all concerned, these problems can be overcome to assure that nuclear insurance and indemnity continue to provide the coverage required.

THIRD PARTY LIABILITY INSURANCE AND GOVERNMENT INDEMNITY
ASSOCIATED WITH THE TRANSPORTATION OF RADIOACTIVE MATERIAL

L. G. Cummings

ABSTRACT

The increasing frequency of radioactive material shipments through out the world with its potential risk of financial loss to shippers and transporters has focused considerable attention on the means to minimize or eliminate this risk. One important contribution to meet this challenge in the United States was accomplished by the formation of two nuclear liability insurance pools which make available to private industry third party nuclear liability insurance coverage up to \$82,000,000 for Bodily Injury and Property Damage arising out of a nuclear incident. In addition, the Federal Government provides, via legislation known as the Price-Anderson Act, indemnification at certain nuclear facilities including transportation up to \$500,000,000 excess of the private insurance coverage. An understanding of the interface between the conventional and nuclear liability insurance policies together with applicable government indemnity is essential if the transportation risk is to be properly insured at minimum cost. Foreign nuclear material shipments also involve unique insurance questions, and certain insurance coverages are available in this field where it is determined that existing international regulations and local liability legislation do not adequately protect the shipper's or transporter's interest. This paper will provide an analysis of the above areas of interest and will include liability cost factors and loss experience data associated with the shipments of nuclear material.

INTRODUCTION

At the time the Atomic Energy Act of 1954 was enacted by the United States Congress few people would have envisioned that by March 1971 the United States electric utility industry would be operating or committed to constructing over 110 nuclear reactors for commercial operation with an estimated total capacity of over 90,000 MWe. In addition there are approximately 50 commercial production facilities engaged in the nuclear fuel supply chain.

The free world, excluding the United States, is committed to approximately 100 reactors with an estimated total capacity of nearly 50,000 MWe.

Considering the fact that only 21 United States and 40 foreign reactors are scheduled to be on line in 1971 it is obvious that the next few years will involve a very substantial growth in the broad spectrum of commercial nuclear activities. This activity will present an increasing challenge to the transportation industry to assure the safe and on-schedule delivery of nuclear material.

Shipments of nuclear material will involve transport by truck, rail, air and water, each with its own unique problems of labor relations, over-taxed traffic lanes and terminals, equipment failures and so called "Acts of God", any of which may contribute to financial losses for the shipper and/or transporter.

It is the primary objective of this paper to examine the availability and applicability of nuclear liability insurance and government indemnity as they relate to (1) domestic shipments and (2) shipments to and from the United States.

DEVELOPMENT OF NUCLEAR LIABILITY INSURANCE AND
GOVERNMENT INDEMNITY IN THE UNITED STATES

Following enactment of the Atomic Energy Act of 1954 private industry expressed the need for a substantial amount of third party liability insurance to protect their financial interest against the unknown risks of nuclear energy and also indicated that the lack of adequate insurance could be a major obstacle to the commercial development of nuclear technology. A number of factors complicated the problem for the insurance industry in trying to provide for the requested high liability insurance limits with the very remote, but theoretically possible, catastrophe risk. It was recognized that the number of nuclear facilities to be insured would be limited for several years precluding the insurers from utilizing a basic insurance requisite of spreading risks over a broad base of exposure. In addition, the lack of loss experience data and the anticipated rapid technological development in nuclear science presented additional concern to insurance underwriters.

Despite these obstacles the resources of the insurance industry were joined together in 1957 in the formation of two nuclear liability insurance pools, the stock company pool called the Nuclear Energy Liability Insurance Association (NELIA) and the mutual company pool called the Mutual Atomic Energy Liability Underwriters (MAELU) with an initial combined insurance capacity of \$60,000,000 applicable to any one nuclear incident. Subsequently, this capacity has been increased to the \$82,000,000 we have today. The NELIA and MAELU pools issue identical liability insurance policies to cover domestic nuclear facility and transportation exposures and function on a cooperative basis with joint rating, underwriting and engineering committees.

At the same time the nuclear insurance pools were organized it was deemed advisable by Congress to make available additional limits of coverage for certain types of facilities. This resulted in the passage of the Price-Anderson Act in 1957. This act provided government indemnity in the maxi-

mun amount of \$500,000,000 excess of the \$50,000,000 available from the insurance pools for a combined total of \$550,000,000. It should be noted that the available amount of government indemnity is decreased by the same dollar amount whenever the nuclear insurance capacity is increased. Consequently the total combined liability limit remains at \$550,000,000 and thus today is comprised of \$82,000,000 by insurance and \$478,000,000 by indemnity.

With only minor exceptions, the liability protection afforded by the insurance pools and Price-Anderson are identical minimizing administrative and loss adjustment problems.

RELATIONSHIP OF CONVENTIONAL AND NUCLEAR LIABILITY INSURANCE

Several points should be commented upon prior to examining the nuclear insurance and indemnity available to shippers and transporters.

In drafting the nuclear liability insurance policies it was decided that the conventional insurance market should continue to provide liability insurance for the non-nuclear hazard. Thus, the basic insuring agreement in the nuclear liability policy only provides Bodily Injury and Property Damage liability coverage caused by the "nuclear energy hazard" thereby making it necessary for a policy holder to insure the non-nuclear hazards in the conventional fashion.

An examination of the Broad Form Nuclear Energy Liability Exclusion endorsement that is attached to all United States Comprehensive Automobile General Liability Policies will indicate that certain nuclear hazards continue to be insurable under the liability policy. Specifically, these deal with operations involving (1) source material (i.e. natural uranium, thorium) (2) by-product material commonly referred to as "commercial radioactive isotopes" and (3) laboratory quantities of special nuclear material limited to 25 grams of Plutonium or Uranium 233 and 250 grams of Uranium 235. Consequently we repeat, shipments of material in these three

categories remain insurable in the conventional manner.

It is important to note that the terminology used in the above mentioned Broad Form Nuclear Exclusion and the NELIA-MAELU policies are identical in order to avoid questions of coverage interpretation.

With regard to the availability of government indemnity only two types of privately owned nuclear facilities are presently eligible. These facilities, designated as indemnified nuclear facilities, are (1) nuclear reactors and (2) fuel reprocessing plants. They are required to furnish financial protection pursuant to the Atomic Energy Act of 1954, as amended, and to date, have elected to satisfy this requirement by purchasing insurance from the nuclear pools. The facility operator can then enter into an indemnification agreement with the Atomic Energy Commission to obtain benefits afforded by the Price-Anderson Act.

NUCLEAR LIABILITY INSURANCE POLICY (FACILITY FORM)

Let us now examine certain features of the NELIA/MAELU Nuclear Liability Policy (Facility Form) which pertain to the liability insurance afforded during the transit of nuclear material as defined therein.

Insuring Agreement I.-- Coverage A stipulates the basic coverage afforded for the nuclear energy hazard as previously mentioned. Coverage B provides insurance for contamination to a named insureds property away from the facility. Coverage C applies to the Workmen's Compensation carrier who has paid off-site employees for injuries caused by the nuclear energy hazard.

Insuring Agreement II.-- pertains to the Definition of Insured which in addition to the named insured includes anyone having legal liability for the loss, except the United States Government or its agencies. The comparable clause in the Price-Anderson Indemnity Agreement is designated "Person Indemnified" and is similarly defined.

Insuring Agreement III. -- includes a definition of "insured shipment" which deserves special attention in this paper. The "insured shipment"

FIG. I. INSURED SHIPMENTS BETWEEN VARIOUS TYPES OF NUCLEAR FACILITIES.

defined in the basic policy applies to an insured non-indemnified nuclear facility which reads in part as follows:

"Insured Shipment means a shipment of source material, special nuclear material, spent fuel or waste, herein called material, (1) to the facility from a nuclear facility owned by the United States of America.. or (2) from the facility to any other location except an indemnified nuclear facility..."

The above definition is broadened by endorsement when the policy applies to an insured indemnified nuclear facility and reads in part as follows:

"Insured Shipment means a shipment of source material, special nuclear material, spent fuel or waste, herein called material, (1) to the facility from any location except an indemnified nuclear facility.. or (2) from the facility to any other location..."

The application of the above Insured Shipment clauses to shipments of material between various types of facilities is illustrated in Figure 1.

NUCLEAR LIABILITY INSURANCE POLICY
(SUPPLIER'S AND TRANSPORTER'S FORM)

In addition to the Facility Liability Policy NELIA/MAELU also issue a Supplier's and Transporter's Liability Policy which is generally purchased as excess insurance by manufacturers and suppliers of equipment to nuclear facilities and by shippers and transporters of nuclear material.

This policy affords third party liability insurance resulting from nuclear incidents and arising out of the furnishing of equipment, supplies and services to nuclear facilities and including coverage during the course of transporting nuclear material. The basic policy covers only "the named insured" except that during the course of transporting source material, special nuclear material, spent fuel or waste, the definition of insured has been broadened by endorsement to include anyone (other than the United States or its agencies) with respect to his legal responsibility.

All liability policies issued by the nuclear insurance pools contain an aggregate limit of liability clause which in effect establishes a maximum limit equal to the pools capacity on any one nuclear incident irrespective of the number of policies which might apply. Consequently, if an insured shipment is covered by a facility policy affording the pool capacity any separate Supplier's and Transporter's policy purchased by the carrier would not provide additional limits. However, in instances where an insured shipment is primarily covered by a facility policy with a limit less than the pool's capacity, for example, a fuel fabricator, an applicable Supplier's and Transporter's policy would afford coverage excess of the existing facility policy limit.

One other important point applicable to all domestic nuclear liability policies is the territorial limitation to insured shipments. These policies only afford coverage that is limited to the territorial limits of the United States of America, its territories or possessions, Puerto Rico or the Canal Zone. For example, no coverage is provided between the continental limits of the United States and territorial limits of Puerto Rico even though coverage is provided within these limits in both instances.

NUCLEAR LIABILITY INSURANCE FOR INSURING SHIPMENTS OUTSIDE THE UNITED STATES

In many cases American shippers and transporters may find that their interests are not adequately protected under existing international regulations and should examine the availability of private insurance. In some instances the conventional liability policies specially negotiated for the airlines and steamship lines have included protection for both nuclear and non-nuclear hazards. However there appears to be a trend among the conventional insurance underwriters to limit or exclude protection arising from the nuclear risk. To the extent that United States firms are unsatisfied with the protection afforded above by the certain transporters they

may consider the purchase of a Supplier's and Transporter's Foreign Liability policy designed to cover outside of the United States territorial limits. \$10,000,000 of this coverage is available from NELIA and \$5,000,000 from MAELU. It will be recalled that these pools operate in close cooperation with respect to domestic insurances but work independently and in competition when providing the foreign coverage. Consequently a few differences exist in the policy coverage and exclusions as well as rating procedures. In addition to the domestic pools the London Insurance Market will provide competitive coverage for nuclear liability on foreign shipments with an available capacity of approximately \$15,000,000. In summary, United States firms may purchase a combined total of some \$30,000,000 of limits to insure their foreign transit liability at varying coverages and costs.

In addition to the above insurance the United States Congress has considered amendments to the Price-Anderson Act to provide limits up to \$100,000,000 covering foreign shipments of nuclear material via ships of United States registry. A bill was introduced during the Ninety First Congress to extend this indemnity for private Atomic Energy Commission licensees and establish a financial protection requirement of \$15,000,000 from private insurance with the Price-Anderson indemnity being excess. Although this proposed legislation did not pass it is anticipated similar legislation will be introduced again this year.

It would be appropriate for American shippers and trnasporters of nuclear material outside of the United States to examine the liabilities associated with shipments on international waters, air routes and within foreign countries. This paper cannot embrace an analysis of these areas but the following sources of information can be recommended to those interested.

1. Selected Materials on Atomic Energy Indemnity Legislation Joint Committee Print (June 1965) available from the U.S. Government Printing Office.

2. Foreign Third Party Legal Liability - an analytical study prepared by the Secretariat of the European Nuclear Energy Agency, (1969) available from offices of the Organization of Economic Co-operation and Development.
  3. Third Party Liability and Insurance in the field of Maritime Carriage of Nuclear Substances - Monaco Symposium (1968) available from the offices of the Organization of Economic Co-operation and Development.
- We understand another similar symposium is contemplated this year.

INSURANCE COSTS

We will now examine the nuclear liability insurance costs associated with the shipments of nuclear material. As previously discussed shipments of material between domestic nuclear facilities are automatically included under the NELIA and MAELU liability facility policies. This being the case no separate premium is chargeable to shippers or transporters since they are "omnibus insureds" under policies purchased by the facility owners. The annual cost of a facility policy insuring a power reactor for \$82,000,000 will develop premium in the area of \$250,000 which will vary depending upon such rating considerations as the power level, containment and proximity to population centers and other property values. With respect to fuel re-processing facilities, the annual facility policy may be estimated to be \$44,000 based upon the annual separation of 1000 kgs of fissionable material and required financial protection limits of \$20,000,000. The facility insurance for fuel fabrication plants is based upon the maximum inventory of nuclear material, plant location and policy limits selected by the owner. For example, a rural location with 300 kgs. Uranium and a \$10,000,000 limit could develop an annual premium of approximately \$14,000.

Domestic Supplier's and Transporter's Liability policies are individually rated after consideration of the type and quantity of material shipped, packaging, annual number of shipments, mode of transit and desired

coverage. A basic rate is established for the first \$1,000,000 limit and a percentage of this rate assigned for higher limits subject to specified minimums. For example, a risk with nominal exposure might develop a premium of \$750. for the first million limit, \$375. for the second million and \$250 each million for higher limits.

An important aspect of the domestic nuclear liability insurance is the realization of a substantial premium refund if the overall industry experience proves favorable under the Industry Credit Rating Plan. Approximately 67% of every premium dollar is set aside for the payment of losses and loss expense and is designated as the Reserve Premium. After a ten year waiting period the total industry premiums and losses are evaluated and a Reserve Premium Refund determined which is shared by all policy holders for the year in a direct ratio of policy premium to total industry premiums. Accordingly, refunds were made in 1967 on those premiums paid in 1957. The four consecutive annual Industry Credit Rating Plan refunds to date have averaged 97% of the reserve premium. This is indicative of the very favorable industry operating experience to date.

A less sophisticated method is involved in developing the cost to facility owners for the Price-Anderson Indemnity. The reactor facility charge is based upon a fee of \$30 per Megawatt Thermal rating of the reactor. For Fuel Reprocessing Plants, the fee is determined by the Atomic Energy Commission in consideration of the processing capacity, waste storage exposure and amount of financial protection required. The most recently proposed annual charge for this type of facility is \$4,000.

The insurance costs associated with insuring foreign shipments are also developed on an individual risk basis after evaluating the type and quantity of the material, packaging, mode of transit, points of departure and destination and special cover required with respect to inclusion of the transporter's interest, damage to companion cargo and port facilities. The final rating

is usually expressed by a premium per shipment for a specified limit of liability (e.g. \$5,000 per shipment for a 5,000,000 limit.)

LOSS EXPERIENCE

This paper would be incomplete with-out providing a few observations concerning the loss experience associated with the shipments of nuclear materials. Generally speaking it can be stated that the experience has been most favorable with no known major contamination losses. However the transportation of nuclear materials is subject to the same risks as conventional shipments.

It may be of interest to summarize certain experience data available from the United States Department of Commerce¹⁻² which includes reports of accidents involving USAEC shipments of radioactive material during the period 1949 through 1961. Information shown in the following tables was tabulated from the above reports.

Table 1. Accident Experience by Mode or Location

Mode	No. of Accidents
Truck Accidents	32
Rail Accidents	13
Air Accidents	2
Terminal Accidents	<u>15</u>
Total	62

1. A summary of Transportation Incidents in Atomic Energy Activities 1949-1956 AECU-3613 (1957)
2. Health and Safety - A summary of Incidents Involving USAEC Shipments of Radioactive Material 1957-1961 by D.E. Paterson and V.P. DeFaha TID (1962)

Table 2. Proximate Cause of Accident
Irrespective of Mode

Type	No. of Accidents
Handling (in terminals)	15
Collision	16
Fire	6
Vehicle Equipment Failure	7
Leakage of Containers	4
Derailment (Rail)	5
Tiedown Failure	2
Aerial Dispersal	1
Overturn (Vehicle)	4
Air Crash	2
Total	62

Table 3. Accident Experience by type
of Material Involved

Type of Material	No. of Accidents
Source Material	20
Radioisotopes	11
Irradiated Fuel Elements	8
Special Nuclear Material	4
Radioactive Waste	4
Empty Containers	3
Contaminated Machinery	3
Water	1
Sand	1
Not Available	7
Total	62

The nuclear liability loss experience incurred since 1957 by NELIA/MAELU for commercial operations has developed a total of sixteen reported incidents, most of them minor. Of these sixteen, nine have involved transportation.

Five of the transportation incidents concerned contamination to property and incurred an average loss payment including loss expense of \$1706, the most costly loss being \$3519. The sixth claim involved minor Uranium contamination to a warehouse and truck weigh station which was settled without payment. A seventh report alleged bodily injury resulting from a small quantity of depleted uranium delivered to a wrong address with a final determination that the injury, if any, was covered by conventional liability insurance. The eighth and ninth reported incidents each involved alleged bodily injury to a worker in the transportation industry. Both of these cases are presently in litigation and it is questionable if any radiation injury is involved.

CURRENT INDUSTRY PROBLEMS

A few observations should be made relative to areas which will receive further attention for insuring the shipments of nuclear material.

You will recall the initial concern of underwriters was the lack of loss experience data and the unknown effect of anticipated technological advances on overall loss experience. The limited number of operating reactors, the lack of operative experience with the large size units together with the relative low frequency of new and spent fuel shipments, makes it necessary to continue using considerable judgement in rating the nuclear hazard.

There has been increasing criticism among nuclear industry opponents of the Price-Anderson Act which comes up for renewal in 1977. The critics contend that the industry should assume its own risks and are particularly vocal with respect to government indemnity being provided for certain licensees where the required financial protection is less than the available liability insurance. It is expected the upcoming hearings for Price-Anderson renewal will generate considerable public interest. Recent issues of INFO, a Public Affairs and Information Program publication by the Atomic Industrial Forum has included articles on this subject.

Several problems presently face shippers of spent fuel due to the weight and size of the transporter casks. Some railroads are refusing to haul spent fuel shipments except under very restrictive conditions. State Motor Vehicle Bureaus have become increasingly alarmed at the frequency of requests for overweight permits on their highways and many are increasingly refusing to issue them. The above rail and motor vehicle restrictions have focused attention on the possible use of barge or specially designed self-propelled vessels. This alternative appears to have merit since the majority of reactors are located on or in close proximity to navigable waterways.

One other area receiving attention involves the potential theft and hijack risk which may necessitate new safeguard and monitoring techniques to minimize loss of material during transit and the attendant liability exposure this could create. An article appearing in the April 26, 1971 issue of Newsweek Magazine discussed this subject.

Both the nuclear industry and transportation industry can take pride in the excellent record of transporting nuclear materials to date. This fact should provide the industry with an additional incentive to meet the increasing challenges that will surely develop during the next several years of projected rapid growth for nuclear energy activities.

PROCESSING RADIOACTIVE SHIPMENTS FROM
THE STANDPOINT OF RADIATION SAFETY*

Carl W. Buckland Jr.
University of California, Los Alamos Scientific Laboratory
Los Alamos, New Mexico 87544

ABSTRACT

The complexity of current shipping regulations applied to the transport of radioactive materials justifies a procedure that minimizes the chance for human error. The procedures exercised by the Health Physics group of the Los Alamos Scientific Laboratory will be detailed in the presentation and paper.

Compliance with the regulations has been assured by developing a data sheet to be processed by the Health Physics Surveyor that provides a reminder for each pertinent requirement of the regulations. Once the surveyor has become familiar with the regulations and associates these rules with the corresponding item in the data flow sheet, the chance for overlooking some requirement is minimal.

Radioactive shipments from a laboratory are extremely varied in nature. For this reason, the operating groups are encouraged to seek advice from traffic, criticality, health physics and engineering specialists before attempting shipment. Advisory information is provided so that the final shipment will conform to the regulations. An "Advance Information Form" is filled in and distributed to the appropriate individuals so that all persons involved in the final shipment will be cognizant of the details. It also provides a method of pre-planning and check list to assure that no requirement has been overlooked. Time consuming gram to curie conversions, dose rate measurements to curie determinations, fissile class and special form declarations are among some of the items included in the form.

The operating group packages their material according to the special requirements for the package specification, Paragraph 173.396 or special permit. At the point of origin, the material packager is required to sign a package certification form for certain containers indicating his conformance with special inner packaging requirements such as ratio of hydrogen to fissile material, special inner capsules, heat load, void space, poisons, absorbers and physical form.

All radioactive shipments are indicated as such on the shipping request. Upon receipt of a shipping request that includes radioactive material, the Health Physics group is notified by the shipping department to process the comprehensive data sheet after making the necessary radiation measurements, smears, label selection and double check of curie determinations.

Every effort is made by the surveyor to conscientiously examine his finished product for errors or omissions to the point of duplicating prior calculations provided in the "Advance Information Form." He also refers to special permit requirements for completeness of compliance.

The data sheet is then reviewed by the traffic manager for violation of any regulation and if found to be satisfactory, signs the certification for receipt by the carrier. A file copy of the data sheet is then reviewed by three supervisory personnel in the Health Physics group for possible further correction and to furnish any constructive criticism to the Health Physics Surveyor for his future guidance in processing other radioactive shipments.

*This work was performed under the auspices of the US Atomic Energy Commission.

INTRODUCTION

The complexity of current shipping regulations applied to the transport of radioactive material justifies a procedure that minimizes the chance for human error. The procedures exercised by the Health Physics group of the Los Alamos Scientific Laboratory are detailed in this paper for the benefit of those organizations faced with comparable administrative complexities.

PRE-PLANNING

Radioactive shipments from a laboratory are extremely varied in nature. For this reason, the operating groups are encouraged to seek advice from traffic, criticality, health physics and engineering specialists before attempting shipment. Pre-planning is especially desirable when the more exotic specification containers are used or when there is a need to apply for a Department of Transportation special permit through the AEC in lieu of a specification container. Advisory information is provided so that the final shipment will conform to the regulations.

Data received from the operating group are recorded and developed in the "Advance Information Form" (Figure 1). Processing the form provides a preliminary check list to assure that no important requirement has been overlooked. Time consuming weight to curie conversions, dose rate to curie determinations, fissile class and special form declarations are prepared in advance. The form is distributed to the appropriate individuals so that all persons involved in the final shipment will be cognizant of the details.

INITIAL PACKAGE CERTIFICATION

The operating group packages their material according to the special requirements of the DOT specification or DOT special permit or/and DOT Paragraph 173.396. At the point of origin, the material packager is required to sign a package certification form for certain containers indicating his conformance with special inner packaging requirements such as the ratio of hydrogen to fissile material, special inner capsules, heat load, void space, poisons, absorbers and physical form.

Past experience indicated that operating groups were not packaging special purpose containers according to the specification or special permit requirements. There was no assurance that limitations on quantity, heat load and ratio of hydrogen to fissile material were not being exceeded. Once the

RADIOACTIVE SHIPMENT
"Advance Information Form"

GROUP SHIPPING _____ DATE _____

DESTINATION _____

MATERIAL _____ FORM _____

AMOUNT _____

TRANSPORT GROUP _____ A, B, or LARGE _____

FISSILE CLASS _____ TRANSPORT INDEX _____

CONTAINER PROVIDED _____

REMARKS _____

_____

_____

Fig. 1. Advance Information Form

container was received in the shipping department there was no way to determine whether the detailed requirements had been met. It was therefore decided that "Certification of Packaging" forms would be prepared to include all of the pertinent requirements on one sheet of paper and thereby place the initial responsibility with the operating group. Forms for three DOT specification and eleven DOT Special Permit containers used most frequently were prepared for routine use. They are as follows:

- DOT Specification Number 55
- DOT Specification Number 6L
- DOT Specification Number 6M
- DOT Special Permit Number 5097
- DOT Special Permit Number 5246
- DOT Special Permit Number 5660
- DOT Special Permit Number 5885
- DOT Special Permit Number 6286
- DOT Special Permit Number 5915
- DOT Special Permit Number 6208
- DOT Special Permit Number 5645
- DOT Special Permit Number 4960
- DOT Special Permit Number 6000
- DOT Special Permit Number 4824

Additional certification forms will be prepared as dictated by need. Figure 2 illustrates a typical certification form. No such forms are deemed necessary for most DOT Specification drums or boxes.

SHIPPING REQUEST FORM

The operating group must process a shipping request form and indicate when the contents of their package are radioactive. Upon receipt of such a form by the shipping department, the Health Physics group is requested to survey the package for dose rate and removable contamination; to select the proper labels, determine curie content and process a comprehensive data sheet.

PROCEDURAL DETAIL FOR PROCESSING A COMPREHENSIVE DATA SHEET

Each Health Physics Surveyor has a working knowledge of the shipping regulations for radioactive material. By a process of association, each pertinent part of the regulation is considered with the data sheet (Figure 3) and in such an order that the information gathered clearly indicates the packaging, marking

CERTIFICATION OF PACKAGING BY A LASL OPERATING GROUP

Covers DOT Specification Number _____

For Shipping Request Number _____

To assure that the container referred to above is packaged according to the DOT Specification 178.104 and DOT Paragraph 173.396, the following check list has been adhered to:

	<u>Initials of Packager</u>
1. There will be no mixture of gases or vapors in the package which could, through any credible spontaneous increase of heat or pressure, or through an explosion, significantly reduce the effectiveness of the packaging -	_____
2. The contents are such that there will be no significant chemical or galvanic reaction among any of the materials in the package -	_____
3. Material being shipped is solid and will not decompose at temperatures up to 250°F -	_____
4. Thermal decay heat does not exceed 10 watts (20 g for ^{238}Pu and 4.5 kg for ^{239}Pu) -	_____
5. For Fissile Class I quantities (1.6 kg ^{235}U , .9 kg ^{239}Pu , .5 kg ^{233}U or less) the maximum ratio of hydrogen to fissile material does not exceed three, all sources of hydrogen within the Specification 2R being considered -	_____
6. ^{235}U enrichment does not exceed 93% -	_____
7. For Fissile Class II shipments the following quantities have not been exceeded:	

U-235

*Metal or alloy	H/X = 0	13.5 kg
*Compounds	H/X = 0	32.0 kg
*Compounds	H/X ≤ 3	19.5 kg

Pu

*Metal or alloy	H/X = 0	4.5 kg
*Compounds	H/X = 0	4.5 kg
*Compounds	H/X ≤ 3	4.5 kg

*Indicate which ratio and form on SR (or circle above) due to different Transport Index assignments!

NOTE: Only the hydrogen interspersed with the fissile material need be considered.

8. "Large Quantity" in "Normal Form" (both by definition) is packaged in one or more sealed and leak-tight metal cans or polyethylene bottles within the Specification 2R containment vessel -	_____
9. If shielding is needed, it is either inside of the 2R inner vessel or protected in all directions by at least the thickness of the thermal insulating material prescribed in DOT 178.104-3(a) -	_____

10. Pipe threads of inner 2R vessel are luted with an appropriate non-hardening compound and the cap has been tightened - _____
11. All of the machined discs or rings used as centering devices are in place - _____
12. Closure ring bolt has been secured with a lock nut or equivalent device - _____
13. A tamper seal has been applied - _____
14. The overall container has been checked for obvious structural failures from repeated handling - _____

The purpose of this check list is to facilitate the task of shipping radioactive materials in specialized containers. Suggestions for improving this form should be directed to the H-1 General Monitoring Section.

This is to certify that the radioactive contents of the above described container offered for transportation have been packaged according to the Department of Transportation requirements as indicated above.

_____
Signature of Group Leader

Packaging Date: _____

NOTE: Attach this completed form to Shipping Request in duplicate.

FORM-HSC-12-(12/70)

Fig. 2. Certification Form

SR NO. _____ DATE _____
GROUP SHIPPING _____ DESTINATION _____

LOS ALAMOS SCIENTIFIC LABORATORY
RADIOACTIVE MATERIALS SHIPPING DATA SHEET

METHOD OF TRANSPORT _____ Gross Weight: _____
MATERIAL _____ Exempt per 173.391(a)1-4 _____
FORM OF MATERIAL - R.A. White I _____ R.A. Yellow II _____
Liquid - Gas - R.A. Yellow III _____ Postal _____
Solid - Powder - Maximum Surface Activity on Container
Special Form* - Metal - (gamma plus neutrons in mrem/h) _____
Normal Form - Gram Weight - Maximum Activity 3' from Container
Ci _____ (gamma plus neutrons in mrem/h) _____
Transport Group _____ Type A or B Quantity Source of Criticality Evaluation and
Transport Index Assignment _____
"Large Quantity" - Yes No Fissile - Yes No _____
2R Container and properly marked _____
Shock or Liquid
Absorbing Material _____
Secured Lead, Paraffin
DOT 55 Container _____ Transport Index Used: _____
EXTERNAL CONTAINER: Tamper Seal: Yes _____ No _____
Size _____ Fissile Class I _____ II _____ III _____
Type A or B _____ Remarks: _____
No Significant
Removable Contamination: _____
(Initials) _____
Heat Load: _____
DOT and Other Marking Required: _____
_____
_____
_____

*Basis for and approved by:

_____
(Health Physics Surveyor's Signature)

Fig. 3. Data Sheet

and labeling required.

Obvious items of the data sheet are not included in the discussion to follow. Because of their brevity, the not so obvious will be developed in this paper.

Method of Transport

The method of transport is taken into consideration since certain shipments are not allowed on passenger aircraft. Packages going by mail are under postal regulations rather than the DOT. "National Security Exempt" shipments require that only one certain label be applied. Exclusive use of vehicle permits higher radiation levels; therefore, the method of shipment dictates the manner in which one evaluates his subsequent findings.

Material

It is important to seek out and record the correct isotope and mass number since the curie content, heat load, fissile class, transport group, quantity designation, removable external contamination and ultimate package requirements are all based on this information. The word plutonium, for example, is not sufficient.

Form of Material

The physical form of the material is indicated so that judgment can be exercised concerning "Special Form" requirements. For example, ^{239}Pu as metal or oxide will pass the "Special Form" criteria when enclosed in a standard DOT 2R inner container. ^{239}Pu in solution would not pass the "Special Form" requirement, whether in an inner 2R container or not. Material in liquid form requires special consideration as described in DOT 173.393(g). Material in gaseous form requires a different method of curie determination.

Curie Determination

Most of the curie determinations are made from weight conversions. The activity-mass relationship is found with the following equation:

$$\text{Ci/g} = \frac{1.308 \times 10^8}{AT}$$

Where A = atomic weight

and T = radioactive half life in days

For convenience, the Ci/g value for most of the isotopes can be found in LA-4400, "Los Alamos Handbook of Radiation Monitoring 1970."

The number of curies of tritium gas is obtained by multiplying the number of cubic centimeters at STP by 2.6.

As a rule of thumb, the curie content for fission products over 30 minutes of age but not greater than 200 days, is determined by measuring the

gamma rate at one meter in r/h and multiplying by 3. If the number of fissions are known as well as the age, curies are obtained from published tables.

For radiation measurements at measured distances from known isotopes, curies can also be determined from published tables or by using commercially available isotope handling calculators (Figure 4).

Transport Group

The transport group is determined from DOT 173.390(a) or (b) or declared as special form for each isotope and recorded since the quantity type designation is established from the curie content of a specific transport group. The correct packaging is then confirmed from this information. (Exempt, Type A or Type B.) When a "large quantity" is encountered, the data sheet is indicated to this effect as a reminder to use Radioactive Yellow III labels and vehicle placards regardless of radiation levels encountered.

Fissile

For purposes of shipping, the material has to be declared fissile or non-fissile according to the DOT regulations, Paragraph 173.396(a)(1) through (5). Small quantities of ^{235}U (16 grams total uranium with 93% ^{235}U enrichment) are considered non-fissile since the ^{235}U content would be <15 grams. When the quantity of fissile material exceeds 15 g and there is doubt concerning an exemption based on the atomic ratio of hydrogen to fissile material, the matter is resolved by contacting a criticality expert. When the material is obviously fissile, the data sheet is identified as such as a reminder that either DOT Paragraph 173.396(b)(3)(i) and (ii) restrictions are imposed or that a DOT 6L, 6M or Special Permit container is utilized. The ultimate container is selected on the basis of type and weight of fissile material authorized.

2R Container and Properly Marked

A DOT 2R inner container is sometimes used to satisfy "Special Form." Such an arrangement permits a Type A container over a Type B thus making the shipment more economical; viz, 10 g of ^{239}Pu as "Special Form." This item in the data sheet is also a reminder to see that the 2R has been marked as required by the specification.

Shock and Liquid Absorbing Material

This item merely records the fact that an absorber or absorbent was used as required by DOT 173.393(g)(1)(2) for liquid radioactive material.

Secured Lead, Paraffin - DOT 55 Container

The purpose of this item is to remind the Health Physics Surveyor to determine if the internal lead shield plug has been secured according to

DOT 173.393(d)(1). The security of the shield plug should not be solely dependent upon internal wood bracing to keep in place. The mention of paraffin and a DOT 55 container provides a record as to whether either of this type of shielding was required. Other types of neutron moderator and gamma shielding may be mentioned here for informational purposes.

External Container Size

The main objective of this item is to remind the Health Physics Surveyor that no dimension of the container can be less than four inches and provide a record.

External Container Type A or B

This item is to remind us that a Type A container must be used as a minimum containment for a non-exempt Type A quantity. A Type B container must be used for either a Type B or Large Quantity of material, taking into consideration the 300 Ci limit in "Special Form" with the DOT 55 for domestic use only which is a Type B quantity. It should be noted that a DOT 6M can be used as either a Type A or Type B container. The type container declared in this latter case is usually determined by the form of the material; viz, "Normal" or "Special." DOT 173.396(c)(1) authorizes fissile radioactive materials containing Type B quantities in a 6L; however, only 14 kg ^{235}U can be shipped in a 6L which is a Type A quantity even in "Normal Form." Most of the DOT Special Permit containers are Type B and are used when no specification container is authorized for the contents. A check of the permit is made to be certain that the contents are authorized and that the permit has been extended to our use.

No Significant Removable Contamination

This item reminds the Health Physics Surveyor to determine if the removable contamination from the external package surface by smear technique is significant according to DOT 173.397(a)(1) and (2). Smears are sometimes taken from numbered 100 cm² chalked squares (4" x 4") so that a relationship between the count and the monitored area can be established (see Fig. 5. Smear Technique). Smears are counted with Ludlum Model 20A portable scalers (one is equipped with an end window beta-gamma probe and the other with an alpha scintillation probe) located in the offices of the shipping department. (See Fig. 6. Counting Smears.) Smear techniques can miss potentially contaminated areas. For this reason, direct instrument surveys are also accomplished when background radiation from the package permits, otherwise a general overall rough swipe is taken for the detection of loose contamination.

Heat Load

With more ^{238}Pu being shipped these days, care is taken to determine that

Fig. 5. Smear Technique

Fig. 6. Counting Smears

the heat load limit for the container in question is not exceeded. A heat load of .5 watts/g is assumed for ^{238}Pu . Some of the old DOT Special Permit containers (such as 4960) express the content limit in kilograms of plutonium without specifying the atomic weight or heat load limit. Care is exercised not to misinterpret the spirit and intent of the permit. It is rather obvious that DOT 4960 is not designed for 2250 watts from 4500 g of ^{238}Pu oxide (4.5 kg of plutonium oxide is permitted in the container).

DOT and Other Marking Required

The Health Physics Surveyor determines the correct marking required for the package to be shipped from the container specifications or Special Permit. Reusable containers having permanent markings are altered when necessary. For example, the word fissile on the 6M is blocked out with tape when non-fissile material is enclosed. Figure 7 illustrates the typical marking required and additional optional information that may be included. The specification number and authorized gross weight of the container are required inside of a rectangle. Name of the maker is included under the rectangle as letters if registered with the Bureau of Explosives. The "Special Form" marking is not required but has been included for whom it may concern. In this instance, one would otherwise consider the shipment in violation since .123 Ci of ^{239}Pu (2 grams as non-fissile material per DOT 173.396(a)(1) in "Normal Form" requires a Type B container. As "Special Form" Type A quantity, the Type A, DOT 19B is authorized.

Basis For and Approved By

The name of the person that authorized "Special Form" is recorded for reference purposes. This practice prevents indiscriminate use without authorization from individuals designated for this purpose. For example, only the DOT approves Special Form criteria for IAEA shipments as the competent authority.

Gross Weight

The weight is tabulated for record purposes and is a reminder to determine whether the gross weight limitation of the container specification has been exceeded. This is especially relevant when shipping heavy pieces of depleted uranium. For example, 200 lbs. of D-38 cannot be shipped in a DOT 19B even though the curie content is a Type A quantity. In the case of the DOT 55, certain weights require skids.

Exempt per 173.391(a)1-4

This item is checked when the package meets all five main requirements of the referenced DOT paragraphs. On occasion, 173.391(b) and (c) may be

Fig. 7. Typical Marking and Tamper Seal

applicable and indicated in place of (a)1-4.

Labels

For the guidance of the Health Physics Surveyor, a label display with pertinent instructions is posted in the shipping department (see Figure 8). Since the label required by the regulations is not always indicative of radiation levels, abbreviated guidelines are necessary for immediate reference. For example, the Transport Index assignment for one special permit container is 1.9 for reasons of the fissile material content. A Radioactive Yellow III label is required even though the radiation levels would otherwise allow a Radioactive White label. This is also the case with 173.396(b)(1) and (3)(i) packages. Radioactive Yellow III labels are required for National Security Exempt shipments regardless of radiation levels. The same is true for "large quantities" of radioactive material and for DOT 173.393(j). For postal shipments, a sticker with the following statement is affixed to the package:

RADIOACTIVE MATERIAL - Gamma
Radiation at Surface of Parcel Less
than 10 Milliroentgens for 24 hours
No significant Alpha, Beta, or Neutron
Radiation

Maximum Surface Activity on Container

The external surfaces of the shipping container are surveyed with portable instruments as shown in Figure 9. Neutron measurements taken with an instrument that reads in n/cm^2 -sec are converted to mrem/h for the energy encountered and added to the gamma reading.

Maximum Activity 3' from Container

The same principle holds true for this reading as at the surface. Figure 10 illustrates a reading being taken with an energy independent neutron detector that reads directly in mrem/h.

Source of Criticality Evaluation and Transport Index Assignments

In the case of Fissile Class II shipments, the source of information providing the Transport Index is recorded for reference purposes. Such information is obtained from various DOT paragraphs, special permits and criticality experts in the case of "National Security Exempt" shipments.

Transport Index Used

The larger Transport Index of the two methods for determination is recorded here and on the labels.

Fig. 8. Label Display

Fig. 9. Surveying Package Surface

Fig. 10. Surveying Package at 3 Feet

Tamper Seal

This item acts as a reminder since all non-exempt packages require a tamper seal. One type of tamper seal is shown in Figure 7.

Fissile Class I   II   III

The fissile class is determined from the various DOT paragraphs or DOT special permits based on quantity, form and atomic ratio of hydrogen to fissile material. The class is recorded as a reminder that the larger of the two methods of determining transport index is utilized in the case of Fissile Class II. It also acts as a double reminder concerning the proper use of labels.

The completed data sheet is then reviewed by the traffic manager for violation of any regulation. He signs the certification for receipt by the carrier if the shipment is found to be satisfactory. A file copy of the data sheet is then reviewed by three supervisory personnel of the Health Physics group for any additional correction.

✓ RADIATION SURVEY OF THE TRANSPORTATION OF RADIOACTIVE MATERIALS

Gail D. Schmidt

ABSTRACT

The current standards for the maximum allowable radiation from packages of radioactive material in transportation were published by the Interstate Commerce Commission in 1947. The surface dose rate limitation of 200 mrem/hr was designed to prevent transportation worker from receiving more than 100 mrem per day during package handling, while the limit of 10 mrem/hr at 3 feet was based upon considerations of exposure to photographic film. Although there has been an excellent record of safety in transporting radioactive materials there is little data available in the literature regarding radiation exposure of transportation workers and the general public or the degree of compliance of radioactive material shipments with the transport regulations.

Based upon these considerations the Atomic Energy Commission, Department of Transportation, and Department of Health, Education, and Welfare undertook a joint project to study the matter in greater depth. The survey conducted at 6 geographical locations (and 23 transportation terminals), included a radiation survey of about 300 packages (primarily radiopharmaceuticals). The results of the survey reported in this paper include (1) data on radiation levels from radioactive material packages, (2) an evaluation of radiation exposure to transportation workers, and (3) information on compliance by carriers and shippers with the transport regulations, including, labeling, transport index (TI) and separation distances.

It was the conclusion of the survey that the radiation exposure of transportation workers under the conditions observed were within permissible limits. A possible exception was noted in the case of radium shipments which involved only 8.2% of the packages surveyed but represented 56% of the total exposure rate at 3 feet and 39% of the total surface exposure rate. The acceptability of radiation exposure of individuals resulted from the short transit times of packages and the low exposure rates external to the package, rather than carrier compliance with stowage/separation distance requirements.

INTRODUCTION

Background

The several studies conducted of the occurrence of accidents and incidents during the shipment of radioactive materials have indicated that no known injuries or significant overexposures to persons have occurred.¹ The literature however, contains only limited data that can be used to estimate the exposure of transportation workers (or the public) and the degree of compliance with the transportation regulations. A 1958 survey of the British European Airways cargo warehouse personnel at London Airport over a two-month period indicated an average dose to transport workers of 182 mrem/yr with a maximum of 770 mrem/yr.²

A study by Davis et al.³ using sealed sources to simulate radionuclide shipments involved the measurement of exposure rates at various positions in commercial aircraft. The study showed maximum exposure rates for a loading of 40 TI of about 5 to 25 mR/hr in passenger areas for most possible stowage conditions. The maximum dose rate observed in passenger areas was 400 mR/hr and the study indicated that specific cargo loading limitations might be necessary for certain of these 1960 vintage aircraft. A theoretical and practical discussion of considerations peculiar to air transport of radioactive materials packages is contained in a report by Evans.⁴ There is also an unpublished survey by a Field Inspector of the Bureau of Explosives, Association of American Railroads, during March 1962 at the Railroad Express Agency terminal in Knoxville, Tennessee.⁵ The results of this five day study of radiation levels within the terminal and from packages has not been analyzed but may be of interest to those who desire to study this subject in greater detail.

In recognition of this fact, the three Federal agencies having a regulatory or program interest in this matter, the U.S. Atomic Energy Commission, U.S. Department of Health, Education, and Welfare, and the Interstate Commerce Commission (now the Department of Transportation), began discussions in early 1967 on a proposed joint project to study the matter in greater depth. These discussions culminated in early 1969 with the approval and signature by each agency of a project plan, entitled "Joint Survey of Radioactive Materials Shipments."⁶

Purpose of the Project

The major objectives of this project were:

1. To gather factual information and data concerning radiation levels in the transportation environment from packages of radioactive materials.

2. To obtain information on the status of compliance with the regulations for transport of radioactive materials by shippers and carriers, as related to such factors as package labeling, assignment of transport indices, and observation of prescribed separation distances.

3. To evaluate the effectiveness of the transportation regulations, with regard to the safety of the general public and transportation workers in the radioactive materials transport environment.

Scope of Project

This project involved field surveys in six selected geographical areas in the eastern part of the United States through which significant numbers of radioactive materials packages pass during normal operations. These facilities were selected after a consideration of the proximity of nuclear facilities such as radioisotope suppliers, nuclear fuel facilities, and radiographers.

SURVEY METHODS

Instrumentation

The primary instrumentation used for the survey was the Eberline E-500B beta-gamma survey meter with side-window probe which was calibrated with an NBS calibrated radium source. All measurements were made with the shield closed (i.e. gamma radiation) and as such may be lower than emitted radiation in rems where the package radiations were beta and/or low energy bremsstrahlung. An alpha survey meter was also available (but used infrequently) and smear samples of packages were counted with an end-window GM counter and an alpha scintillation detector. Commercially supplied film badges and TLD finger badges were worn by the surveyors.

Data Collection

The primary data collected during the survey related to radioactive material packages and included radiation levels at contact and 3 feet, wipe tests for surface contamination, shipper compliance with regulatory requirements (such as labeling, package specification, transport indices (TI), etc.), package condition, shipper and destination. Radiation measurements were also made of terminal areas and transport vehicles and data was collected relating to storage conditions, separation of radioactive material packages from film and personnel and total transport index of material present.

It was decided not to monitor workers with film badges nor to conduct radiation surveys inside passenger carrying aircraft. Accordingly, estimation of exposure to transportation workers had to be based upon radiation levels

observed under working conditions. Film badges were used however in a few cases for area monitoring of terminal areas and vehicles.

SURVEY FINDINGS

Package Characteristics

Volume of Radioactive Material Packages.--During the course of this study 398 individual radioactive material packages were observed at six geographical locations and 293 of these packages were surveyed for radiation levels. Estimates based upon carriers' and suppliers' data, and observation indicated that about 1,100 radioactive material packages per day may originate at or pass through these facilities. Projections on an annual basis indicate perhaps 250,000 packages pass through these terminals which might be compared to the estimate of Patterson¹ of over 300,000 packages per year in 1968 for the total U.S. traffic.

Container Types.--Most of the packages surveyed were type A cardboard boxes, only 3 were type B and no "large quantity" or fissile materials" packages were observed. About 50% of the specification packages observed were DOT-7A, Type A, while the DOT-7A-55 and ICC-12B together represented about 33%.

Radiation Levels.--The radiation levels at the surface and at 3 feet from the surface are summarized in Table 1. The average radiation level per package is 56 mR/hr at the surface and 1.27 mR/hr at 3 feet.

The average ratio of the surface exposure rate to that at 3 feet as seen in Table 1 varied from 34 to 68 at different locations with the overall average of 44 compared to a ratio of 20 between the package limits of 200 mrem/hr at the surface and 10 mrem/hr at 3 feet. The observed ratio is consistent with the observed size of the typical package (about 1 cubic foot) and results in an activity limitation based upon surface dose rate. The optimum size to meet both limits requires a package that is about 22 inches on each side. With regard to the radiation exposure aspects, the smaller package results in a higher extremity exposure to handling personnel and a lower exposure from stored material. It was observed that a significant factor in the ratio was the number of radium packages surveyed at a given location. When the radium packages are considered separately the ratio is 31 vs. 60 for all other packages (primarily radiopharmaceuticals).

Table 1. Package Characteristics

Location	No. of Pkgs.	Total Transport Index	Total mR/hr 3 Ft.	Total mR/hr Surface	Average TI Per Pkg.	Ratio Surface to 3 Ft.
Knoxville	38	33.4	23.7	1590	0.9	67
Washington (National)	64	61.8	32.8	2012	0.97	61
Boston	42	36.3	33.2	1596	0.86	48
New York (LaGuardia)	6	21.0	22.7	1405	3.5	62
Newark	25	16.7	25.6	1742	0.7	68
Chicago	118	179.0	234.0	8059	1.5	34
Total	293	348.2	372.0	16,404	1.19	44
Total Ra	24	139.0	207.2	6,465	5.8	31
Total Other	269	209.2	165.0	9,939	0.78	60

It is of interest to note that radium which involved 8.2% of the packages represented 56% of the cumulative exposure rate at 3 feet and 39% of the surface exposure rate.

Ten packages exhibited radiation levels at their surfaces in excess of the allowable 200 mR/hr. Only 3 packages had radiation levels at 3 feet that exceeded the allowable 10 mR/hr. Maximum levels observed at the six locations were:

<u>Location</u>	<u>Surface (mR/hr)</u>	<u>3 feet from Surface (mR/hr)</u>
Knoxville	200	5
Washington, D.C.	175 no label	4.5
Newark	250	9.0
New York	300	7.0
Boston	350	25.0
Chicago	500	60.0

It should be noted that the surface levels which exceeded 200 mR/hr were measured with a detector chamber located inside the survey instrument and at a distance of about 2 inches. Thus for a typical package with dimensions of 8 to 12 inches, the actual surface levels would be higher by about a factor of two.

Transport Index.--The transport index (which is the number of mrem/hr at 3 feet from the surface of the package) as stated on the package label is also tabulated in Table 1. Based on the totals for all packages it is observed that labelled transport index compares closely with the measured exposure rate at 3 feet while the specific locations show values that varied by up to a factor of two. The higher measured exposure rate at Chicago was due to two packages with streaming and when these are eliminated, the labelled transport index is only slightly greater than that measured, which might be expected, due to decay of short-lived radiopharmaceuticals.

There were a total of 72 packages out of 293 where the measured exposure rate at 3 feet exceeded the labelled TI value. In the case of 28 packages or 9.6 percent of the 293 packages the labelled TI was low by 0.5 mR/hr or more. Two of the three packages which exceeded 10 mR/hr at 3 feet involved streaming apparently due to poor packaging, while the third was overloaded and had a labelled TI of twenty-six (26).

Package Compliance.--Of the 293 containers surveyed, 85 or 29% were incorrectly labelled. The most frequent error (63 packages) involves the use of a Yellow-III where a White-I or Yellow-II was adequate. Three packages or about 1% had no label and 6 packages were under labelled.

All packages examined meet the 4 inch minimum dimension requirement specified in the DOT regulations.

The requirement for a security seal (DOT regulations Sec. 173.393(b))-- "a feature such as a seal, which is not readily breakable and which, while intact, will be evidence that the package has not been illicitly opened" appeared to be the single item most frequently violated or misunderstood. Most packages being cardboard were sealed with tape and/or staples and thus may be easily opened and such packages also, are frequently re-used. In most cases where a security seal was used it did not fulfill the requirement of being "not readily breakable," and only in the case of original package use was there reasonably good "evidence that the package has not been illicitly opened." In addition to the standard cardboard box closure using tape and staples, lead wires, and twisted copper wires were observed.

It would appear that a steel security seal with the number recorded on the address label or the radioactive material label would meet the intent of the requirement although not complying completely with the "not readily breakable" provision. Such a seal is not easily applicable to cardboard boxes.

Although packages were smeared at only one geographical location all of the smears were background for both beta-gamma and alpha activity.

Personnel Exposure

In general, ambient radiation levels in the terminals did not exceed background levels. This results from the fact that the radioactive material packages pass through air freight terminals in a pulsed stream which is geared to the user's demand. These shipments most frequently are short half-life radioisotopes for use in nuclear medicine and are shipped to arrive at the destination for an early morning use. While the transit times of packages through terminals were not timed, it was observed that a package spends about 2-4 hours on the average in a terminal. These packages are offered for shipment in the late afternoon and will have been delivered to the user about 16 hours later, i.e. by 8:00 a.m. the next day. A buildup of multiple groups of packages on a transfer dock or in a terminal for extended times was not observed at any of the sites surveyed. The highest radiation levels found for groups of packages was at Chicago in a storage cage or room called the "value room." Packages, containing radium and being briefly held in this room prior to re-shipment, gave radiation levels in the room of 5 to 20 mR/hr. On this evening between the hours of 6:00 p.m. to 9:00 p.m. 12 radium packages with TI of 82 and measured exposure rate of 155.5 mR/hr at 3 feet were held temporarily in this area.

The principal radiation exposure due to radioactive material shipments appears to be the result of the physical handling of radioactive materials. Practices observed at Chicago were quite typical of those followed at the other sites where surveys were conducted. At Chicago it was observed that each package is physically handled many times during the loading, unloading, sorting, labelling, and dispatching process. At the terminal each package was handled as follows:

1. Unloaded from truck or trailer to a conveyor at loading dock (or offered over the counter for shipment).
2. Sorted at two points on the conveyor as to destination.
3. Additional labels or marking were then added to the packages, when appropriate, or where required for a signature shipment.

4. Next, packages were segregated by destination and placed on a hand truck for carrier pickup.

5. Then carrier personnel wheeled the hand truck to the airport tractor--trailer and loaded the trailer.

It was not determined whether the airline carrier loaded directly from this trailer onto the plane or if packages went through a further sorting process at the carrier's terminal. In any case, packages were handled about 6 times at the air express terminal and at least twice more by the carrier prior to flight departure (that is unloading of the trailer onto a portable conveyor at the plane and stowage in the cargo hold).

It was also observed in the survey at Washington, D.C. that about half of the packages handled were being transferred to another flight. In these cases the radioactive material package will be handled in a least three air express (and airline) terminals and by two ground delivery services.

Although no studies were made of the actual worker exposures, two of the survey team members wore TLD finger badges and film badges for whole body exposure measurement on three occasions. The results of the finger and whole body monitoring are given in Table 2. This data indicates a finger dose of 2-3 mrem per package handled.

Table 2. Summary of TLD Finger Monitor

Surveyor	No. of Packages	Total TI	Measured TI	Total Surface Exposure Rate	Finger* Dose (mrem)
#1	23	11.8	6.7	500 mR/hr	50
#1	56	60.6	136	4300	150
#2	40	68.3	52	>2300	110

*Whole body doses were less than 10 mrem in all cases.

Observations of radiation exposure rates and exposure duration were made at Knoxville during handling of radioactive material packages by the ground carrier and airline personnel. Exposure during all operations at Knoxville involved in receiving, sorting, labelling, distributing and loading a package were estimated to be about 1.2 man-millirem per Transport Index and a hand exposure of 6 man-millirem.⁶ The estimated whole body exposure is relatively

higher than the experience of the surveyors (Table 2) which indicates the ratio of finger to whole body exposures to range from 5 to 15. Unfortunately the data collected are not adequate to resolve these differences or to predict the radiation exposure of transportation workers.

OTHER OBSERVATIONS AND SUMMARY

Carrier Compliance

While only limited data was collected regarding carrier compliance with the radioactive material regulations, general observations can be made with regard to two aspects, (1) the stowage limitation of a Transport Index of 50 and (2) placarding of vehicles. In general the carriers did not appear to take precautions to meet the stowage/separation distances requirements for package aggregates. As noted earlier the radioactive material packages were not segregated, but flowed through the terminal in response to a rapid delivery to the final destination. In at least one case as noted earlier the cumulative Transport Index at a terminal exceeded 50.

With regard to placarding, two instances were observed where vehicles were not placarded and one case where the old placard format was used. During the survey about 20 vehicles carrying radioactive material were observed, however, data on placarding was not recorded in all instances.

Package Consolidation

At Washington, D.C. and Chicago it was observed that the individual packages were being combined in a plastic bag consigned to a local delivery service. A tag attached to the bag, bore one Yellow-III label, listed the isotopes included and the total activity, but insofar as we could determine did not state the total transport index. Where such outer containers were encountered during the survey they were not considered to be a package and the enclosed packages were examined individually.

A similar practice was observed at Newark where small packages, addressed to one receiver or destined for one terminal, were placed by the carrier into larger cardboard cartons. The appropriate labels were placed on the exterior of the large package. The total transport index was written on the new labels.

The practice of combining smaller packages into a large package for ease of handling raises questions of the standards for the outer large package and the labelling and marking requirements for the outer package. It appears that the outer container is not required to meet the packaging standards, since all the packages within the larger container meet the standards. The labelling and marking of the outer container is the responsibility of the person, either the shipper or carrier, who puts the smaller packages into the large outer package.

Summary

1. The study examined about 300 radioactive material packages or about 1/1000 th of the annual traffic. The majority of these packages involved the shipment of radiopharmaceuticals or small quantities of radioisotopes. Three type B packages and no nuclear fuel shipments were surveyed.
2. The typical package had a surface exposure rate of about 50 mR/hr or 1/4 of 200 mrem/hr limit and a Transport Index of 1.0 (the maximum T.I. allowed is 10).
3. Packages containing radium with an average surface exposure rate of 270 mR/hr and Transport Index of 5.8 had much higher radiation levels than packages containing other radioactive materials.
4. Although, almost one-third of all packages observed were labelled incorrectly, these deficiencies did not seem to represent any significant reduction in safety. The use of a yellow-III label when not required was the most frequent infraction.
5. The requirement for a security seal which was not readily breakable and while intact presented evidence that a package had not been illicitly opened was not met in the majority of shipments.
6. No contaminated packages were observed.
7. Packages were often consolidated and overpacked in large bags or boxes. These outer containers were not completely identified in some cases with regard to the total transport index and the radioactive materials involved.
8. Carriers are generally unaware of or did not observe stowage and loading restrictions for radioactive material, and the observed total transport index in one terminal exceeded 50.
9. Radiation levels in terminal areas usually did not exceed the normal background levels except in the immediate vicinity of packages.

CONCLUSIONS AND RECOMMENDATIONS

This survey of 293 radioactive material packages has indicated that in general these packages are being shipped in a safe manner and do not present an unacceptable radiation exposure to the public or transportation workers. While a significant number of packages were found that were not in strict compliance with the regulations these noncompliance items were largely administrative in nature and did not in most cases appear to affect the radiation safety of transport.

The radiation limits at the surface and at 3 feet from the package were generally met within acceptable tolerances and radiation levels in terminal areas appeared to be acceptable. Direct monitoring of transportation workers was not undertaken, and thus estimates must be based upon observed radiation levels under working conditions. This data indicates possible cumulative doses to the ground and airline carrier personnel at one location of 1 man-mrem whole body and a hand dose of 6 man-mrem per Transport Index. With this dose shared by several individuals the radiation exposures do not appear excessive. At one terminal, however, it was observed that personnel could be exposed to levels of 5-20 mR/hr for at least 1-2 hours per day which could lead to personnel doses near the occupational limit.

Therefore, it is recommended that a detailed study be undertaken to determine actual personnel exposure of both transportation workers and the general public. Such a survey should utilize personnel monitoring devices, as well as, area monitoring.

The study also indicated several areas where compliance with the transportation regulations was deficient that should be evaluated with regard to possible modification or enforcement of the provisions. With regard to the shipment of small "radiopharmaceutical type" packages these items should be examined further:

- (1) Shipping papers. Frequently a complete description was not given and the shipping paper was not carried separately by the vehicle operator but as a part of the destination label.
- (2) Security seals. The gummed tape or staple closure of cardboard boxes gives some evidence that a box has not been opened but is readily breakable. There does not appear to be a major justification for this requirement.
- (3) Package labels. In view of the significant mislabeling perhaps only one type label should be used for these packages (rather than the current three).
- (4) Vehicle loading and stowage restrictions. The enforcement of stowage limits and separation distances in airfreight terminals would significantly affect the rapid delivery of radiopharmaceuticals which is important for short half-life radionuclides. If the exposure study recommended above shows acceptable individual doses, it is felt this requirement could be relaxed or deleted in terminals and/or vehicles.

REFERENCES

1. PATTERSON, D.C. The accident experience of the USAEC in the shipment of radioactive material. Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials. Pg. 199-210, Gatlinburg, Tennessee, October 14-18, 1968.
2. A Radiological Survey of British European Airways (BEA) Cargo Warehouse and Personnel-London Airport, 1959. Regulations for the Safe Transport of Radioactive Materials--Notes on Certain Aspects of the Regulations Pg. 103-105, Safety Series No. 7, International Atomic Energy Agency, Vienna, 1961.
3. DAVIS, D. M., J. C. Hart, and A. D. Warden. Radiation dose received by passengers and crew on planes carrying the maximum number of radiation units. American Industrial Hygiene Journal, Pg. 497-503, Vol. 22, No. 6, Dec. 1961.
4. EVANS, R. D. Remarks on space limitations in air transport, Chapter X, Physical, Biological, and Administrative Problems associated with the Transportation of Radioactive Substances. Pub. No. 205, Nuclear Sciences Series, Preliminary Report No. 11, 1951.
5. Survey of Handling of Radioactive Materials at the Railway Express Agency Terminal in Knoxville, Tennessee, March 1962. Unpublished Report by R. M. Sharp, Field Inspector, Bureau of Explosives, American Association of Railroads.
6. Joint Survey of Radioactive Materials Shipments. U.S. Atomic Energy Commission, Department of Transportation, and Public Health Service, in preparation.

A METHOD OF CONTROLLING RADIATION EXPOSURE OF PERSONS IN AIRCRAFT
DURING TRANSPORTATION OF RADIOACTIVE MATERIALS

P. A. Lecomte

ABSTRACT

This paper gives a method to calculate segregation distances to be observed between persons and yellow-labelled packages of radioactive materials in aircraft. The method is based on the inverse square rule applied to arrangements of spherical packages with point source located at the centre. Surface dose rate and transport index of the package are the two parameters which have been taken into account for computing the segregation distances. An IBM computer 360 was used for the method of calculation and to prepare the graphs.

INTRODUCTION

For the purpose of adequately protecting persons from being dangerously exposed to external radiation during air transportation of radioactive materials, the IATA Regulations ¹ prescribe that under any circumstances the dose rate of radiation originating from a package shall not exceed :

200 mr/hr or equivalent at any point on the external surface; and

10 mr/hr or equivalent at one metre (40 inches) from the centre of the package, or a Transport Index of 10.

The segregation distances prescribed in Part 2, Section 10 of the IATA Regulations are based on these limits.

However, in the latest draft revision of the IAEA Regulations ² the Transport Index of a package is broadly defined as the number expressing the maximum radiation dose-equivalent expressed in millirem per hour at one metre from the external surface of the package.

Since it is assumed that the surface measurement reference of the Transport Index will be internationally adopted in the near future, particularly considering that this concept was already adopted in the national regulations

of some important isotopes-producing countries, we have based our method of computing the segregation distances on the surface measurement reference of the Transport Index.

BASIC PRINCIPLES

The basic principles used to determine formulae for calculating segregation distances between persons and yellow-labelled packages of radioactive materials in aircraft, were already set out by Aspinall, Gibson and Morley³ in 1963. They consist of deriving a permissible mean dose rate for crew and passengers from recognized factors.

The first factor to be taken into account is the yearly maximum dose for persons. The current IAEA Basic Safety Standards⁴ recommend the following maximum whole body exposure doses :

- for individual members of the public : 500 mrem/year
- for workers not requiring individual
monitoring and special health
supervision : 1,500 mrem/year

Therefore the value of 500 mrem/year has been chosen for controlling radiation exposure of passengers and 1,500 mrem/year for crews.

The second factor to be considered is the maximum foreseeable annual flying period. For passengers it is assumed that 500 hours per annum may be taken as the maximum mean flying time. For crews, in practice, 1,000 hours per annum can be taken as the maximum flying period.

The derivation of a permissible mean dose rate also depends on a third factor which has been called "radioactive traffic factor". It is used to denote the ratio of flights made in company with yellow-labelled packages of radioactive materials to all flights when averaged over a considerable period. From statistics available some years ago for London Heathrow Airport, which is one of the international airports in the world where the movement of radioactive materials is the highest, it has been demonstrated that a radioactive traffic factor of one-in four is unlikely to be reached. However with the increasing movement of radioactive materials, it is considered that a radioactive traffic factor of 1:3 may now safely be used for air transportation throughout the world.

Multiplying the maximum foreseeable annual flying period by the radioactive traffic factor we obtain the annual mean exposure time and the permissible mean dose rate to be achieved by segregation of yellow-labelled packages of

radioactive materials will be the quotient of the annual maximum permissible dose over the annual mean exposure time. We obtain therefore the following permissible mean dose rates :

$$\text{for passengers : } \frac{500}{500 \times \frac{1}{3}} = 3 \text{ mrem/hour}$$

$$\text{for crews : } \frac{1,500}{1,000 \times \frac{1}{3}} = 4.5 \text{ mrem/hour}$$

Another basic principle which was also emphasized by Aspinall, Gibson and Morley³, is the fact that segregation distances for passengers have not to be calculated so that the permissible mean dose rate will never be exceeded at any passenger seat, but that they must be such that the mean dose rate along the cabin floor, i.e. within the passenger-carrying volume, does not exceed the permissible mean dose rate. As a matter of fact the mean dose received by any particular passenger over the years is considerably reduced with the assumption of his random seating within the cabin space and of the random stowage positioning of radioactive packages along the cabin in the aircraft holds.

In other words such random positioning factor must be taken into account in the method of computing the correct segregation distances for passengers. For crews however, in their fixed position in the flight deck, the random positioning factor is not available and the segregation distances must be so calculated that the permissible mean dose rate will never be exceeded at the flight deck boundary.

THE REFERENCE SOURCE

Packages of radioactive materials of the Yellow-II or Yellow-III Category are in various shapes and sizes. The dose rate at their surface will vary from 0.5 to 200 mrem/h and their transport index from 0 to 10. Though in practice the shape of packages will never be a sphere, to facilitate the computing of segregation distances we have chosen as our reference source a package or an arrangement of packages of spherical shape, the source being ponctual and located at the centre of the sphere. In that case the radius of the spherical package, applying the simple "inverse square rule" is found according to the following formula :

$$R = \frac{1}{\sqrt{\frac{NSURF}{UNIT} - 1}} \quad (1)$$

where

R^* = radius of the package, metre,
 $NSURF^*$ = dose rate at the surface, mrem/h,
 $UNIT^*$ = transport index of the package.

Graph 0 shows the value of R with respect to the surface dose rate for given transport indexes.

Arrangement of Packages

Since the Transport Index must not exceed 10 per package, to calculate segregation distances applicable to a sum of transport indexes above 10, we have considered a plane arrangement of packages as our reference source. We have built two kinds of plane arrangement of packages: a compact arrangement where packages are arranged in spiral starting from the first one, and a line arrangement where packages are arranged in a single row. The appropriate arrangement for a given total number of packages shall be that one which generates the most restrictive segregation distance between the plane tangent to the external surface of the packages and the point of interest located perpendicularly to this plane from the centre of gravity of the arrangement.

Compact arrangement

Let us consider a plane spiral of spherical packages. Knowing the co-ordinates x and y of packages number p and n , we have to find the co-ordinates x_{n+1} and y_{n+1} of packages number $n+1$

Fig.1. Compact arrangement of packages

* terms used in the computer program

From Fig.1 the following equations are easily found :

$$\cos \alpha = \frac{x_p - x_n}{2R} \quad (2)$$

$$\sin \alpha = \frac{y_n - y_p}{2R} \quad (3)$$

$$\beta = \alpha - 60^\circ \quad (4)$$

$$x_{n+1} = x_n + 2R \cos \beta \quad (5)$$

$$y_{n+1} = y_n - 2R \sin \beta \quad (6)$$

where

α and β are the angles shown in the figure,

R = radius of packages number p , n , $n+1$.

From Eq. (4) we can derive :

$$\cos \beta = \cos \alpha \cos 60^\circ + \sin \alpha \sin 60^\circ \quad (7)$$

$$\sin \beta = \sin \alpha \cos 60^\circ - \cos \alpha \sin 60^\circ \quad (8)$$

Substituting $\frac{1}{2}$ and $\frac{\sqrt{3}}{2}$ for $\cos 60^\circ$ and $\sin 60^\circ$ respectively in Eq. (7) and (8), the latter two combined with Eq. (5) and (6), we find the co-ordinates x_{n+1} and y_{n+1} which give us the rule to build a compact arrangement of packages :

$$x_{n+1} = \frac{1}{2} (x_n + x_p + \sqrt{3} y_n - \sqrt{3} y_p) \quad (9)$$

$$y_{n+1} = \frac{1}{2} (y_n + y_p + \sqrt{3} x_p - \sqrt{3} x_n) \quad (10)$$

To start the arrangement we also have to impose that co-ordinates of packages number 1 and 2 are as follows :

$$x_1 = y_1 = 0 \quad (11)$$

$$x_2 = 2R \quad (12)$$

$$y_2 = 0 \quad (13)$$

Line arrangement

For line arrangement of packages formulae giving the co-ordinates are very simple :

$$y_{n+1} = y_n = 0 \quad (14)$$

$$x_{n+1} = n2R \quad (15)$$

where

R = radius of packages number n , $n+1$.

Fig.2. Line arrangement of packages

Eq. (11), (12) and (13) remain valid of course for line arrangement of packages.

Centre of gravity of arrangements of packages

Co-ordinates x_M and y_M of the centre of gravity of arrangements of packages will be determined by the following formulae :

$$x_M = \frac{\sum_{n=1}^{NTEST} x_n}{NTEST} \quad (16)$$

$$y_M = \frac{\sum_{n=1}^{NTEST} y_n}{NTEST} \quad (17)$$

where

$NTEST^*$ = total number of packages involved in the arrangement.

THE SEGREGATION DISTANCE

Segregation distances have been calculated by a computer. Consequently general formulae covering any total number of packages in the two kinds of arrangement, compact and line, for both fixed point of interest and random positioning of point of interest along the cabin floor, were needed.

For a single package, using the inverse square rule, the following formula applies when a fixed point of interest is considered :

$$DOSR = \frac{UNIT (1 + R)^2}{(S + R)^2} \quad (18)$$

* term used in the computer program

for random positioning of point of interest along the cabin floor, Eq. (18) becomes :

$$\text{DOSR} = \frac{1}{L} \int_{l=0}^{l=L} \frac{\text{UNIT} (1 + R)^2}{(S + R)^2 + l^2} dl \quad (19)$$

where

- DOSR* = reference mean dose rate at the point of interest, mrem/h,
- R* = radius of the package, metre,
- S* = segregation distance between the surface of the package and the point of interest, metre,
- UNIT* = transport index of the package,
- 2L = cabin length, metre.

For an arrangement of identical packages, when the fixed point of interest is located perpendicularly above the centre of gravity of the arrangement, Eq. (18) becomes :

$$\text{DOSR} = \frac{\sum_{n=1}^{\text{NTEST}} \frac{\text{UNIT} (1 + R)^2}{(S+R)^2 + (y_n - y_M)^2 + (x_n - x_M)^2}}{\sum_{n=1}^{\text{NTEST}}} \quad (20)$$

For an arrangement of identical packages, when the point of interest is moving along an axis parallel to the long axis X of the arrangement of packages and passing by the perpendicular to the centre of gravity of this arrangement, Eq. (19) becomes :

$$\text{DOSR} = \frac{1}{L} \int_{l=0}^{l=L} \left[\sum_{n=1}^{\text{NTEST}} \frac{\text{UNIT} (1 + R)^2}{(S+R)^2 + (y_n - y_M)^2 + (l - x_n + x_M)^2} \right] dl \quad (21)$$

where

- DOSR* = reference mean dose rate at the point of interest, mrem/h,
- R* = radius of packages, metre,
- S* = segregation distance between the plane tangent to the surface of the packages and the point of interest, or the axis passing by this point, metre,
- UNIT* = transport index of the packages,
- NTEST* = total number of packages involved in the arrangement,
- x_n, y_n = co-ordinates of the packages, metre,
- x_M, y_M = co-ordinates of the centre of gravity of the arrangement, metre,
- 2L = cabin length, metre.

Resolving Eq. (21) we find :

$$\text{DOSR} = \sum_{n=1}^{\text{NTEST}} \frac{\text{UNIT} (1+R)^2}{L \sqrt{(S+R)^2 + (y_n - y_M)^2}} \left(\tan^{-1} \frac{L - x_n + x_M}{\sqrt{(S+R)^2 + (y_n - y_M)^2}} + \tan^{-1} \frac{x_n - x_M}{\sqrt{(S+R)^2 + (y_n - y_M)^2}} \right) \quad (22)$$

* terms used in the computer program

Eq. (1), (9), (10), (11), (12), (13), (14), (15), (16), (17), (20) and (22) are the basic formulae which were used in the computer program to calculate the segregation distances (see attachment A). This program permits to obtain the segregation distances applicable to any number of identical packages (of selected surface dose rate and transport index) in both compact and line arrangements, for fixed point of interest (flight deck - 4.5 mrem/h mean dose rate) and random positioning of point of interest (cabin length from 0 to 200 feet - 3 mrem/h mean dose rate).

Attachment B gives an example of a listing from the computer. Note that for fixed point of interest or short cabin length the segregation distance is more restrictive in compact arrangement than in line arrangement, but for large cabin length the distance is more restrictive in line than in compact. In this example for 45 foot cabin length the distance is about the same for both compact and line arrangement.

THE INFLUENCE OF THE RADIUS

According to selected surface dose rate and transport index various package radius can be obtained, as shown in Graph 0. A lower limit of 0.05 metre is to be settled since the IAEA Regulations² prescribe that the smallest overall external dimension of the packaging shall not be less than 10 cm. On the other hand packages with external dimensions exceeding 2 metres are unlikely to frequently move in air transportation. Therefore, an upper limit of 1 metre seems to be realistic.

We fed the computer in order to get the segregation distances applicable to arrangements of packages with different radius, but with the same total sum of transport indexes. Graphs 1, 2 and 3 illustrate the results for cabin lengths 0, 100 and 200 feet respectively. They apply to a total sum of transport indexes 30. In other words the segregation distances corresponding to 10 transport index apply to 3 packages of 10 transport index each, whilst the distances corresponding to 0.5 transport index apply to 60 packages of 0.5 transport index each. It should be noted that the graph computer program selects the most restrictive segregation distance obtained either by compact or by line arrangement of packages.

We see from these graphs that for a given total sum of transport indexes :

- (i) segregation distances are more restrictive when large packages are involved,
- (ii) for short cabin lengths, limited number of packages generates the most restrictive segregation distances,

(iii) for large cabin lengths, the number of packages has practically no influence on the segregation distances.

Since no statistics about dimensions of yellow-labelled packages of radioactive materials were available to us, for the remnant of our demonstration we chose a package radius of 0.288 metre. Though this choice is arbitrary (any other radius could have been chosen), it was felt that it could represent a good average of the dimension of packages encountered in air transportation, without being too optimistic in the segregation distances. On the other hand a 0.288 metre radius corresponds to the typical Yellow-II category package (10 mrem/h at the surface and 0.5 transport index, or 50 mrem/h at the surface and 2.5 transport index according to the revised IAEA Regulations ²). A package of 0.288 metre radius also corresponds to the typical Yellow-III category (200 mrem/h at the surface and 10 transport index).

SELECTION OF THE CORRECT SEGREGATION DISTANCES

Graph 4, together with its listing, indicates the segregation distances applicable from yellow-labelled packages to the boundary of the passenger cabin. These distances were calculated from packages of 0.288 metre radius with the following surface dose rate and transport index in order to obtain the specified total sum of transport indexes.

Table 1. Total sum of TI. with packages characteristics

Total Sum of TI.	Total number of packages	Surface dose rate, mrem/h	TI.
1	1	20	1
2	1	40	2
5	1	100	5
10	1	200	10
15	3	100	5
20	2	200	10
25	5	100	5
30	3	200	10
35	7	100	5
40	4	200	10
45	9	100	5
50	5	200	10
55	11	100	5
60	6	200	10

If applied correctly these distances will always be on the safe side since no correction for intervening shielding material between packages and persons has been made. For example the attenuation of radiation by loading of dead cargo or the shielding properties of partitions or other structural items of the aircraft have not been taken into account. On the other hand, no reduction of segregation distances for the assumption that say the first foot of the cabin space will not be occupied by other than body extremities, has been allowed.

It should also be remembered that the derivation of a permissible mean dose rate depends, amongst other factors, on the foreseeable annual flying period of the inveterate traveller. It may conceivably be assumed that such a passenger is always flying first class which would considerably reduce his random seating within the cabin space. For that reason it is felt that when selecting the appropriate segregation distance to be applied for a mixed first class/economy class cabin aircraft, the length of both types of cabin must be considered separately.

For the airline the application of the correct segregation distance will be in most cases the problem of knowing roughly the total sum of transport indexes which can be loaded in the particular holds of the aircraft located below the first class and/or the economy class cabin, since the distance available between the bottom of the hold and the cabin floor is invariable and fairly short in most instances (for the B-747 this distance is 80 inches maximum). To solve this problem Graphs 5 to 9 inclusive show, in a format directly usable by air cargo staff, the total sum of transport indexes which can be loaded in a passenger aircraft from selected available distance and cabin length.

Finally Graphs 10 and 11 show the segregation distances to be applied to the flight deck, for normal and full load conditions respectively.

X
X   X

Acknowledgments

The author acknowledges the assistance he has received from Mr. C. L. Martin in the preparation of the computer programs.

REFERENCES

1. International Air Transport Association (IATA) Restricted Articles Regulations - 14th Edition, effective 1st June 1971.
2. IAEA Regulations for the Safe Transport of Radioactive Materials - Third Revised Draft, November 1970.
3. K. J. Aspinall, R. Gibson, F. Morley, The Control of exposure to external radiation during the Transport of Radioactive Materials, 1963 - United Kingdom Atomic Energy Authority, Report AHSB (RP) R 31.
4. IAEA Basic Safety Standards for Radiation Protection, Safety Series No. 9, 1967 Edition.

```

// EXEC FFORTRAN
  DIMENSION X(100),Y(100)
  1 FORMAT(I3,1X,I3,1X,F4.1,1X,I1)
  2 FORMAT(///,6X,'SEGREGATION DISTANCE COMPUTING FOR',I4,' PACKAGE',A
    11,/,6X,'OF',F8.5,' METRE RADIUS',I5,' SURFACE DOSE RATE',F6.1,' TR
    2ANSPORT INDEX',/)
  3 FORMAT(///,1X,I3,' FOOT CABIN LENGTH',//4X,'DISTANCE ',4X,' DOSE',/
    1/2X,'FEET INCHES',/)
  4 FORMAT(2X,I3,3X,F5.2,3X,F7.4)
  5 FORMAT(///,1X,'FLIGHT DECK ',F3.1,' MREM/H MEAN DOSE RATE',//3X,' D
    1STANCE ',4X,' DOSE',//2X,'FEET INCHES',/)
  41 FORMAT(' X(',I3,') = ',F8.5,' METRE Y(',I3,') = ',F8.5,' METRE')
  50 FORMAT(1X,'COMPACT ARRANGEMENT OF PACKAGES',/,1X,31(1H*),/)
  51 FORMAT(///,1X,'LINE ARRANGEMENT OF PACKAGES',/,1X,28(1H*),/)
  60 FORMAT(A1,I3,I3,F4.1,F8.5,F3.1,I3,F8.3,F7.4,A1)
  30 READ(1,1)NTEST,NSURF,UNIT,KRO
    IC=-1019199424
    IF(NTEST-999)7,6,6
  7 R=NSURF/UNIT
    R=1./(SQRT(R)-1.)
    IF(R-0.05)56,57,57
  56 R=0.05
  57 IF(NTEST-1)33,33,34
  33 I=1077952576
    GO TO 35
  34 I=-499105728
  35 WRITE(3,2)NTEST,I,R,NSURF,UNIT
    SQ=SQRT(3.)
    KOMP=1
    IF(NTEST-2)53,53,32
  32 WRITE(3,50)
    X(1)=0.
    Y(1)=0.
    X(2)=2.*R
    Y(2)=0.
    DO 8 I=3,NTEST
      IR=I-1
      IP=IR-1
      IPP=IP
      12 XN=(X(IR)+X(IP)+Y(IR)*SQ-Y(IP)*SQ)/2.
        YN=(Y(IR)+Y(IP)+X(IP)*SQ-X(IR)*SQ)/2.
        IXN=XN*1000
        IYN=YN*1000
        DO 9 J=1,IPP
          IK=IR-J
          KRE=X(IK)*1000
          IF(IXN-KRE)9,10,9
        10 KRY=Y(IK)*1000
          IF(IYN-KRY)9,11,9
        11 IP=IK
          GO TO 12
        9 CONTINUE
        X(I)=XN
        8 Y(I)=YN
  55 XM=0.
    YM=0.
    DO 13 I=1,NTEST
      XM=XM+X(I)

```

```

13  YM=YM+Y(I)
    XM=XM/NTEST
    YM=YM/NTEST
    IF(KRO)14,14,15
15  DO 40 K=1,NTEST
    I=K+NTEST
40  WRITE(3,41)K,X(K),K,Y(K)
14  F=(1+R)**2
    DOSR=4.5
23  STEP=1000.
    ND=1
    SI=2004.
    WRITE(3,5)DOSR
19  DOS=0.
    S=SI*0.0254
    IFT=0
    DO 16 I=1,NTEST
16  DOS=DOS+UNIT*F/((S+R)**2+(X(I)-XM)**2+(Y(I)-YM)**2)
    ISI=SI
45  IF(ISI-12)43,44,44
44  ISI=ISI-12
    IFT=IFT+1
    GO TO 45
43  SI=SI-12*IFT
    IF(KRO-1)61,61,62
62  WRITE(3,4)IFT,SII,DOS
61  IF(DOS-DOSR)17,17,18
17  SI=SI-STEP
    IF(SI)18,18,19
18  SI=SI+STEP
    STEP=STEP/10.
    SI=SI-STEP
    ND=ND+1
    IF(ND-6)19,19,20
20  WRITE(3,4)IFT,SII,DOS
    IFF=-968867776
    IGRAN=0
    SI=SII+12*IFT
    WRITE(2,60)IFF,NTEST,NSURF,UNIT,R,DOSR,IGRAN,SI,DOS,IC
    IF(DOSR-3)21,22,21
21  DOSR=3.0
    GO TO 23
22  XR=2.5
    DO 24 I=1,40
    XL=I*XR*0.3048
    STEP=1000.
    ND=1
    IGRAN=2*XR*I
    WRITE(3,3)IGRAN
28  DOS=0.
    IFT=0
    S=SI*0.0254
    DO 25 J=1,NTEST
    DIV=(S+R)**2+(Y(J)-YM)**2
    DIV=SQRT(DIV)
    ARC=ATAN((XL-X(J)+XM)/DIV)+ATAN((X(J)-XM)/DIV)
25  DOS=DOS+UNIT*F*ARC/(DIV*XL)
    ISI=SI
48  IF(ISI-12)46,47,47
47  ISI=ISI-12

```

```

      IFT=IFT+1
      GO TO 48
46  SII=SI-12*IFT
      IF(KRO-1)58,58,59
59  WRITE(3,4)IFT,SII,DOS
58  IF(DOS-DOSR)26,26,27
26  SI=SI-STEP
      IF(SI)27,27,28
27  SI=SI+STEP
      STEP=STEP/10.
      ND=ND+1
      IF(ND-6)28,28,29
29  WRITE(3,4)IFT,SII,DOS
      SI= SII+12*IFT
24  WRITE(2,60)IFF,NTEST,NSURF,UNIT,R,DOSR,IGRAN,SI,DOS,IC
      IF(KOMP-1)30,53,30
53  WRITE(3,51)
      KOMP=KOMP+1
      XRE=2*R
      DO 54 I=1,NTEST
      IN=I-1
      X(I)=0.+IN*XRE
54  Y(I)=0.
      IC=-750763963
      GO TO 55
6  CALL EXIT
      END

```

SEGREGATION DISTANCE COMPUTING FOR 12 PACKAGES
OF 0.28801 METRE RADIUS 50 SURFACE DOSE RATE 2.5 TRANSPORT INDEX

COMPACT ARRANGEMENT OF PACKAGES

*****

X(1) = 0.0	METRE	Y(1) = 0.0	METRE
X(2) = 0.57601	METRE	Y(2) = 0.0	METRE
X(3) = 0.28801	METRE	Y(3) = -0.49884	METRE
X(4) = -0.28801	METRE	Y(4) = -0.49884	METRE
X(5) = -0.57601	METRE	Y(5) = -0.00000	METRE
X(6) = -0.28801	METRE	Y(6) = 0.49884	METRE
X(7) = 0.28801	METRE	Y(7) = 0.49884	METRE
X(8) = 0.86402	METRE	Y(8) = 0.49884	METRE
X(9) = 1.15203	METRE	Y(9) = 0.00000	METRE
X(10) = 0.86402	METRE	Y(10) = -0.49884	METRE
X(11) = 0.57602	METRE	Y(11) = -0.99768	METRE
X(12) = 0.00000	METRE	Y(12) = -0.99769	METRE

FLIGHT DECK 4.5 MREM/H MEAN DOSE RATE

DISTANCE		DOSE
FEET	INCHES	
9	8.48	4.5004

FLIGHT DECK 3.0 MREM/H MEAN DOSE RATE

DISTANCE		DOSE
FEET	INCHES	
12	2.48	3.0000

5 FOOT CABIN LENGTH

DISTANCE		DOSE
FEET	INCHES	
12	1.59	3.0001

45 FOOT CABIN LENGTH

DISTANCE		DOSE
FEET	INCHES	
8	2.85	3.0003

200 FOOT CABIN LENGTH

DISTANCE		DOSE
FEET	INCHES	
1	4.45	3.0003

LINE ARRANGEMENT OF PACKAGES

*****

X(1) = 0.0	METRE	Y(1) = 0.0	METRE
X(2) = 0.57601	METRE	Y(2) = 0.0	METRE
X(3) = 1.15203	METRE	Y(3) = 0.0	METRE
X(4) = 1.72804	METRE	Y(4) = 0.0	METRE
X(5) = 2.30406	METRE	Y(5) = 0.0	METRE
X(6) = 2.88007	METRE	Y(6) = 0.0	METRE
X(7) = 3.45609	METRE	Y(7) = 0.0	METRE
X(8) = 4.03210	METRE	Y(8) = 0.0	METRE
X(9) = 4.60811	METRE	Y(9) = 0.0	METRE
X(10) = 5.18413	METRE	Y(10) = 0.0	METRE
X(11) = 5.76014	METRE	Y(11) = 0.0	METRE
X(12) = 6.33616	METRE	Y(12) = 0.0	METRE

FLIGHT DECK 4.5 MREM/H MEAN DOSE RATE

DISTANCE		DOSE
FEET	INCHES	
8	4.15	4.5006

FLIGHT DECK 3.0 MREM/H MEAN DOSE RATE

DISTANCE		DOSE
FEET	INCHES	
11	0.16	3.0001

5 FOOT CABIN LENGTH

DISTANCE		DOSE
FEET	INCHES	
10	11.72	3.0002

45 FOOT CABIN LENGTH

DISTANCE		DOSE
FEET	INCHES	
8	2.75	3.0000

200 FOOT CABIN LENGTH

DISTANCE		DOSE
FEET	INCHES	
1	9.72	3.0007

GRAPH 0

PAC. RADIUS VERSUS SURF. DOSE FOR TRANS. IND. CONSTANT
*****

1	CURVE	IS	FOR	TRANS. IND. =	0.100
2	CURVE	IS	FOR	TRANS. IND. =	0.200
3	CURVE	IS	FOR	TRANS. IND. =	0.500
4	CURVE	IS	FOR	TRANS. IND. =	1.000
5	CURVE	IS	FOR	TRANS. IND. =	1.500
6	CURVE	IS	FOR	TRANS. IND. =	2.000
7	CURVE	IS	FOR	TRANS. IND. =	2.500
8	CURVE	IS	FOR	TRANS. IND. =	3.000
9	CURVE	IS	FOR	TRANS. IND. =	10.000

GRAPH 1

SEGR. DIST. VERSUS TRANS. IND. FOR RADIUS CONSTANT

CAS. LENGTH = 0.0

3 MREM/H MEAN DOSE RATE - TOTAL SUM OF TI. 30

1	CURVE IS FOR RADIUS	=	1.000
2	CURVE IS FOR RADIUS	=	0.809
3	CURVE IS FOR RADIUS	=	0.547
4	CURVE IS FOR RADIUS	=	0.462
5	CURVE IS FOR RADIUS	=	0.288
6	CURVE IS FOR RADIUS	=	0.188
7	CURVE IS FOR RADIUS	=	0.111

GRAPH 2

10.0000

6 5 4 3 2 1

7.5000

5 5 4 3 1

5.0000

7 6 5 4 3 2 1

2.5000

7 6 5 4 3 1

2.0000

7 6 5 4 3 2 1

1.0000

7 6 5 42 1

0.5000

4 7 6

0.0 10.0000 20.0000 30.0000 40.0000 50.0000 60.0000 70.0000 80.0000 90.0000 100.0000

INCHES

T1.

SEGR.DIST. VERSUS TRANS.IND. FOR RADIUS CONSTANT

CAB.LENGTH = 100.000

*****

*****

3 MREM/H MEAN DOSE RATE TOTAL SUM OF T1. 30

1 CURVE IS FOR RADIUS = 1.000
2 CURVE IS FOR RADIUS = 0.809
3 CURVE IS FOR RADIUS = 0.547
4 CURVE IS FOR RADIUS = 0.462
5 CURVE IS FOR RADIUS = 0.288
6 CURVE IS FOR RADIUS = 0.188
7 CURVE IS FOR RADIUS = 0.111

GRAPH 3

SEGR.DIST. VERSUS TRANS. IND. FOR RADIUS CONSTANT CAB. LENGTH = 200.000
 *****

3 MREM/H MEAN DOSE RATE - TOTAL SUM OF TI. 30

1	CURVE IS FOR RADIUS	=	1.000
2	CURVE IS FOR RADIUS	=	0.809
3	CURVE IS FOR RADIUS	=	0.547
4	CURVE IS FOR RADIUS	=	0.462
5	CURVE IS FOR RADIUS	=	0.288
6	CURVE IS FOR RADIUS	=	0.188
7	CURVE IS FOR RADIUS	=	0.111

X	Y1	Y2	Y3	Y4	Y5	Y6	Y7	Y8	Y9
0.0	17.9300	20.0600	54.1200	91.2400	119.3300	148.4800	173.1200	194.8300	
5.0000	13.6800	26.7600	51.9200	79.4500	117.9300	147.5600	172.3300	194.1200	
10.0000	6.8800	19.8400	46.2200	75.2300	114.7900	144.8600	169.9900	192.5300	
15.0000	2.1900	13.4500	39.1200	68.9000	109.8300	140.5900	166.2300	188.6400	
20.0000	0.0	8.7300	32.3000	61.7700	103.6000	135.0400	161.2500	184.1000	
25.0000	0.0	5.3400	26.4800	54.7100	96.6500	128.5800	155.3100	178.6000	199.4100
30.0000	0.0	2.9700	21.7300	48.2300	89.5200	121.6700	148.6900	172.3500	193.4700
35.0000	0.0	1.0000	17.8600	42.5200	82.5000	114.5600	141.6500	165.5900	186.9400
40.0000	0.0	0.0	14.7700	37.5700	75.9000	107.4900	134.5200	158.5100	180.0200
45.0000	0.0	0.0	12.2100	33.3200	69.8200	100.6700	127.4590	151.3300	172.8900
50.0000	0.0	0.0	10.0800	29.6500	64.2900	94.2100	120.5600	144.1900	165.6810
55.0000	0.0	0.0	8.2800	26.4900	59.2900	88.1700	113.9690	137.3000	158.6300
60.0000	0.0	0.0	6.7610	23.7400	54.7900	82.5800	107.7000	130.6500	151.7600
65.0000	0.0	0.0	5.4410	21.3400	50.7500	77.4400	101.8200	124.2900	145.1000
70.0000	0.0	0.0	4.3010	19.2310	47.1200	72.7100	96.3290	118.2500	138.6900
75.0000	0.0	0.0	3.3010	17.3610	43.8400	68.3800	91.2000	112.5600	132.5700
80.0000	0.0	0.0	2.4210	15.7010	40.8800	64.4100	86.4590	107.2000	126.7600
85.0000	0.0	0.0	1.6410	14.2110	38.2000	60.7600	82.0490	102.1800	121.2700
90.0000	0.0	0.0	0.9410	12.8810	35.7600	57.4200	77.9500	97.4800	116.0800
95.0000	0.0	0.0	0.3110	11.6710	33.5300	54.3400	74.1690	93.0900	111.1300
100.0000	0.0	0.0	0.0	10.5710	31.5000	51.5000	70.6390	88.9900	106.5900
105.0000	0.0	0.0	0.0	9.5710	29.6300	48.9800	67.3600	85.1500	102.2700
110.0000	0.0	0.0	0.0	8.6610	27.9100	46.4500	64.3190	81.5500	98.2000
115.0000	0.0	0.0	0.0	7.8210	26.3300	44.2000	61.4700	78.1900	94.3700
120.0000	0.0	0.0	0.0	7.0510	24.8600	42.1100	58.8100	75.0200	90.7600
125.0000	0.0	0.0	0.0	6.3410	23.5000	40.1600	56.3390	72.0700	87.3700
130.0000	0.0	0.0	0.0	5.6810	22.2300	38.3400	54.0100	69.2900	84.1700
135.0000	0.0	0.0	0.0	5.0610	21.0500	36.6400	51.8400	66.6800	81.1500
140.0000	0.0	0.0	0.0	4.4920	19.9500	35.0500	49.7900	64.2100	78.3000
145.0000	0.0	0.0	0.0	3.9620	18.9200	33.5600	47.9700	61.8900	75.6100
150.0000	0.0	0.0	0.0	3.4620	17.9600	32.1500	46.0690	59.7010	73.0600
155.0000	0.0	0.0	0.0	2.9920	17.0500	30.8300	44.3500	57.6210	70.6500
160.0000	0.0	0.0	0.0	2.5520	16.1900	29.5900	42.7390	55.6610	68.3600
165.0000	0.0	0.0	0.0	2.1420	15.3900	28.4100	41.2190	53.8100	66.1900
170.0000	0.0	0.0	0.0	1.7520	14.6300	27.3010	39.7690	52.0410	64.1300
175.0000	0.0	0.0	0.0	1.3820	13.9100	26.2510	38.3990	50.3710	62.1600
180.0000	0.0	0.0	0.0	1.0320	13.2300	25.2510	37.1000	48.7810	60.3000
185.0000	0.0	0.0	0.0	0.7020	12.5800	24.3010	35.8600	47.2610	58.5200
190.0000	0.0	0.0	0.0	0.3930	11.9710	23.4010	34.6800	45.8210	56.8200
195.0000	0.0	0.0	0.0	0.0930	11.3910	22.5410	33.5600	44.4510	55.2000
200.0000	0.0	0.0	0.0	0.0	10.8310	21.7200	32.4900	43.1310	53.6500

GRAPH 4

SEGR.DIST. VERSUS CAB.LENGTH FOR SUM OF TI. CONSTANT
*****

MEAN DOSE = 3.000
*****

PACKAGES OF 0.288 METRE RADIUS

1	CURVE IS FOR SUM OF TI. =	1.000
2	CURVE IS FOR SUM OF TI. =	2.000
3	CURVE IS FOR SUM OF TI. =	5.000
4	CURVE IS FOR SUM OF TI. =	10.000
5	CURVE IS FOR SUM OF TI. =	20.000
6	CURVE IS FOR SUM OF TI. =	30.000
7	CURVE IS FOR SUM OF TI. =	40.000
8	CURVE IS FOR SUM OF TI. =	50.000
9	CURVE IS FOR SUM OF TI. =	60.000

GRAPH 5

STCR.DIST. VERSUS SUM OF TI. FOR CAB.LENGTH CONSTANT
*****

MEAN DISE = 3.000
*****

PACKAGES OF 0.226 METRE RADIUS

1 CURVE IS FOR CAB.LENGTH = 0.0
2 CURVE IS FOR CAB.LENGTH = 5.000
3 CURVE IS FOR CAB.LENGTH = 10.000
4 CURVE IS FOR CAB.LENGTH = 15.000
5 CURVE IS FOR CAB.LENGTH = 20.000
6 CURVE IS FOR CAB.LENGTH = 25.000
7 CURVE IS FOR CAB.LENGTH = 30.000
8 CURVE IS FOR CAB.LENGTH = 35.000

GRAPH - 6

TI.SUM	0.0	20.0000	40.0000	60.0000	80.0000	100.0000	120.0000	140.0000	160.0000	180.0000	200.0000				
60.0000								8	7	6	5	4	3	2	1
55.0000								8	7	6	5		3	2	1
50.0000								8	7	5	5	4	3	2	1
45.0000								8	7	6	5	4	3	2	1
40.0000								8	7	6	5	4	3	2	1
35.0000								8	7	6	5	4	3	2	1
30.0000								8	7	6	5	4	3	2	1
25.0000								8	7	6	5	4	3	2	1
20.0000								8	7	6	5	4	3	2	1
15.0000								8	7	6	5	4	3	2	1
10.0000								8	7	6	5	4	3	2	1
5.0000								8	7	6	5	4	3	2	1

SEGR.DIST. VERSUS SUM OF TI. FOR CAB LENGTH CONSTANT
*****

MEAN DOSE = 3.000
*****

PACKAGES OF 0.288 METRE RADIUS

2	CURVE	IS	FOR	CAB.LENGTH	==	40.000
3	CURVE	IS	FOR	CAB.LENGTH	==	45.000
4	CURVE	IS	FOR	CAB.LENGTH	==	50.000
5	CURVE	IS	FOR	CAB.LENGTH	==	55.000
6	CURVE	IS	FOR	CAB.LENGTH	==	60.000
7	CURVE	IS	FOR	CAB.LENGTH	==	65.000
8	CURVE	IS	FOR	CAB.LENGTH	==	70.000
9	CURVE	IS	FOR	CAB.LENGTH	==	75.000

GRAPH

SEGR.DIST. VERSUS SUM OF TI. FOR CAB.LENGTH CONSTANT
*****

MEAN DOSE = 3.000
*****

PACKAGES OF 0.286 METRE RADIUS

1	CURVE IS FOR CAB.LENGTH =	80.000
2	CURVE IS FOR CAB.LENGTH =	85.000
3	CURVE IS FOR CAB.LENGTH =	90.000
4	CURVE IS FOR CAB.LENGTH =	95.000
5	CURVE IS FOR CAB.LENGTH =	100.000
6	CURVE IS FOR CAB.LENGTH =	105.000
7	CURVE IS FOR CAB.LENGTH =	110.000
8	CURVE IS FOR CAB.LENGTH =	115.000

GRAPH 9

SEGR.DIST. VERSUS SUM OF TI. FOR CAB.LENGTH CONSTANT
*****

MEAN DOSE = 3.000
*****

PACKAGES OF 0.288 METRE RADIUS

1 CURVE IS FOR CAB.LENGTH =	120.000
2 CURVE IS FOR CAB.LENGTH =	125.000
3 CURVE IS FOR CAB.LENGTH =	130.000
4 CURVE IS FOR CAB.LENGTH =	135.000
5 CURVE IS FOR CAB.LENGTH =	140.000
6 CURVE IS FOR CAB.LENGTH =	145.000
7 CURVE IS FOR CAB.LENGTH =	150.000
8 CURVE IS FOR CAB.LENGTH =	155.000

GRAPH 9

SEGR.DIST. VERSUS SUM OF TI. FOR CAB.LENGTH CONSTANT
 *****

MEAN DOSE = 3.000
 *****

PACKAGES OF 0.288 METRE RADIUS

1	CURVE IS FOR CAB.LENGTH =	160.000
2	CURVE IS FOR CAB.LENGTH =	165.000
3	CURVE IS FOR CAB.LENGTH =	170.000
4	CURVE IS FOR CAB.LENGTH =	175.000
5	CURVE IS FOR CAB.LENGTH =	180.000
6	CURVE IS FOR CAB.LENGTH =	185.000
7	CURVE IS FOR CAB.LENGTH =	190.000
8	CURVE IS FOR CAB.LENGTH =	195.000
9	CURVE IS FOR CAB.LENGTH =	200.000

GRAPH 10

SEGR.DIST. VERSUS SUM OF TI. FOR PAC.RADIUS CONSTANT
 *****

MEAN DOSE = 4.500
 *****

FLIGHT DECK NORMAL CONDITION

1 CURVE IS FOR PAC.RADIUS = 0.288

GRAPH 11

SEGR.DIST. VERSUS SUM OF TI. FOR PAC.RADIUS CONSTANT
*****

MEAN DOSE = 4.500
*****

FLIGHT DECK FULL LOAD CONDITION

1 CURVE IS FOR PAC.RADIUS = 0.288

✓ ANALYSIS OF PACKAGING COMPLIANCE
TO TRANSPORT REGULATIONS

M. Labrousse
H. Lucas

ABSTRACT

This report shows the methods that can be used to prove that a packaging complies with the Type B drop and fire test requirements: calculations, tests on components, tests on scale models. As regards fire resistance, besides the classical methods, an analytical method giving an approximate solution to Fourier equation is mentioned. As regards drop test, we show how to determine the reduction of shielding following an impact and how to determine the elongation of the screws of a closing device. The tests on components allow one to fill in gaps on some badly known particular points. The tests on scale models look promising. However, there are some difficulties in carrying them out. In conclusion, we hope for an international cooperation with a view to developing solutions to these problems.

INTRODUCTION

Since 1967, the shipments of radioactive materials carried out by the CEA in France have remained at a rather constant level corresponding to about 140,000 packages weighing between 40,000 and 50,000 tons and a total distance covered of 3 million kilometers. The shipments carried out for private firms and the shipments of radionuclides manufactured by the CEA are not included in these figures, though they are increasing steadily.

A new increase of these quantities is to be expected, in particular because of the raise in power of the latest gas graphite power plants (St Laurent II, Bugey, Vandellós) and of the starting of a program of light water power plants; the increase will mainly concern the shipments of natural and enriched UF 6 and of irradiated fuel.

The total turnover per year is presently of about 4 million dollars, of which 3 million dollars are administered by the Section des Transports Spéciaux of the CEA. The duties of the Section des Transports Spéciaux are of two kinds and include on the one hand the execution of the transports themselves and on the other hand, studies and orders for various materials (packagings, vehicles, handling equipments, etc.).

Though rather variable, the mean amount of the study and testing expenses involved in packagings, particularly that spent to insure that these packagings meet

the existing regulations, may be assumed on an average of \$ 100,000 per year for the last five years. This amount will tend to increase in the coming years, mainly because of the particular difficulties presented by the packagings for the fuel of the light water or fast reactors.

METHODS OF ANALYSIS OF PACKAGING COMPLIANCE TO TRANSPORT REGULATIONS

We shall study only the Type B or large source packagings and consider only the compliance to point C-2.3 (Annexe IV, § 1.4.2 and 1.4.3) of the IAEA Recommendations, without examining the problems of criticality or shielding. In other words, we shall only consider the resistance to fire and to 9 m drop.

Real tests performed on one or several packagings of the same type are evidently always possible. This was the method applied at first. It seemed to be beyond question. We dropped the packaging from a 9 m height at a judiciously selected impact angle; then the packaging was submitted to an open fire of 800° C for 30 minutes. We checked, after both tests, whether or not the packaging met the requirements of § C-2.3*. The designer should have been satisfied. But who can warrant that this judiciously selected angle - selected in fact on rather subjective criteria - is the angle giving the maximum damage? A difference of a few degrees may change the impact surface and therefore the resistance of the packaging by a large amount. Who can warrant that the flames remained between 0.7 and 3 m thick during the 30 minutes of the test, and that the packaging was exactly in the position most damaging to it?

With much experience and a little good faith, the designer realizes that the test performed does not offer a total guarantee. He also sees that the test is expensive. We do not mention the testing facilities, which may be amortized on several years, but except for some cases, each test means a packaging damaged, and on small series the cost becomes prohibitive.

In any case, the large packagings that are considered for the transport of irradiated fuel and whose weight may reach or even exceed 100 tons are not easy to test.

There are three reasons: scientific accuracy, economics, and technical possibilities, which militate in favour of the use of other solutions.

We are going to examine successively three of these solutions, showing both their advantages and disadvantages.

---

*i.e. no loss or dispersion of the radioactive contents could happen, and the shielding would reduce the dose rate at 1 meter to at most 1 R/hr.

CALCULATIONS

Resistance to fire

The IAEA Recommendations, considering the open fire test or implicitly the oven test as a means of demonstration only, define the thermal test in terms of quantities of heat brought to the packaging. Therefore, it is normal to use this definition directly.

Rough calculations

A rough estimate, but one which is very easy to use, consists in writing an equation of the following form:

$$Q = M C_p \Delta \theta \quad (1)$$

The quantity, Q , of heat absorbed is calculated assuming a constant temperature T_c on the outer shell, intermediate between the initial temperature and 800°C

$$P = 23765 \left[1 - \left(\frac{T}{1073} \right)^4 \right] \text{ Kcal/hr/m}^2 \quad (2)$$

where

C_p = the heat capacity of the packaging

$\Delta \theta$ = the rise in temperature of the packaging.

Thus, we obtain the mean temperature at which the packaging mass will be at the end of the test. If this temperature were to approach the acceptable limit temperature for the material employed, the packaging must very probably be rejected. If we obtain, for example, 300°C for a packaging with a lead shielding, a certain mass of lead will melt. In this case, special arrangements must be used to avoid a rupture of the shell due to melting of lead. However, this method, used only at the beginning, is not quite correct. The choice of the temperature of the outer shell used to calculate Q is rather large:

for $T = 273 + 200$, the packaging will receive (for 1 m^2 surface) 22,635 Kcal

for $T = 273 + 400$, the packaging will receive (for 1 m^2 surface) 20,085 Kcal

for $T = 273 + 600$, the packaging will receive (for 1 m^2 surface) 13,665 Kcal

Experience shows that the mean temperature of the outer shell is below 400°C , but the choice remains arbitrary and the error is at least 10%. More important is the lack of information on the value of the gradient inside the shielding.

Finite difference method

A more elaborate method, the method of the diagram of Binder and Schmidt, allows one to obtain the distribution of the temperatures within the shielding thickness. We are not going to describe this method, which is well known. It also permits one to take into account the variation of flux versus time, since we have

the temperature of the shell at any time. However, this method is rather tiresome to use, and the more so as we want to have a great accuracy. The numerical method of Dusiemberre, belonging to the same class of techniques, presents the same disadvantages.

Use of computers

The use of computers to solve the equations of Fourier allows one to go still further. For example, we shall be able to take into account the dilatation that can create gaps between the different materials or increase those already existing. We use currently such methods either with analog computers or with digital computers.

Analytic method

It is well known that the problem of fire cannot be solved mathematically. But it can be given an approximate solution, quite satisfactory. In fact, it is a matter of defining in terms of the time t :

- the temperature in any point of a hollow solid $\theta(x, t)$ and in particular on its surface i.e. $\theta(t)$,
- the flux received by this surface, i.e. $\varphi(t)$.

The relation between temperature and flux is given by Stefan's law

$$\varphi = R(\theta) \quad (3)$$

Without going into all the details, it can be shown that the Fourier equation imposes a relation of convolution between θ and φ :

$$\theta = h * \varphi \quad (4)$$

We make the approximation

$$\varphi = 1 - \frac{\theta * \theta'}{a^2} \quad (\text{normalized flux}) \quad (5)$$

Those last two equations define $\varphi(t)$ by its Laplace transform, which is the positive solution of the equation of 2nd degree

$$H^2 \Phi^2 + \frac{a^2}{p} \Phi - \frac{a^2}{p^2} = 0 \quad (6)$$

where

a = the final temperature: $a = \theta(\infty)$

H and Φ = the Laplace transform of h and φ

$$\Phi(p, x) = \int_0^\infty \varphi(x, t) e^{-pt} dt \quad (7)$$

However, all these solutions apply only to comparatively simple geometrical shapes. The ideal would be to have spherical packagings. But this is not the general case; far from it. The cylinder, if it is long enough to enable one to disregard the influence of the ends, will still be amenable to such solutions. With parallelepipeds, we are going to have difficulties on edges and particularly on corners. But their influence, which can be assessed with a certain approximation, remains local; on a large packaging, it does not much change the value of the absorbed flux.

Methods suitable for heat barriers

The problem of fire resistance may be rather easily solved by incorporating a thermal insulator between an outer steel shell and the lead shielding. If the packaging contains radioactive materials with high residual power, whose outward disposal must be insured, the heat barrier must be effective in one direction only: from outside to inside. If the residual power is not too high, the use of plaster, good conductor in normal conditions when it is moist, and insulating when heated above 100° C, gives the solution. A simple calculation may then be performed.

The 30 minutes are divided into two parts:

- in the first part, using a conservative method, the outer shell is assumed to be at a rather low constant temperature. The flux is assumed to be absorbed only by water vaporizing. For a packaging of the design used for the irradiated fuel of gas graphite reactors, including a layer, e, of plaster 12.5 mm thick between lead shielding and outer shell, we find a duration of 6 to 7 min. for a shell temperature of 150° C;

- in the second part, account is taken of the thermal conductivity of dry plaster, which decreases to 0.2 Kcal m/hr/m²/°C (instead of 0.6 when it is moist), and the flux going through is calculated, assuming that the maximum temperature of the outer shell is 800° C. We introduce thereby another factor of safety, because the temperature is in practice below 700° C. (If the temperature had reached 800° C, there would not have been any heat flux between the fire and the shell).

We have on the one hand:

$$dQ = MC_p d\theta \quad (8)$$

where

MC_p = the heat capacity of the shielding

and on the other hand:

$$dQ = \frac{\lambda S}{e} (800 - \theta) \quad (9)$$

where

S = the external surface area

from which, we draw by integration:

$$\log \frac{800 - \theta}{800 - \theta_0} = \frac{\lambda}{e} \frac{S}{MC_p} T_2 \quad (10)$$

where

T_2 = the duration of the second part.

Concerning the irradiated fuel packaging previously taken as an example, we find that starting from an initial temperature $\theta_0 = 120^\circ \text{C}$, we reach 177°C . This is of course a mean temperature. The temperature gradient in the lead being low because of the thermal protection, and the mean temperature being far enough from the melting temperature of lead, we can be sure that melting will not occur.

Lead crushing

In regard to overall fracture of container, the use of lead in most of our packagings with an outer shell made of stainless steel having a good ductility gives us a good degree of safety. But the problem of the lid closing device behaviour has not been touched upon. It is bound up with the deceleration suffered by the packaging on crushing. Though the crushing itself is generally light (comparatively light, and for big packagings only), we had however to make sure that it did not reduce the shielding thickness beyond the limit of the dose rate specified by the Recommendations for accident conditions (see § C-2.3.1).

We tried to solve those problems by a purely mathematical approach, in the particular case of a cylinder, starting from the assumption that drop energy was absorbed entirely by creeping of the material shifted on crushing.

Though this assumption is not rigorous and the value of this "creeping energy" varies in accordance with a certain number of parameters: material quality (e.g. antimony content of lead), material temperature, thickness and quality of the steel sheath surrounding the lead; it is still constant enough to give the value of the angle of drop involving the maximum reduction or deceleration (the most important thing to know), even though it does not give the precise absolute value of the thickness reduction or of the deceleration versus time.

We have then at our disposal curves known as "hull curves", forming the family of the envelopes of the crushing plans, plans limiting the crushed volume V . These curves, designed for a standardized cylinder, are functions of a single parameter: the relation $\frac{V}{V_0}$ of the crushed volume to the external volume of the packaging. V takes into account both the height of drop and the "creeping energy".

Analysis of lid closure behaviour

Calculations of the maximum static force required to reach the elastic limit of lid closure screws give an idea of the safety offered by a lid closing device during an impact. To this force will correspond a deceleration δ_L . Therefore, if it can be shown that during the drop test the deceleration supported by the packaging is below δ_L , the lid closing device will be satisfactory. This method of approach is conservative. In fact, this method involves applying the maximum strength during an infinite time, though the actual deceleration peaks are quite short (some 10^{-5} sec.) while the natural period of the mass-spring system made up by the lid and the screws is much longer (some 10^{-3} sec.). Therefore, the effect of those peaks on the screws is greatly lessened.

The application field of this method is limited to light packagings and to packagings provided with shock absorbing device. For heavy packagings without shock absorbing device, the deceleration found by this method is much too low to be consistent with the deceleration encountered during the test.

For these latter packagings, which account for the most problems, we use a method which takes into account the transient character of the shock and the elastic and plastic properties of the material. A closing device with screws may be shown schematically by the following figure,

where

m = plug mass divided by number of screws,

x = abscissa of screw head,

u = abscissa of a point rigidly bound to the anchorage of the screw and such that at the starting moment, selected just as the shock begins, $x = u$,

F = binding strength of the screw.

The basic relation of dynamics applied to the mass m allows us to write:

$$\frac{m}{dt^2} \frac{d^2 x}{dt^2} = -F + mg \quad (11)$$

Disregarding the effect of gravity and introducing the stress-strain relation of the material used, it is easy to check that this equation can be written in the following form:

$$l_0 \frac{d^2 \epsilon}{dt^2} + \frac{S}{m} \sigma(\epsilon) = - \frac{d^2 u}{dt^2} \quad (12)$$

where

l_0 = free length of screw (length which can be deformed),

S = cross-section of screw,

ϵ = relative elongation,

$\sigma(\epsilon)$ = stress-strain relation of the material used.

This equation with x versus t is the one to be solved.

The solution $\mathcal{E}(t)$ depends finally:

- on the parameters free length l_0 and relation $\frac{S}{M}$,
- on the shape of the stress-strain curve of steel,
- on deceleration of the anchorage point of the screw.

Because of the term $\sigma(\mathcal{E})$, the equation is not linear. Therefore, an analog computer was used to solve this equation, and to study the effect of previously mentioned parameters. Although the study is not completed, the prominent part played by deceleration on the anchorage point may be underlined.

TESTING ON COMPONENTS

Drop test

As will be seen in Part 3, the general behaviour of a packaging during a drop test may be determined by means of scale models. However, when the effects of impact are localized, it may be interesting to study the behaviour during impact of a full size packaging component (lead, thermal insulation, shock absorber, screws, etc.).

Without losing sight of the interesting aspects of the study of lead deformation or the adjusting of shock absorber, we turned our attention to the impact behaviour of the screws used to hold a packaging lid in position.

For this purpose, we have built a testing device reproducing the dynamic loads to which the screws of the closing devices are generally subjected. Our first concern has been to observe the dispersion of permanent elongations of screws tested in seemingly identical conditions. We have also taken advantage of this series of tests to control the similitude laws by reproducing the testing device to $\frac{1}{2}$ scale. Without going into the details, the results of the tests have been as follows: the distribution of the elongations obtained with the first device follows fairly a Gaussian, whereas the distribution obtained with the $\frac{1}{2}$ scaled device divided into two families. This last behaviour seemed abnormal. Moreover, the comparatively large range of dispersion meant that each kind of test must be performed a great number of times, which made the continuation of the study impossible, with the means we had at our disposal. The search for the causes of these anomalies led us to think that they were due to comparatively small variations of the impact angle. Since then, the addition of a guiding system during the greater part of the drop reduced appreciably the dispersion, and we are going to proceed with this experimental study.

Thermal test

As a whole, the heat transfer phenomena are well known. Nevertheless, some points may require particular studies, e.g. the effect of fins or the motion of the vaporization face inside a thermal protection including water. The tests on components are well adjusted to the experimental study of this kind of problem.

TESTS ON SCALE MODELS

The similitude laws are well known. If $1/S$ is the scaling factor of a scale model compared with the prototype, for a certain dropping height, the parameters such as deformation, shock duration, acceleration will be in the following ratio:

	Model	Prototype
Deformation	d	sd
Acceleration	δ	δ/S
Shock duration	t	st

The method can almost be called ideal. The obvious advantage is the savings which result from the fact that the mass of the model is divided by S^3 . Consequently, the number of tests can be much higher than on a prototype, and by changing the impact angle, it is possible to have a rather complete idea of the resistance offered by a packaging. The weight reduction also permits an easier handling: the tests can be done with a smaller number of operators.

However, when applying this procedure, some difficulties are encountered:

- in the construction of the model,
- in the carrying out of the test.

The development of the model requires theoretically the reproducing to scale of:

- plate thickness,
- weld-seams,
- clearance between lid and body,
- piping.

For commercially available screws, in particular, there is no exact similarity between one screw diameter and another. Thread, pitch and diameter do not vary in the same ratios. From the foregoing, the result is that the exact reproduction of all the characteristics of a prototype requires a great deal of care when using quite large scaling factors. So the savings on raw materials are more or less counterbalanced by the cost of practical application.

In free drop test, below a certain weight, it becomes difficult to guarantee the incidence of the angle of drop, particularly if the tests are carried out in the open. Because of this, the similitude of the deformation can be obscured. This was the case with the previously mentioned screw tests. It may then be necessary to guide the models during a part of their travel, what requires more elaborate equipment.

We think that the experimental study of the behaviour of a packaging during drop test may be roughed out by the use of a certain number of models on a rather small scale and more thoroughly studied with a smaller number of models on a greater scale.

To this end, we are presently preparing a series of 9 m drop tests that will include:

- testing of five 1/8 scale models of a 10 ton packaging prototype with different angles of drop,
- testing of three 1/5 scale models of the same prototype with the most destructive angles of drop following the previous tests,
- testing of the 10 ton prototype.

All these tests will include deceleration measurements.

It seems interesting to use the scale model method for other purposes than the simple statement of the deformation suffered by the model.

The simplicity of the similitude law relating acceleration and time leads us to consider the scale model as a "physical operator" giving the time-history of the deceleration suffered by the prototype during impact. A change in scale of amplitudes and times allows us to change from the model to the prototype.

So, in the analog computer solution of the problem of screw behaviour, the deceleration - $\frac{d^2 u}{dt^2}$ at the level of the anchorage point may be put into the computer starting from a signal recorded during a real test on a model.

However, this way of dealing with the data gives rise to problems of adaptation of the various materials, and a certain specialization of the personnel is required.

CONCLUSION

In order to establish that the packaging meets the regulation requirements, the designer is induced on the one hand to work out "patterns" giving the loads: stress, temperature, etc. to which the different components of a packaging are submitted, and on the other hand, to analyse the reactions of these components to these loads.

Taking into account the increasingly high performances required from the packagings, these become more and more complicated and make use of less and less ordinary materials. The problems to be solved require the use of advanced techniques involving special skills in very specialized fields. Such skills are necessarily limited and therefore progress is slow.

Progress should be much greater if the engineers of different countries working at these problems were better informed of each others work. Duplication of studies, as those appearing too clearly on each international meeting, would then be avoided.

A typical example of this lack of coordination is found in the past: to determine the thermal test resistance, computer codes are generally used. Each country, not to say each research department, has worked out its own "code". All these codes are no doubt of same value and probably give results that are not much different - though this has to be confirmed - but it would have been better if a general conference had led to the acceptance by everyone of a unique code, which would be used in all the agreements using this "proof" of compliance.

Transport regulations of radioactive materials had the good fortune to come into existence in an international sphere and to be common to all countries. This is a great advantage, making international transport easier. It would be good if this international cooperation were to be extended to the field of analysis of the compliance of packagings to regulations.

Without trying to establish "regulation standards" for the IAEA Recommendations, or even a coordination of the basic studies concerning safety, a better communication among the designers of all countries is to be desired.

This exchange of information could include:

- (1) Exchange of physical laws applying to the phenomena under consideration, e.g. impact strength, release of gaseous fission products from fuel elements whose casings are not leakproof, neutron shieldings, etc.
- (2) Exchange of "standard codes", computer codes, etc. capable of being used in the safety analyses of the agreements.

REVIEW OF EXPERIMENTAL STUDIES ON THE FISSLE
FUEL SHIPPING CASK IN JAPAN

S. Aoki

ABSTRACT

In order to obtain the data necessary to design spent fuel shipping cask, some experimental studies have been performed in Japan. The Research Subcommittee for Design of Spent Fuel Shipping Cask in the Japan Society of Mechanical Engineers JSME performed the experimental study on the heat removal from lead shielding cask for the various decay heat generation rate, which was simulated by means of electric heaters loaded in the cask under steady and hypothetical accident condition. The temperature distributions in the case of "with full coolant", "with half coolant" and "without coolant" were measured and the temperature change of cask and fuel assemblies were obtained during loss of coolant also. The experimental studies on the drop impact were carried out by JSME since 1968. Assuming that the prototype cask for light-water reactor would be of horizontal cylinder type weighing about 80 ton, the 1/16, 1/8, 1/4, 1/2.5 and 1/2 scale models were fabricated and tested. In these tests deceleration at the moment of impact were measured with accelerometers and deformation of cast were also observed. The relation between drop height and deformation was analysed.

From the fire test by the Fire Institute of Japan, the coefficients of heat transmission on the surface of cask and between steel shell and lead shielding were obtained.

The over-the-road tests of a shipping container for PWR type new fuels conducted by Mitsubishi Atomic Power Industries, Inc. (MAPI) gave some valuable data for the compatibility with road condition. The drop tests of PWR type new fuel container were also performed by MAPI, that was presented in this symposium by itself.

INTRODUCTION

Nowadays in Japan the three light-water cooled nuclear power stations are in operation and more than ten in process of application or construction. According to the programme, the electric output from power station will come to about 16,000 MW in 1975 and 60,000 MW in 1985. This means that the amount of spent fuel is estimated to run into about 60, 300 and 500 t/y after 5, 10 and 15 years respectively. On the other hand the reprocessing plant will be completed in 1973. therefore the transportation of a large amount of spent fuels from light-water reactors will start at the same time. In the case of Japan all nuclear power stations are located on the seashore and consequently it seems to be most convenient to transport by ship directly from the pier of station site to the nearest harbour to the reprocessing plant and on road only in the short distance between them.

Providing for such situation, the Japanese National Standard, "the Safety Criterion for Shipping Container of Nuclear Fissile Fuel" based on the IAEA Regulations(1967) was enacted in 1968. In this standard the shipping casks were classified as follows:

- L type is equal to A type packaging for low specific activity materials in IAEA Regulations.

- A type is equal to A type defined by IAEA.

- B type is equal to B type defined by IAEA.

- H type corresponds to B type for large radioactive source in IAEA Regulations and is very similar to B(M) type proposed in the third revised draft of IAEA Regulation in 1970.

When the revision of IAEA Regulation will be finished, the Japanese National Standard should be revised inevitably too.

Apart from the administrative affair, we have no experience in the test and no design data for such large cask. As a matter of course it is very difficult to obtain the design data from foreign countries except from few published papers. In view of such situation, we decided to perform the tests prescribed in the IAEA Regulations in order to collect the mechanical design data for shipping cask and also to investigate the reasonability of test methods. In 1967 the Japan Society of Mechanical Engineers (JSME) established the research Subcommittee for design of Spent Fuel Shipping Cask to carry out a series of tests supported by the Atomic Energy Bureau in the Agency of Science and Technology of Japan. The test series consisted of following themes:

- (1) An Experimental Study on the Heat Removal from Spent Fuel Shipping Cask (1967)
- (2) An Experimental Study on the Drop Impact of Spent Fuel Shipping Cask (I) (1968)
- (3) An Experimental Study on the Drop Impact of Spent Fuel Shipping Cask (II) (1969)
- (4) An Experimental Study on the Drop Impact of Spent Fuel Shipping Cask (III) (1970)

Separately from these programmes by JSME, some experimental and analytical studies on shipping cask have been done as follows:

- (a) Fire Test of Spent Fuel Shipping Cask by the Fire Institute,
  - (b) Over-the-road Test of Shipping Container for PWR type New Fuel by Mitsubishi Atomic Power Industries, Inc.(MAPI),
  - (c) Drop Test of Shipping Container for PWR type New Fuel by MAPI.
- Here I would like to introduce these works briefly.

AN EXPERIMENTAL STUDY ON THE HEAT REMOVAL FROM SPENT FUEL SHIPPING CASK IN 1967

The Purpose of Experiment and Experimental Condition

The purpose of this experiment was to investigate the temperature change of lead-steel cask under loss of coolant accident.

The experiments were carried out for the various heat generation rates simulated to the decay heat of spent fuels. The experimental condition were, (1) steady condition (a) with full coolant, (b) with half coolant and (c) without coolant, and (2) transient condition with gradually lost coolant (initially with full coolant).

Cask Model and Simulated Fuel Assembly

The cask model designed and fabricated had inside diameter 500 mm; inside length 1,200 mm; lead shield thickness 190 mm; Maximum Pressure 7 kg/cm²; Maximum Temperature 165°C; dry Weight 8 ton with longitudinal fins. Fig.1 shows the cask model.

The simulated fuel assemblies were made similar to those of the actual fuel except the length. The assembly was composed of 96 sheathed heaters (10 mm ϕ) and 4 structural dummy rods arranged in 10 x 10 square array with the dummy rods in every corners. In a cask model, four assemblies were installed. Fig.2 shows the simulated fuel assemblies.

Fig. 1 Cask Model for Heat Removal Test

Fig. 2 Simulated Fuel Assemblies

Considering the fact that the heat generation rate of spent fuel varied widely depending on the burn-up and cooling period, the rate was estimated to be 0.5 to 1.0 kW per assembly, if its length was equal to that of the simulated assembly. In order to cover this range, the test assembly was designed to generate 0.2 to 1.5 kW per assembly. At the several points the temperature were measured by sheathed Alumel - Chromel thermocouples.

Experiment

Temperature Measurement for Steady Condition

The temperature change and maximum temperature of the cask with full coolant, with half coolant and without coolant, were measured for the various heat generation rates, as shown in Fig. 3.

Temperature Measurement for Transient Condition

First the cask filled with coolant was closed and the electric input was switched in. After the temperature and pressure became steady state, the valve at the bottom were opened quickly. Then the coolant flew out and lost. The temperature rise at several points were recorded. All these experiments were conducted with model cask laid horizontally assuming overturning accident.

Fig.4 shows the temperature change during loss of coolant.

Experimental Results

(1) The steel-lead combination has fairly high thermal resistance and particularly between the outer-shell and lead.

(2) The maximum temperature of cask and fuel assemblies in the case of half coolant are fairly lower than that in the case of no coolant. As shown in Fig.3. In the former case the coolant boils and generated steam cools fuel assemblies and then condensates on the inner surface of cask. Therefore, the heat transmission from fuel assemblies to cask wall may be promoted very much by boiling and condensation. On the contrary, if there is no coolant, both convection of air and steam and radiation play a main role for heat transmission, and it seems to be insufficient to transport the generated heat. For such case some special structure to promote heat conduction should be necessary.

(3) The record of temperature change of cask and fuel assemblies has been obtained at loss of coolant lost. As shown in Fig.4, the temperature rise is not so rapid after having finished loss of coolant. It seems that the remained water and steam can remove generated heat considerably.

Fig. 3 Maximum Temperature for "With Full Coolant" and "With Half Coolant"

Fig. 4 Temperature Change at " Loss Of Coolant "

AN EXPERIMENTAL STUDY ON THE DROP IMPACT OF SPENT FUEL SHIPPING CASK SINCE 1968

Drop Test Facilities

Lifting and Falling Device

The drop test facilities were constructed in the site of the Government Mechanical Laboratory in 1968. The main design concept was came from the IAEA Regulation as follows;

- (1) the capacity of lifting device : 10 ton
- (2) the maximum fall height : 9 m
- (3) the clamping device was designed to operate mechanically and also electro-magnetically.

Target

The target was made of a reinforced concrete block with a steel plate on its upper surface weighing about 210 t (6 m x 6 m x 2.5 m). The steel plate on the surface had the thickness 12.5 mm in 1968, but 50 mm after 1969.

Scale Model for Test

In reference to the present situation in Japan as mentioned above, we imagined as the prototype a horizontal-type cylindrical cask, weighing about 80 t and having steel-shells and lead shielding. The scale model was simplified in shape so that the measurement of deformation and the analysis of motion at drop impact might be made easier. The reduced scales were chosen as 1/16 (about 20 kg), 1/8.5 (114 kg) in 1968, 1/8 (about 0.15 t), 1/4 (about 1.25 t), 1/2.5 (about 5 t) and 1/2 (10 t) in 1970, in order to confirm whether the similarity existed between the scale model and the prototype or not. In addition to these models, lead spheres and cylinders weighing 20 kg were prepared as the test pieces for obtaining data of lead at drop impact in 1968, and the various kinds of shock-absorber for 1/4 scale model were fabricated for testing the shock-absorber performance in 1970.

Experimental Results

According to the IAEA Regulations, a shipping cask for large radioactive source must be able to pass the free drop test from 9 m onto a rigid target. But in 1968 we accomplished the horizontal drop test from different height, namely 3, 6 and 9 m, to examine the effect on deformation of fall height and weight. The measurement of deceleration was done by means of accelerometers mounted in the middle of

the top surface of model in horizontal position. A high-speed cinematography was used to record and to analyse the motion of model during drop, at collision and during rebound. Deformation was measured by means of contact meter and other suitable measuring tools.

Table 1. The maximum Deceleration

Lead Cylinder 1043 x 208	No.8	3 m drop	347 g	strain gage
	No.9	6 "	874 g	"
	No.10	9 "	defected	
1/16 Scale Model 100 O.D x 325		3 m drop	860 g	piezo type
		6 "	1,842 g	"
		9 "	2,170 g	"
1/8 Scale Model 185.7 O.D. x 610		3 m drop	760 g	"
		3	840 g	strain gage
		6	977 g	piezo type
		9	1,238 g	"

Deceleration

The maximum deceleration in the case of almost flat collision shown in Table 1. The deceleration and the rebound behavior were strongly affected by the attitude of the model at the moment of impact

Motion of Test Model

The motion of model during test was recorded by high-speed 16 mm motion pictures (film speed 3,000 - 3,500 pps) and was analysed by a motion picture analyser. An example for Model No.1, picked up from the films, is given in Fig.5. From these analysis it was made clear that the duration time of pulse ranged from 0.3 to 3 msec.

Deformation

Fig.6 shows the deformation of model No.1 (1/8.5 scale). The theoretical analysis was applicable to reveal the relation between fall height and resulted deformation, the detail of that was presented in the paper IAEA/SM-147/1 submitted in the Seminar On Test Requirement For Packaging For The Transport Of Radioactive Materials on Feb. in 1971.

Experiment in 1969

In 1969 the experimental study with 1/8, 1/4, 1/2.5 and 1/2 scale models was performed. The test was accomplished only in the case of horizontal drop from 9 m, but both without shock-absorber and with it. The result of experiment are separately reported in this symposium.

(a)

(b)

Fig. 5 Motion of Models at Drop Impact
(a) 1/8.5 Scale Model, (b) 1/16 Scale Model

Fig. 6 Deformation of 1/8.5 Scale Model

Experiment in 1970

In 1970 the corner drop tests for 1/4, 1/2.5 and 1/2 scale models and the performance tests of shock-absorber for 1/4 scale model were carried out. The two different sized shock-absorber for each of the four different type were disgned and fabricated. Horizontal drop and corner drop both for lid side and for bottom side were done. The fall height was always 9 m. The deceleration and deformation at impact were measured and the motion of models was recorded.

Test Programme in Future

In 1971 the pucture test will be performed using the 1/2.5 and 1/4 scale models.

FIRE TEST OF SPENT FUEL SHIPPING CASK

The fire test has been carried out by the Fire Institute of Japan since several years ago and some of them have been reported in the 1st and 2nd International Symposium on Packaging and Transportation of Radioactive Materials by T. Moriya. Now the heating equipment for large cask has been completed and the test are being carried out to investigate the fire resistance characteristics of spent fuel shipping cask and to obtain the fundamental data for its safety evaluation. In this test series the four kinds of cask model have been used, which have the outer-diameter 800, 400, 400, 200 mm, inner-diameter 500, 74, 254, 126 mm, lead thickness 125, 150, 60, 30 mm, length 800, 600, 600, 400 mm, and weight 3,000, 636, 530, 85 kg respectively. These cask models are placed and heated in a large furnace (1.6x1.6x1.6 m) at 800°C during 30 minutes. The temperature in the furnace and that of outer-shell, lead-shielding and inner-shell of cask model are measured by means of Alumel-Chromel thermocouples at twenty or twenty-four points.

Based on the experimental results, the coefficients of heat transmission at the outer surface of cask and between steel and lead are obtained. And, using these results, the thermal behavior of large shipping cask for practical use will be analysed with analog-computer theoretically.

OVER-THE-ROAD TESTS OF A SHIPPING CONTAINER FOR PWR TYPE NEW FUELS

A long distance over-the-road tests of shipping container for

PWR type new fuel assemblies were conducted by Mitsubishi Atomic Power Industries, Inc. (MAPI), in order to examine the compatibility of PWR type fuel shipping container with road condition. The container tested consisted basically of a split type cylindrical steel shell reinforced with circumferential ribs and of a fuel assembly receptacle being capable of accommodating two PWR type fuel assemblies of 10 ft long. In order to protect the assemblies from shocks and vibrations encountered during transportation and handling, the fuel assembly receptacle was suspended from the shell by means of shock-absorber. The container with two sets of dummy fuels, weighing about 2.5 t, was loaded and tied on an 8 t capacity truck. The tests were carried out on the route from the suburbs of Tokyo to a nuclear power station in Tsuruga, some 350 miles west of Tokyo. The road condition ranged from modern super highway to unpaved local rough roads. The vehicle speed was up to 60 mph.

From this test, the spectra of accelerations induced in the fuel shipping container due to shocks and vibrations encountered under the test conditions over the variety of road phase were obtained. It was the most interesting result that the maximum shocks at the container shell were observed at the surface cracks and holes on the paved roads rather than on the unpaved roads, whereas the maximum deflection (derived from the acceleration) of the fuel receptacle were observed on the unpaved rough roads.

DROP TESTS OF SHIPPING CONTAINER FOR PWR TYPE NEW FUEL

This test also was accomplished by MAPI. The test procedure and test results are presented precisely in the paper titled "DROP TESTS OF SHIPPING CONTAINER FOR PWR NEW FUEL" in this symposium.

ACKNOWLEDGMENT

The author wishes to express his gratitude to the Atomic Energy Bureau and the Japan Society of Mechanical Engineers, which supported the series of experiments. He also wishes to express his thanks to the members of the Research Subcommittee For Design Of Spent Fuel Shipping Cask in JSME, who conducted these experiments for a long time, and also to the Fire Institute and MAPI, who offered me their experimental informations.

EVALUATION OF METAL SEAL CRITERIA
FOR SPENT FUEL CASK CLOSURES

R. E. Latham*

R. T. Brown**

ABSTRACT

Reliability of the gasket used for sealing the end closure of fuel casks and mating appurtenances of similar fluid containing piping is of prime importance in overall system evaluation. The degree of reliability will be dependent upon the gaskets ability to respond to relative displacements of the mating seal surfaces during both operational pressure and thermal cycles and superimposed mechanical shock loads that occur during maximum credible incidents.

Basic sealing parameters for metal-to-metal gasketed joints which will result in leakage rates less than 1×10^{-6} cc/sec. helium are summarized. The mode of achieving initial minimum seal interface loading to meet this leak rate criteria is evaluated with regard to the material properties of mating connectors and gaskets.

The effects of seal geometry and mode of interface loading on the reusability of seals and seal surfaces to minimize replacement and repair is discussed. Economic considerations of manufacturing tolerances for both initial installation and rework operations are reviewed. Coating and plating effects on initial sealing, reusability and reliability are considered. Overall sealing criteria are summarized for various seal types providing a guideline for designers in gasket selection.

* Chief Engineer, Gray Tool Company, Houston, Texas.

** Supervisor, General Products Dept., Engineering Div.,
Gray Tool Company, Houston, Texas.

INTRODUCTION

Cask Applications

Transportation of radioactive materials is accomplished by the use of shielded containers to protect against contamination of equipment and personnel. These containers consist of an internal vessel in which the radioactive material is placed, and an outer vessel which contains the shielding material¹. The internal vessel is usually cooled by jacketing.

During normal operating conditions, the radioactive materials are placed into the internal vessel and a lead plug is placed between this material and the end closure. A closure blind is then made up to the internal vessel, usually remotely, and the shielded cover is put in place. Pressures and temperatures remain near ambient as long as cooling system is functioning. If the cooling system fails, however, the radioactive material usually begins to heat up causing an increase in both pressure and temperature. In liquid metal fuel cells, 1,000 psig and 1100°-1200°F. can be reached in a matter of minutes. Not only must the internal closure seal maintain its integrity during this thermal and pressure shock loading, but its reliability during the cool down phase required to bring the system back to equilibrium.

Although failure of the cooling system can and does occur because of mechanical equipment failure, the prime credible incident occurs during transportation of these casks. Whether the transporting is being done by truck or rail, there exists the possibility of an accident which can subject the cask to a high impact load. The problem of designing a closure seal that can withstand the loads imposed by a 5,000 pound cask being thrown from a truck traveling 60 miles per hour as it rolls up

an embankment or into a ditch, is compounded when we must consider that this incident will, in all probability, cause the cooling system to fail also.

Even if no catastrophic events occur, the closure seal must be opened and its contents removed a significant number of times during the lifetime of the cask. One estimate places the number of full opening and closing operations at 500-600 times. Therefore the seal should be easily adaptable to assembly operations, keeping in mind that remote handling is at best, difficult and time consuming.

Cask Closure Criteria

Considering the applications for which the containment casks are to be used, we can establish certain criteria for the closure seals.

1. Seals will be required to seal relatively low pressures but must be capable of withstanding a fairly rapid pressurization and heat up without leaking or plastically deforming such that they will not seal upon depressurization. This requires that the seal be both self energized and pressure energized and of a material and design which will impose maximum residual elasticity in the gasket. This will insure maintenance of minimum gasket loading during all phases of operation.
2. Seals must withstand severe axial and radial impact loads without deformation or loss in gasket loading. This will require a seal which is insensitive to axial unloading even to the extent of some separation of the connector faces. The seal must be of sufficient cross section to be essentially self supporting against radial or shear loading.
3. Seals must be reasonably adaptable to quick assembly and remote handling. This will require a self aligning seal or

a seal which is confined within the connector.

4. Seals must be able to achieve a maximum number of make-break operations without damage to the seal surfaces on the cask containment vessel. This requires a seal geometry which can achieve reliable sealing with minimum gasket interface loading and a seal surface readily accessible for rework if required.

Economics and availability are relatively minor considerations of the overall cost of a cask containment system, especially when usage of an improper seal design creates the consequences of radioactive contamination.

BASIC SEALING PARAMETERS

Sealing Interface Criteria

The interface geometry between two metal sealing surfaces is the major factor influencing the leakage across the seal. Several conditions contribute to this factor.

1. Initial surface roughness of the two sealing surfaces.
2. Minimum contact load required to obtain superficial plastic deformation at the interface and the residual load required to maintain the resulting geometry.

Surface Finish

The resultant surface finish required to seal a particular fluid may be approximated from the equation for laminar flow between parallel flat plates and D'Arcy's equation for viscous flow through a permeable media².

The required surface finishes (in microinches arithmetic average (AA)) may be calculated for various media using this simplified method to obtain a type of sealability index. This is a relative value since the above procedure assumes uniform surface finishes and neglects compressibility effects. Figure 1 graphically depicts

Fig. 1. Surface Finish Requirements versus Leak Rates for Various Media.

theoretically required surface finishes for a 1,000 psi pressure differential and a 6 inch seal diameter of 1/2 inch width for a 1×10^{-3} cc/sec. leak rate, which corresponds to a bubbletight joint, and a 1×10^{-6} cc/sec. leak rate which is generally considered a maximum leak rate for nuclear containment systems.

We are normally concerned with obtaining much lower leak rates than can be economically attained by surface finish control alone. It is necessary to look at how this controlling characteristic can be varied, if needed, since all other sealing parameters remain constant for a given application.

It is apparent that the resultant surface finish can be created by either superfinishing techniques³, or by mechanical loading of the seal surfaces to obtain plastic flow of the mating parts whereby the residual effective surface finish is sufficient to maintain required leak rates.

Repeatable leak rates not exceeding 1×10^{-6} std. cc/sec. of helium have been obtained using a combination of superfinished seal surfaces and average mechanical seal loading stresses below the yield of the seal material tested. However, based on past experimental work, it may be concluded that control of leak rates by superfinishing alone is not the practical answer to seal design⁴.

Initial Interface Loading

Since plastic deformation of one or both of the sealing surfaces is required for a reliable leaktight joint, the need for defining the magnitude and type of loading needed to obtain this deformation becomes apparent.

There are two basic ways to obtain plastic flow: The first by direct bearing or normal loading of the seal surfaces and, second, by a combination of normal and shearing loading. The

first type of gasket loading has been investigated extensively, and minimum yield values and minimum loading constants for conventional gasket materials and geometries have been established and widely published, similar to those shown in Table UA-49.1⁵. Normal seal loading requires contact bearing stresses over the entire gasket area from two to six times the tensile yield strength of the material used, depending upon the surface finish of the original parts and the degree to which the materials are susceptible to work hardening^{3,6}.

A comprehensive description of the five modes of plastic deformation for normal loading is summarized as follows³.

1. An initial yielding of the peaks of the surface asperities occurs at a load which produces average mean contact stresses of only a fraction of the yield strength of the seal material. The remaining deformation is essentially elastic.
2. Further yielding of the individual asperities occurs to the point that additional deformation begins to interfere with adjacent asperities, at loading producing average mean contact stresses equal to the yield strength of the softer material.
3. Additional loading initiates bulk flow of the softer material, resulting in a rapid increase in resistance to further deformation as the contact area increases.
4. Further loading causes bulk flow in the softer material and maximum strain hardening due to the resistance to movement imposed by containment in the harder material.
5. Additional loading will cause bulk flow in the harder material which is the final failure mode.

It should be noted that from observations made during several test programs, loadings of mode 3 or higher were required to obtain reliable sealing^{3,7}. Also, with mode 3 or higher loadings, a

shearing type failure occurred at the outer and inner edges of the gasket being tested. These shearing plastic deformations resulted in a more uniform and complete seal area.

It is the shearing type of deformation that has had very little correlation to test results. However, from the previously discussed work done on a few seal materials, it can be initially concluded that this combination shearing and normal loading requires only one to three times the tensile yield strength of the material used as the seating bearing load.

Loading levels equal to and less than the bottom range of values have been observed during the design and the manufacture of the GRAYLOC® Seal Ring, a pressure energized bore seal used in vessel closures and piping connectors². This gasket obtains its seal by axially drawing tapered seats together, wedging and flexing the sealing surfaces inwardly, thereby causing a shearing movement normal to the load on the seal surfaces. From comparison of measured makeup loadings for this type of gasket over the past ten years, it is apparent that the gasket loading constants for the conventional type loading are too restrictive for this application. A new parameter (the method of loading the sealing surfaces) must therefore be considered in establishing design seal criteria.

Residual Interface Loading

The next sealing criterion to be investigated is that of the minimum residual loading required to maintain sealability under operating conditions, gasket unloading due to joint separation, and seal surface rotation. Conventional gasket guidelines require unit loading up to 6-1/2 times the pressure as a minimum, however,

---

*GRAYLOC® is a registered trademark owned by Gray Tool Company.

in actual testing, loads as low as 50 per cent and as high as five times the initial unit loading were observed to be required to maintain sealability^{7,8}. Factors influencing this value are the seal material, the initial surface finish, and the degree and mode of deformation of the seal surfaces. For seal surface finishes in the range of 16 AA to 63 AA and plastic deformation of the seal surface of mode 3 or greater, the required residual seal loading has been found to be only that required to prevent unloading of the seal surfaces⁷. This characteristic has been utilized by the manufacturers and users of pressure energized seals to reduce bolting and connector requirements. If severe cyclic movement occurs between the seal surfaces resulting from either pressure, temperature, or external mechanical loading, it is usually necessary to use soft gasket materials to prevent seal surface fretting. Seal interface loading high enough to obtain plastic flow of this softer material will also be required during all phases of joint loading if leakage is to be prevented.

Interface loading may be maintained by the elastic properties of the gasket itself ("springback") and/or a pressure multiplier due to gasket geometry and location. It is recommended that higher yield strength materials be used where possible for the gasket, in order to provide a greater range of elastic movement. Platings or coatings may be required for lubrication during initial makeup to prevent seal surface microgalling.

Effect of Components and Operating Conditions

The relative movement of the mating flanges of a gasketed joint has a significant influence on the action of the gasket. As an example, consider the case of the closure on a vessel. Typical configurations for cask closures are shown in Figures 2 and 3. The dimensions of the gasket and the seating surfaces for a pressure

Fig. 2. Schematic of Cask Closure for Semi-remote Handling.

Fig. 3. Photograph of 3" and 4-1/2" I.D. Threaded Cap Containment Vessels.

energizing gasket should be established to insure that the joint is tight without significant pressure application. This is necessary in order to insure a seal when the vessel is first placed in service. However, as pressure is applied the movement of the seating surfaces requires that the gasket be deformed if the seal is to be maintained. If the operational conditions are cyclic, the deformation of the gasket must remain essentially elastic during the variations in the service conditions.

The deformation of a gasket in an assembly of identical flanges, rather than the nonsymmetric combination illustrated, would result in only translational gasket deformations. Because of the symmetric characteristics of the flange combinations, relative radial movements would not occur and therefore the gasket would not be forced to rotate².

The influence of temperature on the elastic action of the gasket should not be neglected. The influence of differing flange and gasket materials, with different coefficients of thermal expansion, must be considered along with the effect of thermal transients which may heat or cool components of the assembly at different rates.

If the deformation of the gasket caused by the action of the cover, the flange, any differential thermal movements, and the pressure acting on the gasket itself are sufficient to cause large plastic strains in the gasket, its ability to seal at lower pressures will be lost, as well as the reusability of the assembly. Therefore, it is essential in designing a gasket for a nonsymmetric flange combination that recognition be given to the action of each of the components and a sufficient flexibility designed into the gasket to maintain essentially elastic action.

BASIC SEAL CONFIGURATIONS

A brief review of what is available, both commercially and as special designs for seals in piping systems, sharply points out that sealing problems have been prevalent for a long time. Metal seals can be categorized as those that seal on the face of the connector and those that seal on the bore. Seals in either of these two categories may achieve sealing by essentially crushing or yielding the gasket; by elastically loading a flexible gasket utilizing a soft coating or plating on the seal surface; or by a combination of the above. Seals may or may not be pressure energized as dictated by their geometry and location in the connector⁹. Figures 4 and 5 show some of the more accepted seal designs in use today.

The crush type face seals require either high preloading, such as the flat and ring joint gaskets, or close manufacturing tolerances, such as the solid and tubular "O" rings. Both designs dictate the use of relatively heavy, rigid connector systems to impose and maintain seal loading when subjected to internal pressure and/or temperature fluctuations. Because of high bolt loading and high localized stresses in the connector, these seals do not lend themselves to high temperature applications, as the materials will usually creep allowing axial unloading of the seal surfaces.

The concept of the pressure energized versions of the face seals is to offer a higher degree of reliability by compensating for the reduction in the seal loading caused by creep and bolt elongation. The gasket loading under pressure is increased by providing a larger gasket area subjected to the internal pressure than the area over which the sealing load acts. The effect of pressure energizing is dependent upon the inherent stiffness of the gasket which, in itself, is greatly affected by the diameter of the seal.

FACE SEALS

CRUSH TYPE	FLEXIBLE TYPE
 FLAT METAL	 DELTA SEAL *
 RING JOINT	 CONOSEAL *
 "O" RING	 "V"- SEAL *
 TUBULAR "O" RING *	 "K"- SEAL *
 FOUR POINT	 BOBBIN *
 INDENTED SEAL	 BX SERIES RING JOINT *

* PRESSURE ENERGIZED SEAL

Fig. 4. Typical Face Seal Configurations.

BORE SEALS

CRUSH TYPE	FLEXIBLE TYPE
TAPERED METAL SLEEVE	DOUBLE CONE *
LENS RING *	MODIFIED * BRIDGEMAN
LAPPED TAPERED JOINT	GRAYLOC SEAL *
CONE *	LAURENT *
	WAVE RING *

* PRESSURE ENERGIZED SEAL

Fig. 5. Typical Bore Seal Configurations.

For liquid metal sealing applications which normally concern relatively low pressures, the pressure energizing advantages are insignificant except to compensate for sudden pressure surges. Therefore, all crush type face sealing gaskets can be expected to respond essentially the same for nuclear fuel handling applications.

The flexible face seals differ from the crush seals in that their geometry consists of a relatively long gasket with a very small seal contact area. Since the gasket is also relatively thin, it is usually confined to prevent nonuniform gasket distortion during makeup which would prevent uniform seal interface loading. These gaskets are usually coated with a soft metallic coating to prevent damage to the connector, as seal unit contact loading is extremely high. These seals perform best when manufactured from high yield strength materials which impose maximum residual seal loading. Many of these materials are age hardened or precipitation hardened to resist creep and prevent reduction of contact loading at high temperatures.

Because of the relatively small cross section of these gaskets, the initial seating loads are low allowing the use of a smaller connector and less bolt preload. These gaskets are, however, extremely sensitive to misalignment, and care must be taken to make up the union square and uniformly. If this is not done, the gasket will become cocked in its seat and either experience a localized failure of the seal surface or the gasket itself. Therefore, some external means must be provided to center the union and hold faces square in the made up condition. Also in high temperature applications, the high unit seal loading can tend to promote pressure welding of the seal interfaces. This is especially true in the handling of liquid sodium which promotes mass transfer of elements¹⁰. All of the face type flexible gaskets require a rigid control of tolerances and surface finishes. When correctly handled, these seals

have performed with reasonable success.

The crush type bore seals have essentially the same characteristics as the crush type face seals - high preloading, rigid connectors and generally not suitable for high temperature applications. Most of the bore seals, with exception of the Bridgeman, do have the advantages of being self centering; however, only one geometry has been used extensively in nuclear application, including liquid sodium systems. This is the GRAYLOC seal which uses a combination of shearing and bearing loading resulting in the lowest possible combination of seal interface loading, interface permeability, and maximum residual seal elasticity. The GRAYLOC seal is especially adaptable because it utilizes a tee shaped cross section wherein the center portion provides a centering feature and a restraint against radial displacement during preloading and operational impact loads. This allows the use of thin flexible sealing lips which, when manufactured from high yield strength material, gives maximum resistance to thermal shock. It has been experimentally shown that this type of gasket can repeatedly seal with interface loading less than the yield strength of either the gasket or the mating connector². Therefore, the tendency to pressure weld in high temperature helium, sodium vapor, or liquid sodium is substantially reduced. Lubrication is necessary during makeup for low leak rate applications of low molecular weight gases because of the sliding action of the seal surfaces.

All of the bore seals require considerably more bolt up loading because of their more massive cross sections but utilize more economical manufacturing tolerancing. The flexible bore seals are readily adaptable to high temperature use because they are radially compressed during makeup with residual seal loading, a function of yield strength, and modulus of elasticity.

CONCLUSION

In summary, only flexible seals should be considered for containment of nuclear systems where leakage cannot be tolerated. Additional considerations concerning the seal geometry to be used are, of course, economy, availability, ease of seal replacement, maintenance of sealing surfaces, to include repair of same if damage occurs during handling or seal replacement. The latter is especially important since many of the flexible seal geometries utilizing small seal areas and high contact unit loading, require hard facing of the seal seats if numerous makeups are to be a part of the design requirements. This will, of course, complicate any necessary repair as most radioactive systems require remote handling. Seal materials should be as high a yield strength as seal geometry permits. For high temperature applications aged or precipitation hardened, materials will give maximum life and reliability. These heat treatable materials are available for temperatures ranging up to 1300°-1350°F. For higher temperature applications, high nickel-high cobalt or tungsten alloys will provide maximum service life but will usually require a planned program of seal replacement because of their lower yield strengths. An interface of a soft material between the gasket and seal surface is desirable to provide lubrication during assembly and act as a sacrificial barrier to prevent detrimental pressure welding. This material may be a soft metallic plating, such as nickel, silver, or gold, or a high temperature coating, such as molybdenum disulphide or graphite. Coating or plating compatibility must be evaluated with system requirements. Any plating or coating process should be halogen free to prevent contamination of the system components which are usually austenitic stainless steels.

Review of all of the sealing parameters prior to the specifying of a gasket for nuclear containment systems will aid the designer in making the proper selection.

REFERENCES

1. Directory of Shipping Containers for Radioactive Materials, United States Atomic Energy Commission, Washington, D. C. (1969).
2. R. T. Brown and W. R. Mikesell, "Application of Primary Sealing Criteria to a Self Energized Gasket", 68--WA/PVP-4, Journal of Engineering for Industry, TRANS. ASME (1969).
3. F. O. Rathbun, Jr. and R. S. White, "Superfinished Surfaces as a Means for Sealing", Conference on Design of Leak-Tight Fluid Connectors, George C. Marshall Space Flight Center, Propulsion Division (1965).
4. F. O. Rathbun, Jr. "Five Regimes of Metal to Metal Sealing," Conference on Design of Leak-Tight Separable Fluid Connectors, George C. Marshall Space Flight Center, Propulsion Division (1964).
5. ASME Boiler and Pressure Vessel Code, Section VIII, (1965) and latest Addendum.
6. Irving Roberts, "Gaskets and Bolted Joints", Pressure Vessel and Piping Design, collected papers 1927-1959, by The American Society of Mechanical Engineers (1960).
7. F. O. Rathbun, Jr. "Experimental Leakage Rate Experiments", Conference on Design of Leak-Tight Separable Fluid Connectors, George C. Marshall Space Flight Center, Propulsion Division (1964).
9. Bernard Goobich, "Helium Leak-Tight Seal; Functional Analysis and Development", Conference on Design of Leak-Tight Separable Fluid Connectors, George C. Marshall Space Flight Center, Propulsion Division (1964).
10. U. S. Atomic Energy Commission, "Mass Transfer in Sodium", Fundamental Nuclear Energy Research - 1964 (U. S. Government Printing Office, 1964) p. 146 LM-00112

RECENT DEVELOPMENTS IN LEAD SHIELDS FOR TRANSPORT
CONTAINERS IN THE UNITED KINGDOM

R. Smith

ABSTRACT

The manufacture of the earliest transport containers in the UK suffered from lack of liaison between the various interests involved and the absence of any really satisfactory regulations and testing procedures sometimes had unfortunate results. The introduction of regulations for safe transport tended to make for improved liaison but at the same time brought new problems. While the UKAEA solved the direct problems associated with the new regulations the lead industry was busy developing new techniques of shield manufacture. The use of simple cast shields was gradually being replaced by shields fabricated from rolled lead sheet and slab. Further increases in the size and activity of sources being carried introduced new problems in that the conduction of 'source heat' through this type of shield was not very efficient and further development became necessary resulting ultimately in the production of shields made up from a number of smaller cast units thus taking advantage of the best features of both techniques. The cast units provide an efficient transfer of source heat to the outer steel where it is dissipated while the fabrication technique simplifies manufacture and handling problems. The introduction of this combination technique coincided conveniently with the introduction of the 'heat sink' theory by UKAEA eliminating the need for external packaging on the largest containers. Liaison throughout has now been improved so that designs facilitate the use of the most efficient techniques. Regulations and testing have thus become matters of guidance rather than restriction. Production efficiency is now such that the rigorous testing currently applied is almost unnecessary and is seen in light of testing for approval rather than for rejection.

INTRODUCTION

In the early days of the Nuclear Age a demand arose for lead for radiation shielding. The demand came from the nuclear scientist, who understood the ability of lead to protect him from the inherent dangers of the new scientific tool he was developing, but knew little about the production metallurgy of lead. Unfortunately the little knowledge he had became the proverbial dangerous thing; he knew that lead had a relatively low melting point, and mistakenly took this to be synonymous with ease of casting.

As a result there seemed no need for specialist shielding companies. Anyone could make a lead shield and often did; with some unfortunate results. One large London hospital commissioned their plumber to make some small transport containers for nurses to carry radioactive materials around the hospital. Fortunately, by the nature of the use made of these materials, detecting devices were in use and the inefficiency of these shields soon became obvious. Consequently expert lead companies soon entered this field, quickly learning the required nuclear knowledge to match their existing abilities in the metallurgy and handling of lead.

The transport of larger quantities of radioactive materials began in a similarly haphazard way and for similar reasons. A large shielded container was seen as an important item of steelwork requiring - almost as an after-thought - some shielding made of lead. At the design stage the lead was considered only in terms of the space it would occupy and the weight it would add to the finished container. Most designers, being engineers were familiar with the first class steel fabricators and these companies were therefore invited to quote for the manufacture of shielded containers.

At this time there began, in the UK, what is sometimes cynically referred to as the 'Golden Egg Age'. Steel fabricators were being offered contracts to produce a steel shell and then to fill it with a relatively expensive metal which seemed to them quite simple to handle. For this, it appeared at the time, quite fabulous prices could be quoted and the new Nuclear Industry with seemingly unlimited finances would be happy to pay. However, the Golden Eggs soon produced chickens which equally quickly 'came home to roost'.

It soon became apparent that molten lead poured into a cold steel shell did not produce a satisfactory solid lead shield, but resulted in large voids and

widespread porosity. These unexpected problems proved very costly to the steel fabricators.

The immediate result was two-fold. Specialist lead companies were sub-contracted by the steel fabricators to undertake the lead filling. At first even this change bore little fruit. The initial design produced by the nuclear engineer was often based on the assumption that a four inch hole in the top of a large steel shell was sufficient for the introduction of several tons of molten lead. At first with all the mystery surrounding the whole subject of Atomic Energy, the lead shield maker accepted this design and tried to make the best of an almost impossible situation. In these circumstances, even the best of lead companies produced imperfect shields from time to time. Early attempts to heat the steel while the lead was being poured caused distortion of the steel.

Even the testing of shields was not as rigorous as it is today. One set of six medium sized containers crossed the Atlantic fully laden and on arrival at their destination two were found to have faulty shields and consequently to be unsafe.

Slowly but surely the lead industry began to make its voice heard. When a bad design was received for quotation, the companies spoke against it. The designer was told what was necessary to facilitate the production of a first class shield. Safety was accepted as being more important than the possibility of a lost contract - and with this acceptance it was found that the designer was only too pleased to be guided. In addition, the company which had helped him redesign one type of container often found that the next design followed the principles they had suggested so that they were in a position of some advantage over their competitors whose favoured techniques might be somewhat different.

At this point Lead Development Association, having at heart the continued and increased use of lead as a shielding material, rather than the advantage of any individual company, issued a series of publications on the use of lead for radiation shielding. The second publication in this series, 'Design and Construction of Lead Shielded Containers' outlined the problems involved in casting lead and described recommended methods for producing perfect shields either by casting or by fabrication from such semi-finished products

as sheet and slab.

The reasons for issuing this book were two-fold. First it was felt that if designers understood the problems faced by the shield manufacturer and the techniques employed by him to overcome these problems, then future designs would tend to facilitate the use of such approved techniques. Secondly it was thought advisable to ensure that the steel fabricating companies should be made aware of the problems involved in shield manufacture so that when they decided whether to produce the shield themselves or to subcontract to a lead company, at least the decision would not have been made in ignorance of what was involved.

THE EFFECT OF THE INTRODUCTION OF IAEA REGULATIONS

At this point lead shields for transport containers were generally satisfactory as far as their shielding duty was concerned. The established techniques of the chemical engineering industry, for bonding lead to steel were now being employed and developed to facilitate the transport of radioactive materials which emitted heat. Shields now provided a continuous metal path for the transmission of heat from the internal containment vessel to the outside shell from which it could be dissipated.

With the introduction of the first safety regulations based on the Recommendations of the International Atomic Energy Agency, came a number of new problems for both the shield and the steel shell to cope with. From the beginning the two requirements which caused the most trouble were the thermal requirement and the shock requirement.

To deal with the shock requirements; this problem has always seemed the least difficult as far as lead shields are concerned. Indeed, the ability to withstand shock is often related to the degree of 'unification' of the steel and the lead. A container whose shield completely fills its outer steel and for preference is completely bonded to it will be much less at risk than one in which the lead is free to move on impact.

Because of this apparent solubility of the shock problem it has been usual to try to solve the thermal problems first hoping that the solution to the shock problem would follow, either automatically or at least with minimal adaptations.

In the UK some designers looked at the new requirements which specified a test at 800°C for 30 minutes, considered the melting point of lead (327°C), and turned to steel shields as the solution to this new problem. At the same time however, they continued to design lead shielded containers for private use where it was not necessary to meet the requirements of the new regulations. This, naturally, was seen by the lead industry as an admission that the lead shielded container was acceptable as the superior type if only it could be protected from the heat generated in the fire test.

In the event the invention and development of this protection came not from the lead industry but from the UKAEA. While the lead shielding industry comprised a number of relatively small companies each with a small share in the annual production of shields, the UKAEA was a vast organisation with a very large share in the annual purchase of shielded containers. They foresaw the need for research to ensure that their existing containers would remain in use in spite of the new regulations and to ensure that they could continue to purchase lead shielded containers which offered the advantages of minimum size and weight at a reasonable cost.

Initially the department entrusted with this work was handicapped by the lack of either a metallurgist or a lead specialist, and some of the earliest attempts to design a lead shielded container which would withstand the thermal test demonstrated this handicap.

The designers soon discovered the expansion and contraction problems likely to be encountered during melting and resolidification of the lead shield. Early tests demonstrated not only that expansion of the lead on melting could build up immense pressures causing stretching of the steel and perhaps fracture of welds, but also that resolidification would normally be accompanied by the formation of a large void either within the body of the lead or between the lead and the newly enlarged steel.

The lack of metallurgical knowledge showed in an early attempt to allow for both these problems by introducing a void of appropriate size into the original lead shield in a position where the reduction in shield thickness would cause no danger. This was done by casting in a hollow cylinder having less strength than the steel shell. It was assumed that expansion on melting would crush this cylinder and take up the space it had occupied and that on

resolidification contraction of the lead would produce a new void in this same, harmless, position.

While the first half of this theory usually proved acceptable in practice, the second half was invariably proved wrong, and the new position of the resolidification void depended on such variables as the orientation of the shield during resolidification and the thermal conditions, particularly the direction of maximum cooling.

The engineers involved in this programme quickly came to realise that their best chance of success was to insulate the lead shield from the external source of heat, and once this objective had been recognised, progress was rapid.

The detailed description of this progress is not a matter for consideration in this paper and in any case it has been fully documented by F. E. Dixon^{1,2}

The result of this work, however, has been to produce a whole range of standard packages, for application to lead shielded containers of all types and sizes, by means of which the continued use of lead shields for this purpose has been assured.

In common with all modern research there has been some unexpected 'fall-out'. One development has been a series of packages which can be used with 'families' of containers. Each family is a number of close tolerance concentric cylindrical bare lead containers. By omitting the appropriate number of containers from the centre, a source of any physical size may be accommodated, while by adding additional units to the outside the required amount of shielding may be built up. Ultimately the built-up unit is placed in the appropriate protective package for transport safety.

Another interesting development is the production of 'throw-away' packages. As a result of the UKAEA programme, costs have been cut to a minimum so that with the smaller sizes of lead containers, it is often cheaper to abandon them in a far country than to accept the paperwork and transport costs of retrieving them. Certainly the image of lead as an expensive metal has been shattered at last.

DEVELOPMENTS IN MANUFACTURING METHODS

Many of the early containers used for the transport of radioactive

materials employed lead not merely for shielding but for the whole container with perhaps a few minor additions such as steel supporting straps, handles etc. These were always produced by casting and when made by experienced lead manufacturers they were perfectly satisfactory. Most small containers today are made in the same way except that some of the smallest are now produced by a cold pressing technique. This technique, originally introduced for the mass production of shielding bricks, is, however, only economically practicable where identical containers are required in large numbers.

Of more interest here is the type of container which is mainly a steel fabrication and employs lead for shielding purposes only. The various stages through which the casting processes progressed are now a matter of history and it is only of current interest to describe those techniques ultimately employed which are still in use today.

Casting techniques

All modern casting techniques are based on consideration of the metallurgical processes involved in the solidification of lead. The simplest way of describing these processes is to consider a cube shaped steel box, completely filled with liquid lead, and having a closed top.

Initially cooling will occur from all faces of the cube, and as the temperature of the lead in contact with the steel drops to 327°C so a thin layer of solid lead will form all round. At the four sides and the base, the pressure of the molten lead will keep this layer in contact with the steel; further cooling will take place and the solid layer will increase in thickness.

At the top surface however the state of affairs will be very different. When molten lead solidifies it contracts by 3.85%. Thus even with the first small solidification there will be a decrease in the volume of lead in the box. The very thin layer of solid lead formed at the top will fall with the surface of the contracting liquid lead since it has no strength at this thickness. Immediately this happens it loses contact with the steel top, and its cooling rate will be considerably reduced. At the same time convection within the liquid lead will tend to warm the top surface, so that further growth of solid will be retarded there.

For some time therefore the growth of the solid layers around the sides

and the base will be more rapid than that of the top layer. Nevertheless after a time it will become thick enough to have some strength and then it will no longer fall continuously as the level of the liquid beneath it falls. At this time, when this top layer holds its position while the liquid lead level continues to fall a void will be formed. This will be small at first, but now that there is no contact between the solid lead and the top of the liquid the effect of convection will no longer keep the solid top warm; it will cool more rapidly and gain strength. At the same time further solidification on the inner faces of the solid at the sides and base will cause continuing contraction and the liquid level will fall still further, increasing the size of the cavity.

In the simplest case, the ultimate result will be a cube of solid lead with a sizeable dent in its top surface and a sizeable cavity somewhere beneath the dent.

In practical castings there will obviously be variations on this theoretical case. Nevertheless modern shield casting methods are designed to avoid the type of situation described here.

A number of techniques are applicable and all are described in detail elsewhere.³ All are designed to prevent the formation of the top solid layer of lead. It is essential in all cases that the steel shell shall have a completely open top. This ensures that whatever filling technique is used the top surface of the lead can be kept in the liquid state until the rest of the shield is completely solid. At the same time efforts are made to ensure that the direction of maximum cooling is vertically downwards to maintain a fairly flat horizontal interface on which the growth of solid lead occurs. If this is successfully achieved, the void formation described earlier is impossible since that requires a 'box' of solid with liquid lead inside, whereas this technique produces a flat layer of solid lead with a mass of liquid on top. To maintain the horizontal interface requires bottom cooling, and side and top heating, the side heating being removed upwards from the base as solidification proceeds.

The top heating may be applied with gas torches playing on the top of the whole mass of lead required to form the shield, or it may be achieved by employing very slow pouring, so that at any time there is only about two inches of liquid lead on top of the solid lead. In this way as more lead is

added very slowly so it effectively provides the continuous top heating. This slow pouring may be continuous; or it may be intermittent, each separate pouring being sufficient to add about a six inch height to the shield and then this is allowed to solidify. With this technique where a number of 6 inch shield sections are produced successively, even if by some misfortune a void forms it cannot be of serious proportions due to the limited mass of metal from which it is formed.

Fabricated shields

Where a shielded container is of large size but relatively simple design it is often less costly, and technically more sure, to fabricate the shield from semi-finished products such as sheet and slab. Most of the casting techniques are time consuming and therefore result in high labour costs. In addition there is always a risk of cavitation or porosity however small with the need for costly remedial action.

Fabrication of shields, by which we mean the building up of a shield from a number of pieces of solid metal, is the surest and cheapest method provided that the design is suitable. For example there should be a minimum of changes in section, and a minimum of controls, fittings etc. which have to protrude into or pass through the shield.

Given these requirements there are a number of relatively simple techniques available. Rectangular shields may be built up from flat slabs of lead. Cylindrical shields can be wrapped with a continuous sheet of lead (Fig. 1). At the time when these techniques were being developed one of the most interesting applications was in the shielding of the spent fuel container for nuclear submarines. This demonstrated the use of lead for all the best reasons. A simple cylindrical shield was required for an immense container which was to travel by rail and road on its own special vehicle, and only by using lead shielding was it possible to keep down the size and weight to within the limits which were acceptable. This was particularly so with the rail transport section of the journey where the outside diameter of the shielded container came within an inch or two of the maximum size which could be carried without danger to the traffic on neighbouring tracks.

At this time the technique of continuous wrapping with lead sheet was not new but thin sheet was always employed for ease of handling. With the

Fig. 1 One of the earliest containers to be shielded using the continuously wrapped lead sheet technique. The uniform cross section along the whole length of the container made this technique the obvious choice. (Photo. W.G. Jenkinson Ltd.)

size problem in this application it was necessary to rethink the whole technique.

However efficiently the wrapping is done, there will always be a slight wastage of space between layers, and with thin sheet and many layers this can eventually add up to a significant space. It is therefore necessary to employ an outer steel shell of sufficient thickness and strength to compress

these layers.

The design for this container was based on the idea of making the lead nearly self-supporting, and this was achieved by wrapping one-inch slabs around the central containment cylinder, and lead burning these to the steel at both ends and to each other so that the final 40 ton shield behaved almost as a single cylinder. As a result it was only necessary to add a thin skin of steel to the outer surface of the shield to prevent accidental damage, and the size limits on the finished container were not exceeded. There was no space available for an outer package and so the container had to be transported under special conditions to eliminate any danger of an accident.

Development of this technique of shield manufacture has been so efficient that protrusions can be accommodated by cutting holes in the sheet as required and then lead burning around the protrusions when wrapping is completed, thus avoiding shine paths through the shield (Fig. 2).

Thermal conductivity

As the materials being carried have grown in size and activity a further problem has arisen. The efforts made to insulate the lead shield from external heat in an accidental fire, have also tended to prevent the dissipation of source-produced heat from the container, with the prospect of a thermal build-up inside the package.

The solution to this problem has also been devised by the UKAEA. The standard form of package for the larger containers in the UK has for some years been a timber packing case type of construction (Fig. 3). This has met all the test requirements and given ample protection from both shock and fire.

To cope with source-produced heat, experiments were tried with gaps in the timber casing near the bottom and top of each side, these allowing convection currents to be set up, by the heat emitted from the container, causing a through draught which cooled the container in normal circumstances. The discovery that, in a fire, the chimney effect caused ignition of the inside surface of the wood was in turn met by covering this surface with a thin steel sheet (Fig. 4).

The success of this design gave rise to further progress. When this

Fig. 2. A modern continuously wrapped lead sheet shield. Experience with this technique now permits change in section and protrusions to be accommodated. Shielding is lead burned around protrusions to prevent shine paths. (Photo. Lead Fabricators Ltd.)

Fig. 3 Timber packing case type of protective package being assembled around lead shielded container. (Photo. Graviner Manufacturing Co. Ltd.)

design was under consideration there was a fear that the ingress of heat through the ventilating gaps would nullify the protective effect of the package during a fire. In practical tests it was found that this was only partly true. The insulating effect of the package was only slightly reduced and it was discovered that the heat penetrating the package was not sufficient to cause damage to the shield of the large container inside. The containers had by now become so large, that the quantity of heat required to raise the container temperature to 327°C was approaching comparability to the quantity of heat produced in the test fire.

The next step was a matter of calculation. A sufficiently massive container could be used as a 'heat sink'. Where a source required a shield of around 11 inches of lead, it would be possible to employ, say, 10 inches of lead with an outer steel shell about 2 or 3 inches thick. The total mass would 'swallow up' the heat of the fire test without excessive temperature increase, the

Fig. 4 Later design of packaging, showing ventilating slots for expulsion of source-produced heat. (Photo. UKAEA)

massive steel would contain the thermal expansion forces developed in the lead over the temperature increase experienced and the massive steel would withstand the shock requirements imposed by the regulations.

This has tended to halt the progress of fabricated shields of the wrapped type since the interfaces and small air gaps between layers reduce conductivity, which is now even more essential than before. In these immense containers, the sources being carried may emit large quantities of heat from which build-up at thermal barriers can be quite immense. In a fire these interfaces become even more important for what may be termed 'geographical' reasons.

Heat emitted from the sources passes through the steel inner shell to the first layer of lead. Beyond this first layer there may be a thermal barrier in the form of an interface or air gap. This will cause the first layer of lead to

heat up, but in doing so it will expand and press against the next layer, thus reducing the effectiveness of the thermal barrier. Eventually there will be a form of dynamic equilibrium through the shield to the outer steel shell allowing the transmission of heat to this shell for dissipation.

A fire situation is 'geographically' the opposite of this. The outer steel is the first to heat up and will expand away from the outer layer of lead shielding. However the insulation of the thermal barrier between steel and lead is not perfect and therefore the outer layer of lead will experience some increase in temperature. At first it may not be as hot as the steel, but the coefficient of thermal expansion of lead is $1\frac{1}{2}$ to 2 times that of most steels and so eventually the lead will begin to press more closely against the steel and will accept heat from it more freely.

Unfortunately no such advantage exists between the first and second layers of lead, which have equal coefficients of expansion and will therefore move apart under the influence of external heat. At this point the 'heat sink' has been reduced to the outer steel shell and one layer of lead, and it is unlikely that the spiral path through the lead shield will provide a large enough 'heat channel' to prevent the outer layer of lead from melting.

This therefore has given rise to a return to the solid shield, but in this case it is not a simple casting manufactured in the steel shell.

CURRENT PRACTICE

In the days when fabricated shields were most widely used, one technique was occasionally employed for cylindrical shields which has not yet been described here. Where a shield was required in the form of a pure cylinder, some manufacturers would form such a cylinder from thick lead slab and then push it into the required position inside the steel shell. In doing this use was made of the softness and malleability of lead, which enabled any dimensional imperfections to be overcome by brute force.

Since that time there has been a move in the lead industry to establish lead as a recognised 'engineering material', which can be machined⁴ and handled just like any other metal and one of the results of this has been the development of a technique of engineering a lead shield which suits all the current requirements.

Cylindrical shields are now often produced as a series of shorter cylinders, made by casting, which are then machined to very accurate tolerances (Fig. 5) and fitted together around the inner steel shell by 'feeding' them on, one at a time from one end before the outer steel shell is assembled.

Thus in the radial direction the shield is a single solid casting permitting the efficient transfer outwards of heat from the source and, because of its accurate inner and outer circumferential dimensions giving good lead to steel contact. For this reason it also acts as an efficient heat sink in conjunction with the thick outer steel shell.

Because of the limited size of the cast and machined components these can be produced in a moderate sized lead works, so that the production of even the largest shields is no longer restricted to the largest companies.

The accuracy with which the mating faces of the individual units are machined ensures that even with a simple single-step interlocking between the units there will be no lack of shielding at the joints.

Finally it is now convenient for the steelwork to be manufactured in one works while the individual shielding cylinders may be cast and machined in another. Since transport of these relatively small cylinders is no great problem they are then brought to the engineering works for assembly. Radiographic testing is carried out before fitting the steel end plate and outer shell so that in the unlikely event of remedial action being necessary this can be undertaken with minimum inconvenience.

SUMMARY

Thus the problems which originally beset the lead shielding industry and the new ones arising from the safety regulations have been overcome.

Originally most designers knew nothing of lead's properties nor of the techniques of manufacture necessary to produce efficient shields. Education of design engineers has eliminated this problem, and most designers now consult shield manufacturers before finalising their designs.

Originally, many shields were cast by companies with little or no lead experience often with most unfortunate results. Currently shield manufacture is entirely in the hands of experts.

Fig. 5. Cast and machined lead cylinder (2 ft. deep) demonstrating high quality machined finish obtainable. Note single step interlocking design and one of two diametrically opposite holes for crane lifting. (Photo. James Girdler and Co. Ltd.)

Originally all shields were cast. Now combination techniques prevail such as fabrication from cast and machined short cylinders.

In the early days of the IAEA recommendations, there were some doubts as to whether lead would still be the ideal material for container shields. Now even those who had no faith are returning to lead. This is due to the combined efforts of the UKAEA which has developed the techniques of protecting the container and the UK lead industry which has developed appropriate designs of shield to parallel the developments in protective packaging.

Finally, we now have rigorous radiographic and physical testing of shielded containers which by virtue of the efficiency of their design and manufacture, are less in need of testing than were the earlier somewhat hopeful products which did not always receive a detailed examination.

The overall picture of this industry is now one of modern technology at its best. Instead of regulatory bodies devising means of rejecting failures, we have complete cooperation throughout to produce an efficient product for which rigorous testing is now seen rather as a means of applying a seal of approval.

REFERENCES

1. F.E. Dixon, Thermal Shields for Lead Shielded Containers for Radio-active Materials, Proc. Lead Shielding and Nuclear Safety Conference (1969), Lead Development Association, London.
2. F.E. Dixon, Packaging for Large Lead Shielded Containers for Radio-active Material, Proc. Lead Shielding and Nuclear Safety Conference (1969), Lead Development Association, London.
3. Lead for Radiation Shielding, Lead Development Association, London (1965).
4. Lead Machining, Polishing, Etching, Lead Development Association, London.

SHIELDING ASPECTS OF LWR SPENT-FUEL SHIPPING CASKS

B. A. Engholm

ABSTRACT

Shielding requirements for light-water-reactor spent-fuel shipping casks were investigated as part of the Gulf/Allied joint program on nuclear fuel reprocessing. Comparisons were first made with GE IF 300 cask design. The GGA PATH gamma-shielding code, which can handle three-dimensional multi-material configurations, was utilized for the gamma dose-rate calculations, while both transport and diffusion techniques were applied to neutron analyses. Gamma dose buildup factors for multi-layer configurations were estimated from earlier work performed by GGA for the USAEC. The important nuclide contributors to external dose rates were identified. Following these studies, dry casks were investigated, ranging from a single PWR element capacity to 24 BWR element capacity. Combinations of lead and depleted uranium were studied for gamma shielding, and either borated water or Uniroyal 3807 assumed for neutron shielding. Radiation levels around a dry 4-PWR cask are shown.

INTRODUCTION

In recent months Gulf General Atomic has been engaged in the analysis and conceptual design of several types of light-water-reactor spent-fuel shipping casks. The need for neutron shielding in such casks when carrying high-burnup fuel has complicated the overall shielding analysis and optimization. Hence, several preliminary shielding studies were performed in order to check out the available source information, GGA's calculational methods, and the effects of changing the shield configuration. The General Electric IF 300 cask design¹ was used as a model for some of these studies, and a concept in the ORNL "Cask Designer's Handbook"² for others.

The purpose of this paper is to define the gamma and neutron source terms, to describe GGA's calculational methods, to present comparisons with GE's published figures, and to show the dose rates around a dry 4-PWR cask.

SOURCE TERMS

Fission Product Decay Gammas

Past experience with HTGR fuel has shown that the important shielded gamma emitters after 90 days cooling are La^{140} and Pr^{144} . In the case of LWR fuel containing a substantial amount of plutonium, Rh^{106} also becomes important. PWR fission-product inventories³ for a burnup of 35,000 MWD/tonne, a specific power of 40 Kw/kg, and a cooling time of 90 days are as follows: Rh^{106} 675,000 Ci/MTU, La^{140} 16,460 Ci/MTU, and Pr^{144} 1,078,000 Ci/MTU.

The GGA PATH code⁴ library contains the following hard-gamma emission data for these fission products.

Table 1. Gamma Emission Data in PATH Code
(Mev/disintegration)

Nuclide	Average Photon Energy (Mev)						
	1.0	1.25	1.6	2.0	2.4	2.9	3.5
Rh^{106}	0.0215	0.0049	0.0053	0.0018	0.0008	0.0002	0
La^{140}	0.1	0	1.58	0	0.075	0.001	0
Pr^{144}	0	0	0	0.0218	0	0	0

An axial power peak, if present, is assumed to have a direct multiplicative effect upon these fission-product concentrations.

Curium Neutrons

The basic mechanism of curium buildup in nuclear fuel and subsequent neutron emission is well covered in the literature, and will not be repeated here except to point out that this source is unimportant in HTGR casks. The production of Cm^{242} plus Cm^{244} in HTGR fuel is only ~3 grams per tonne of metal, and the graphite fuel blocks have considerable self-shielding for neutrons. Returning to LWR fuel, in the particular case of the IF 300 cask design General Electric adopted a source term of 2×10^9 neutrons/sec for four PWR elements and 90 days cooling, which works out to $\sim 1.1 \times 10^9$ n/sec-MTU. Pertinent data from Refs. 1 and 3 are listed below.

Table 2. Curium Inventories and Emission Data for PWR Fuel, 35,000 MWD/MTU Burnup, and 90 Days Cooling

Inventories		
Cm^{242}	48,210 Ci/MTU	
Cm^{244}	3,856	
Emission Data	Yield (n/sec-g)	Yield (n/sec-Ci)
Cm^{242} (SF)	2.02(+7)	6060.
$\text{Cm}^{242}\text{O}_2$ (α ,n)	2.34(+7)	7030.
Cm^{244} (SF)	1.05(+7)	1.29(+5)
$\text{Cm}^{244}\text{O}_2$ (α ,n)	5.05(+5)	6200.

Ref. 5 provides neutron spectra for all four cases above. The spontaneous fission spectra for both nuclides are almost identical, as are the two (α ,n) spectra. These are plotted in Figure 1, normalized to unity. Multiplication of data in Table 2 indicates that about two-thirds of the neutrons in PWR spent fuel comes from spontaneous fission and one-third from (α ,n) reactions. A spectrum representing this split is also shown in Figure 1. Finally, the Cf^{252} fission spectrum is plotted for comparison. Because of the similarity, this Cf^{252} spectrum was used for cross-section averaging in the GGC-4 code.⁶

Axial power peaking is assumed to have no effect upon neutron source intensity.

Other Source Terms

Secondary gammas from neutron captures in hydrogen (photon energy = 2.2 Mev) are sometimes a problem when the neutron shield is not borated.

Secondary neutrons can arise from (1) thermal fissions in the ~ 0.2 w/o U^{235} present in depleted uranium (DU), (2) fast fissions in the U^{238} in DU, and (3) fissions in the fuel assemblies.

NEUTRON CALCULATIONS — IF 300 CASK

Geometry

For purposes of the neutron calculations, the geometry of the IF 300 cask design was simplified to the one-dimensional configuration shown in Figure 2.

Materials

For the neutron transport calculations the homogenized fuel region consisted of 0.0386 atoms/barn-cm* H, 0.031 at/b-cm O, 0.012 at/b-cm Zr, and 0.00516 at/b-cm U^{238} , the boronated copper separators being represented by extra O and Zr. In the accident case, the H was removed and the O concentration reduced to 0.0117 at/b-cm.

The diffusion calculations utilized 0.0386 at/b-cm H, 0.00136 at/b-cm B_{nat} , 0.0296 at/b-cm O, 0.012 at/b-cm Zr, and 0.00516 at/b-cm U^{238} . The cask water was not borated.

Calculations

Fast-neutron dose rates were calculated using the 1DF neutron-transport code⁷ in the P-2, S-8 approximation with 10^{-6} convergence. Eight neutron-energy groups were utilized, the lowest energy boundary being 0.111 Mev. Results are shown in Figure 3.

Intermediate and thermal dose rates in the cask were computed by the GAZE diffusion code,⁸ using as source terms the down-scattered neutrons from the 1DF runs. Four intermediate and four thermal energy groups were used. These results are also plotted in Figure 3.

Secondary neutron dose rates were estimated separately.

Discussion

Figure 3a, the normal case, shows a total surface neutron dose rate of 5.7 mrem/hr, consisting of 4.5 mrem/hr fast and 1.2 mrem/hr intermediate and thermal. Secondary fission neutrons created in the DU add 0.8 mrem/hr to this surface dose rate (boration of the water would essentially eliminate this contribution). The geometrical reduction factor in moving the detector from the cask surface to 10 ft from the centerline was found to be 0.214 for a flat power distribution. The resulting 10-ft neutron dose rate of 1.4 mrem/hr can be compared with the published¹ GE value of 5.1 mrem/hr, which is a factor of 3.6 higher than the GGA result.

In the accident case (Figure 3b) the surface dose rate increases to 1180 mrem/hr. A geometry correction to 3 ft from the surface — a factor of 0.41 — yields an accident neutron dose rate of 480 mrem/hr, as compared

---

*Density in g/cc = density in at/b-cm multiplied by $\frac{\text{atomic weight}}{0.6023}$.

with the GE value of 134 mrem/hr, which is a factor of 3.6 lower. Thus, GGA's studies indicate an accident neutron dose rate 340 times higher than the normal operating neutron dose rate, whereas GE finds only a factor of 26. No explanation for this large discrepancy can be offered at this time.

The advantages of situating the neutron shielding outside the gamma shielding are demonstrated in Figure 3c. Fast neutrons are inelastically scattered in the DU region, reducing their energies to the range where hydrogen has a very high scattering cross section. The primary neutron surface dose rate is thereby reduced from 5.7 mrem/hr to 1.5 mrem/hr with no change in actual material thicknesses. Furthermore, the DU weight is reduced by 35%. In this altered configuration, secondary fission neutrons from thermal captures in DU would add ~ 0.75 mrem/hr to the surface dose rate; therefore, the water should be borated. Fast fissions in the DU add only ~ 0.13 mrem to the surface dose rate.

GAMMA CALCULATIONS — IF 300 CASK

Geometry and Materials

The PATH gamma-shielding code⁴ is capable of treating three-dimensional geometries in almost any degree of complication, except where streaming dose rates would predominate over direct dose rates (such as a labyrinth). Hence, the IF 300 cask geometry did not have to be simplified for the gamma calculations. Materials were also represented accurately, except that Cu — not available in the PATH library — was simulated by 9-g/cc Mo.

Calculations

Gamma dose rates were calculated both at the cask surface and at 10 ft from its centerline. Computing time was reduced by the knowledge that the remote two assemblies would contribute only 1% to the dose rate, owing to shielding by the near two assemblies.

The PATH code includes a routine for the calculation of buildup factors for any desired single material which characterizes the shield, as represented by its equivalent atomic number Z . The GE IF 300 cask design is characterized by a DU gamma shield; therefore, the PATH calculations were performed for uranium buildup ($Z=92$). However, as has been shown in numerous studies such as Ref. 9, following the heavy material with a lighter material can substantially increase the overall buildup of gammas. Therefore, correction factors were

derived from the curves of Figure 4. As will be shown below, 2 Mev is the predominant gamma source energy for purposes of this adjustment. The dose correction factor was generally about 1.8.

Results and Discussion

The corrected gamma dose rate at dose point "A" in Figure 2 was found to be 13.1 mrem/hr for a flat power distribution, or 19.6 with the axial peak. At 10 ft from the centerline along this same radius, the values were 2.81 and 3.62 mrem/hr, respectively. Moving 45° to this radius, so the detector is opposite a corner of the basket, PATH gives dose rates of 1.76 and 2.27 mrem/hr. General Electric publishes¹ a gamma dose rate of 4.24 mrem/hr, in good agreement with the maximum GGA value of 3.62 mrem/hr.

It is of interest to determine which source energies and isotopes contribute the most to these dose rates. Study of the PATH results shows that among the photon energies listed in Table 1, the 2-Mev source group contributes 60% of the dose rate, the 1.6-Mev group 22%, and the 2.4-Mev group 15%. By isotope, one finds that Pr^{144} contributes 61% of the dose rate, La^{140} 29%, and Rh^{106} 10%.

Under accident conditions, some gamma attenuation provided by water is lost. It is estimated that the accident gamma dose rate 3 ft from the cask surface would then be 11.2 mrem/hr without the axial peak or 15.2 mrem/hr with the axial peak. GE's value is 7.0 mrem/hr, which seems somewhat low.

DOSE RATE CALCULATIONS — DRY 4-PWR CASK

Typical of the many cask designs investigated by GGA following the IF 300 comparison studies is the National Lead concept shown in Ref. 2, and redrawn with some modifications in Figure 5 of this paper. The configuration merely illustrates the type of complicated geometries which can be handled by the PATH gamma-shielding code, and should by no means be considered as engineered design.

Neutron calculations were performed as described before, after simplification of the geometry to equivalent cylindrical regions. For a neutron source strength of 1.1×10^9 n/sec-MTU, a neutron shield of 6 inches of Uniroyal 3807 or borated paraffin, or 7 inches of borated water, will be adequate to keep the neutron dose rate at 6 feet from the cask surface well below 5 mrem/hr.

Gamma calculations involved buildup corrections for lead/iron as well as uranium/iron, using curves similar to those in Figure 4. Shielding was arranged so that the gamma dose rate six feet from the cask surface, on the midplane, was about 5 mrem/hr at any polar angle. The shell diameter is determined by the gamma shielding thickness needed at the corners, in this case 3 inches of DU plus 1 inch of lead. If only DU were used, the thickness would be 3.6 inches.

ACKNOWLEDGEMENT

The author gratefully acknowledges the participation of C. R. Davis, W. C. Finch, and D. E. Hurd in this work.

REFERENCES

1. Design and Analysis Report - IF 300 Shipping Cask, General Electric Company (September 26, 1969).
2. L. B. Shappert, Cask Designers Guide, Figure 8.3, ORNL-NSIC-68, (February, 1970).
3. Barnwell Nuclear Fuel Plant Safety Analysis Report, Table III-1, Part 4, Section III, Vol. I (August 15, 1969).
4. S. S. Clark and B. A. Engholm, PATH, A Highly Flexible, General Purpose Gamma Shielding Program, GA-9908 (December 10, 1969).
5. S. J. Rimshaw and E. E. Ketchen, Curium Data Sheets, ORNL-4187 (December, 1967).
6. J. Adir and K. D. Lathrop, Theory of Methods Used in the GGC-4 Multigroup Cross Section Code, GA-9021 (October 1, 1968).
7. K. D. Lathrop, DTF-IV, A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering, LA-3267 (April 23, 1965).
8. S. R. Lenihan, GAZE-2, A One-Dimensional, Multigroup, Neutron Diffusion Theory Code for the IBM-7090, GA-7156 (August 3, 1962).
9. B. A. Engholm, Gamma Buildup in Heterogeneous Media, GA-8741 (September 16, 1968).

SOURCE
NEUTRONS
PER MEV
(NORMALIZED
TO ONE
NEUTRON)

LC-88337-A

Fig. 1. Neutron Spectrum Comparison.

LC-88333-A

Fig. 2. Original IF 300 Cross Section (top), and Idealized IF 300 Cross Section for Neutron Calculations (bottom).

NEUTRON
DOSE RATE
(MREM/HR)

1 FUEL ASSEMBLIES	3 DU	5 WATER	7 OUTSIDE ATMOSPHERE	— INTERMEDIATE & THERMAL NEUTRONS - - - FAST NEUTRONS (E > 0.111 MEV)
2 OUTER BASKET	4 STEEL CLAD	6 STEEL SHELL	8 AIR	

LC-88334-A

Figs. 3a, 3b, 3c. Primary Neutron Dose Rates in
GE IF 300 Cask Design as Calculated by 1DF and GAZE.

LC-88336-A

Fig. 5. Typical Dry 4-PWR Cask Design Investigated by GGA.

✓ LEAD REMELT EXPERIENCE WITH 60-TON CASK

E. C. Lusk

BATTELLE
Columbus Laboratories

ABSTRACT

The fabrication of large-size spent fuel shipping containers can result in a variety of unanticipated crises. During the fabrication cycle of the NFS-100 cask, (a 60-ton gross weight cask) there was a failure of the lead pouring equipment which necessitated the remelting of 80,000 lb of lead. Since the "lead pour" phase is near the end of a cask fabrication cycle, it was necessary to quickly establish a guaranteed success remelt operation which would result in a nonporous gamma shield. The size of this particular cask and a large number of heat rejection fins necessitated the set up of shroud-type furnace within the metal fabrication shop which would bring the entire cask up to the desired lead remelt temperature of approximately 680 F. Five 750,000 Btu rated liquid propane burners (oriented so the flame was directed tangential to the cask finned exterior) were positioned at several vertical levels and a successful remelt operation was carried out. This paper presents the details of that operation, as well as the temperature history of the cask. Temperatures were recorded by means of both external thermocouples welded to the cask and dip-type thermocouples employed within the lead volume to establish the liquid/solidus interface and its movement with time. The remelt operation required 27 hours to complete, was completed on site, and resulted in a void-free gamma shield. Specifics of this operation and associated temperature data are detailed in this paper.

CASK DESCRIPTION

The spent fuel shipping cask discussed in this paper is the NFS-M100 cask system which is shown mounted on a rail car in Figure 1. This cask is a multipurpose shipping cask designed and licensed by BCL, and which is capable of shipping either 12 Big Rock Point, 18 Pathfinder, or 18 Humboldt Bay spent fuel assemblies. Figure 2 provides an exploded view of the NFS-M100 cask and illustrates the major components. The cask weight, exclusive of the skid, is 60 tons and has an external diametric envelope of 70 inches (as established by the heat rejection fins) and an overall length of 130 inches. The biological lead shield weighs approximately 40 tons and has a radial thickness of 8.75 inches.

The heat rejection fins, which are required to dissipate an anticipated decay heat load of approximately 28 kw thermal, increase the cask surface area by more than a factor of three and were fabricated from a continuous strip of four-inch wide x 1/4-inch thick stainless steel. Attachment of the fins to the outer shell was accomplished by a continuous welding operation which resulted in 1/4-inch fillet welds on both sides. The fins are four inches high and are set on 1.75-inch axial centers. The outer shell of the cask is a composite plate comprised of a 1.75-inch thickness mild steel plate to which a roll bond clad 1/4-inch thickness Type 304 stainless steel plate has been joined. The interior fuel cavity has a 40 inch ID and is 103 inches deep. Eighteen square type permanent fuel assembly cavities are established by a means of the inner basket which is comprised of 1/2-inch thick copper plates which are bonded to themselves as well as to the walls of the inner cavity.

Lead Melt Background

The lead fill procedure utilized called for setting the cask in a vertical attitude and then positioning a lead kettle plus furnace apparatus at a higher level to effect a gravity feed of molten lead. During the first lead filling operation, a flow control valve on the lead fill line between the lead source kettle and the furnace froze in the closed position. During the time required to replace the stuck valve, the auxiliary heating equipment (which normally would maintain the overall cask system at the desired casting temperature) was unable to supply sufficient heat to hold the lead (which had already been cast) in a molten condition. Upon resumption of the lead filling (after the defective valve had been replaced) the cooling rate of the lead was too rapid and a void was formed at approximately 24 inches from the

bottom of the cask surface. Information obtained from the thermocouples which were placed on the cask to monitor the lead pour operation, indicated that the heat rejection fins were being too effective in shadow shielding the cask surface from the flame being produced by the liquid propane burners (as oriented for this initial casting operation). Since the cask would require extensive rework if the situation were not remedied quickly, the decision was made to set up an on-site remelt furnace. This is described in the next section.

Description of Remelt Equipment

A shroud-type furnace composed of 0.187-inch thick mild steel plates was erected on-site for the remelt operation. Scheduling and cost requirements precluded moving the cask. The furnace arrangement is shown in Figure 3, and since the cask fins increase the cask envelope by 8 inches resulted in a chimney diameter of approximately 80 inches. The cask was positioned vertically with the open end of the fuel cavity facing downward to permit heating of the cavity interior. A cover employing a controllable vent was provided for the top of the furnace shroud and the entire furnace assembly was then supported by steel legs resting on the floor of the fabrication shop. Ten burner openings were provided for in the sides of the furnace shell and were equally spaced circumferentially and vertically to permit desired burner placement for the purpose of heating up zones in a uniform fashion. These burner ports were also provided with a controlled air inlet vent and provision was made to direct the burner flame at a tangential angle with respect to the outer shell of the cask, thus utilizing the circumferential fins to, in effect, direct the flame front (see Figure 3). Baffle-type deflection plates were provided for the burner flame to impinge upon and to then provide for a swirling motion around the cask.

Details of Remelt Operation

Five 750,000 Btu rated propane burners were positioned within the furnace shroud to commence a uniform remelt heating cycle. In addition, a ring burner was positioned beneath the cask to heat the fuel assembly cavity and a standard "blast" burner was positioned at the top to heat the lead pour port. In addition, eight chromel-alumel thermocouples were welded to the outer shell of the cask between the fins for purposes of temperature monitoring during the heatup and cooldown cycles since the preceeding experience had shown the fins shadow shield the flame. These couples were positioned such

that there would not be direct burner flame impingement and were located at approximately one foot axial increments from bottom to top. A portable dip-type thermocouple was utilized to monitor temperatures within the lead cavity after the lead remelt had begun. As another backup, a portable contact pyrometer with a range of 0 to 1,200 F was utilized to check available surface temperatures of the furnace and cask. The pyrometer employed was a compensated, direct reading type pyrometer which had been calibrated prior to the operation against a bureau of standards related instrument.

The burners were lit at approximately 7 a.m. and commencement of temperature data recordings were initiated at 8 a.m. Burners were adjusted for maximum heat output and within a short time period were capable of maintaining the flame baffle plates and the outer edges of the cask fins in the area of the flame impingement at a cherry red color, which was measured via a thermocouple to read approximately 1,400 F. The operation proceeded quite smoothly with very little burner adjustment being required. The vent opening at the top of the furnace configuration was adjusted upward to provide for a blast burner to be directed down at the center of the vent opening onto the lead pour hole. Despite all of this deliberate heating, lead melt temperatures did not occur until approximately 9-1/2 hours after starting the heatup. The first indication of lead melting on the top was some 30 minutes later, or at 5 p.m. At six o'clock p.m. the internal "dip" thermocouple indicated molten lead at a depth of four feet. At this time, the top burners were cut back and at 12:15 a.m., or 17 hours after starting, all of the lead within the NFS cask was assessed to be in a molten condition.

The solidification process which followed required approximately ten hours. This operation was accomplished by opening the top vent, turning the burners down, and removing vertical sections of the furnace wall commencing at the bottom and proceeding upward. Following the cooldown period, a complete gamma scan of the cask surface was made on a three-inch grid. The results of this test proved that a solid condition of remelted lead gamma shielding existed. The detailed thermal history is summarized in Table 1.

Postmelt Evaluations and Comments

The preceding sections have described the successful remelting of a large volume of lead at a fabricator's shop. However, some additional insight as to the severity of postulated industrial fires, and their effect on a cask system, can be gleaned from this experience.

First of all, the remelting process took 17 hours, with the cask outer shell temperature reaching 620 F after ten hours. In addition, this heat up took place in a specially erected furnace which directed propane gases to sweep between the fins, with the gases having a very high energy content. This can be illustrated by the fact that 800 gallons of liquid propane (heating value of propane is 91,690 Btu/gal) were burned over a 17-hour period for a potential heat load of 73.4×10^6 Btu/hr.

The hypothetical fire which is defined in 10-CFR-Part 71 as a 1475 F radiant source having an effective emissivity of 0.9, and which regulation also stipulates that the receiver surface has an effective absorptivity of 0.8, permits utilization of the Stefan-Boltzmann equation to describe the heat transfer coupling, or

$$q_{\text{Fire}} = (\Sigma F_G A) (F_\epsilon) (\sigma T_{\text{Fire}}^4 - \sigma T_{\text{surface}}^4) \quad (1)$$

where F_G = geometric view factor

A = area receiving radiation energy

$$F_\epsilon = \text{emissivity factor} = \frac{1}{\frac{1}{\epsilon_{\text{fire}}} + \frac{1}{\epsilon_{\text{surface}}} - 1} .$$

If we neglect the countering effect of an increasing surface temperature, and utilize Equation (1) to calculate a "cold wall" heat input, we obtain the following heat inputs resulting from the hypothetical fire:

$$q_{\text{RAD}} = 2.12 \times 10^6 \text{ Btu/hr at } \epsilon_{\text{surface}} = 0.8 \quad (2a)$$

$$= 0.54 \times 10^6 \text{ Btu/hr at } \epsilon_{\text{surface}} = 0.2 \quad (2b)$$

$$= 0.4 \times 10^6 \text{ Btu/hr at } \epsilon_{\text{surface}} = 0.15 . \quad (2c)$$

The results shown in Equation (2a) when compared to 4.3×10^6 Btu/hr, as calculated from the heating value of liquid propane, would permit me to say that the remelt fire was more severe than the prescribed fire test. The latter two values illustrate the effect of a lower emissivity and since the cask fins and shell had already been polished to a 63 G micro finish, the lower values are justifiable for 304 stainless steel. More will be said regarding these lower values below.

Consideration of the thermal capacitance of the NFS-100 cask itself leads to the following tabulation and resulting thermal capacitance values:

$$80,000 \text{ lb of lead} \times 0.031 \text{ Btu/lb-F} = 2480 \text{ Btu/F} \quad (3a)$$

$$27,000 \text{ lb of steel} \times 0.11 \text{ Btu/lb-F} = 2970 \text{ Btu/F} \quad (3b)$$

$$4,000 \text{ lb of copper} \times 0.0915 \text{ Btu/F} = 366 \text{ Btu/F} \quad (3c)$$

$$(\rho \text{ VCp})_{\text{total}} = 5816 \text{ Btu/F} \quad (3d)$$

Exclusive of the heat of fusion of lead, the total heat required to raise this cask to 600 F above room temperature is then

$$Q = \rho \text{ VCp} \Delta T = 5816 \text{ Btu/F} \times 600 \text{ F} = 3.48 \times 10^6 \text{ Btu} \quad (4)$$

In addition, the heat of fusion of lead (10.5 Btu/lb) raises the total heat to 4.32×10^6 Btu, which is significantly lower than the 73.4×10^6 Btu cited for the propane previously. The logical question is where did this heat go? However, before we go on, if the capacitance calculated via Equation (4) is averaged over a ten-hour period (time lead melt temperature is reached, see Table 1), and a 17-hour period, heat transfer levels of 0.35×10^6 Btu/hr and 0.25×10^6 Btu/hr, respectively, are calculated. These latter numbers should be compared with the results shown in Equations (2b) and (2c), and the next questions is - how realistic is the absorptivity value of 0.8 (to be held constant through the 30-minute fire) stipulated by the regulations? It took ten hours in a specially designed furnace to reach melt temperatures, and another seven hours to complete melting of the lead. On the other hand, Serkiz⁽¹⁾ shows melt temperatures being approached in approximately 30 minutes with the fins "blocking" and lead melt occurring if the fins are removed. There would appear to be an inconsistency.

In an attempt to consolidate the above remarks, consider the following tabulation:

Heat load required to raise cask temperature 600 F and melt lead	= 4.32×10^6 Btu
Heat load contained in 800 gal of propane	= 73×10^6 Btu
Heat load dumped to external environment by 200 ft ² of furnace shell at 900 F*	= 18.5×10^6 Btu
	$\Delta = 54.5 \times 10^6$ Btu.

With 54.5×10^6 Btu of heat energy being potentially available to heat up the cask (and it is recognized that the burner gases took considerable energy out through the stack opening) why did it take ten hours to reach the lead

---

* Radiative coupling to the ambient environment only.

melting temperature, and then another seven hours to melt the 40,000 lb of lead? The hypothetical 30-minute fire heat level* (in comparison) calculated to be $0.27 \leftrightarrow 1.06 \times 10^6$ Btu. Obviously, convective coupling was not as effective as had been hoped for in a special furnace. Further speculation is left to the reader, but this remelt experience does raise some questions as to the actual interpretation of the effect of the fire test conditions on a shipping container. Needless to say, our analytical approaches appear to be quite conservative.

ACKNOWLEDGMENTS

Acknowledgment is extended to the Edward Lead Company, Scientific Advances, Inc., and Nuclear Fuel Services, Inc., for permission to use in this paper the information generated on this project.

REFERENCES

- (1) Serkiz, A.W., "Safety Analysis Report NFS-M-100 Cask", Battelle's Columbus Laboratories.
- (2) Perry, R., "Log of Lead Pour Operation NFS-M-100", Edward Lead Company.
- (3) Jones, S., Pictures and Drawings and NFS-M-100 Cask, Scientific Advances, Inc.

Table 1. Thermal History of Remelt Operation (Temperature, F)

Time	Thermocouple Number								Dip	Comments
	1	2	3	4	5	6	7	8		
8:00 a.m.	150	150	150	185	250	310	370	310		
10:00 a.m.	170	175	185	220	290	350	410	350		
12:00 a.m.	190	200	225	290	360	440	480	380		
2:00 p.m.	222	240	280	370	470	542	590	470		
4:00 p.m.	255	290	345	450	550	600	605	580		
5:00 p.m.	365	418	460	570	640	635	630	640		First lead melting
6:00 p.m.	430	470	515	600	645	640	630	615	610	Dip 4 ft. down
7:00 p.m.	460	540	570	640	655	660	655	615	610	Dip 7 ft. down
7:30 p.m.	500	670	630	630	635	665	670	650	620	Dip 9 ft. down
8:00 p.m.	520	682	650	635	645	675	680	635	655	
9:05 p.m.	545	635	640	635	660	690	690	605	670	Moved burner No. 3 to No. 4 and No. 2 to No. 3
10:00 p.m.	515	610	665	720	720	710	720	700	700	
11:00 p.m.	495	610	715	750	740	730	730	660	725	
12:25 a.m.	630	718	732	732	735	745	732	675	730	All lead melted Burners raised and set back
1:00 a.m.	550	680	755	770	770	760	740	675	740	
2:00 a.m.	500	635	720	760	770	765	760	670		Removed bottom panel
3:00 a.m.	--	602	675	730	755	760	750	560		Covered top vent hole
4:00 a.m.	--	--	660	730	770	780	780	670		Removed 2nd section of furnace
5:00 a.m.	--	--	620	715	765	780	780	675	780	
6:00 a.m.	--	--	593	660	725	755	760	760	760	Shut off No. 3 and 4 burner More sections of furnace removed
10:00 a.m.										Final lead solidification

Fig. 1. NFS-M100 Cask System

Fig. 2. Exploded View of NFS-M100 Cask System

Fig. 3. Furnace Arrangement.

WEIGHT MINIMIZATION OF CASKS FOR SHIPMENT OF HIGH
BURNUP AND RECYCLE LWR FUELS

J. L. Ridihalgh
R. E. Best

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

IMPACT TESTING OF CHEMICAL LEAD,
CASK MODELS, AND AN ENERGY ABSORBING DEVICE

J. H. Evans
Oak Ridge National Laboratory*
Oak Ridge, Tennessee

ABSTRACT

Several impact test programs have been conducted at Oak Ridge National Laboratory, and three recent programs directed toward the development of design and computational data relevant to the 30-ft free drop hypothetical accident condition specified in regulations governing the design and performance of containers in which radioactive materials are shipped are discussed herein. Chemical lead specimens were impact tested, and the stress-strain properties of lead undergoing gross impact-produced deformation at the strain levels and strain-rate ranges experienced by lead-shielded casks as a result of the 30-ft free drop and impact were determined. Models of a new energy absorbing device, the tubes-in-pipe absorber, were impact tested; and the data and techniques required to predict the behavior of full-size casks equipped with this device were developed. Scale models of the proposed steel-shielded LMFBR shipping cask were impact tested, and the data and techniques necessary to predict the response of the full-size cask were developed and the feasibility of using metal-to-metal seals in such a cask was demonstrated.

INTRODUCTION

A series of impact testing programs directed toward answering some of the many perplexing and complex questions associated with the potential damage to shipping casks (Type-B containers) and their contents that might result from the free-drop condition of the hypothetical accident specified in current regulations governing the design and performance of containers in which radioactive materials are shipped have been conducted at Oak Ridge National Laboratory (ORNL). The results of two investigations^{1,2} that essentially solved the problem of predicting cladding puncture in lead-shielded casks were reported³ at the Second International Symposium held in Gatlinburg, Tennessee, in 1968. Since that time, four distinct impact testing programs have been completed and a fifth is under way. Three of these programs are discussed herein.

---

*Oak Ridge National Laboratory is operated by Union Carbide Corporation under contract with the United States Atomic Energy Commission.

The first of the programs discussed herein involved an experimental study⁴ of the stress-strain properties of lead undergoing gross impact-produced deformation of the magnitude experienced by lead-shielded casks during the impact resulting from a 30-ft free drop that was successfully concluded in the spring of 1970. Models of an energy absorbing device, the tubes-in-pipe absorber devised by A. R. Irvine of ORNL, were tested⁵ to verify the applicability of the device and to develop the data and analytical relationships necessary to calculate the response of casks equipped with this device during an impact resulting from the 30-ft free drop. The third program discussed herein was concluded in the spring of 1971, and it consisted of impact testing scale models of the proposed LMFBR shipping cask,⁶ which employs steel as the biological shielding material. The data and analytical relationships required to calculate the response of full-size casks and their contents and the potential damage resulting from the 30-ft-free-drop impact were developed during this program. The ability of a metal-to-metal seal to withstand the impact resulting from a 30-ft free drop was also demonstrated.

An in-depth study⁷ of the dynamic energy absorption properties of cask heat transfer fins conducted at ORNL was completed in the fall of 1970, and this program is discussed by F. C. Davis elsewhere in these proceedings. An experimental investigation of the energy absorption properties, dynamic load capacity, and elongation properties of threaded stainless steel fasteners is now being conducted by D. D. Cannon of ORNL.

TEST FACILITIES, INSTRUMENTATION, AND DATA TRANSFORMATION TECHNIQUES

The physical impact testing reported here was conducted at the ORNL Drop Tower, which is a steel structure 40 ft tall that has a free-drop load capacity of about 5000 lb. The tower drop pad is essentially the unyielding surface specified in the regulations by virtue of its mass, support from bedrock, and the hardness and toughness of the armor-plate impact surface. The free-drop passage down the center of the tower is square, with each side being about 5 ft long. An angle-iron track installed at the rear of this passage was used for controlled-orientation specimen drops and to impact specimens with free-falling drop hammers (weights). The operating range of this system is from 25 to 500 lb. A low-capacity system installed on one side of the free-drop passage was used to impact specimens with 2-in.-diameter cylinders whose weights did not exceed 25 lb.

A totally automatic ball-detent release mechanism was used to release free-falling weights and specimens at specific heights. Load cells or accelerometers or both were used to measure forces or accelerations associated with the impacts. Load cells with capacities of up to 2,500,000 lb of force were designed,

fabricated, instrumented, and calibrated at ORNL. The output of the load cell during each impact was amplified, and filtered where accelerometers were used, and fed to a dual-beam cathode-ray oscilloscope. A schematic diagram of the instrumentation system is illustrated in Fig. 1. A Polaroid camera was used to photograph the trace resulting from each impact, and a typical photograph made during an impact test is shown in Fig. 2.

Fig. 1. Typical Load Cell and Schematic Diagram of Instrumentation Used to Measure Force-Time Relationships During the Impact Tests

Fig. 2. Typical Force-Time and Acceleration-Time Photograph Recorded During An Impact Test.

A Fortran program for cask impact data reduction (CIDR) applicable to the IBM 360 computer was used to transform the force-time data into force-deflection data. The values of force were scaled from the photographs made during the impact tests. These values were transferred to computer data cards as were data for other such test parameters as the drop height, weight of specimen or hammer dropped, initial diameter and/or length of specimen, and the final deformed length of the specimen. Force was converted to acceleration by using Newton's second law, and the impact velocity (V_0) was computed from the well-known expression for the velocity of a free-falling weight.

The CIDR computer program utilizes a numerical integration technique to convert the acceleration-time data to velocity-time and deflection-time data. That is,

$$V = V_0 - \int_0^t a \, dt \quad (1)$$

and

$$X = \int_0^t V \, dt, \quad (2)$$

where

V = velocity at any time and

X = displacement of the model or specimen.

Since the value of the acceleration at specific points in time can be obtained from the oscilloscope readings, both of the time integrals can be evaluated by numerical integration techniques. An energy balance effected by using test data and computed values that compared the potential or kinetic energy of the system with the work done in plastically deforming the model or specimen was used as a measure of the accuracy of the data acquisition and reduction system. Additions were made to this basic computer program to obtain data in a more usable form and to calculate other parameters by using the laws and theories of engineering mechanics as dictated by the requirements of the individual testing programs.

IMPACT TESTING OF CHEMICAL LEAD

Analyses of lead shielded containers have generally resulted in ultraconservative designs in that most of these analyses have been based on material properties determined by static tests. However, the impact properties of many materials vary significantly from their static properties. Generally, the stress for a given strain will be higher for the higher strain rates that occur under

impact conditions. Since lead is frequently used as the biological shielding material in many shipping containers for radioactive materials, the determination of its stress-strain properties under impact conditions is of importance.

A survey of the technical literature revealed that a considerable amount of work has been done by several investigators⁸⁻¹⁴ to determine the impact properties of lead. However, the published results of this work did not fill the needs of the cask designer because of the wide variation in these results, the lack of documentation of the material tested in some cases, and because the strain rates were generally out of the range of those applicable to the free-drop accident condition and the maximum strains were generally low.

Sixty-eight specimens made from chemical lead, ASTM B 29, were impact tested by using the low-capacity system at the ORNL Drop Tower. Cylindrical specimens with nominal diameters of 1.0 and 1.25 in.; length-to-diameter ratios of 1, 1.5, and 2.0; and weights of 6.94, 8.69, and 10.56 lb were dropped from heights ranging from 10 to 36 ft. Specimens before and after impact testing are shown in Fig. 3 for visual comparison. Force-time data for each impact were measured and photographically recorded with the equipment previously described. The force-time data were reduced through the use of the CIDR computer program, which was enlarged to effect such basic calculations as area with respect to time, engineering stress, true stress, strain rate, etc., and to perform certain grouping operations to simplify evaluation of the output.

Fig. 3. Typical Lead Specimens Before and After Impact Testing.

A least squares method was used to derive the equation

$$\sigma_t = 20,900 - 14,250\epsilon^{-0.093}, \quad (3)$$

where

σ_t = true stress and

ϵ = strain ($0.03 \leq \epsilon \leq 0.5$),

which describes the experimentally measured true stress for all measured strain rates with an average error of less than 267 psi. The relationship between the engineering stress and strain for a constant volume specimen is given by the expression

$$\sigma_e = \frac{20,900 - 14250\epsilon^{-0.093}}{1 - \epsilon}, \quad (4)$$

where σ_e = engineering or apparent stress = force/original area. Equation 4 is nonlinear and can result in awkward mathematical manipulations. When a lower level of computational accuracy can be tolerated, the following linear equation can be used.

$$\sigma_e = 17,470\epsilon + 1914. \quad (5)$$

The effect that the strain rate had on the stress-strain relationship was investigated and found to be negligible in the test strain rate range of 100 to 800 in./in..sec. However, when the dynamic stress-strain properties are compared with static and quasi-static properties determined from specimens of identical material, chemical lead exhibits a significant degree of rate sensitivity.

IMPACT TESTING OF TUBES-IN-PIPE ENERGY ABSORBER SPECIMENS

Shipping casks can frequently be designed to meet the requirements of the hypothetical accident without the addition of a structure or device whose sole purpose is the storage of energy or the reduction of acceleration forces. However, the addition of such a device is sometimes necessary, and this is the case when the mass of the closure is large when compared with the total mass of the cask. This situation can be further aggravated when the contents of the cask bear directly on the closure or when significant operating pressures are encountered. Since there is a practical limit to the level of integrity that can be built into a shipping cask, particularly in the closure and seal areas, protective devices will probably receive more widespread use in future designs.

Various protective devices have been proposed and used to protect shipping casks during impact, but most of these are bulky and massive. However, one such device, the tubes-in-pipe energy absorber, is compact, efficient, and insensitive to the direction of load application. As the name implies, this device consists of several small-diameter tubes held in close proximity within a pipe. The tubes

in-pipe absorber was devised primarily as a protective device for the seal and closure area of a specific shipping cask, but the unique characteristics of the device were recognized and the principle of the device was investigated to determine its applicability to the protection of shipping casks in general.

Prior to actual specimen testing, a literature search was conducted and some preliminary tests were made to determine the readily available optimum material of fabrication from the standpoint of energy absorption efficiency. Dynamic stress-strain curves presented by Clark¹⁵ indicated that 300-series stainless steel was probably the proper material, and a series of uninstrumented impact tests was made on single-tube specimens to verify this. A second series of tests was made to determine a feasible geometric relationship between the tubes and the outer pipe of the energy absorber. The results of these preliminary tests showed that an arrangement consisting of tubing with an outside diameter of 3/8 in. and a wall thickness of 0.065 in. contained within a Sch.-40 pipe was feasible.

Thirty tubes-in-pipe absorber specimens were impact tested, and the force-time data for each impact were measured and photographically recorded with the equipment previously described. A specimen length of 4 in. was selected since this was the maximum length which could be accommodated by existing test equipment. Specimens of four different geometries were impact tested with weights ranging from 126 to 498 lb dropped from a height of 30 ft. The size of the tubes (3/8 in. with a wall thickness of 0.065 in.) used was the same for each specimen, but the outside diameter of the 4-in.-long Schedule-40 pipe and the number of tubes contained within the pipe were varied. The pipe sizes tested were 1 1/4, 2, 2 1/2, and 3 in. OD; and the tubes used in each size pipe were 9, 22, 33, and 51, respectively. Typical specimens before and after impact testing are shown in Fig. 4.

The CIDR computer program was used to reduce the force-time data, but the force, energy, and deflection data developed were usable only for the particular pipe sizes tested. Scaling or modeling theory was applied to the data to develop relationships that would be valid for all reasonable pipe sizes that might be used for tubes-in-pipe energy absorbers. The force and energy data obtained from the impact tests on the 4-in.-long specimens were converted to unit force and energy or force per inch of length and energy per inch of length. Deflection was converted to a dimensionless parameter, percent deflection, by dividing its value by the actual outside diameter of the pipe. Force per linear inch was converted to a scalable quantity, defined as specific force, by dividing it by the outside diameter of the pipe. Energy per linear inch was converted to a scalable quantity, defined as specific energy, by dividing it by the number of tubes contained within the pipe. An existing graphical plotting computer program, XYPLOT,¹⁶ was used to plot curves of the specific force versus percent deflection and the

Fig. 4. Typical Tubes-in-Pipe Energy Absorber Specimens Before and After Impact Testing.

specific energy versus percent deflection. The curves shown in Fig. 5 are an example of this plotting technique.

Fig. 5. XY Plot of Specific Energy Versus Percent Deflection From Impact Test Data for Tubes-in-Pipe Energy Absorber.

Equations which adequately describe the mean of these plots were written. These are

$$U' = 0.121\lambda^{2.25} \quad (6)$$

and

$$F' = 18.7\lambda^{1.75} + 600, \quad (7)$$

where

U' = specific energy = energy per linear inch/number of tubes contained within pipe,

λ = percent deflection = deflection/outside diameter of pipe, and

F' = specific force = force per linear inch/outside diameter of pipe.

Equations 6 and 7 may be converted to the equations

$$U = 0.121\left(\frac{X}{D} \times 100\right)^{2.25}LN \quad (8)$$

and

$$F = \left[18.7\left(\frac{X}{D} \times 100\right)^{1.75} + 600\right]LD \quad (9)$$

by using the definitions

$$U' = \frac{U}{LN}, \quad (10)$$

$$\lambda = \frac{X}{D}(100), \quad (11)$$

and

$$F' = \frac{F}{LD} \quad (12)$$

where

U = energy stored in full-size tubes-in-pipe energy absorber,

L = length of pipe,

N = number of tubes contained within pipe,

X = actual deflection of full-size tubes-in-pipe energy absorber,

D = outside diameter of pipe, and

F = force applied to full-size tubes-in-pipe energy absorber.

Equations 8 and 9 are valid for predictions of the energy stored and the force applied to a tubes-in-pipe energy absorber of a particular geometry for a known or assumed deflection experienced as the result of an impact normal to the longitudinal axis of the outer pipe. When the impact is not normal to the longitudinal axis of the outer pipe of the energy absorber, the solution is more complex and is best accomplished by using computer methods. Examples of simple programs that may be used to accomplish this type of computation are given in Ref. 5.

IMPACT TESTING OF STEEL CASK MODELS

Twenty-two all-steel models of the proposed LMFBR shipping cask with the geometries shown in Fig. 6 were free dropped from a height of 30 ft in a controlled orientation so that the cask model impacted on a corner with the plane containing the corner and the center of gravity vertical. Three models were dropped in an end impact orientation. Force-time histories of the impacts were recorded with the load cell, oscilloscope, and camera equipment described previously. Acceleration-time histories were also recorded by using the accelerometer system for three of the 6-to-1 models dropped in the corner impact orientation and for the three 6-to-1 models dropped in the end impact orientation.

IMPACT ORIENTATION	SCALE FACTOR	WEIGHT (LB)	NO. OF TESTS	DIMENSIONS IN INCHES				
				OD	ID	OL	IL	T
CORNER	18.75/1	37.9	9	4.00	2.83	13.24	6.49	0.85
CORNER	9.375/1	295	5	8.00	5.65	26.50	13.00	1.71
CORNER	8.34/1	425	1	9.00	4.75	32 ^a	32 ^a	2.0
CORNER	6.0/1	1195	7	12.5	8.83	41.67	14.67	2.67
END	6.0/1	1195	3	12.5	8.83	41.67	14.67	2.67

^aTHE 8.34/1 MODEL WAS A CYLINDER WITH ONE CLOSED END, ONE OPEN END, AND NO MASS COMPENSATOR.

Fig. 6. Geometry of Steel Cask Models Impact Tested.

The CIDR computer program was used in conjunction with model theory to reduce the impact test data to scalable relationships, and the XYPLOT computer program¹⁸ was used to plot these relationships for the corner impacts, as illustrated in Fig. 7. The spread of the data is significant in that it is approximately +20% in the area where all the energy possessed by a large cask would be absorbed. However, these data are still the most valid tool the designer has at

Fig. 7. XY Plot of Force Versus Deflection for Steel Casks in Corner-Oriented Impacts.

his disposal for predicting the behavior of steel casks with geometries similar to those of the test models.

Observation of the plot shown in Fig. 7 indicates that a reasonable approximation of the data would be a straight line through the origin and the point $F = 53.0 \times 10^5$ lb, $\Delta = 3.6$ in. An energy balance obtained by using the expression

$$-U = -wh = W = f \int F d\Delta \approx \frac{1}{2} f F \Delta_{\max} \approx \frac{f F^2}{2m}, \quad (13)$$

where

U = energy stored in cask,

w = weight of cask,

h = height or elevation from which cask was dropped = 360 in.,

W = work required to decelerate cask,

f = scale factor = $W/248,000$,

F = maximum applied force,

Δ = maximum deformation, and

m = slope of approximate curve = $F/\Delta = (53 \times 10^5)/3.6 = 14.7 \times 10^5$;

may be used to estimate the forces and deflections experienced by casks with configurations similar to those of the models tested.

The cask models that were dropped in an end impact orientation failed to achieve a perfect end impact, and the test data obtained are not valid for a true end impact. A calculation based on the dimensions of the deformed area

indicated that the axis of the model was inclined approximately 3° . A basis for estimating the maximum forces involved in an end impact was formulated by computing the average force per unit area based on the measured final deformed area and the acceleration at the time the computer solution indicated that the velocity was zero. The resulting average force per unit area experienced by the three models dropped in the end impact orientation was 98,400 lb/in.² A similar calculation made for the models dropped in the corner impact orientation revealed that the average force per unit area for the 22 models was 90,530 lb/in.² Until better data are available, an estimate of the maximum force applied in a perfect end impact can be obtained by multiplying values such as these by the cross-sectional area of the cask. For the proposed LMFBR shipping cask with a scale factor of 1, the maximum acceleration computed in this manner would be 1781 G's.

CONCLUSIONS

Impact testing, particularly when conducted in conjunction with adequate instrumentation, computer data reduction techniques, and modeling theory, has been an invaluable tool in providing the answers to some of the complex questions associated with the free-drop portions of the regulations governing the design and performance of shipping casks. The empirical equations developed to describe the dynamic stress-strain properties of lead can be used to calculate the deformation of the lead shielding in a cask and the energy stored in the lead with accuracy levels superior to those previously possible. Tubes-in-pipe energy absorbers may be installed on full-size casks, and the energy stored and force applied to the absorber for a known or assumed deflection experienced as the result of an impact normal to the longitudinal axis of the outer pipe can be calculated within the bounds of engineering accuracy by using the relationships developed in conjunction with the impact testing program. The response of steel casks to the impact resulting from the 30-ft free drop can be determined by using modeling theory and the model test data.

It is anticipated that impact testing of models and cask components directed toward answering specific questions and developing empirical relationships will be continued during the next several years. A large portion of this testing will probably be directed toward demonstrating the adequacy of primary containment and the response of various neutron shielding schemes.

REFERENCES

1. A. E. Spaller, "Structural Analysis of Shipping Casks, Vol. 2: Resistance to Puncture," USAEC Report ORNL TM-1312, Vol. 2, Oak Ridge National Laboratory, September 1966.
2. H. A. Nelms, "Structural Analysis of Shipping Casks, Vol. 3: Effects of Jacket Physical Properties and Curvature on Puncture Resistance," USAEC Report ORNL TM-1312, Vol. 3, Oak Ridge National Laboratory, June 1968.
3. J. H. Evans, H. A. Nelms, and W. C. Stoddart, "Experimental and Analytical Developments at Oak Ridge National Laboratory for Demonstrating Shipping Cask Compliance With Federal Regulations," Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, USAEC Report CONF 681001, October 1968.
4. J. H. Evans, "Structural Analysis of Shipping Casks, Vol. 8: Experimental Study of the Stress-Strain Properties of Lead Under Specified Impact Conditions," USAEC Report ORNL TM-1312, Vol. 8, Oak Ridge National Laboratory, August 1970.
5. J. H. Evans, "Structural Analysis of Shipping Casks, Vol. 10: An Experimental and Analytical Evaluation of the Tubes-in-Pipe Energy Absorber," USAEC Report ORNL TM-1312, Vol. 10, Oak Ridge National Laboratory, 1971.
6. J. H. Evans, "Structural Analysis of Shipping Casks, Vol. 11: An Experimental and Analytical Evaluation of the Impact Properties of Mild-Steel Cask Models," USAEC Report ORNL TM-1312, Vol. 11, Oak Ridge National Laboratory (to be published).
7. F. C. Davis, "Structural Analysis of Shipping Casks, Vol. 9: Energy Absorption Capabilities of Plastically Deformed Struts Under Specified Impact Loading Conditions," USAEC Report ORNL TM-1312, Vol. 9, Oak Ridge National Laboratory, February 1971.
8. J. H. Vincent, "Experiments on Impact," Proceedings of the Cambridge Philosophical Society, Vol. X, p. 332, 1898-1900.
9. D. Tabor, pp. 120-138 in The Hardness of Metals, Oxford University Press, London, 1951.
10. A. W. Crook, "A Study of Some Impacts Between Metal Bodies By a Piezo Electrical Method," Proceedings of the Royal Society, Vol. A212, p. 377, 1952.
11. C. H. Mok and J. Duffy, "The Behavior of Metals at Elevated Temperatures Under Impact With a Bouncing Ball," International Journal of Mechanical Science, Vol. 6, pp. 161-175, 1963.
12. C. H. Mok and J. Duffy, "The Dynamic Stress-Strain Relation of Metals as Determined From Impact Tests With a Hard Ball," International Journal of Mechanical Science, Vol. 7, pp. 355-371, 1964.
13. J. M. Gondusky and J. Duffy, "The Dynamic Stress-Strain Relationship of Lead and Its Dependence on Grain Structure," Division of Engineering Technical Report 53, Brown University, Providence, Rhode Island, May 1967.
14. R. A. C. Slater, W. Johnson, and S. Y. Aku, "Experiments in the Fast Upsetting of Short Pure Lead Cylinders and a Tentative Analysis," International Journal of Mechanical Science, Vol. 10, pp. 169-186, 1968.
15. D. S. Clark, "The Influence of Impact Velocity on the Tensile Characteristics of Some Aircraft Metals and Alloys," National Advisory Committee for Aeronautics Technical Note No. 868, October 1942.
16. M. Tobias and L. Jung, "XYPIOT, An Elementary Graphical Preparation Routine for the CRT Plotters," USAEC Report ORNL TM-2867, Oak Ridge National Laboratory, January 1970.

Y ADMINISTRATION OF THE RADIOACTIVE MATERIALS TRANSPORTATION
SAFETY PROGRAM - OAK RIDGE OPERATIONS OFFICE

W. A. Pryor

ABSTRACT

Transportation safety administration within the Oak Ridge Operations Office is concerned with the implementation of domestic and international regulations. The progress and versatility in shipping container design and utilization is discussed. An account of the planning and experience of shipping irradiated fuel and control rods from a decommissioned reactor is presented.

The Oak Ridge Operations Office (ORO) is a major field office of the U.S. Atomic Energy Commission. Figure 1 shows the locations of the principal installations reporting to Oak Ridge, which have significant shipments of radioactive material. There are three gaseous diffusion plants producing enriched uranium located at Oak Ridge, Tennessee, and Paducah, Kentucky, which are operated by Union Carbide Corporation and at Portsmouth, Ohio, which is operated by Goodyear Atomic Corporation. The Feed Materials Production Plant which produces fuel elements for the Hanford and Savannah River production reactors is located at Fernald, Ohio, and is operated by the National Lead Company of Ohio. Also tied to this fuel element production chain is the Extrusion Plant located at Ashtabula, Ohio. The other major installations located in Oak Ridge include the Y-12 Plant, which is a weapons production facility, and the Oak Ridge National Laboratory, which is a major radioisotope producer, a reactor research and development facility and holds the national stockpile of U-233. These installations are operated by Union Carbide Corporation. Also under ORO are the AEC operated New Brunswick Laboratory in New Brunswick, New Jersey, and the Puerto Rico Nuclear Center in Rio Piedras, Puerto Rico, which is operated by the University of Puerto Rico. The second major installation in Puerto Rico was the Boiling Nuclear Superheater Reactor (BONUS), a joint venture of the AEC and the Puerto Rico Water Resources Authority, which was shut down in 1966. ORO is concerned with the administration of radioactive materials transport from these installations.

In the few years that we have had our upgraded radioactive materials transport regulations, considerable change has occurred in transport safety administration. Initially, the principal concern was the "tooling up" period

where new containers had to be designed, tested against the accident test series, approved under AECM-0529 and permitted by the Department of Transportation. After those first containers were approved and permitted, most of the difficulty and confusion had been eliminated.

From September 1967 through April 1971, the prime contractors under Oak Ridge Operations have compiled an impressive number of approved shipping containers. Table 1 summarizes the Oak Ridge experience:

Table 1. Summary of Containers Approved by ORO

Type of Radioactive Material	<u>No. AECM-0529 Approvals</u>							
	FY '68		FY '69		FY '70		FY '71 - 4/71	
	Initial	Rev.	Initial	Rev.	Initial	Rev.	Initial	Rev.
Fissile - unshielded	5	-	6	6	3	1	2	4
Fissile - shielded	3	-	4	3	5	-	-	-
Non-Fissile - shielded	8	-	4	4	2	-	1	2
Subtotals	16	-	14	13	10	1	3	6
Totals	Initial - 43		Revisions - 20					

We were also concerned with the upgrading of old containers and/or the building of new containers in order to meet the requirements of the then new regulations. Gradually, we are broadening the approvals so that our shipping containers are now quite versatile. Four overpacks and shipping containers approved by ORO under AECM-0529 will illustrate some of this versatility. Table 2 summarizes the initially approved usage and the subsequent applications:

Our progress in improved transportation safety can also be measured by the accident experience of the industry. For Type B containers, the number of accidents overall in a transportation environment has been small.¹ A "before" and an "after" implementation of our current regulations will demonstrate how far we have moved. Figure 3 shows a "before." This is an empty birdcage, which was received prior to 1968 at the Portsmouth Gaseous Diffusion Plant to be loaded with a 5-inch diameter cylinder of UF₆. The cylinder was held under pressure between the two cups. The birdcage was junked.

Figure 4 shows the "after" sequence. It involves an Oak Ridge Gaseous Diffusion Plant type protective package, containing an empty 30-inch diameter UF₆ cylinder during an actual fire at a Knoxville, Tennessee, motor carrier

Table 2. ORO Broad Usage Shipping Containers

<u>Part A - Overpacks</u>			
DOT SP No.	Design Instal.	Initial Usage	Subsequent Applications
4909 ^a	ORGDP	Overpack for family of 5-in., 8-in., 12-in., and 30-in. cylinders of UF ₆	1. 50,000 Ci of Tritium and all Noble gases. 2. Highly enriched U-metal scrap 3. Scrap Fuel Element 4. Unirradiated BONUS fuel elements 5. Slightly enriched U-metal for critical experiments 6. UF₄ paraffin blocks used in critical experiments
5752 ^a	ORNL	Overpack for Sugarman Radio-isotope Cask	1. New Sugarman Cask 2. Gamma Field Irradiator - ¹³⁷Cs 3. Thermoelectric Generator - ⁹⁰Sr 4. NBL Hot Sample Cask^b 5. PRNC Irradiator - ¹³⁷Cs
<u>Part B - Shipping Containers</u>			
5765	ORNL Y-12 ORGDP	Vermiculite container for highly enriched uranium metal and compounds-unmoderated	1. Unmoderated Pu 2. Unmoderated ²³³U 3. Uranium Moderated to H/²³⁵U-15 4. U up to 10% ²³⁵U (proposed)
5795	ORNL	Foamglas container for ²³³ U solution	1. Unmoderated Pu, ²³⁵U and ²³³U 2. Moderated Pu and ²³⁵U

^aThese overpacks have been submitted to the Department of Transportation as specification containers.

^bFigure 2 shows the NBL Hot Sample Cask, which was specially designed by ORNL, in a Sugarman type overpack.²

terminal in March 1969. There were two protective packages in the truck. This incident was not highly publicized with the photograph being printed on Page 5 of the newspaper. The \$20,000 statue had more descriptive words than the protective packages.

Figure 5 shows the more damaged protective package and its undamaged cylinder. Note the burned gasket. This one was cleaned and regasketed prior to the shipment. The other required only superficial cleaning and was shipped out immediately with a filled cylinder.

One of the most interesting situations involving transportation safety was the movement of irradiated elements and control rods from a reactor, which was being decommissioned. The basic problem was to ship from the BONUS Reactor at Rincon, Puerto Rico, 64 boiler (2.4% ^{235}U) elements, 36 superheater (3.25% ^{235}U) elements, 148 shim rods and 10 control rods, to the U.S. The fuel elements were to be shipped to Nuclear Fuel Services at West Valley, New York, for recovery, and the control rods were to be shipped to ORNL for burial. Figure 7 indicates part of the rather complicated and involved logistics. At least two modes of transport would be required to make the move. In addition, the fuel and control rod removal would intimately affect the overall schedule for decommissioning. The Laboratory and University Division, ORO, coordinated the entire operation. Staff assistance was provided by the ORO Safety and Environmental Control and the Supply Divisions, while engineering support was provided by the ORGDP Engineering Division.

The overall problem was divided into five basic tasks:

1. Cask Selection - criteria included:
  - a. ability to be approved under AECM-0529
  - b. economics
  - c. logistics
2. Preparation of Cask and Equipment - criteria included:
  - a. Engineering Design
  - b. Procurement
  - c. Fabrication of fuel baskets, control rod basket, cask modification, tools and handling equipment
3. Transportation Arrangements
Arrangements with commercial carriers
4. Documentation
  - a. Safety Analysis
  - b. AECM-0529 Review and Certification
  - c. DOT Special Permit

5. Shipment Execution - audit for:

- a. Safety
- b. Transportation Management
- c. Accountability

Tasks 2 through 4 were completed concurrently.

The two casks which were considered were the HNPF Six-Element Irradiated Fuel Casks (Hallam) and the ATCOR Vandenberg Cask. Table 3 shows some of their characteristics.

Table 3. Cask Characteristics

	<u>HNPF</u>	<u>ATCOR-VNDB</u>
Shield Thickness	Lead Equivalent - 8-1/2 in.	7-3/4 in. Lead
Cavity	18-1/2 in. dia. x 197-1/2 in. long	36 in. dia. x 117 in. long
Tare Weight	71,000 lb.	57,000 lb.
Loading	Top	Bottom
Authorized Transport	Rail	Motor Freight ^a and Rail

^aThe ATCOR Cask with its trailer could be used in the "drive on - drive off" service available by sea to Puerto Rico.

During the cask selection period, the ATCOR Cask was physically located at the Dresden, Illinois, Reactor. A team from ORO visited the site to see the cask and to see how the Dresden crew handled the cask. We learned several very important lessons that alleviated problems later at the BONUS. Figure 7 shows such a lesson where the cask lifting yoke would not fit on the crane hook. While this could have been a minor problem at Dresden, it would have been a major problem at BONUS. Specific checks were made to assure that the yoke would fit on the BONUS crane hooks. In this particular case, the opening in the yoke was enlarged; however, this necessitated verification that the yoke had not been weakened.

From the standpoint of both time and economics, it became fairly obvious that the ATCOR Cask was the prime candidate for the job. The Hallam Cask would require additional modifications and significantly more trips to complete the move. Thus, the ATCOR Cask was formally selected.

The Chicago Operations Office had a current contract with ATCOR for use of the cask, and the contract was extended to ORO to cover the BONUS move. The safety analysis was also under review and was expected shortly. Since

there was some question in regard to the structural integrity of cask under accident conditions, ORO sent a team of structural engineers to work with the Chicago Operations Office, Battelle Memorial Institute and Franklin Institute, and ATCOR, Inc., in resolving any questions.

The Engineering Division of ORGDP provided technical support to design, evaluate, and fabricate both a fuel basket and a control rod basket.^{3*4} As a part of the evaluation, a critical experiment was conducted at the Y-12 Critical Experiments Laboratory with unirradiated fuel. The fuel elements were spares which were shipped from BONUS before the irradiated fuel was shipped. Figure 8 shows the fuel elements at the Laboratory. Figure 9 shows the cask covered with a sunshade on its heavy duty trailer prior to leaving Oak Ridge. Auxiliary equipment included the lifting yoke, the turning fixture, and a spreader base. The spreader base was used to spread the weight of the cask over a larger area of the reactor pool floor. Figure 10 shows the fuel basket sitting in the storage pool at the BONUS site. The grid in the basket contained BORAL, a neutron absorber, so that the actual shipment was made as Fissile Class I.

The control rods were to be buried at ORNL in a disposable basket. Dry runs were conducted to minimize problems with both the fuel basket and the control rod basket. Figure 11 shows the empty cask and the control rod basket at the burial site during a dry run. The basket was lowered by a crane into a deep hole and covered. The robot was brought to the site and included in the dry run. This unit is operated from a control panel, which in this case was approximately 1,000 feet away. Note the two television cameras. In the actual burial operation, its services were not required, although it was on standby. This unit, which is operated by the Y-12 Plant, is part of the emergency equipment maintained by ORO.

Table 4 lists the shipments from Puerto Rico and the material shipped.

During the period of the move, there were several "roadblocks" which affected the schedule. A threatened, and subsequently an actual, longshoremen's strike required obtaining clearance for temporary storage of the loaded cask at the U.S. Naval Base in San Juan. The actual strike lasted about 2-1/2 months which accounted for most of the delays involved. A breakdown at sea required the ship to be towed in once. This necessitated a changeover to a second steamship line.

The move was made with no significant radiation and nuclear safety problems. It logged over 30,000 miles and transportation costs were about \$38,000.

Table 4. BONUS Shipments

Shipment No.	Date	Material Shipped		
		Fuel Shipments		Control and Shim
		Boiler	Superheater	Rods
1	12-68	21		
2 ^a	3-69	21		
3	4-69	22		
4 ^b	5-69		12	
5 ^b	6-69		12	
6	7-69		12	
7 ^c	9-69			x

^aDelay due to strike of Longshoremen.

^bAll sea shipments by Seatrain Lines except shipments 4 and 5 by Transamerican Transport; change due to malfunction of equipment.

^cShipped to Oak Ridge for removal of fuel basket and installation of control rod basket.

As a result of the experiences at ORC, three specific recommendations are made:

1. Radioactive material shipping containers should be designed and evaluated for maximum utilization.
2. The smaller containers should be permitted for all modes of transportation under domestic and IAEA regulations.
3. As a part of the basic critical experiments programs for development of new reactor fuel configurations, a specific portion of the program should be devoted to safety outside of reactors, particularly for transportation safety.

ACKNOWLEDGEMENTS

The author gratefully acknowledges the helpful assistance of Messrs. J. A. Lamb and L. G. Blalock, members of the Transportation Branch, ORC Supply Division.

REFERENCES

1. A. W. Grella, Accident Experiences in the USA Involving Type B Packaging, International Atomic Energy Agency "Seminar on Best Requirements for Packaging for the Transport of Radioactive Materials," Vienna, Austria, IAEA/EM-147/19 (February 2, 1971).
2. J. H. Evans, Analysis of the New Brunswick Laboratory Sample Cask, ORNL-TM-3298 (March 1971)
3. C. E. Newlon, Nuclear Criticality Safety Analysis of BONUS Irradiated Fuel Assemblies in ATCOR Shipping Cask, K-L-3045, (October 1, 1968).
4. C. E. Newlon and J. C. Bailey, Supplemental Radiation Shield for 1100-Curie Cobalt Shipment, K-II-37 (May 23, 1969).

Figure 1. Oak Ridge Operations Installations.

Figure 2. New Brunswick Laboratory Hot Sample Cask.

Figure 3. Old Style Birdcage - Damaged

Figure 4. Motor Carrier Terminal Fire - Knoxville, Tennessee, March 1969

Figure 5. Protective Package after Fire

Figure 6.

Figure 7. Dresden Reactor Crane Hook

Figure 8. BONU, Fuel Elements at Y-12 Critical Experiments Laboratory

Figure 9. ATCOR Cask

Figure 10. Fuel Basket in Storage Pool - at BONUS Reactor

Figure 11. ATCOR Cask and BONUS Control Rod Basket at ORNL Burial Site

✓ THE RESPONSIBILITIES OF THE CONSIGNOR
AND THE CONSIGNEE

K. R. Schendel

ABSTRACT

The responsibilities of the consignor and the consignee with respect to the packaging and transportation of radioactive materials, are divided into three general categories--legal, practical and social. The legal responsibilities are relatively simply stated. However, the regulatory requirements imposed on the consignor and the consignee in preparation for and follow up after the actual movement also are briefly enumerated and indicated to be quite extensive. The practical responsibilities, as discussed, are generally related to financial considerations and to specific properties of or conditions associated with radioactive materials. The social responsibilities are described as relating principally to an overriding requirement for a degree of safety adequate to satisfy the professional standards of the consignor, particularly when those standards exceed the legal minimum. The responsibility for an effective, positive attitude to help to combat the unwarranted distrust of radioactive materials that appears to inflict the general public is also asserted.

The responsibilities of the consignor and the consignee, in connection with the packaging and transportation of radioactive material, can be reduced to the same essential terms as the responsibilities of any individual with an assignment to carry out, that is, to "Do the job right." Now this is the crux of the matter, the "ultimate truth". However, it really is of value only to those who already know what is required of them, but who can still profit from the reminder. To the newcomer, the uninitiated, it merely provides the questions, "Who must do the job? What is the job? When is it done right?"

The "Who" part is very straightforward. My subject clearly specifies it,

although, at this time, I would like to establish my definition of the consignor and consignee. As a convenience, I propose to consider them as "black boxes" (to use some electronic engineering terminology.) What I mean is that the method used by the consignor to carry out his assigned responsibilities is not defined. Inside his company organization, there may possibly be a traffic group, a health physics group, a shipping department, etc. Or just as possibly, a single person may have to do everything. Either way, for purposes of this discussion, whatever must be done by the consignor or consignee, does in fact get done in some manner.

The "What is the job" part is almost as simple--to assure the physical transfer of some radioactive material from one place to another.

So that leaves the final question, "When is it done right?" The various actions which contribute to the answer to this question are what I choose to regard as the responsibilities of the consignor and the consignee, and are what I shall discuss in the remainder of my presentation.

It seems logical to divide the responsibilities of the consignor and the consignee into three major categories--legal responsibilities as established by applicable government licensing regulations; practical responsibilities; and social responsibilities. I have deliberately placed a limitation on the term "legal responsibilities" because I don't intend to include such considerations as financial liability, torts, and contracts.

The legal responsibilities of the consignor are deceptively simple and straightforward. He is required to present to the carrier, an acceptable package, properly documented and properly labelled. Under certain circumstances he may be required to specify the need for placarding the transporting vehicle, and to provide some special loading assistance or some special instructions.

Correspondingly, all that the consignee has to do is accept, or occasionally unload, the package, and possibly to assure the carrier that there has been no contamination of the vehicle.

There have been previous discussion of the various regulations promulgated by the Atomic Energy Commission and the Department of Transportation. Various states also have equivalent and compatible regulations, where applicable. So when you consider the multiplicity of rules that exist, it will come as no surprise to you when I say that the consignor works like a dog to fulfill his

his responsibilities.

First, I'm simply going to list the sequence of actions required of him, basically by various government regulations. Then I'll go back and discuss each item a little more thoroughly. The list looks like this:

1. Ascertain that consignee is properly authorized to receive the material.
2. Define the package.
3. Preplan the route.
4. Select a carrier.
5. Prepare the package.
6. Load (or supervise the loading of) the vehicle.
7. Provide the necessary documentation.
8. Provide special instructions (if required).
9. Follow in transit (if required).
10. Process material transfer papers (if required).

This list is respectable, but is it really as impressive as you have been led to expect? If not, just wait.

Item 1 on our list is perfectly straightforward. The AEC and comparable state licensing regulations state that no radioactive material subject to the regulations may be transferred to a person not authorized to receive it. Small quantities and materials in forms that are either exempt or generally licensed present no problems. For any other situation, the consignor must contact the consignee and obtain some type of adequate assurance that the consignee is duly authorized to receive the material.

Item 2 is the sleeper on our list. "Define the package" it says, just three little words. However, behind them lay all of Part 71 of the AEC's regulations, and almost all of the Hazardous Materials Regulations of the DOT. Now, not everyone is going to be concerned with all of these regulations. Many consignors have a quite simple, repetitive operation which soon becomes routine. On the other hand, there is always a first time, and surprisingly often a change in activity, form, weight, volume, destination or some other aspect of the packages makes a shipment a "first time."

The word package, in the industry, means both the contents and the packaging (or container). Different loads, different quantities, different materials, even though they are all in the same packaging, make different packages. So, let's look at the material. Obviously, it's radioactive material, but is it anything more--a corrosive liquid, a corrosive solid, or possibly a flammable solid? These additional classes of hazardous materials also have their packing and labeling requirements. If it is solely radioactive, is it a small, readily packaged, Type A quantity; a larger, Type B quantity that requires a package that will withstand the hypothetical accidents; or a large quantity that requires specific licensing,--or is it an exempt quantity? Special form material (material that doesn't disperse readily) and low specific activity material are subject to less strict requirements. Don't forget to determine which Transport Group normal form material belongs to. Or maybe it's fissile material that involves nuclear criticality safety controls; what is its enrichment, how much can be put in a single package, what is the material concentration, or the moderating ratio?

When the expert gets all those questions straightened out, the consignor can start looking for packagings. For non-fissile materials in small, Type A quantities, it's fairly simple. Numerous specification packagings are available, such as 17H metal drums, 15A, B, C or D nailed wooden boxes, or the Spec 12 series fiberboard boxes. However, once the Type A quantity limits are exceeded, the pickings become very slim, and usually special permits are required. Then, a whole series of questions arise, such as, "Who has an approved packaging? Can I get listed on the permit? Does the permit have to be modified for the contents I'm interested in shipping? How long will it take to get a revised permit? Has the AEC's Division of Materials Licensing approved it? Where can I procure enough of the containers? What labels do I use?" For a shipment of fissile materials, "Should it be shipped Fissile Class I, Fissile Class II, or Fissile Class III? How many packages make a load?" The questions mount up again.

Once the consignor has assured himself of an acceptable package he really is over the hump. Item 3, preplanning the move, merely involves such items as "Does the special permit (when required) limit the mode of transportation which may be used? Are there any bridges, tunnels, turnpikes, etc. enroute which forbid or limit the movement of radioactive materials? Will the load exceed local weight, height, or width limitations on roads or bridges, etc.?"

Item 4 involves selecting a carrier whose tariffs permit him to carry the material involved, within any routing limitations that may have been determined in Item 3, to the destination desired. This may develop into a problem if a transfer to another carrier is involved, as we'll see in Item 9.

Item 5 reveals how well the consignor carried out Item 4. Are the packagings suitable? Is each one complete? Is each one in usable condition? How can I repair that puncture (or crack or whatever)? Are the contents within any expressed limitations on activity, form, thermal output, quantity, dryness or other property? Is it properly sealed? Does it meet required external radiation levels, or surface contamination levels? What Transport Index must be assigned? (The Transport Index is an indication of either the external radiation levels associated with a package or the nuclear criticality reactivity of a package. It is determined such that when a vehicle loading is maintained at or below 50 units, no further radioactive materials controls are required.) Ultimately, Item 5 determines whether or not the package can be certified.

Item 6 is intended to assure that a carrier observes the 50 unit rule (that was just described) on a load of radioactive material. It is also significant if the shipment is going "exclusive use of vehicle" to take advantage of certain higher radiation levels allowed by DOT regulations, or if it is a Fissile Class III shipment. The Fissile Class III designation applies to the vehicle loading as a whole, where the number, arrangement, and spacing of the packages on the vehicle are important considerations in the nuclear criticality safety of the array. Of course, simple considerations like avoiding leakage by assuring proper orientation and tie-down of the packages can be involved. Furthermore, at this point a decision must be made on whether placarding is required on the vehicle and if so, what placard should be used.

Item 7 is no big problem for the experienced shipper, or even for the inexperienced shipper who reads his DOT regulations. He hands the carrier's representative a bill of lading properly describing the load, the quantity, the labels required, and the placards required, along with a certification that everything is in accordance with regulations. Certain transportation modes may require that copies of any special permits which are involved also accompany the shipment.

Item 8 is generally applicable only when the material involved in the shipment requires the exclusive use of the vehicle. If the material is subject to the requirements of Part 73 of the AEC's regulations (the Part which deals

with the physical security of shipments) signature service on transfers, call-in procedures and times, and other very important instructions may have to be prepared, given to the carrier, and impressed on him to assure some degree of understanding of their importance.

Item 9 also is basically restricted to security shipments under 10CFR73, or to accidents and/or misroutings. Security shipments are too new, few and unusual to warrant extended discussion. Nevertheless, it should be indicated that the AEC's intent, as expressed in Part 73, is to require a shipper to undertake to arrange with a carrier to move a shipment using certain preselected connecting carriers and predetermined routes and to arrive at the destination at a specified time. This intent may well be in direct opposition to a common carrier's normal legal freedom from any such restrictions. Presently, the best that the shipper can do, apparently, is make his arrangements, cross his fingers, and hope. Accident reporting and cargo handling are really up to the carrier. However, the consignor normally retains an interest and responsibility in a shipment of this type until it reaches its destination. In the event of an accident, particularly, the shipper is peculiarly qualified to provide advice and assistance in handling damaged packages.

Item 10 is a routine requirement applicable to shipments involving more than 1 gram of special nuclear materials, and more recently, to large amounts of source materials (over a metric ton). These reports are part of the AEC's financial and safeguards accountability systems.

With this expanded understanding, let's go back and review those ten basic steps which the consignor performs to meet his legal responsibility.

1. Assure consignee's authorization
2. Define the package
3. Preplan the route
4. Select a carrier
5. Prepare the package
6. Load the vehicle
7. Provide documentation
8. Provide special instructions
9. Follow in transit
10. Process transfer papers

There is an adage that says it is more blessed to give than to receive. However, in transporting radioactive materials, it seems to be more blessed to receive than to ship. Certainly, the consignee's expressed responsibility to accept the package and possibly certify the absence of contamination on the vehicle can be fulfilled in far fewer steps.

The consignee's activities can be expressed as:

1. Provide authorization to receive the material.
2. Monitor for arrival of the shipment.
3. Survey load and check for damage.
4. Unload (or supervise the unloading of) the vehicle.
5. Survey and release the vehicle.
6. Process transfer papers (if required).

The consignee's activities are relatively simple. Item 1 merely requires that, at some time before the shipment is made, the consignor be provided a license number, or a copy of a license or whatever similar proof is required to satisfy him. Of course, if the consignee doesn't have a license and he wants the material, he'd better get busy.

Item 2 is urgent primarily when the shipment is subject to Part 73 of the AEC's regulations. Contact with the consignor should provide a gradually refined estimated time of arrival, and if the vehicle fails to put in its appearance promptly, specified notification to the consignor must be made. Of course, any shipment which is expected, but simply doesn't arrive, is eventually going to be questioned. No one is going to completely ignore a missing shipment of radioactive materials, it's just that sometimes it takes a little bit longer.

Item 3 is performed primarily as a routine radiation protection measure. Some data must be accumulated concerning the possible radiation exposures which would be received by the consignee's employees in the course of handling the package. Radiation levels or surface contamination above specification limits could also be indications of in-transit damage to the package.

Item 4 is equivalent to the Item 6 performed by the consignor, and generally is required under the same circumstances. Consignee unloading is a requirement for certain "exclusive use of vehicle" shipments and for shipments

of fissile radioactive material where nuclear criticality safety is a consideration.

Item 5 can present the nastiest problem to the consignee. The DOT specifies that radiation and radioactive contamination levels in the carrier's vehicle be reduced below specified levels before the vehicle can be used for non-radioactive applications. In the vast majority of movements, there is no evidence of residual radioactivity. In those few instances where it occurs, however, decontamination can be an expensive, tedious process.

Item 6, the final item on the list, again is the complement of Item 10 on the consignor's list. It is the formality which says, "I received the material you sent. I accept responsibility for it."

The items I have discussed are representative of the actions required to appropriately fulfill the responsibilities imposed by law, rule and regulation on the consignor and the consignee. I offer no warranty whatsoever that my descriptions are complete. What I have attempted is merely to create an impression of the complex nature of the requirements imposed by law on persons connected with the shipment of radioactive materials, due to their classification as hazardous materials. Now, let's turn to what I referred to earlier as the practical aspects of the responsibilities of the consignor and the consignee.

The practical aspects basically revolve around money. In our economy, practically everything revolves around money. It seems self-evident, but always bears repeating, that the consignor should endeavor to accomplish his required task with a minimum of expenditures. Now please, I'm not saying, "Ship it the cheapest way." Notice that I said "accomplish his required task." Furthermore, if you recall, at the start of my talk I said the consignor was to be treated as a black box, without specifying any of his internal operations. The consignor as a whole obviously has many needs, some measurable in dollars, such as storage space; and some measurable in customer good will, such as meeting delivery schedules. The preferred practical shipping choice balances all of these internal needs so that the transportation of the material involves the minimum overall expenditures consistent with management objectives.

Other practical considerations are associated with the nature of the radioactive material itself. Many members of the audience, I'm sure, remember the reaction of the industry when it was proposed to pretty much exclude by-product materials from transport by air in passenger carrying aircraft. Our

industry developed a highly effective technique of using the regularly scheduled passenger airlines to move byproduct materials that have very short half-lives. Such materials are particularly popular in many fields because a relatively short time after being applied, they have decayed to the point that the samples can be considered to be no longer active. But as the result of this same property, any delay during the transfer between the producer and the customer involves either ordering unreasonably large quantities or trying to use orders-of-magnitude smaller activities in the experiments. In view of this very real, very practical problem, the proposed regulations were modified to assure the continued movement of such samples by passenger aircraft. That is an ideal example of how the transfer mechanism is partly dictated by the material's properties. Another obvious example is the transportation of spent reactor fuel between the reactor and the reprocessing plant. The lead encased casks used for such purposes range up to 75 tons in weight. Their movement by air, or possibly, even by highway, is out of the question. Special rail cars or barges are indicated here.

Having discussed some of the legal and practical aspects of the responsibilities of the consignor and consignee, what can be said of the social aspects? And possibly, "social" is not even the proper word. The first and most important item under this heading is safety. The primary purpose of the majority of the legal requirements that have been imposed is to attempt to assure that an adequate degree of safety must be exercised during the shipment. Further, we have all been taught that safety is good business and that safety is everybody's business.

But safety is a particularly significant responsibility of the consignor of radioactive materials. (And I might add, of the carrier also.) When it is necessary to satisfy his own professional opinion of what constitutes a "safe" package, a consignor should feel obligated to exceed the minimum requirements of the regulations or to spend extra dollars.

This specific obligation is partly due to the nature of what might be called the "radiation hazard." Most hazardous materials represent a stored energy situation, that is, an explosive does or doesn't explode or a flammable material does or doesn't burn. Furthermore, after the explosion or the fire, the particular hazard no longer exists. Most of what are left in the way of hazardous materials produce a definitely noticeable (and usually unpleasant) physiological reaction such as pain, nausea, unconsciousness and possibly death. For all of these materials, once they have been identified, emergency actions

following an accident can proceed based on what the emergency crew can perceive using the normal five senses. In addition, suitable corrective or protective devices for such hazardous situations (such as respirators, fire extinguishers, insulated or non-sparking tools, etc.) are relatively standard equipment for emergency crews.

As classes, only radioactive materials and etiological materials give no warning, produce no immediate symptoms, and exhibit relatively prolonged and continuing effectiveness. Of these two classes, the etiological materials usually involve some type of physical contact, and the eventual physical reaction is customarily responsive to some type of medical treatment if diagnosed in time. As a class, only radioactive materials present a continuing hazard which requires special instruments and expertise to determine the extent to which they can be safely approached, or moved; and which can produce a range of effects which may not be readily detectable, let alone treatable.

Viewed in this light, radioactive material transfers require every appropriate safety device dictated by good shipping practices. However, it is useless, and possibly unsafe to keep on piling up precaution after precaution beyond that point.

The implications of this radiation hazard, plus radioactive materials' "Hiroshima" reputation, lead to the second aspect of the social responsibility I would assign to all persons who process, transfer or use radioactive materials. This responsibility can roughly be described as a public relations job. I feel that a continuing expansion in the use (and therefore, in the transportation) of radioactive materials requires that the general public be educated to the degree of safety that our industry maintains. Every individual in our industry, by which I mean the nuclear industry in all its phases, has a responsibility to help to overcome the bad public image the industry is cursed with.

We must strive to reduce to an absolute minimum the hysteria (and that's a term I use advisedly) that has become associated with radioactive materials. No one denies that they are properly classified as hazardous, but with due care and understanding it is what I consider a normal, minimum degree of hazard. Certainly, I feel much safer in one of our plants, surrounded by radioactive materials, than I do crossing a busy city street.

The most frequent and intimate association with radioactive materials that most members of the public endure (?) is while the material is outside of its

carefully controlled application sites during its transportation between sites. One significant exception would be that group of people who have been introduced to radioactive materials in the course of medical diagnosis or treatment. It is notable that in this instance, the authority of the M.D. degree and the general mystique associated with materia medica really tends to overshadow the radiation aspects. This same aura of professional confidence must always be communicated from the consignor and the consignee to the carrier's representative and, hopefully, on to any other individual he contacts.

I feel very strongly that every personal contact, every interface (if you will) between the consignor and the consignee, and any other person should involve this effort to sell the acceptability of the material we handle.

In summary, then, my interpretation of the responsibilities of the consignor and consignee includes legal, practical and social aspects. The legal aspects are typified by the various written rules and regulations. There are a confusing plenitude of such regulations, but careful research and normal good sense can contribute to effective compliance. The practical aspects involve mostly a financial balance, which customarily includes judgment factors requiring knowledge and experience of phases of the consignor's operations other than simple transportation charges. Finally, the social aspects deal primarily with assuring the safety of the shipment to the maximum practical degree, to provide continuing convincing evidence to all the rest of the world that the nuclear business is the world's finest.

DEVELOPMENT OF DOCUMENTATION FOR THE TRANSPORT OF RADIOACTIVE MATERIALS

J. Fairey

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

✓
TRANSPORTATION AND HANDLING OF
RADIOACTIVE PHARMACEUTICALS AND RADIOACTIVE CHEMICALS

S. H. Sanger

ABSTRACT

The movement of Radioactive Pharmaceuticals and Radioactive Chemicals, either by surface or air or a combination thereof, has been a monumental problem from the inception of the original processing of nuclides due to the short half-life of the commodity. Many precautionary measures must be taken to insure a safe and timely delivery of these radioactive products. From the moment the nuclide has been processed, the problem of packaging and transporting, coupled with the services needed to insure fast portal to portal delivery, is extensive. Both the surface carrier and the air carrier must handle this material with precision-like speed. Packaging, handling, shipping and transporting are an integral factor in connection with the supply of this product.

INTRODUCTION

During the past five years, we have specialized in the timely and efficient delivery of medical nuclides. Originally we transported radioactive generators that had a five-day half-life and had a two-day pre-calibration factor built therein. From this early beginning, we moved through the full spectrum of nuclides, which included sodium pertechnetate with a six-hour half life. Within the past six months, we have transported the newer and even shorter half-lived nuclides, such as F^{18} with its

unusually short half-life of less than two hours. As the scope of the Radioactive Pharmaceutical Industry and Nuclear Medicine progressed, a larger geographical area had to be covered. This, in itself, has created many logistical problems. In previous years, nuclides with a long half-life have come into disfavor due to the lingering after-effects within the human body. Therefore, the future of the more exotic short half-lived nuclides is most promising, but the problems connected with their distribution have become more difficult.

In the early stages of transporting radioactive pharmaceuticals, the people connected with packaging, shipping and transporting were completely unfamiliar with the commodity they were handling. During these early stages, many of the truck drivers, airline pilots, cargo handlers, cargo agents and various other people connected with the radioactive pharmaceutical products had great misgivings about any commodity whose derivative was "radioactive". This superstition has continued unabated throughout the past six or seven years. However, it is encouraging to note that with the aid of training films sponsored by various manufacturers and with numerous safety meetings throughout the country, the average person coming in contact with radioactive pharmaceuticals has become more oriented with the product, and, therefore, handles his particular phase of this industry at a more efficient level.

The very fact that all nuclides are delivered on a nationwide basis prior to 8:00 AM, is a tribute to the many people who have taken the trouble to learn the proper handling and transportation of radioactive pharmaceuticals. The hospitals, research centers, universities and government facilities operate more efficiently due to the fact that they receive the source material on a timely basis.

We frequently take for granted the fact that the product arrives on time without ever considering the myriad steps needed to effect this delivery. It is necessary for the shipper to notify the carrier prior to shipment, either by telephone or teletype, the origin of the shipment, the name of the airline, flight, airbill number, name and destination of the consignee and weight and number of pieces involved in the shipment. In order for the shipper to supply the carrier with this informa-

tion, it is necessary for him to make arrangements for reserved air freight (RAF) space. He must know how much lead time is necessary for the shipment to be at the airline prior to takeoff. In many instances, this varies from one to three hours. The coordination between the Order Department, production line, Shipping Department and surface carrier used to transport the material to the airport is one of great responsibility. In the event that there are production problems or the material does not pass the required tests, it is incumbent upon the shipper to relay this information to the destination carrier. Often the shipper or shipper's agent must get proof on board from the airline. Weather conditions are always creating problems, and, therefore, certain backup flights, destinations and airlines must be considered prior to delivery of the nuclide to the originating airport. With each shipment, the restricted articles manifest must accompany the package. This indicates the particular nuclide and the activity contained therein. It is the shipper's responsibility to see that he does not exceed the transport index allowed for one airline at one time. In the event that the activity is too great and the transport index is too high, it is necessary for the shipper to book space on another flight or conceivably another airline.

Since the receiving carrier has been notified of the impending arrival of a shipment, it is necessary that he check with the airlines repeatedly to ascertain if the shipment is on the manifest of the scheduled flight. Also, he must take into consideration any delay due to weather or recovery time. Recovery time can vary from one to five hours, depending on the volume of air freight that the airline is handling, which in many instances can vary due to the time of year, holidays or type of aircraft in use. Invariably, a radioactive shipment that comes on a freighter aircraft will be delayed in recovery at the destination. A very important phase of the overall transportation picture in the pharmaceutical industry is the transshipping of a shipment from a major air carrier to a second or third level air carrier. The surface carriers at the major destination stations throughout the United States have the responsibility of seeing that these shipments are carried out effectively in order to supply hospitals and other institu-

tions in remote sections of the country.

The surface carrier at the destination is the all important cog in this very large wheel of various services. The processor can pre-calibrate his material to insure against any eventuality in the tardiness of the distribution of this product. Also, the airline can negotiate 3,000 miles in five hours, but the surface carrier conceivably must cover 150 to 200 miles on the ground in a maximum time of four hours. This factor alone necessitates the pre-routing of each vehicle from this carrier. Pre-routing has certain built-in safety factors that in the event of an accident or breakdown, there will be a minimum of difficulty in locating the vehicle. Each delivery must be taken in exact rotation as described in the driver's pre-routing manifest. Surface carriers who do not have two-way radios must require the driver to telephone the dispatcher at various checkpoints along the route. At all times, the driver of the vehicle must have in his possession the restricted articles manifest and should be aware of the nearest decontamination or radioactive waste-disposal location in the event of an accident or in the event a nuclide package becomes a "leaker". Needless to say, all drivers must wear radiation-detection film badges, and all vehicles must be placarded on all four sides. Since radioactive material can not be seen, touched, and is odorless, it is very important to have no comingling of this commodity with any other freight in the vehicle. Exposed or unprocessed film could receive irreparable damage by being placed in the same vehicle with a package of nuclides. If this is an interstate movement, the driver must also keep an hour log. This log is of a positive nature. It records normal procedures as well as any irregularities. If a surface carrier transports any quantity of radioactive pharmaceuticals, a counter should be readily available at the terminal. Each vehicle should be checked after daily use to see if there has been any background activity or spillage during the previous shift. It goes without saying that a vehicle that has only the driver's seat is a most suitable vehicle for this type of operation. Transporting nuclides is unique in the sense that there is more paper work connected with its transportation, and, therefore, requires the driver's undivided attention with no distraction by either an authorized or unauthorized passenger.

Since all hospitals, research centers, universities and government facilities receive these shipments on a regular or semi-regular basis, the driver of the delivery vehicle should be supplied with a loose-leaf book with photostatic copies of the floor plan of each delivery location. In many instances, it is necessary to indicate corridors that must be used and the location at which he must park his vehicle. Since each location has its own peculiarities, regarding the method needed to effect an expedited delivery, his instruction sheet must carefully enumerate the various steps he must follow in order to adhere to the local instructions that must be complied with at all times. In some cases, the driver will have the key to the nuclear medicine laboratory or will have to leave the package with the Security Guard or some other employee of the consignee. In any case, it is most important that he receive a signature and denote the time and date that the delivery was effected. It is also most helpful to have the individual who signed, print his name on the proof-of-delivery (POD) receipt so that it is legible and can be traced at a later time. A facsimile copy of a standard airbill plus numerous shipping instructions should be photocopied and added to the book along with maps and any other pictorial information that could be deemed useful.

With regard to safety precautions, a placarded vehicle with the standard black and yellow radioactive sign shown on four sides should deter the average person. Nevertheless, it is mandatory that the key be removed from the ignition and the vehicle be locked while making a delivery. At all times, fire extinguishers should be required on all vehicles. Additionally, it is of great importance for all personnel who handle radioactive material to understand the various types of labeling. Radioactive pharmaceuticals come with either a Type I, II, or III label. There are enumerable symbols and signs that have been standardized by the Federal Government, a copy of which should be readily available to all persons handling any phase of the radioactive pharmaceutical business. The strong element of safety can be best illustrated by a thorough knowledge of these symbols and labels. Color coding is a part of the safety precautions required for this material. In the event a terminal is used for in-transit storage of these radioactive pharmaceuticals, it has

been advisable to paint a certain area on the floor of the terminal in yellow paint so that all packages that are labeled as radioactive material will be placed at that spot and all personnel will endeavor to keep any other freight from this area for fear of contamination.

How to select drivers is always a difficult question to answer. It has been proven over the years that a mature driver, physically fit, is most psychologically suited for the transportation of this commodity. If this material is transported in interstate commerce, it is necessary for the driver to take a special ICC physical. Contacting the recent past employer and having the driver's finger prints checked by the local Police Department helps in the selection of a competent driver for dangerous or hazardous material. Prior to his appointment as a permanent employee, the driver should be trained over the various routes for a minimum of one week by a seasoned driver or manager. As this commodity could have certain side effects, it is most important that no driver work more than 40 hours per week. Not only does his efficiency drop, but he is prone to errors and accidents during a longer week.

In order for anybody to do a satisfactory job in the transportation of this unique commodity, it is necessary to hold meetings and continuously talk and compare ideas about the delivery of radioactive pharmaceuticals and allied products. From time to time, there are innovations in packaging and handling materials. These should be shown to the drivers prior to their usage. During the past few years, the vinyl plastic bag has been used in the consolidation of various smaller corrugated boxes. This helps eliminate the loss of an individual package, or in the event of spillage, it confines it to a specific area.

Should there be any delays or irregularities in the transportation of this commodity, it is mandatory the shipper and consignee be notified immediately. Since the surface delivery carrier must remain open 24 hours a day, 365 days a year, he is considered the fountainhead of information. Both the shipper and the consignee resort to his knowledge of the immediate whereabouts of the shipment. It is his responsibility to incept a tracer or gather any other information that might help in expediting the shipment. There is nothing more disturbing to a

consignee than to not know the immediate whereabouts of the radioactive pharmaceutical that had been ordered. By keeping in constant touch with the shipper and airline alike, we are able to supply this information immediately. A close liaison between the destination carrier and the shipper's technical representative in the locale become an important part of the handling of radioactive pharmaceuticals. Since the technical representative sells this material to the various institutions, he must be fully cognizant of the problems connected with the transportation of his product. The destination carrier should keep this individual informed in order that he may pass on any information to the consignee whereby the case-load or diagnostic study schedules can be altered in order to coordinate with the time of delivery of the radioactive pharmaceutical.

The handling of radioactive pharmaceuticals has become a highly specialized facet of the pickup and delivery business, and it goes without saying that the finest processed nuclides popularly priced but delivered on an untimely basis are not as sought after as overpriced nuclides of a poorer quality delivered on a timely basis.

Since this industry is still in its infancy, there will no doubt be improved methods of effecting and expediting delivery as well as packaging improvements. Such innovations as telecopier machines in the vehicle, whereby the driver would receive a copy of the original airbill, manifest, or any other pertinent information are in the future. Newer and more compact vehicles with greater mobility are also being experimented with and conceivably could improve the service. With the development of more efficient surface and air transportation, the use of radioactive pharmaceuticals in the nuclear medicine profession could be as far reaching as every town and hamlet in America.

AN EXPERIMENTAL STUDY ON THE DROP IMPACT
OF SPENT FUEL SHIPPING CASK

S.Aoki, S.Nakata, S.Shimamura, Y.Kanae

A.Muramatsu, H.Nakazawa

Research Subcommittee for Design of
Spent Fuel Shipping Cask,
Japan Society of Mechanical Engineers

ABSTRACT

Since 1968 the experimental study on the drop impact test has been performed in Japan. The test done in 1968 are mentioned in the other paper. In 1969 the 1/8, 1/4, 1/2.5 and 1/2 scale models of the assumed prototype of 80 t in weight were fabricated and tested. In this test deceleration at the moment of drop impact were measured with accelerometer mounted on the top surface of models. Deformations were examined, too. In addition, a high speed 16 mm motion picture was used to analyse the motion of model at drop impact. The result from this study showed that the relation between the maximum deceleration and reduced scale was approximately linear, and gave about 600 g as the value of the maximum deceleration for the 80 t cask dropped horizontally. The deformation mode of the 1/8 scale model, also of 1/16, 1/8.5 scale model in 1968, were significantly different from that of the 1/4, 1/2.5 and 1/2 scale models. Therefore, a model larger than the 1/4 scale should be used to obtain a reliable estimation for prototype from model test.

INTRODUCTION

The experimental studies on the spent fuel shipping cask have been planned, scheduled and conducted by the Research Subcommittee for Design of Spent Fuel Shipping Cask in the Japan Society of Mechanical Engineers (JSME) since 1967, under the sponsorship of the Atomic Energy Bureau in Agency of Science and Technology. In this program, the drop test facilities was built in 1968 on the basis of

the IAEA Regulations. The capacity of lifting device was 10 t and the nominal fall height 9 m. The target was made of reinforced concrete block weighing 210 t with a steel plate of 50 mm thick on the surface. In 1968, using this facilities, the 1/16 and 1/8.5 scale model were tested and some data were obtained. Successively in 1969, larger scale models was tested.

DESIGN AND FABRICATION OF SCALE MODEL

In order to confirm the similarity between practical cask and scale model and the dependency of reduced scale on the estimation of integrity, the four kinds of scale model for the 80 t prototype cask were prepared. The dimensions of prototype and scale models are shown in Table 1 and Fig. 1.

The shape of casks are horizontal cylinder-type consisted of inner-shell (stainless steel), lead shielding and outer-shell (low carbon steel). The design of prototype was based on ASME Pressure Vessel Code - Section VIII and Japanese Industrial Standard to conform the IAEA Regulations and Part 72 of 10 CFR of USA. The dimension and number of bolt for lid assuming that pulse time was 0.3 msec on the basis of the experimental result in 1968.

The scale models were painted in white and the black reference lines were drawn so as to be convenient for measuring the deformation as shown in Fig. 2.

Table 1. Dimension of Prototype and Scale Model Cask

Scale		1	1/2	1/2.5	1/4	1/8
Weight t		80.	10.	5.	1.25	0.15
D	mm	1,624	812	650	406	203
d	mm	1,000	500	400	250	125
L	mm	5,312	2,864	2,290	1,438	717.5
l	mm	5,000	2,500	2,000	1,250	631
t ₁	mm	56	28	22	14	7
t ₂	mm	56	28	22	14	7
t ₃	mm	16	8	7	4	2
t ₄	mm	16	24	1919	12	6
t ₅	mm	240	120	96	60	30

Fig. 1 Tank Model Dimension

Fig. 2 1:2 Scale Model 10 t in Weight, showing Reference Lines

TEST PROCEDURE

The drop tests were always performed from 9 m in conformity with the IAEA Regulations (but only for the 1/8 scale model with shock-absorber, 3 and 6 m drop tests also were done). Fig. 3 shows the drop test of the 1/2 scale model just before collision.

In order to grasp satisfactorily the external force added on the model at the moment of collision and the deformations and damages caused by it, the following measurements were done;

- (1) measurement of deceleration: three strain gage type accelerometers mounted on the top surface of model,
- (2) measurement of deformation: by means of suitable measuring tools, the size and shape were recorded before and after every tests,
- (3) damage of lid: damage of bolts and nuts for lid were examined,
- (4) record of motion at impact: in order to record and analyse the attitude of model during drop, at collision, and the behavior during rebound, high speed 16 mm motion pictures were taken.

EXPERIMENT AND RESULT

Deceleration

There were two kinds of accelerometer which could be used to measure the deceleration at impact, namely strain gage type and piezo type. However, as the latter were strongly affected by weather condition and had some other defects, only strain gage type accelerometer was adopted in this experiment. And, to make clear the effect of measuring position on the magnitude and shape of acceleration pulse, though the attitude of model was slant, three accelerometers were mounted at the three position, namely on bottom side, center and lid side, on top surface of model by an adhesive agent.

The data of deceleration detected by accelerometers were once record on a data-recorder and later they were reproduced on an oscilloscope and an electro-magnetic oscillograph. Fig.4 shows an example of the pulse on oscilloscope for the 1/2 scale model. Table 2 shows decelerations on each position for every scale models.

Deformation

As above-mentioned, the reference lines were drawn on outer surface, lid surface and bottom surface of models to locate the points of deformation. The diameter, length and radial deformation were

Fig. 3 1/2 Scale Model just before Collision

Fig. 4 Pulse Wave on Oscilloscope for
1/2 Scale Model Drop Test

Table 2. Deceleration for Scale Model

Scale		Deceleration		
Scale	Weight t	Lid Side	Center	Bottom Side
1/2	10	830 g		630 g
1/2.5	5	1,410 g	1,100 g	450 g
1/4 a.	1.25	2,080 g	2 2,200 g	570 g
b.	1.25	2,220 g	920 g	1,070 g
1/8 a.	0.15	1,710 g	1,920 g	1,360 g
b.	0.15	2,550 g	1,880 g	2,230 g

measured before and after every tests for all models.

External Examination

The diameter change and deformation of diameter before and after test were shown in Fig.5 - 8. The deformation of diameter on the reference lines were shown in Fig.9 - 12 for 1/2, 1/2.5, 1/4 and 1/8 scale models respectively.

Internal Examination

In the case of 1/2 and 1/2.5 scale models, leak tests were conducted with water at 5 kg/cm²g and with air at 2 kg/cm²g during 30 minutes. In both cases no pressure drop was detected. The radial deformation of internal diameter were given in Fig.13 and 14 for 1/2 and 1/2.5 scale models.

CONSIDERATION

(1) The side reference lines swelled horizontally at the center as shown δ_2 and δ_4 in Fig.9 - 12 and Fig.13 and 14. It was common between all models.

(2) The highest reference line deformed always convexly below for all models, as shown by δ_1 distribution.

(3) The lowest reference line deformed concavely below except the regions of both ends for the 1/2, 1/2.5 and 1/4 models, as shown by δ_3 , but in the case of 1/8 scale model, as shown in Fig.12, it deformed convexly below on the contrary to the other case, which was in agreement with the results in 1968.

(4) Therefore, as it seems that the deformation of the small sized scale model is quite different from that of large sized model, the scale model should be larger than a limited sized scale model so that the prototype cask may be simulated by model test.

Fig. 5 Diameter Change ofor
1/2 Scale Model

Fig. 6 Diameter Change for
1/2.5 Scale Model

Fig. 7 Diameter Change for
1/4 Scale Model

Fig. 8 Diameter Change for
1/8 Scale Model

Fig. 9 Deformation of Diameter for 1/2 Scale Model

Fig.10 Deformation of Diameter for 1/2.5 Scale Model

Fig.11 Deformation of Diameter for 1/4 Scale Model

Fig.12 Deformation of Diameter for 1/8 Scale Model

Fig. 13 Deformation of Internal Diameter for 1/2 Scale Model

Fig. 14 Deformation of Internal Diameter for 1/2.5 Scale Model

MOTION OF MODEL AT DROP IMPACT TEST

The motion of models at drop test was recorded by a high speed motion picture (film speed 3,000 pps) and was analysed by a motion picture analyser. Fig.15 shows an example for the 1/2 scale model when rebound began, after 147 msec from the contact with target.

The models rebounded after collision and rested after few seconds. It moved always in the direction of bottom side from the first contact point. The results as follows;

	T_r msec	v m/s	T' msec	l_s msec
1/2 scale	50	0.57	360	123.8
1/2.5	33	1.46	495	152
1/4	25	1.82	585	147

Here; T_r : time until rebound begins after contact with target

v : average velocity at rebound

T' : time required before rest

l_s : distance of movement

ANALYSIS

Relation Between Deceleration and Reduced Scale

Assuming stress-strain relation for perfect plastic body and that whole potential energy of body is converted into crush deformation energy, the relation between fall height H and external deformation δ for horizontal drop of cylinder may be given as

$$\delta = 1.4 R \left(\frac{\gamma H}{\sigma_y} \right)^{2/3} \quad (1)$$

And, assuming that impulse is equal to momentum change, deceleration may be given as follows:

$$\left(\frac{dv}{dt} \right)_{\max} = - \frac{\pi g}{2.8} \frac{H^{1/3}}{R} \left(\frac{\sigma_y}{\gamma} \right)^{2/3} \quad (2)$$

Depending Eq.(2), the maximum deceleration is inversely proportional to the reduced scale. In Fig.16 the experimental results are plotted. Calculating from the dotted line, which is drawn to fit the experimental data, it seems that the maximum deceleration of prototype (80 t) at horizontal drop impact comes to about 600 g or

Fig. 15 The 1/2 Scale Model when Rebound begins

less than it.

Relation Between External Deformation and Scale Model Diameter D_1

If deformation δ at the center of model versus diameter D_1 is plotted as Fig.17, it seems to be quite all right to consider that the linear relation exists between δ and D_1 . From this result, the external deformation of diameter in the vertical direction at the center of prototype cask (80 t) may be assumed to be approximately 80 mm.

CONCLUSION

(1) The theoretical analysis has been applied to reveal the existence of linear relation between the maximum deceleration and reduced scale, which has been proven by the experimental results using the 1/2, 1/2.5, 1/4 and 1/8 scale models.

(2) It has been found that the deformation of cask model at drop impact strongly depends upon the reduced scale. That is, the deformation of the 1/4 scale model, which is same as the result in 1967 using 1/8.5 and 1/16 scale models, is significantly different from the deformation of other larger scale models. Deducing from this fact, it is desirable to use the scale model larger than at least 1/4 scale, in order to estimate the deformation of prototype from a scale model test.

ACKNOWLEDGMENT

The authors wish to express their gratitude to the Atomic Energy Bureau and the Japan Society of Mechanical Engineers. They also wish to express their thanks to the member of the Research Subcommittee for Design of Spent Fuel Shipping Cask and to the staffs of Government Mechanical Laboratory and Institute of Medical and Dental Engineering in Tokyo Medical and Dental University, who accomplished the series of drop impact test for a long time.

Fig. 16 Relation between Deceleration and Reduced Scale

Fig. 17 Relation between External Deformation and Scale Model Diameter

IMPACT TESTING OF CASK FIN SPECIMENS

F. C. Davis
Oak Ridge National Laboratory*
Oak Ridge, Tennessee

ABSTRACT

An impact test program was conducted at Oak Ridge National Laboratory to determine the energy absorption characteristics of typical heat transfer or structural fins on shipping casks as these fins undergo dynamic, gross plastic deformation. It was assumed that the energy absorption characteristics of fins welded parallel to the longitudinal axis of a cylindrical cask are linear with respect to their length, and the test program was designed to determine the energy absorption capability per inch of fin length along the cask body. Specimens of typical fin geometries were impacted with free-falling weights dropped from known heights, and the force-time relationships occurring during each impact were recorded. A computer program was used to convert the test data into deformation-time and energy-time relationships. These relationships were correlated in a set of curves that enable the cask designer or analyst to compute the energy absorption capacity of a longitudinal fin on a cask as a function of its geometry and the percentage of deformation it is expected to experience as well as the peak force the fin would experience during the impact resulting from a 30-ft free drop onto an essentially unyielding horizontal surface.

INTRODUCTION

When a package containing radioactive or fissile material is shipped from one location to another, the package is subject to regulations governing its structural integrity and its shielding and heat dissipating capabilities. Many complex questions have arisen in connection with demonstrating the conformance of shipping packages or casks to the free-drop portion of the hypothetical accident conditions specified in these regulations. Many shipping casks have external ribs or fins welded to the outer shell to provide additional strength or for heat transfer purposes. An impact such as that specified in the first condition of the hypothetical accident will result in gross deformation of these fins, and performing a theoretical analysis to determine the energy absorbed by the fins

---

*Oak Ridge National Laboratory is operated by Union Carbide Corporation under contract with the United States Atomic Energy Commission.

during the gross deformation would be very difficult. Use of analyses based on static material properties generally results in ultraconservative design because the stress for a given strain generally will be much higher under impact conditions. Thus, the impact testing program reported here was undertaken to develop a reliable method for computing the energy absorption capacity of a longitudinal fin on a cylindrical cask subjected to an impact resulting from the 30-ft free drop.

EXPERIMENTAL PROGRAM AND APPARATUS

The general approach adopted for the experimental program involved dropping a hammer of known weight from a predetermined height onto a specimen and recording the force-time relationship experienced by the specimen throughout the duration of the resulting impact. Force with respect to time can be converted to acceleration with respect to time by using Newton's second law. This acceleration-time relationship can be converted to a velocity-time relationship by using the equation

$$v_t = v_o - \int_{t=0}^{t=t} a \, dt , \quad (1)$$

where

- v_t = velocity at a point in time,
- v_o = initial velocity at time equal zero,
- t = time, and
- a = acceleration.

Similarly, the velocity-time relationship can be used to determine the deformation-time relationship by using the expression

$$X_t = \int_{t=0}^{t=t} v_t \, dt = \int_{t=0}^{t=t} v_o \, dt - \int_{t=0}^{t=t} \int_{t=0}^{t=t} a \, dt \, dt , \quad (2)$$

where X_t = the displacement at a point in time. Knowing the force-time and the deformation-time relationships, the energy absorbed can be calculated from the expression

$$U = \int_0^X P \, dX , \quad (3)$$

where

- U = energy absorbed for a specific amount of deformation,

X = deformation, and

P = applied force.

At the end of the impact duration, at which time the velocity has become zero, the calculated absorbed energy should be approximately equal to the kinetic energy of the impacting hammer at time equal zero. This gives a time at which the experimental data can be verified.

The impact tests were conducted at the Oak Ridge National Laboratory (ORNL) 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 embedded in steel-reinforced concrete. Two 3-by-3-in. angles 3/16 in. thick were attached to the tower vertically to guide the impact hammer during its free fall. The basic structure of this hammer weighed 99 lb, but this basic weight could be increased in 21-lb increments by the addition of steel blocks to the hammer and a maximum weight of about 500 lb could be obtained. The guide rails, weights, impact hammer, and other equipment used in the impact tests are shown in Fig. 2.

Fig. 1. The ORNL Drop Tower.

Fig. 2. Equipment Used in Gask Fin Specimen Impact Tests at the ORNL Drop Tower.

Three load cells were designed to provide different ranges of sensitivity from 0 to 300,000 lb necessary to cover the fin geometries of interest. Bonded resistance strain gages were secured to the periphery of the load cells with epoxy resin and wired into a full Wheatstone bridge arrangement. The output of the bridge was fed to one channel of a dual-beam oscilloscope with a dual-trace differential amplifier. During the latter half of the tests, acceleration-time relationships were also obtained by attaching a piezoelectric accelerometer to the hammer and feeding its output signal through a charge amplifier and a low-pass filter to the second channel of the oscilloscope. With the incorporation

of this instrumentation, a force-time relationship and an acceleration-time relationship were recorded simultaneously for each impact.

DESIGN AND FABRICATION OF TEST SPECIMENS

Consider the cross section of a cylindrical cask with longitudinal fins along its axis impacting a horizontal level surface, as shown in Fig. 3. It is

Fig. 3. Cross Section of Cylindrical Cask With Radial Fins Impacting a Horizontal Surface.

obvious that only one fin may have an angle of inclination of zero degrees, while the angle of inclination of adjacent fins will vary according to the position of the fins around the circumference of the cask. The decision was made to test fin specimens at five different angles of inclination: 0, 10, 20, 30, and 40°. The energy absorption characteristics of longitudinal fins on shipping casks were considered a linear function of the length of the fin in this program. Five specimens of each configuration shown in Fig. 4 were fabricated from ASTM A 285, Grade C, fire box quality steel, which has a yield strength of 30,000 psi.

PERFORMANCE OF TESTS

Impact tests were conducted on 275 specimens of shipping cask fins. In each test, a fin specimen of the desired geometry was securely clamped to the top of the load cell, and the appropriate sweep rate and vertical gain were set on the oscilloscope. These settings were a function of the drop height, weight dropped, specimen geometry, and sensitivity of the load cell. The mechanical trigger was set a small distance above the top of the fin specimen. The hammer with an appropriate amount of added weight was raised to the desired drop height and released. The free-falling hammer closed the contacts on the mechanical trigger, initiating a single sweep on the oscilloscope. Impact of the hammer on the fin

T (in.)	θ																								
	0°					10°					20°					30°					40°				
	H (in.)																								
	3½	4	6	8	9	4	6	8	10	4	6	8	10	4	6	8	10	4	6	8	10				
1/4		X	X				X	X	X		X	X	X		X	X	X		X	X	X				
3/8						X	X	X	X	X	X	X	X	X	X	X		X	X	X	X				
1/2	X		X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X				
5/8						X	X	X	X	X	X	X	X	X	X	X		X	X	X	X				
3/4			X		X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X				

Fig. 4. Dimensions of Fin Specimen Configurations Tested.

specimen transmitted a force to the load cell. This force caused the strain gage bridge associated with the load cell to produce an output voltage proportional to the applied force. The voltage, which changed rapidly with time, was displayed on the oscilloscope screen and recorded by a scope camera. A typical force-time photograph recorded during an impact is shown in Fig. 5.

Fig. 5. Force-Time Photograph Taken During Impact Test of a 3.5-in.-High Fin Specimen 1/2 in. Thick With a Zero-Degree Angle of Inclination.

After 157 fin specimens had been impact tested, the decision was made to measure and record the acceleration-time relationships of the hammer during the tests because some ringing, which caused difficulties, had appeared in the load cell signal. An accelerometer was attached to the hammer and its output signal was transmitted to the oscilloscope through a charge amplifier and low-pass filter. Thus, the impact force-time relationship and the acceleration-time relationship for each of the remaining 118 fin specimens tested were recorded simultaneously. A typical photograph of these oscilloscope recordings is shown in Fig. 6.

Fig. 6. Force-Time and Acceleration-Time Photograph Taken During Impact Test of a 4-in.-High Fin Specimen 3/8 in. Thick With a 40° Angle of Inclination.

From the trace or traces on each photograph, coordinates of points along the curves were read, recorded, and converted to force or acceleration values as a function of time by applying the instrumentation settings and calibration values. The acceleration-time data recorded in this manner were analyzed by applying the methods used to analyze the force-time data except that the force-to-acceleration computation was not required. The results obtained should be theoretically identical to those obtained by using the force-time data.

DATA TRANSFORMATION

A Fortran program applicable to the IBM 360 computer at ORNL was prepared to transform the force-time and acceleration-time data obtained manually from the oscilloscope photographs into absorbed energy and deformation data. The computer output included a printed copy of the final computational results and a deck of IBM cards that served as input data cards for an existing computer plotting program.

EVALUATION OF TEST RESULTS

Sets of curves, one of which is shown in Fig. 7, were derived from the force-time data obtained from the load cells during the impact tests, and the heights (H) of the specimens tested are given for their respective curves. These curves are plots of the absorbed energy divided by the plastic moment of each specimen tested versus the vertical deformation of the specimen multiplied by 100 and divided by the original height of the specimen. This abscissa dimension is essentially the percentage of total deformation possible. The plastic moment values used were calculated from the expression

$$M_p = \sigma_y \left(\frac{bt^2}{4} \right), \quad (4)$$

where

- M_p = static plastic moment of specimen,
- σ_y = yield stress of specimen material = 30,000 psi,
- b = length of specimen = 2 in., and
- t = thickness of specimen in inches.

Therefore, the static plastic moment

$$M_p = 15,000t^2 \text{ in.-lb.} \quad (5)$$

Fig. 7. The Absorbed Energy Derived From Load Cell Test Data Divided by the Plastic Moment as a Function of the Percentage of Deformation Experienced by Impacted Specimens With a 20° Angle of Inclination.

Other sets of curves, one of which is shown in Fig. 8, were derived from the acceleration-time data obtained from the accelerometer during the impact tests.

Fig. 8. The absorbed Energy Derived From Accelerometer Test Data Divided by the Plastic Moment as a Function of the Percentage of Deformation Experienced by Impacted Specimens With a 20° Angle of Inclination.

These curves are also plots of the absorbed energy divided by the plastic moment of each specimen tested versus the percentage of deformation experienced by the impacted specimen. The curves derived from the accelerometer data and those derived from the load cell data should theoretically be identical. However, careful analysis of these curves indicated that those derived from the accelerometer data show approximately 10% less energy absorption than those derived from the load cell data. Although not large, this discrepancy remained fairly constant, and it may be the result of a calibration error. The accelerometer was not calibrated by using the instrumentation used during performance of the tests. The calibration data received from the manufacturer were used in the analysis.

There is a vast difference between the energy absorption capabilities of the stiff (angles of inclination of 0 and 10°) specimens and the less stiff (angles of inclination of 20, 30, and 40°) specimens. The reason for this vast difference is that all of the specimens with zero-degree angles of inclination and most of the specimens with 10° angles of inclination failed in a double-hinge mode similar to the buckling type of failure expected of a column restrained at one end, whereas most of the specimens with 20° angles of inclination and all of the specimens with 30 and 40° angles of inclination failed in a single-hinge mode. This indicates that the transition point between single- and double-hinge modes of failure lies between 10 and 20° of inclination.

The typical modes of failure experienced by the test specimens are shown in Fig. 9. Shown at the top of this photograph are three impact tested specimens

Fig. 9. Typical Fin Specimens With Angles of Inclination of 0, 10, 20, 30, and 40° Before and After Impact.

that originally had angles of inclination of zero degrees. The left-hand group of three specimens in the second row is comprised of specimens with 10° angles of inclination, and the right-hand group is comprised of specimens with 20° angles of inclination. Specimens with 30° angles of inclination are shown in the left-hand group of three specimens in the bottom row, and specimens with 40° angles of inclination are shown in the right-hand group in the bottom row. In each of these groups of three specimens except the first at the top of the photograph, the first specimen shown has not been impact tested and the other two have been tested.

The time durations of the impact events for the specimens which failed in a double-hinge mode ran from 2.6 to 22.2 msec, with the majority running from 7 to 15 msec. The durations of impact events for specimens which failed in a single-hinge mode were much longer, with some being up to 154 msec. Although the impact durations of single-hinge failures were much longer, the magnitudes of the peak forces were much less than those of the double-hinge failures.

The printed computer output for each specimen tested afforded a means by which the calculated deformation could be compared with the final measured deformation. It also allowed the absorbed energy calculated from the experimental data to be compared with the energy of the falling hammer at the time of initial impact. A review of the computer output data for the specimens failing in a double-hinge mode indicated deviations of up to 30% between the calculated values of deformation and the measured values of final deformation. Larger deviations were observed for specimens that failed in a single-hinge mode. These deviations are at least partially due to the elastic springback of the specimens when the energy had been dissipated and the load removed, especially for those specimens which failed in a single-hinge mode.

The difference between the energy which was applied to the specimens and the absorbed energy accounted for in the test data were very small, being in the order of 2 to 3%. This was true for specimens experiencing either type of failure mode, single or double hinge. A relatively small amount of energy was dissipated by friction in the horizontal motion between the hammer and the top of the specimens as the collapse occurred. The horizontal motions were small for double-hinge failures where the forces between the hammer and the specimens were large, and the horizontal motions were relatively large for single-hinge failures where the forces between the hammer and the specimens were small. Therefore, the energies absorbed by this friction remained relatively small.

The range of data scatter is indicated by the fine vertical lines in the graphical presentation of the computer output, examples of which are shown in Figs. 7 and 8. The curves derived from the accelerometer data were well within the data band derived from the load cell data. This correlation between the

load cell data and accelerometer data provides a degree of assurance that the results are within acceptable engineering accuracy.

Analysis of the curves indicated that the energy absorption capabilities of a specimen are dependent upon the square of the thickness of the specimen. This is evidenced by the fact that when the energy absorbed by the specimen was divided by its plastic moment value, which is basically a constant multiplied by the square of the thickness of the specimen, and the value obtained was plotted against the percentage of deformation resulting from the impact; the result for the specimens tested was a set of curves dependent upon specimen height alone. For deformations of 10% or less, the results were almost independent of the height of the specimen. This is especially true for the specimens with angles of inclination of 0, 10, and 20°. For larger deformations, it is evident that the specimens with greater heights absorbed less energy for the same percentage of deformation experienced by the specimens with lesser heights.

The computer output for the peak force experienced by the specimens during impact was plotted as a function of their height divided by their thickness. A typical curve of this type is shown in Fig. 10. Buckling is the primary mode in double-hinge failures, while bending is the primary mode in single-hinge failures.

Fig. 10. The Peak Force Derived From Load Cell Test Data as a Function of Specimen Height Divided by Thickness for Impacted Specimens With a 20° Angle of Inclination.

The peak force information was derived solely from the load cell data recorded during the impact tests. A low-pass filter was necessary to remove

high-frequency noise from the accelerometer output signal. This filter attenuated the sharp peak acceleration value that actually occurred in the first part of the impact event, and this attenuation may be noted on the typical oscilloscope photograph shown in Fig. 6. Therefore, the peak force values obtained from the accelerometer data were not valid. This attenuation characteristic did not materially affect the energy absorption information derived from the accelerometer data because the energy values were obtained by double integration of the acceleration-time curve.

The peak force values recorded during the impact tests were dependent upon the height-to-thickness ratio of the specimens tested, and the resulting curves are very similar to critical load curves for columns with different length-to-radius-of-gyration ratios. It was observed many times throughout the impact tests that the peak force value for a given specimen geometry was not dependent on the input energy of the hammer as long as the specimen was plastically deformed. The input energy could be doubled by either doubling the weight of the hammer or the drop height, but the peak force would remain essentially constant. The increased energy was absorbed by more deformation that occurred over a longer duration.

CONCLUSIONS

The objectives of this impact testing program were achieved. The energy absorption characteristics of fin specimens undergoing dynamic, gross plastic deformation were determined, and the test data were reduced to a form readily usable for computation of the energy absorption capability of external fins along the longitudinal axis of a cylindrical cask subjected to the 30-ft free-drop condition of the hypothetical accident specified in the regulations governing shipping casks. Detailed information about the testing program and all of the resulting data plots that will enable the cask designer or analyst to compute the energy absorption capacity of a longitudinal fin on a cask as a function of its geometry and the percentage of deformation it is expected to experience during an impact are presented in USAEC Report ORNL TM-1312, Vol. 9 ("Structural Analysis of Shipping Casks, Vol. 9: Energy Absorption Capabilities of Plastically Deformed Struts Under Specified Impact Loading Conditions").

The energy absorption capacity of a fin specimen was shown to be dependent upon the square of its thickness. The greater the height of the specimen, the less the energy that was absorbed for the same percentage of deformation once that deformation had proceeded into a gross range (10% or greater). The fact that the data obtained by two independent methods during the latter portion of

the experimental program reduced to similar curves representing the energy absorbed versus the percentage of deformation experienced by the specimens provides a degree of confidence in the results.

The peak force values observed during the impact tests were dependent upon the height-to-thickness ratio of the specimens tested, and the plots are similar to critical load curves for columns. The peak force experienced by the specimens did not appear to be a function of the impact energy of the hammer as long as plastic deformation was attained. A complete set of plots of the peak force experienced by specimens during impact versus their heights divided by their thicknesses is presented in USAEC Report ORNL TM-1312, Vol. 9. With the weight of the shipping cask known, the peak acceleration that would be experienced by the cask subjected to the 30-ft free-drop condition of the hypothetical accident can be computed.

IMPACT TESTS OF MODELS AT THE SAVANNAH RIVER PLANT

J. W. Langhaar
E. I. du Pont de Nemours and Company

ABSTRACT

Impact testing of small-scale models for assessing gross deformation resistance to puncture, behavior of welds, and certain other physical attributes has been theoretically established and experimentally verified. For evaluation of the strength of bolted fastenings and leakage of closures, the validity of model testing is not so well proven but is nevertheless often considered the most reliable practical method available.

The Savannah River Plant has tested numerous models in recent years and sometimes a full-size mockup of part of a package. Results in some cases have confirmed predictions and other cases have necessitated a change in design. Several of the tests are described. The need for further study of methods of modeling closures is emphasized.

INTRODUCTION

About ten years ago the validity of model testing for assessing the behavior of shipping containers under impact conditions became a subject of serious study. In 1962, Clark and Reddi¹ showed that for a true geometric model the stresses, strains, and percentage deformation are for practical purposes independent of model size provided (a) the impact velocity is unchanged, (b) the resisting force continues to increase while the container comes to a stop, and (c) the stress-strain characteristics of the materials are not affected by how fast the stress is applied. The last of these is important because a quarter-scale model, for example, stops four times as fast as a full-scale model.

The requirements are approximately met in practice for impacts not resulting in rupture,² and test work has verified this even for the puncture-type accident up to the point where shearing and penetration occur.³ A major problem is fabrication of a true model, particularly with respect to welds and fastening devices. Good correlation of results has been obtained with carefully made models as small as one-tenth scale,² but approximately quarter scale is more reliable and easier to fabricate. Variability of physical properties throughout a structural part such as the shell of a cask, variations in details of fabrication, orientation at impact, and other factors may significantly affect a buckling pattern, so that only the general nature and magnitude of buckling can be duplicated even with full-scale testing.

Use of models has become generally accepted as a means of demonstrating structural response to a 30-ft drop. The practice has been extended to assessment of leaktightness, although as discussed later there are some important unanswered questions in this area.

A model may be simplified by omission of features not significant for the test or by modification in some manner judged to be conservative for the test. Thus it may be a mockup of only some part of a container. Also, it may be small-scale or full-scale. Considered in this broad sense, the Savannah River Plant has during the past few years performed several 30-ft drop tests with models of different types. Results in some cases confirmed expectations but in other cases led to modification of design.

Full-size prototypes of relatively inexpensive containers have also been tested. For present purposes these are not classified as models and are not included in the discussion.

DISCUSSION

Curium Cask

This cask (Fig. 1) is essentially a tank of water 5 ft in diameter and 6 ft high, with a one-inch-thick stainless steel wall.⁴ An internal birdcage structure supports the product capsules. External fins help to dissipate heat. The lid is recessed and protected from damage by a cover plate. The closure utilizes a 20-inch-diameter, spiral-wound, "Flexitallic" gasket, with seating surfaces having a finish specified as 32 rms or better. The cask is equipped with 16 fusible plugs so positioned that regardless of cask orientation at least one of these plugs will melt in the specified fire and permit escape of steam without loss of liquid water. The loaded weight is about 7 tons.

It was necessary to demonstrate that rupture of the tank would not occur and that leaktightness of the closure would be maintained after a 30-ft drop to a flat surface and after a 40-inch drop on a 6-inch-diameter penetrator. A model was fabricated to quarter scale except that standard sizes of "Flexitallic" gasket were used and the finish of the seating surfaces was about the same for the model as for the full-size cask. The loaded weight was 220 pounds.

The model was subjected to a series of 14 drops, of which only a few were considered required for demonstration of compliance and the remainder were performed for added information. Results are given in Table 1 and in Figures 2 through 8.

The puncture tests were of particular interest because experimental work at the Franklin Institute and at Oak Ridge, represented by the correlation given in "The Cask Designer's Guide,"³ was restricted to designs with lead behind the steel plates. The degree to which the plate is stiffened by fins or edge restraint or backup material and the energy absorbed by the backup material while being pushed aside appreciably affect the thickness of steel required to avoid puncture. It was found that for a 30-ft drop onto a penetrator, puncture occurred on the bottom but not on the side or top. The fact that the top drop was off-center may have influenced the results.

Leaktightness of the model was assessed before and after tests 1, 2, 5, and 7 by pressurizing with water to 70 psig and checking for visible leaks or loss of pressure during a period of about 15 minutes. No leakage was found.

Bismuth Cask

Two casks of 24-inch inside diameter and 39-inch inside height, with solid stainless steel wall of 5-inch thickness, were constructed for shipping irradiated bismuth slugs (Fig. 9). The cavity was to contain water for effective heat removal and – because of the low melting point of bismuth – maintain a leaktight closure with no escape of coolant or other contents even under fire conditions. The casks have a design pressure of 930 psig at 525°F and are adequate for 1500 psig based on yield strength at the maximum temperature expected in the hypothetical fire. For the design contents, the calculated maximum pressure for the fire condition is about 1000 psig.

As an added safety precaution for fire conditions, a rupture disk assembly rated at 1390 psig at 525°F is provided in the cask cavity. Piping for pressure relief and for cask draining is incorporated in the lid. The lid closure uses a “Viton” O-ring; the flanged connection for piping on the underside of the lid has a “Flexitallic” gasket. The casks were hydrostatically tested at 1500 psig at room temperature with no visible leakage or loss of pressure.

Two 30-ft drop tests were performed, one with a quarter-scale model of the upper portion of the cask to evaluate the closure design and one with a full-scale mockup of the rupture disk assembly to evaluate its resistance to impact. Both tests led to design modifications.

The quarter-scale model (Fig. 10) was not pressurized or gasketed, because it was considered that relative movement of the lid with respect to the body would be a suitable measure of performance. Appreciable sidewise motion was not possible without gross failure. The maximum gap between lid and body after the drop was 4 mils, corresponding to 16 mils for the full-size cask or to only about 25% reduction in the compression of the O-ring. However, because of considerable damage to two of the six lid studs and nuts, the number was increased to twelve for the full-size casks and the nuts were recessed into the lid.

The initial design contemplated a single rupture disk rated at about 1400 psig at 525°F. For the cask upside down at impact, the head of 3 ft of water if decelerated at about 1000 G's might be expected to rupture such a disk. There was no sound basis for estimating the actual deceleration; furthermore, shock phenomena are not amenable to such simple analysis. Consequently, a mockup was fabricated as shown in Fig. 11 except that for the first test the dip tube and union-type rupture disk shown at the upper end of the tube were omitted. An available disk rated at 1322 psig at 72°F was used for the test. The vessel was filled with water except for a two-inch freeboard and pressurized with air to the maximum expected actual operating pressure of about 140 psig. The 30-ft drop caused the rupture disk to fail in what appeared to be a normal manner.

The design was changed to provide two disks in series with air in the tube between them. For the second test, the flange-type assembly contained a disk similar to that for the first test, and the union-type assembly contained a disk rated at 975 psig at 72°F. The mockup was pressurized as before, and dropped once on each end. Disassembly showed that the union-type disk had failed (as expected from the first test) but that the other disk remained intact. The modified design was therefore incorporated in the casks, except that the casks have disks of somewhat higher rating.

PM-3A Casks

The upgrading of two casks (Fig. 12) for shipping reactor cores from McMurdo Sound is described in another paper of this Symposium.⁵ These casks are lead-shielded and were originally equipped with a cylindrical shroud of one-inch-thick steel for added gamma shielding during land and water transport. The shroud was not required for air transport. The shipment is dry.

Because of questionable structural integrity of the shell under hypothetical accident conditions, it was decided to extend the shroud so as to shield the entire side of the cask from radiant heat in a fire and to add similar protection for the top and bottom. The configuration of the O-ring seal was suspect with respect to lateral displacement and possible leakage under impact conditions. This O-ring was replaced by a "Gasko-seal" unit which could accommodate greater lateral movement of the lid.

A quarter-scale simplified model was fabricated to assess the integrity of the shroud, insulating plate and lid closure (Fig. 13). After a 30-ft top-corner drop (Fig. 14), the model was immersed in water, pressurized with air to 30 psig, and checked for leaks. No leakage was observed. The crash frame had protected the cask from any visible damage (Fig. 15), and measurements showed insignificant movement of the lid.

A bottom-corner drop (Fig. 16) did not cause exposure of any part of the cask body and thus did not disqualify the cask. However, there was failure of several attachment bolts of the shroud (Fig. 17). The model was repaired, reassembled, and dropped 30 ft on its side. Two sections of shroud and the bottom insulating plate came off (Fig. 18).

The model was then equipped with cables and clips to hold the shroud together, and a smaller diameter bottom insulating plate to avoid direct impact on the edge of the plate (Fig. 19). The redesign was successful in retaining the shroud (Fig. 20), but the bolts holding the insulating plate failed because the insulating plate was too stiff to conform to the bending of the bottom plate of the cask. The model was again pneumatically leak-tested, with no observable leakage.

For the full-size casks, attachment of the insulating plate was strengthened and the plate was reduced in stiffness by a factor of 5, as measured by section modulus, by making it of two steel sheets each 0.1875-inch thick separated by 0.375 inch of hardboard. It was determined by calculation that this would conform to the heavy bottom of the cask without bolt failure for impact in any orientation on a flat surface.

Both full-size casks were pneumatically leak-tested at 30 psig by immersion in one of the cask handling basins. These tests were considered necessary because of possible slight differences from the model with respect to the gasket and the contact surfaces.

Other Containers

In two recent cases, poisoned inserts of the eggerate type were fabricated for use in existing casks. One of these was for shipment of Borax V fuel from Idaho to SRP in the bismuth casks; the other was for shipment of Spert III fuel from Idaho to SRP in the Paducah uranium-shielded demonstration cask⁶ which a few years ago satisfactorily survived impact and fire tests.

In order to demonstrate structural integrity of the inserts, one-half scale models of six-inch-long sections of cask and insert were subjected to a 30-ft drop. The cask wall was simulated by a weldment of suitable stiffness resembling an I-beam rolled to a ring-shape with one flange serving as the inner cask shell and the other flange as the outer cask shell. The fuel was simulated by steel rods of suitable size and number. The Borax V insert did not distort significantly. In the Spert III case there was appreciable distortion, and it was decided to provide structural reinforcement rather than complicate the nuclear safety analysis.

Leak Testing

A common feature of the model tests for the casks described above was testing for leakage. Methods used were considered appropriate, taking into account the nature of the material which could leak and the regulatory requirements. For the curium cask, loss of a large percentage of the water would result in external radiation levels somewhat above regulatory limits. For the bismuth cask, a large loss of water could lead to melting of the bismuth which contains polonium; this conceivably could escape from slug cladding damaged by impact, and to a minor extent through the opening that permitted escape of water. For the PM-3A cask, the most restrictive limit was the IAEA requirement for unilaterally approved casks that no more than 0.1% of the coolant (i.e., air) should escape in one week following the accident. In all three cases the expected degree of contamination of the coolant is very low.

The tests illustrate the need for study and guidance relating to two basic questions: (1) How tight is leaktight? (2) How should one go about designing a small-scale model of a closure?

It must be recognized that absolute leaktightness or zero escape cannot be achieved. Also, it would be illogical and impractical to require the same degree of leaktightness for all radioactive materials, regardless of toxicity, physical form, or concentration. Tests which might be applied for proving leaktightness for a prototype or a design encompass a range of more than a million in relative sensitivity. Common types of closure may vary that much in their capability. Relative toxicities per unit volume for the potential leakage vary by an even larger factor. A rational basis for design and testing is urgently needed.

Some progress is being made on this. The IAEA has proposed⁷ maximum leak rates which take account of the toxicity and which are considered to represent an insignificant hazard. It is nevertheless intended that good engineering and operating practice will be followed, and that actual leakage will be as low as practical. Subcommittee N-14.2.6 of the American National Standards Institute is developing recommendations on test procedures for prototypes and on the matter of providing adequate assurance that each package presented for shipment is properly assembled. Similar studies are under way in other countries.

With regard to design principles for small-scale modeling of closures, I am not aware of any basic studies in progress. A common opinion or "educated guess" among engineers is that a gasket or O-ring for a model should have the full-size cross section, and that surface roughness and runout should be unchanged. However, the magnitude of leakage and the corresponding mechanism of leakage in a particular case may be important factors. For example, if leakage develops after a 30-ft drop because of damage to the O-ring from movement of the lid, and this movement is only one-fourth as much in a model as in a prototype, then conceivably the damage and leakage could be much greater for the

prototype than for the model. Similarly, the reduction of gasket or O-ring compression in the model would be less than in the prototype.

Suppose, on the other hand, that the closure is linearly modeled in all respects. This presumably includes surface roughness. Since sealing is due to forcing the gasket material to conform to surface roughness sufficiently so that every path for escape is blocked, it is not clear whether the gasket hardness should be changed. Also, if gasket damage from lateral displacement of the lid is partly dependent upon surface roughness, it is not clear that the smoother surface and lesser displacement for a model would yield valid results.

The questions and problems with closures have been mentioned not with the promise or expectation of immediate answers, but rather as a suggestion that this is an important area of shipping container design where there are large gaps in our knowledge.

SUMMARY

The Savannah River Plant has found model testing to be a very useful technique for assessing the performance of containers under impact conditions of the hypothetical accident. For complex and previously untested features of casks, such testing is considered more reliable than calculation, and frequently there is no practical alternative. As time goes on, the larger store of test results and the development of better basic data for reliable and inexpensive calculation should reduce the amount of testing which would otherwise be performed.

Much of the effort has been directed toward demonstration of adequate leaktightness. Unfortunately this is a rather nebulous area with regard to what is adequate and with regard to appropriate methods of modeling a closure. The sensitivity of tests and the degree of confidence to be established should be related to the hazard involved. Further study and technical guidance would be helpful to designers and very likely to competent authorities.

REFERENCES

1. H. G. Clark, Jr. and M. M. Reddi, Structural Integrity of Shipping Containers for Radioactive Materials, Part I: Study of Transport Operations and Container Construction, Appendix A: Feasibility of Applying Model Theory to Simulate Impact Damage of Shipping Containers. The Franklin Institute Interim Report I-A2412-1, U. S. Atomic Energy Commission Report NYO-9859 (July 1962).
2. H. G. Clarke, Jr., Impact Resistance of Casks, American Institute of Chemical Engineers, Nuclear Engineering Part XIV, number 56, volume 61, (1965).
3. L. B. Shappert, Cask Designers Guide, ORNL-NSIC-68. (February 1970.)
4. C. A. Wilkins, et al Design and Testing of Curium Shipping Capsule and Cask, Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, October 14-18, 1968. CONF 681001. p. 344.
5. R. A. Scaggs, Upgrading of PM-3A Casks, Proceedings of Third International Symposium on Packaging and Transportation of Radioactive Materials (August 1971).
6. C. B. Clifford, Demonstration Fuel Element Shipping Cask from Laminated Uranium Metal-Testing Program, Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, October 14-18, 1968. CONF 681001. p. 521.
7. Regulations for the Safe Transport of Radioactive Materials, Third Revised Draft, PL-383. International Atomic Energy Agency. November 1970.

TABLE 1. DROP TESTS OF MODEL OF CURIUM CASK

Test No.	Drop Situation	Damage Results
Drop Series I		
1	30 feet, top edge impact	Landed on lifting lug; lug buckled. Local damage of 5 fins. 3 bolts securing puncture shield broken. No damage to body shell.
2	30 feet, flat top impact	Another lifting lug deformed.
3	40 inches, top puncture	Puncture shield permanently deflected 3/8 inch. No puncture, off-center.
4	40 inches, side puncture	Local deformation of one fin; fin slightly bent. No puncture.
5	30 feet, side impact	Landed squarely on 6 fins. Body shell permanently deflected 7/16 inch max along impact line. One horizontal internal member connected to impact side of shell buckled, product capsules remained centered in shell.
6	30 feet, flat bottom impact	Some deformation of bottom fins.
7	30 feet, bottom edge impact	6 side fins smashed at bottom edge. Cask bottom slightly convex.
Drop Series II		
8	30 feet, top edge impact	No damage to new puncture shield.
9	30 feet, flat top impact	No damage to new puncture shield.
10	40 inches, top puncture	Puncture shield permanently deflected 1/2 inch. Dead-center hit.
11	15 feet, top puncture	Puncture shield permanently deflected 1 1/4 inches. Lid top slightly deflected. No puncture. Dead-center hit.
12	30 feet, side puncture	Sheared one fin. Bent four fins. Permanently deformed side. No puncture.
13	30 feet, top puncture	Off-center hit. Shield permanently deflected 1 1/4 inches max. Two puncture shield bolts sheared. Lid top permanently deflected 1/4 inch. No puncture.
14	30 feet, bottom puncture	Flat hit in center. Two fins sheared. Puncture of bottom clear through about 3/4 of pin circumference.

CURIUM SHIPPING CASK

FIGURE 1

**AFTER
DROP NO. 1**

FIGURE 2

**INTERNALS
AFTER
DROP NO. 1**

FIGURE 3

**AFTER
DROP NO. 7**

FIGURE 4

FIGURE 5

FIGURE 6

FIGURE 7

FIGURE 8

BISMUTH CASK

FIGURE 9

DEFORMED STUDS & NUTS

FIGURE 10

TOTAL WEIGHT WITH WATER 260 POUNDS.

RUPTURE DISK TEST MODEL

FIGURE 11

**PM-3A CASK
BEFORE
MODIFICATION**

FIGURE 12

**MODEL WITH
SHROUDS
REMOVED**

FIGURE 13

DAMAGE FROM TOP CORNER DROP

FIGURE 14

ONE SHROUD SECTION REMOVED

FIGURE 15

BOTTOM CORNER DROP

FIGURE 16

BOLT FAILURE FROM BOTTOM DROP

FIGURE 17

FIGURE 18

**MODIFIED
MODEL
READY FOR DROP**

FIGURE 19

**AFTER
DROP**

FIGURE 20

ON THE PREDICTION OF DEFORMATION AND
DECELERATION OF A COMPOSITE CYLINDRICAL
BODY FOR THE CORNER DROP CASE

L. Lee

ABSTRACT

Using the concept of dynamic flow pressure, an energy balance was developed for the impact condition of a corner drop. The wedge shaped deformation as well as the deceleration of a composite cylindrical body caused by a corner drop was then related to the energy level of the impact. Results predicted by this empirical corner drop correlation were compared with the cask drop tests performed by H.G. Clarke. The agreement is reasonably good.

INTRODUCTION

In the design of a cask for the transport of radioactive materials, the regulations¹ requires that "a free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected", must be investigated. This statement implies two things. One is the direct damage at the impact plane due to deformation and the other is the indirect damage at other locations due to deceleration. From a physical standpoint, the maximum direct damage would occur with the cask so oriented that the line passing between the center of gravity and the point of impact coincides with the direction of fall. The geometrical representation of the corner drop is shown in Figure 1. The idealized deformation, the external damage after impact, is indicated as Z.

No standard method is presently available for the estimation of the value of Z of a composite cylindrical body or its deceleration. H.G. Clarke² has performed some actual corner drop tests using scale models. He suggests that the test data may be used to predict the deformation of a prototype by means of a scaling factor. It is probable that Clarke's scaling approach is the only method now available for a corner drop. L.B. Shappert³ has formulated a procedure to obtain the deformation for both end and side drops. Experiments were performed to support his formulations. The three positions for drop study have one thing in common, material flows or displaces under dynamic conditions. It is therefore logical to present a procedure for the corner drop case similar to those given by Shappert for which Clarke's test data can be used as a check. This has been worked out in a semi-analytic way and the results are promising. In carrying out the correlation for deformation, deceleration can also be obtained. The details of this work is presented in the following presentation.

FIG. I. CORNER DROP REPRESENTATION

Empirical Correlation

When the impact velocity exceeds a certain limit, plastic deformation will take place. The material under impact will shift from its original position, that is, it flows. The displacement of material requires a certain amount of work. Relating the impact energy to the displaced volume, a parameter is obtained as

$$K = \frac{d(E)}{d(Vol)}, \text{ in-lb/in}^3 \quad (1)$$

where

E = impact energy

Vol = displaced volume

L.B. Shappert⁴ has done some impact tests on solid lead. He has found that the slope of E-Vol plot is almost constant. That means eq.(1) can be rearranged as

$$E = K (Vol) \quad (2)$$

where

K is energy expended in displacing one unit volume. Shappert called it the dynamic flow pressure. It will be named here as the energy absorption constant to emphasize its relation to impact energy. It should be noted that K is not a material constant such as strain energy or flow stress but varies with geometrical configurations due to the extra work in displacing the material. As an example, for the same deformed volume, a straight bar absorbs less energy than a wedge-shaped stock.

Equation (2) states that the energy absorbed by the deformation is proportional to the volume displaced. Shappert⁵ has applied this equation to an end drop as well as side drop. His predicted values were found to be very close to the test results. Based on these two cases, it would be expected that a corner drop might show similar agreement.

Consider a solid cylindrical body of weight W, dropped freely from a certain height H. According to equation (2), we have

$$WH = K (Vol) \quad (3)$$

Referring to Figure 1, it is obvious that the displaced volume is related to the deformation or simply

$$Vol = f(Z) \quad (4)$$

Thus a relation of Z and Vol has to be established before Equation

(3) can be applied.

Wedge (Ungula) Volume of Right Circular Cylinder

The displaced volume below the impact plane is shown in dotted lines, Figure 1, whereas Z represents the accompanying deformation. The volume of the wedge can be calculated from various formulas. One such formula⁶ is given below

$$\text{Vol} = R^3 \tan(A) \left[\sin B - \frac{\sin^3 B}{3} - B \cos B \right] \quad (5)$$

where

A = angle of impact

B = half angle of the extended deformed arc

From the geometrical relation of the wedge shown in Figure 2, we have

$$Z = R (1 - \cos B) \sin A \quad (6)$$

Let

$$(\text{Vol})_n = \frac{\text{Vol}}{R^3} \quad (7)$$

$$Z_n = \frac{Z}{R} \quad (8)$$

Then equations (5) and (6) become

$$(\text{Vol})_n = \tan A \left(\sin B - \frac{\sin^3 B}{3} - B \cos B \right) \quad (9)$$

$$Z_n = (1 - \cos B) \sin A \quad (10)$$

$(\text{Vol})_n$ and Z_n are related by the two parametric equations (9) and (10). By eliminating B, the two equations can be combined into one single equation as

$$(\text{Vol})_n = f(A, Z_n)$$

With a given value of A and Z_n , the normalized volume $(\text{Vol})_n$ can be directly calculated by the use of the above equation. However, by direct computation, it has been found that the single equation can have the following simple empirical form

$$(\text{Vol})_n = C (Z_n)^{2.38} \quad (11)$$

where

C = a coefficient which varies with impact angle "A"

FIG. 2. GEOMETRY OF CYLINDRICAL WEDGE

According to equations (7) and (8), equation (11) may be converted back as

$$\text{Vol} = CR^{.62}(Z)^{2.38} \quad (12)$$

This is the empirical equation relating the deformed volume directly to the deformation.

Energy Balance For Composite Circular Cylinder--For a solid circular cylinder, an energy balance can be set up by means of equations (3) and (12) as

$$WH = KCR^{.62}(Z)^{2.38} \quad (13)$$

However, actual casks are usually built as a composite structure with lead shielding enclosed within a steel shell. During a corner drop, the steel will be deformed first with the lead layer as its backing. After the impact starts, it will be a continuous process even if the time duration is very short. The pictures in Clarke's report² indicate that the final shape of the crushed shell seems to be formed from small local plastic changes. During the impact, the deformed portion enlarges gradually and the plastic front moves on at a definite speed until all the impact energy is exhausted. This reveals that deformation of the steel begins

Z_1 = deformation of steel upto the contact point
of steel and lead at the corner, in.

Z_2 = deformation of lead, in.

Z_0 = external deformation, in.

t_s = thickness of shell, in.

R_1 = outside radius of steel shell, in.

R_2 = radius of contact surface between steel and
lead, in.

From the geometry shown in Figure 3, the following relations can
be established:

$$Z_1 = 2 t_s \cos (45^\circ - A) \quad (14)$$

$$A = \tan^{-1} \frac{R}{h} \quad (15)$$

The deformed volume consists of two parts, steel and lead.
For steel, the volume assumed is that under the impact plane X - X
which is

$$(Vol)_{st} = CR_1^{.62} (Z_0)^{2.38} - CR_2^{.62} (Z_0 - Z_1)^{2.38}$$

For lead, the deformed volume is

$$(Vol)_{pb} = CR_2^{.62} (Z_2)^{2.38} = CR_2^{.62} (Z_0 - Z_1 + t_{st})^{2.38}$$

Applying equation (3), we have

$$\begin{aligned} WH = & K_{st} [CR_1^{.62} (Z_0)^{2.38} - CR_2^{.62} (Z_0 - Z_1)^{2.38}] \\ & + K_{pb} CR_2^{.62} (Z_0 - Z_1 + t_{st})^{2.38} \end{aligned} \quad (16)$$

where

K_{st} = energy absorption constant for steel wedge, in-lb/in³

K_{pb} = energy absorption constant for lead wedge, in-lb/in³

From equation (16), the external deformation can be solved by
trial and error.

Comparison with Model Test Results

The energy balance equation (16) may now be tested by apply-
ing it to Clarke's tests. Before any calculation is made, the
values of K_{st} and K_{pb} must be specified. The deformed volume of
lead is a solid cylindrical wedge for which a constant K_{pb} would
be expected. However, the steel shell would not be so simple.
With a deformation up to Z , Figure 3, the deformed volume is a

as plastic deformation and continues as such throughout the duration of the impact. In regard to stress, it may be said that the deformed material fails at same stress condition. Therefore it is reasonable to assume that the impact energy absorbed by a steel shell is directly proportional to the volume of displaced material. This means equation (3) can also be applied to steel corner.

Figure 3 shows a composite cylinder with the dimensions defined in the following:

FIG.3 IDEALIZED DEFORMED COMPOSITE CYLINDER

solid wedge for which K_{st} remains constant. After this limit is exceeded, the geometry is changed. A solid wedge becomes two formed steel plates. For thin shell structure, the solid wedge is only a small portion of the total deformed mass. But for thick shell, the energy absorbed by the solid wedge will predominate in the lumped energy absorbing constant. Therefore it is expected that thin shell would not have the same K_{st} as for a thick shell. In the extreme case, it became a solid wedge if

$$Z_0 < Z_1$$

In order to be as consistent with the test data as possible, the values used for models are tabulated as follows

Table 1. Values of Energy Absorption Constant used for Comparison with Tests

	<u>Energy Absorption Constant, in-lb/in³</u>		
	<u>CL - 0.05</u>	<u>CL - 0.106</u>	<u>CL - 0.173</u>
K_{st}	50,000	60,000	110,000
K_{pb}	8,000	8,000	8,000

Based on the above assumed values, deformations on Clarke's test models are calculated using equation (16). The results have been added to a plot of Clarke's test results to enable comparison (Figure 4). Clarke's test report showed only a few points. Indeed, it may not represent the general trend of deformations since the impact test is hardly reproducible and usually there is much scatter even for low-velocity levels. In Figure 4, test data points are shown and the solid lines represent the calculated values.

DECELERATION

When a mass is dropped freely from a height, the motion will be stopped in a short time after the mass touches the impact plane. During impact the velocity decreases from its initial value at contact because of the energy loss. Without resilience, the velocity would become zero after the impact is completed. The rate of change of the velocity, the deceleration, has an important role in the design of a transportation cask. It is obvious that the deceleration is a function of deformation. Since the deformation has been correlated with deformed volume as given by equation (13), it is possible now to estimate the deceleration directly for a

FIG. 4. EXTERNAL DEFORMATION

corner drop.

The energy loss at a deformation level Z is

$$E_1 = 12 W \frac{V_0^2 - V^2}{2g}, \text{ in-lb} \quad (17)$$

where

V_0 = initial velocity, ft/sec.

V = remaining velocity at position $Z = Z$.

Relating to displaced volume of a solid cylinder, the energy absorbed is

$$E_a = KC(R) \cdot 62(Z)^{2.38} \quad (18)$$

From energy balance,

$$E_1 = E_a$$

$$12 W \frac{V_0^2 - V^2}{2g} = KC(R) \cdot 62(Z)^{2.38} \quad (19)$$

Differentiating equation (19) with respect to time, t

$$12W \frac{(-)2V}{2g} \frac{dv}{dt} = 2.38 KC(R)^{.62} (Z)^{1.38} \frac{dz}{dt} \quad (20)$$

where

$$(-) \frac{dv}{dt} = a, \text{ deceleration, ft/sec}^2$$

$$1/12 \frac{dZ}{dt} = V, \text{ ft/sec}$$

rearrange equation (20) and simplify,

$$a = 2.38 \frac{g}{W} KC(R)^{.62} (Z)^{1.38}$$

in terms of g's, the deceleration is

$$\begin{aligned} N_g &= \frac{a}{g} = \frac{2.38}{W} KC(R)^{.62} (Z)^{1.38} \\ &= \frac{2.38}{W} KC(R)^{.62} \frac{(Z)^{2.38}}{Z} \end{aligned} \quad (21)$$

when the motion is completely stopped, the maximum deformation level is reached, say $Z=Z_0$, then all the impact energy is absorbed. Equations (19) and (21) become

$$WH = KC(R)^{.62} (Z_0)^{2.38} \quad (22)$$

$$\text{and} \quad \text{max. } N_g = \frac{2.38}{W} KC(R)^{.62} \frac{(Z_0)^{2.38}}{Z_0} \quad (23)$$

combine equations (22) and (23), we obtain

$$\text{max. } N_g = 2.38 \frac{H}{Z_0} \quad (24)$$

where

H = height of drop, 360 inches

Z_0 = maximum deformation, in.

By the same procedure, the maximum deceleration for a composite structure of steel and lead (Figure 3) would be

$$N_g = 2.38 \frac{H}{Z_0} (M) \quad (25)$$

where

$$M = \frac{1 - \left(\frac{R_2}{R_1}\right)^{.62} \left(1 - \frac{Z_1}{Z_0}\right)^{1.38} + \frac{K_{pb}}{K_{st}} \left(\frac{R_2}{R_1}\right)^{.62} \left(1 - \frac{Z_1}{Z_0} + \frac{t_s}{Z_0}\right)^{1.38}}{1 - \left(\frac{R_2}{R_1}\right)^{.62} \left(1 - \frac{Z_1}{Z_0}\right)^{2.38} + \frac{K_{pb}}{K_{st}} \left(\frac{R_2}{R_1}\right)^{.62} \left(1 - \frac{Z_1}{Z_0} + \frac{t_s}{Z_0}\right)^{2.38}} \quad (26)$$

= the correction factor which has a maximum value of 1.

Conclusion

For a corner drop, both the direct effect on deformation at impact plane and the indirect effect on deceleration can be estimated by means of the empirical equations. The comparison between the predicted and the test values shows good agreement. In adjusting the curves, the value of the energy absorbing constant has to be varied with shell thickness. The curve for CL-0.05 does not fit very well with the three test points. However, the test data for CL-0.05 seems not to be consistent with the data for the other models. From physical considerations, the thin shell model should have the largest slope for deformation when impact energy is the variable. In other words, the slope of the deformation-energy curve at each impact level should follow the inequality relations

$$\left(\frac{dZ}{dE}\right)_{CL-0.05} > \left(\frac{dZ}{dE}\right)_{CL-0.106} > \left(\frac{dZ}{dE}\right)_{CL-0173}$$

Comparison with the test data shows the three points for CL-0.05 not to be in agreement with the above inequality expression. It may be concluded that this inconsistency in the test data is probably due to scatter. Without the availability of more elaborate tests (including unequal thicknesses of shell and bottom) the present approach offers a handy and reasonably method for the prediction of deformation and deceleration for a corner drop.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 71, Packaging of Radioactive Material for Transport.
2. H.C. Clarke, "Some Studies of Structural Response of Casks to Impact", Proceedings of the Second International Symposium of Packaging and Transportation of Radioactive Materials, pp. 373-398, (1968).
3. L.B. Shappert, Irradiated Fuel Shipping Casks Design Guide, ORNL-TM-2410, pp. 48-53, (1969).
4. L.B. Shappert, A Guide to the Design of Shipping Casks for the Transportation of Radioactive Material, ORNL-TM-681, pp. 101-104, (1965).
5. Reference 3, pp. 54-55.
6. O.W. Eshbach, Handbook of Engineering Fundamentals, Item 32, pp. 2-42, (1936).

X ECONOMICS, COMPOSITIONS AND PROCESS EQUIPMENT FOR
RADIOACTIVE WASTE SOLIDIFICATION

L. Rutland

ABSTRACT

From ecological and safety considerations, solidification of radioactive wastes results in a desirable waste product form for transportation and subsequent disposal. Government regulations may and some commercial burial facilities now require radioactive wastes from nuclear plants to be handled as a free standing solid waste product. Suitable solidified radioactive waste product compositions, economic considerations associated with transportation and disposal, and a waste solidification process system are discussed.

INTRODUCTION

Large volumes of low and intermediate activity levels of radioactive wastes may be generated in nuclear reactor process systems and coolant purification systems. Unlike radioactivity induced in metallic reactor components which are essentially non-corroding, these radioactive wastes when buried for disposal are more susceptible to dispersion in the environment and may therefore be termed "available".

The radioactive waste materials considered in this paper are ion exchange resins, filter sludges, and evaporator concentrates. The radioactive components contained within these materials are better isolated from the biosphere if the wastes are cast in a solidified matrix. Cement or cement containing mixtures have been to date the most popular medium to achieve a free standing solidified waste product for disposal by burial.

Current Department of Transportation regulations concerning the packaging and transportation of radioactive materials do not require a free standing solidified waste product for transportation. It is obvious however that a solidified waste product would be desirable to reduce dispersion and facilitate cleanup in case of accidental rupture of a transportation container bearing these waste materials. On the other hand, most commercial burial facilities do require a free standing solidified radioactive waste material in order to be accepted for burial. It becomes apparent that it would be prudent for nuclear power utilities to provide at least the flexibility of adding solidification capability in the future should Department of Transportation regulations be changed or that all burial sites require a solidified waste product.

Indeed, the location of a nuclear power plant or specific conditions may make the solidification of the radioactive waste materials in question the most economical waste product form for disposal. Vital to the cost of radioactive waste disposal are the compositions of solidified radioactive waste materials. Compositions are discussed in the following sections as well as some economic considerations of solidification. Also discussed is the process system provided by ATCOR Inc. for the solidification of these materials.

COMPOSITIONS OF SOLIDIFIED RADIOACTIVE WASTES

Specifications for cemented radioactive waste have not been issued by governmental agencies or private burial facilities. As a guideline for a satisfactory solidified product, ATCOR Inc. recommends the two following properties for a disposable waste product:

1. the waste product should be free standing
2. the cemented product should remain intact after complete submergence in water.

As a result of experimentation, specific mixtures of the various wastes (resins, filter sludges or evaporator concentrates) and cement have been developed which satisfy the above two conditions. An acceptable mixture suitable for the solid waste

materials, bead resin, powdex, filter sludges or even mixture of these materials is one part of moist dewatered solid to an equal part of dry cement with sufficient water then added for hydration. This is an acceptable composition for unspecified mixtures of the solid waste materials, however, more specific proportions can be specified with a single waste material (rather than mixtures of the various waste materials) to maximize the amount of radioactive waste in the cemented mixture.

For planning purposes it is useful to consider that dewatered solids for disposal comprise about two thirds of the cemented mixture while cement itself comprises the remaining one third, by volume. The volume of a cemented waste product after mixing of the individual components may be obtained from the following expression:

$$\text{Cemented waste volume} = \text{Volume of dewatered waste material} + 0.5 \text{ volume of dry cement}$$

Similarly, one volume of evaporator concentrates with an equal proportion by volume of dry cement results in an acceptable mixture.

The density of the cemented waste products is approximately 100 pounds per cubic foot of mixture.

ECONOMIC CONSIDERATIONS FOR SOLIDIFIED WASTE PRODUCTS

At present the decision to solidify waste is optional since some commercial burial facilities will still accept un-solidified dewatered wastes. Aside from the desirable ecological or safety features of solidified wastes, it will be seen that it may even prove to be cheaper to solidify radioactive wastes. The overall economics depend on the proximity to a particular burial site or whether dewatered wastes (resins or filter sludges) or evaporator concentrates are to be disposed of.

Dewatered Solids (resins or filter sludges)

With dewatered solids, solidification tends to increase disposal costs since the addition of cement or cement mixtures:

1. makes for a heavier waste product

2. reduces the amount of radioactive waste in the transportation package
3. involves the added cost of the cement itself.

If however, the nuclear power plant is located closer to a burial facility which requires solidification, the reduced transportation costs (for example, shipping to Morehead, Kentucky) may more than offset the cost factors noted above. Also, if activity levels are sufficiently high to require a shielded transportation cask then items 1. and 3. above become second order effects and therefore increase costs only slightly.

In this case, the weight of the contents is usually small compared to the weight of the shielding material in the transportation container and the cost of cement is small compared to the transportation costs.

While it appears that solidification reduces the actual volume of radioactive waste in a particular shipment (item 2.), it should be noted that there is a compensating effect with solidification. The addition of cement increases the density of the waste product and thereby increases the self attenuation of radiation by the mixture itself. Whereas the volume of the radioactive waste is increased by about 50% the radiation dose rate on the surface of the transportation container will be reduced by a similar degree. Thus, with a solidified waste product, the number of curies of solidified radioactive material that can be shipped is about the same as with an unsolidified (dewatered) product.

Evaporator Concentrates

Evaporator concentrates cannot be shipped as a liquid and must at least be contained with an absorbent such as vermiculite. Solidification of evaporator concentrates with cement does not impose a penalty by virtue of increased volume since a cemented mixture of evaporator concentrates could contain about as much liquid radioactive waste as a vermiculite absorbed product. In addition the increased self attenuation would be beneficial and solidification in cement would then be the desirable product form for disposal.

THE ATCOR SOLIDIFICATION SYSTEM

Some nuclear power plants have separate systems to collect, concentrate and package for disposal resin, filter sludges and evaporator concentrates. This makes for a diversity of process equipment, transportation packages and shielded transportation containers. Actually, radioactive resins, filter sludges, evaporator concentrates could all be handled in a similar manner and shipped in the same type of transportation container. The result is a savings in capital equipment and greatly simplified handling procedures.

The ATCOR solidification system (presently installed in a major nuclear power utility) accepts resins, filter sludges or evaporator concentrates and conditions waste material. This waste and a separate feed of cement are metered into a continuous small volume mixer. The cemented radioactive waste product is then discharged directly into a transportation package. Controlled discharge of the waste as a moist material precludes the possibility of airborne contamination. An efficient dust collector prevents release of cement dust into the plant or ventilation system.

Full instrumentation is provided on the control panel for complete readout of flow rates and operating condition of components. Interlocks and alarms are provided to prevent operation in an unsafe manner. For example, no cement waste can be discharged unless a waste container is properly in place and no radioactive waste can be fed unless cement is also being provided. While control of individual components is possible under the manual mode of operation, normal operation makes full use of functional controls. Operator error is minimized since controls are automated such that sequential operation of functional pushbuttons and controls cannot be performed unless the previous step has been completed and the system is in the required configuration.

Some advantages of the ATCOR solidification system are listed:

1. A single process system can handle bead resin, powdered resin, filter sludges, evaporator concentrates and can even solidify spent filter cartridges in a transportation container.

2. Proportions of waste and cement can be altered to suit the specific waste material to maximize the amount of radioactive waste in the disposal product. Transportation and disposal costs are reduced.
3. Operation is inherently clean to prevent either radioactive or non-radioactive spatter or airborne contamination.
4. Filling of a package takes minimal time, for example only about five minutes for a 55 gallon drum.
5. No expensive special drums or time consuming pre-filling of drums is required. Prefilling mixtures in other systems can separate and result in a non-free standing waste product.
6. Internal surfaces of process equipment in contact with radioactive cement is small. Equipment is automatically flushed after completion of waste processing such that no cement bearing flush water enters plant piping systems.
7. Fully interlocked functional controls minimize the possibility of operator failure or system operation in an unsafe manner.
8. No manual connections or handling is required so that radiation exposure to personnel is minimized.
9. Filling of transportation packages with highly radioactive waste can be performed remotely without the need for separate shielding for individual packages.

SUMMARY

Present regulations do not require solidification of radioactive waste for transportation and disposal. It was shown however that solidification could actually result in lower waste disposal costs depending on the proximity to neighboring burial facilities and also depending on whether the waste is a dewatered material or evaporator concentrate. With a preponderance of radioactive waste in the form of evaporator concentrates it seems that solidification with cement is clearly preferable to the use of an absorbent. While solidification reduces the quantity of waste shipped in a container self attenuation by the waste itself permits equal curie content in solidified form as in unsolidified form. In any case, the advantages of solidification make it the desired disposal product form even in cases where solidification results in somewhat increased costs.

A solidification system is currently available which can handle bead type resin, powdered resin, filter sludges, evaporator concentrates and even spent filter cartridges. This system can fill cemented waste into standard and readily available containers so that the nuclear power utility can freely select its waste disposal contractor. Operation of this system offers the advantages of

1. reduced manpower requirements for processing of wastes.
2. reduced radiation exposure to operating personnel.
3. maximum waste content in cemented mixtures for lowest disposal costs.

✓ APPLICATION OF FISSILE MATERIAL STORAGE LIMITS TO CRITICALITY
SAFETY ANALYSIS IN TRANSPORT

J. T. Thomas

ABSTRACT

Knowledge of the criticality of spaced subcritical spherical masses in water-reflected cubic arrays can be used to estimate allowable mass limits for containers of fissile material in transportation. Calculated criticality data for representative fissile materials are tabulated. The methods of calculation as well as the techniques required to employ the data in the solution of arbitrary problems of array criticality are described. Monte Carlo-determined critical arrays characterize criticality for various forms of fissile materials and the factors affecting their criticality. Applications of the data and methods to typical transport containers are given.

INTRODUCTION

The problem of providing an acceptable shipping container for fissile materials is complex. Setting aside the practical difficulties of regulatory requirements in 10 CFR 70 and of design, testing, and documentation, there is left the question of criticality evaluation. To the need for determination of the criticality of air-spaced units of fissile materials, one can add the separate influences of reflection, moderation, rearrangement of units, mixtures of fissile materials, and finally, the neutron scattering and absorption properties of the container and insulating materials. Sufficient theories and techniques are available today to satisfy administrative and regulatory requirements in this regard. For economic reasons alone, it is desirable that verification of subcriticality be simple, direct, and easily substantiated.

It seems reasonable to begin with the criticality of hydrogen-reflected, air-spaced units in regular geometry. This will provide both a common orientation for analyses and, incidentally, a basic value for the reactivity of the arrays of fissile material. One can then compensate for the reactivity effect of changes in the array configuration and structure by an acceptable scale, measured in terms of the mass reduction equivalent to the reactivity change. Fortunately, a good beginning to, and understanding of, this basic requirement has recently appeared. It is contained in the output of a group active in the

nuclear standards effort sponsored by the American Nuclear Society. The ANS-8 Standards Subcommittee on "Fissionable Materials Outside of Reactors" has produced a "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials." The preparation of this Guide has necessitated the investigation and evaluation of factors complicating the analysis of criticality of simple arrays, including those mentioned above except for container and insulating materials. While one might, in general, investigate the influence on the neutron multiplication factor of the scattering and absorbing properties of materials occupying the space between the units of the arrays, the required effort lacks both support and interest. An alternative is to be content with two complementing but separate bodies of information: descriptions of critical arrays and the criticality analyses of a limited number of containers. The import of the Guide is the tabulations of masses of fissile materials that may occupy fixed volumes in water-reflected arrays. Following a discussion of these calculated criticality data and some of the interpolative techniques which render them useful to practical problems, the factors affecting criticality will be treated and the application of the result to transport suggested.

ARRAY CRITICALITY

The materials considered were the three principal fissile isotopes, ^{235}U , ^{233}U , and ^{239}Pu , both as metals and as dioxides. The oxides were assumed to contain various quantities of water with concomitant reduction in their theoretical densities. The ^{235}U -enriched uranium considered had concentrations of the fissile isotope ranging from 30 to 100 wt % with the remainder being ^{238}U . Uranium of lower enrichments can be evaluated by similar techniques. The uranium-233 was assumed isotopically pure. The plutonium was assumed to be composed of ^{239}Pu and ^{240}Pu although other plutonium isotopes were appropriately included in them. The notational convention of $\text{U}(x)$ or $\text{Pu}(x)$ was adopted, designating the content of ^{235}U or ^{239}Pu in wt %. The water content was expressed in terms of the ratios of hydrogen atoms to all uranium or to all plutonium atoms without regard to isotopic composition. These ratios ranged from zero to twenty. This treatment of materials may seem idealistic but, nevertheless, is acceptable under the desired margins of subcriticality in practical application and does provide a suitable point of departure for nuclear safety analyses.

The fissile materials were considered as spherical units centered in the cubic cells of an array. It has been demonstrated¹ that spheres define mass limits per cell which may be conservatively applied to units of other shapes

contained in reflected arrays of sizes that may be of concern to most storage and transport problems. The number of calculations necessary to characterize the criticality of a given fissile material is minimized by considering reflected cubic arrays in which each unit is centered in a cubic cell. For each array of a given number of units with specified cell dimension there is determined a unique radius, r_c , for each form of fissile material. The calculations defining criticality were performed with the KENO² Monte Carlo code and the 16-group Hansen-Roach³ neutron cross-section sets. This code and the cross-section sets constitute a validated method⁴ for defining criticality of uniform arrays of fissile materials.

Validation implies evaluation of the method of calculation by correlation with experimental data to establish bias over the area of applicability. Two additional provisions are required. The degree of consistency of the cross-section sets used with differential and with appropriate integral data should be established. Finally, the computer program should be checked to assure that it performs the intended mathematical operations properly. In the absence of confirming experimental data, sufficient conservatism should be employed to assure subcriticality in practical applications.

Table 1 summarizes the calculated critical spherical radii for four fissile materials, U(93), Pu(100), ²³³U, and U(30)O₂ at an H:U of 10. The format is similar to that adopted for presentation in the Guide. Arrays of seven different sizes, each composed of cells of six different dimensions, comprise a total of 42 arrays for each material considered. The arrays ranged from 64 to 1000 units and the cell half dimensions from 12.70 to 30.48 cm.

Let us digress for a moment and consider some relationships among those tabulated data that will enable a more comprehensive understanding and will extend their range of applicability, now limited by fixed radii, dimensions of cells, and number of units. Each entry of Table 1 describes an array by specifying the half dimension of a cell, a_n , where the subscript n is the number of units along an edge of the array, and by the corresponding radius, r_c , of the fissile unit. The reflected cubic array then contains $N (= n^3)$ spheres of radius r_c whose centers are spaced a distance of $2a_n$. The values of n and of a_n for each value of r_c may be related through a coupling parameter NB_N^2 by the relation⁴

$$NB_N^2 = \frac{3\pi^2}{4} \frac{n}{a_n^2} \left(1 - \frac{0.312}{\sqrt{N}}\right)^2 \quad (1)$$

A graphical representation of all the data contained in Table 1 is shown in Fig. 1 where NB_N^2 is given as a function of a_n and the lines labeled $n = 4, 6,$

Table 1. Spherical Radius in Centimeters of the Fissile Material Required for Criticality in Water-Reflected Cubic Arrays

Number of Units in Array	Half Dimensions of Cubic Cell (cm)					
	12.70	15.24	19.05	22.86	25.40	30.48
Uranium Metal; H:U = 0; 18.76 g U/cm ³ Maximum, 93.2 wt % ²³⁵ U, 6.8 wt % ²³⁸ U						
64	5.408	5.780	6.480	6.880	7.160	7.520
125	5.100	5.400	6.190	6.550	6.835	7.250
216	4.836	5.166	5.880	6.330	6.666	7.058
343	4.627	4.977	5.677	6.083	6.440	6.818
512	4.440	4.816	5.520	5.883	6.280	6.698
729	4.248	4.671	5.355	5.751	6.165	6.516
1000	4.100	4.580	5.220	5.637	6.075	6.408
Plutonium Metal; H:U = 0; 19.70 g Pu/cm ³ Maximum, 100 wt % ²³⁹ Pu						
64	3.670	3.920	4.184	4.414	4.460	4.580
125	3.530	3.790	4.015	4.250	4.350	4.460
216	3.399	3.641	3.915	4.156	4.277	4.375
343	3.280	3.528	3.794	4.046	4.183	4.312
512	3.175	3.432	3.723	3.986	4.120	4.285
729	3.060	3.339	3.654	3.906	4.050	4.230
1000	2.970	3.259	3.576	3.844	3.919	4.183
Uranium Metal; H:U = 0; 18.40 g U/cm ³ Maximum, 100 wt % ²³³ U						
64	4.034	4.371	4.736	5.013	5.179	5.395
125	3.865	4.162	4.520	4.795	4.975	5.225
216	3.666	3.983	4.393	4.657	4.817	5.115
343	3.577	3.892	4.270	4.550	4.739	5.005
512	3.415	3.749	4.116	4.472	4.688	4.862
729	3.316	3.600	4.005	4.311	4.500	4.770
1000	3.202	3.512	3.926	4.242	4.416	4.733
Oxides of Uranium; H:U = 10; 2.067 g U/cm ³ ; 30 wt % ²³⁵ U, 70 wt % ²³⁸ U						
64	8.624	9.616	10.846	11.981	12.574	13.636
125	8.000	8.950	10.150	11.235	11.785	12.900
216	7.548	8.419	9.648	10.651	11.237	12.334
343	7.175	8.036	9.128	10.143	10.724	11.830
512	6.867	7.681	8.778	9.781	10.386	11.408
729	6.642	7.416	8.460	9.432	10.044	11.070
1000	6.375	7.170	8.191	9.100	9.724	10.764

Fig. 1. Calculated Criticality or Water-Reflected Array Showing the Relationship Between Half-Dimension of Cubic Cells, Number of Units along Edge of Cubic Array, and Spherical Radius of Fissile Materials.

Fig. 1. Calculated Criticality or Water-Reflected Array Showing the Relationship Between Half-Dimension of Cubic Cells, Number of Units along Edge of Cubic Array, and Spherical Radius of Fissile Materials.

8, and 10 are specified by Eq. (1). Since each of the calculated arrays has a corresponding unique spherical radius for the fissile unit, each value of r_c may be associated with the NB_N^2 value determined by Eq. (1). The relation is presented to the right of the figure as the dimensionless ratio of r_c to the calculated unreflected critical radius r_0 for each of the fissile materials considered. The curves in Fig. 1 allow interpolation of data derived for the specific arrays recorded in Table 1, thereby enabling application to many other situations. Since the relation between n and a_n is invariant to the type of fissile material present in the arrays, as few as seven calculated arrays of a given fissile material, properly selected over a range of interest, are sufficient to characterize that material.

An additional observation to come from the efforts of ANS-8 and, perhaps, a key to establishing uniformity in the selection of safety margins, i.e., a defined degree of subcriticality, is a relation derived from the behavior of the array neutron multiplication factor in response to a reduction in the radius of the spherical units. It was determined that the calculated k_{eff} of the array does not exceed the fraction r/r_c where r is the radius of spherical units in the subcritical array. It was also demonstrated that this relation holds over a wide range of k_{eff} values and under a variety of conditions. Application of this relation between k_{eff} of the array and changes in the ratio of radii permits adjustments in the array necessary to compensate for the effects on reactivity of the perturbations noted previously, provided the magnitudes of the changes in k_{eff} resulting from the perturbations is known. For example, factors which decrease the reactivity are changes in the shape of the unit, the cell, or the array, provided reasonable separation of units is not abandoned. Factors which may increase the reactivity are a change in the arrangement of units in the tiers of an array from square to triangular, adding hydrogenous moderation to the space between units, or replacing the water reflector by concrete. Whereas one does not, in general, take credit for the former, he certainly is required to debit the latter.

APPLICATIONS TO TRANSPORT

Regulations, through the assignment of transport indices for Class II shipments, define the range of array sizes over which the number of containers must be examined for subcriticality when reflected by water. The smallest index that may be assigned to a container is 0.1 and corresponds to an upper limit of 2500 containers since the sum of the indices per shipment cannot exceed 50 and there is a factor of 5 for commingling of shipments. The largest

assignable index of 10 defines the smallest number of containers as 25. These limits extend beyond the range given in Table 1. The larger limit is resolvable through Fig. 1 but the lower limit is outside the valid range of Fig. 1 and should be treated separately by other methods. A demonstration of applicability of the calculated criticality data to transport is facilitated by considering a specific container, say, the 55-gal drum. If the capacity of the drum is utilized to define a_n , the need to consider various arrangements is obviated and the resultant arrays of drums made conservative by a few percent in k_{eff} .

A cube with a half dimension of 29.6 cm has a capacity equal to that of the 55-gal drum. Entering Fig. 1 at $a_n = 29.6$ cm, the corresponding values of NB_N^2 for various values of n are fixed. The radius of the appropriate spherical units can then be determined from the values of r_c/r_0 having the same value of NB_N^2 for each n . The results of these operations are presented in Fig. 2, where the number of units in a reflected cubic array is given as a function of the parameter r_c/n for the four materials. The near linear portrayal shown is typical of such representation. Some discrete values of the transport index are shown, for information, at the right ordinate of the figure. The results of Fig. 2 allow estimates of mass limits corresponding to transport indices. As a particular application, consider the 6-L specification container. This 55-gal drum has an inner container of schedule 40 pipe with an i.d. of 5.25 in., a wall thickness of 0.25 in., and a usable height of 17.95 in. The region between the inner and outer containers is filled with vermiculite at a packed density of 0.114 g/cm^3 . If one were to determine the mass limits allowed per container utilizing the KENO code and including the container materials, a comparison could be made with those limits inferred from the criticality data of Fig. 2. Let us suppose the mass limits required are for cubic arrays of the containers having a $k_{eff} = 0.97$ when reflected by an 8-in. thickness of water and that the masses are further reduced by an amount equivalent to $\sim 3\%$ in k_{eff} to provide a contingency for a triangular pattern of the drums. These mass limits are summarized in Table 2 where they are compared to limits determined from Fig. 2 for the same k_{eff} . It is evident that limits based on the analysis of calculated data for air-spaced units are slightly conservative.

A similar analysis was carried out for the 6-M specification container and the results are compared in Table 3. The 6-M container has a 2R inner container centered in a nominal 10-gal steel drum and utilizes celotex as the insulating material. Unlike the 6-L container, the mass limits for the 6-M container were not established by Monte Carlo calculations and, therefore, do not represent limits having a uniform margin of safety. Monte Carlo calculations were used simply to confirm subcriticality.

Fig. 2. Representation of Calculated Criticality of Water-Reflected Cubic Arrays Having Cubic Cells of Half-Dimension 29.6 cm (55-gal drum).

Table 2. Comparison of Metal Mass Limits Calculated for the 6-L Specification Containers with Those Derived from Calculated Critical Arrays of Air-Spaced Spherical Units

Trans- port Index	Pu(100), kg		²³³ U, kg		U(93.2), kg	
	6-L Con- tainer	Calculated Spaced Units	6-L Con- tainer	Calculated Spaced Units	6-L Con- tainer	Calculated Spaced Units
0.1	5.27	4.83	7.58	6.26	17.3	13.3
0.2	5.61	5.16	8.04	6.89	19.6	17.4
0.3	5.82	5.43	8.31	7.38	20.8	18.6
0.5	6.21	5.86	8.77	7.79	23.0	20.5
1.0	6.50	6.29	9.15	8.94	25.6	23.9
2.0	6.98	6.83	9.86	9.74	28.8	27.3

Table 3. Comparison of Metal Mass Limits Established for the 6-M Specification Container^a with Those Derived from Calculated Critical Arrays of Air-Spaced Spherical Units

Trans- port Index	Pu(100), kg		²³³ U, kg		U(93.2), kg	
	6-M Con- tainer	Calculated Spaced Units	6-M Con- tainer	Calculated Spaced Units	6-M Con- tainer	Calculated Spaced Units
0.1	3.1	2.72	--	3.19	7.2	6.36
0.2	3.4	3.19	--	3.72	8.7	7.66
0.5	4.2 ^b	3.87	--	4.65	11.2	11.66
1.0	4.5 ^b	4.45	--	5.43	13.5	17.43
2.0	--	6.04	--	7.71	--	20.50

^aFederal Register, Vol. 33, No. 194, Oct. 4, 1969.

^bMaximum allowed mass due to 10-watt decay heat limitation.

Although one may be tempted to conclude that the presence of the container and insulating materials allows larger mass limits, it would be an improper generalization. The presence of steel is probably the major cause for the increased mass. This is evidenced by the increasing mass differences as the number of containers in the array increases. A significant advantage of the steel in these containers is that the thickness is sufficient to override⁵ the increase in reactivity that otherwise would result at optimum hydrogenous moderation in the arrays.

Although limits were not developed for the wet oxides of $U(30)O_2$, since they are not permitted in these specification containers because of excess water content, the data were included to demonstrate that the presence of water in a unit does not complicate the analysis. The representations of this material in Figs. 1 and 2 confirm the suitability of the method to characterize the criticality of such materials.

REMARKS AND SUMMARY

At the very least, the utility of the Guide and the simple techniques portrayed above are beneficial in providing both a point of departure and guidance to more detailed criticality analyses of shipping containers. If, through continued effort, the Guide can be expanded to effectively serve as a basic reference for the evaluation of criticality, there will follow a uniformity or ordering in the margins of safety employed. This, in turn, will lead to a needed, evaluated scale for the determination of the degree of risk. Such an evaluated degree of risk is necessary to a more proper economic balance in transportation costs to industry.

It is appropriate to issue a word of caution regarding the Guide. Whereas the information presented represents a correlation among unit masses, cell dimensions, number of units, and the array neutron multiplication factor, it should be realized that the Guide is a supplement to, and does not substitute for, good nuclear safety practices. The mass limits given define recommended capacities for cell volumes in specified arrays and necessitate individual analyses for their implementation. For example, a 26 kg $U(93.2)$ metal limit in 2-ft cubic cells comprising a 64-unit array has a k_{eff} of 0.93 but there are definite possibilities of single-unit criticality which must be considered when handling such a mass.

The tabulated critical parameters given in this presentation are a portion of the data establishing the subcritical limits in the Guide itself. A total

of 40 different forms of fissile materials are presented. These can be augmented in the future and possibly at an accelerated pace were those individuals performing criticality analyses to characterize the materials under consideration in the format given here. It would facilitate the utility of such information were validated methods of analysis also used.

The references cited contain further material and discussion concerning the influence on array reactivity of changes in the various conditions of the reference arrays, such as shape, reflection, moderation, commingling of different units, neutron coupling between arrays, etc. The concepts and results are applicable to criticality analysis in transport.

REFERENCES

1. J. T. Thomas, Uranium Metal Criticality, Monte Carlo Calculations and Nuclear Criticality Safety, Y-CDC-7, UCC, Oak Ridge Y-12 Plant (1970).
2. G. E. Whitesides and N. F. Cross, KENO-A Multigroup Monte Carlo Criticality Program, CTC-5, UCC, Computing Technology Center (1959).
3. Gordon E. Hansen and William H. Roach, Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies, LAMS-2543, Los Alamos Scientific Laboratory (1951).
4. J. T. Thomas, The Criticality of Cubic Arrays of Fissile Material, Y-CDC-10 (to be published).
5. J. T. Thomas, Some Effects of Interspersed Moderation on Array Criticality, Y-CDC-5, UCC, Oak Ridge Y-12 Plant (1959).

CRITICALITY SAFETY EVALUATIONS OF SHIPPING
CONTAINERS USED BY IDAHO NUCLEAR CORPORATION AT NRTS

J. K. Fox
W. G. Morrison

Idaho Nuclear Corporation

ABSTRACT

This report gives the results of several criticality safety evaluations of fuel shipping containers. The containers included casks for shipping irradiated fuels, as well as containers for unirradiated fuels and other fissile materials. These containers all contain poisons such as cadmium and boron to assist in maintaining subcriticality and their presence must be assured. Most of the containers evaluated gave k_{eff} values that are quite low even for the most reactive credible condition. The k_{eff} values obtained for these conditions were between about 0.40 and 0.85 for the containers tested. In some cases, the k_{eff} values were below unity even when poison was omitted. The methods used involve the use of several reactor codes. The accuracy of these methods have been thoroughly checked by comparisons with clean critical experiments.

INTRODUCTION

Criticality safety evaluations are performed routinely by Idaho Nuclear Corporation on all new or special uranium shipping or transfer containers under its jurisdiction at the NRTS. This paper describes the calculational methods used and results obtained for some of these evaluations. Routine shipments include new and irradiated fuel elements for various reactors and both solid and liquid products from the Idaho Chemical Processing Plant. Also, numerous special shipments of fissile materials are made and received in connection with Idaho Nuclear Corporation activities.

Most of the fuel elements handled are alloys containing highly enriched uranium. Shipping boxes for the unirradiated test reactor fuel elements contain planar series of poison sheets and polyethylene between layers of fuel and have been evaluated on the basis of dry and flooded conditions. Casks for transfer or receipt of irradiated fuel elements contain lattices of poison plates to provide for multi-element shipments. These casks are normally loaded and unloaded under water and must be evaluated for the flooded condition. Since all of the containers evaluated contain neutron poisons, it is necessary to verify the presence of the poison for any new container before it can be used at Idaho Nuclear Corporation.

The ICPP product shipping containers are evaluated for both dry and flooded conditions. For flooded conditions, the problem is essentially an evaluation of a single reflected unit because the size of the outer container is such that there is no interaction. Dry arrays are evaluated on the basis of conservative assumptions, e.g., some water present in the free space inside the inner container and array sizes normally larger than will be encountered in shipments or storage.

A number of calculational techniques are used for the criticality evaluations. The KENO-II¹ Monte Carlo criticality program is the most versatile program in use at the NRTS. It has the advantage of treating three dimensions and complicated geometries of single units or arrays of mixed units. The well-known Hansen-Roach 16-group cross section sets² have been used exclusively up to the present with the KENO program. One- and two-dimensional neutron transport programs SCAMP³ and DOT⁴, respectively, are used as appropriate. Normally the 16-group Hansen and Roach cross section sets are used with these programs; however, other group structures using ENDF-B cross sections can be used. The PHROG⁵ and INCITE⁶ programs are used to generate these cross section sets. Both programs use the B-1 approximation to the transport equation for spectrum calculation. PHROG is used for the fast constants and INCITE for the thermal range.

Diffusion theory programs are used in some evaluations. These include the versatile two-dimensional PDQ-7⁷ and the one-dimensional FOG⁸ program. The few-group nuclear constants used with these programs are also generated by the PHROG and INCITE programs. Diffusion theory has been found adequate for well moderated problems where there are no voids. Extensive calculations have been performed with all of these programs in comparing them with clean

critical experiments covering a wide range of conditions. The excellent agreement in results of the comparison provides great confidence in our ability to determine the criticality safety of shipping and transfer containers used by Idaho Nuclear Corporation.

UNIRRADIATED FUEL SHIPPING CONTAINERS

ETR Shipping Boxes

These boxes are intended for shipping unirradiated (new) fuel and are made mainly of wood that has been treated to retard fire. Later, a second box was fitted over the first one to gain additional fire resistance. Thus, the present boxes are double walled. A sketch of one of these boxes is shown in Figure 1. The box holds six elements in two layers. The layers are separated vertically by a sandwich construction containing two layers of cadmium separated by a one-inch polyethylene plate. There is also a poison sandwich at the top and bottom. Infinite arrays of these boxes were calculated using PDQ with the derivative B.C. applied at each outer edge. The XY layout treated each material explicitly as a separate region. Since a density analog treatment shows that dry box arrays are very subcritical, it was assumed that all voids were filled with water. Inside each element the plates and water gaps were homogenized. A sketch of the PDQ layout is shown in Figure 2 (only one quadrant is required because of symmetry). A k_{eff} of 0.64 was obtained with a water density of 1.0 in all voids. If the fuel has full density water and all other voids contain water at a density of 0.3, the k_{eff} is 0.69. Thus, the ETR box arrays are safe under any foreseeable condition.

ATR Shipping Boxes

The ATR new fuel shipping boxes are constructed in a manner similar to that used for ETR fuel. The main difference being that each box contains four fuel elements in one layer. The box contains two cadmium and polyethylene sandwiches so that there are two of these between layers of fuel when boxes are stacked vertically. The PDQ four group analysis was very similar to that used for ETR fuel. Assuming an infinite array to be water flooded, the k_{eff} was calculated based on an element loading of 1500 grams U-235. The actual loading averages about 1075 grams per element. The calculated k_{eff} was 0.58. Reducing the water density in all voids except those between fuel plates again raised the k_{eff} several percent. However, it is not felt that this is a credible condition.

ICPP Oxide Product Shipping Container Evaluation

The solid UO_3 product from ICPP is shipped in Y-12 model FD Foamglass shipping containers (DOT #5463). A sketch of this shipping container is shown in Figure 3. This container has been approved for fissile Class I and Class II with authorized contents of 3.7 and 19.0 kg U-235, respectively, as uranium metal or dry uranium compounds enriched to at least 75 percent in U-235. The maximum physical loading for ICPP product is about 13.6 kg UO_3 or 11.3 kg uranium. Normally the UO_3 product is stored in the vault in the shipping containers. Consequently, the criticality safety evaluations of the packaged UO_3 in the shipping containers were made for both cubic arrays and conditions of vault storage. The KENO II program was used for this evaluation. Vault arrangements, as well as cubic arrays of shipping containers, were calculated. A thick concrete reflector surrounded all of these arrays. Results are as follows:

Vault Array	2 x 20 x 3	$k_{\text{eff}} = 0.393 \pm .005$
Cubic Array	5 x 5 x 5	$k_{\text{eff}} = 0.400 \pm .007$
Cubic Array	8 x 8 x 7	$k_{\text{eff}} = 0.488 \pm .006$

Storage of cans of UO_3 in the shipping containers is quite conservative. However, the capacity of the vault is still adequate. The calculated k_{eff} for the 8 x 8 x 7 cubic array reflects a lower loading than that allowed in the DOT permit. The k_{eff} obtained by Crum⁹, et al, was $0.90 \pm .01$ for the same array with the maximum loading allowed.

CASKS FOR SHIPPING IRRADIATED FUEL

ATR Poisoned Transfer Casks

The ATR transfer cask is used to transfer spent ATR fuel from the reactor canal to the ICPP for reprocessing. A sketch of the cask is shown in Figure 4. The ATR fuel is shaped as a 45° segment of an annulus, and each element fits between Boral radial fins so that eight elements complete an annulus. In the center is a box, sized so that only one element can be inserted. The radial fins contain canned Boral plates 0.125 inches thick. To mockup the cask for a PDQ-7 calculation using XY geometry, the inclined 45° fins had to be converted into small horizontal and vertical steps, as shown in Figure 5. For the Z direction, an input buckling corresponding to the 48-inch fuel length plus reflector savings was used. To assure conservative results, it was assumed that each fuel element contained 1500 grams U-235 and that no boron was present. The actual new fuel loading averages about 1075 grams U-235.

Since the cask is loaded and unloaded under water, the flooded condition was used in the calculations. An infinite reflector of water was assumed in place of the actual lead water combination. The effect of Boral was determined by doing two calculations in which aluminum was substituted for Boral. The results are as follows:

1. Nine elements with Boral present	$k_{eff} = 0.75$
2. Nine elements with Al replacing boron	$k_{eff} = 1.16$
3. Eight elements (Center element out) with Boral present	$k_{eff} = 0.62$
4. Eight elements (Center element out) with Al replacing boron	$k_{eff} = 1.014$

Thus, it is seen that the presence of the Boral is quite important. For this reason, it is standard practice to use a source-detector method to verify the presence of poison in new casks.

ETR Poisoned Transfer Casks

The ETR transfer cask is used to transfer spent ETR fuel from the reactor canal to ICPP. A sketch of the cask is shown in Figure 6. The three by three array of element boxes has a sheet of cadmium around the central box only. This position is blanked off so that only eight elements can be inserted. The array was mocked up for a PDQ calculation in XY geometry. Fuel element, side plates, stainless steel partitions and cadmium were made into explicit regions. A water flooded condition was assumed. The fuel plates were homogenized with the water gaps. Since the elements are smaller than the lattice boxes, they were centered and surrounded by a water gap. To account for leakage in the Z direction, an input buckling corresponding to the fuel length plus reflector savings was used. Four group constants were obtained based on each element containing 788 grams U-235, but no boron. Blackness theory was used for the cadmium membrane. An infinite water reflector was substituted for the water and lead mixture. The calculated k_{eff} was 0.747.

National Lead Casks

Over the past several years, two similar casks designed for MTR-type irradiated fuel have been evaluated. These were designed by the National Lead Company. Each utilized Boral plates between rows of fuel in a rectangular lattice. The plates are of a sandwich construction with the Boral canned in aluminum. The fuel elements are essentially in contact in the rows. The first cask was designed to contain 28 elements in a circular lead cask. This

cask is assigned DOT SP 5786. A PDQ calculation was run with the cask loaded with BM1 type fuel (3.1 x 3.2 inches, 24 inches long). With each element loaded to 220 grams U-235, the calculated k_{eff} is 0.73. If the loading is increased to 550 grams, the k_{eff} is 0.935.

The JMTR cask for Japanese fuel is a more recent version, designed to contain 41 3-inch by 3-inch elements. It is very similar to the above cask, except for size. For this evaluation, the mixed box version of KENO was used. Because of the circular outer container, the rows are of varying length and columns of elements are staggered. This complicates the box layout. Another complication is the gaps in the Boral plates at their center. A simplified sketch of the lattice is shown in Figure 7. By making each element into two or more boxes and using different box widths for the Boral gaps, it is possible to mockup the array quite accurately. Problems of symmetry were such that about 19 different box types are required. Each fuel element has 350 grams U-235 in a 29.5 inch active length. With the cask assumed to be water flooded and reflected, the k_{eff} obtained is $0.71 \pm .008$. If a lead reflector is used, the k_{eff} obtained is $0.68 \pm .008$. In an independent mockup a k_{eff} of $0.74 \pm .01$ was obtained for the water flooded case.

Evaluation of Savannah River 70-Ton Cask for HTRE and Other Fuels

The NRTS inventory of miscellaneous irradiated nichrome and stainless steel fuels was transferred to the Savannah River plant in 1966-1968. One of the 70-ton Savannah River casks described in report DP-357¹⁰ was modified slightly and used for these shipments. B of E permit No. 2189 was assigned to the cask for these transfers. Twelve shipments were made with the total U-235 loading per shipment varying from about 54 kg to 141 kg. The fuels were essentially all 93 percent enriched UO₂ nichrome cermets clad with nichrome except for one shipment which consisted of 3 and 4 percent enriched UO₂ fuel rods clad with stainless steel. Rods, plate, and annular ring-type fuels were all represented. The cask, shown in Figure 8, contained five positions for fuel. These positions were separated by four poison plates made of cadmium clad with stainless steel. Clips for holding stacks of horizontal fuel elements and spacers were fabricated to fit into the positions, so that almost exact fuel arrangement dimensions could be maintained. Depending on the size and U-235 loading of the various fuel elements, slabs of fuel elements could be placed in either positions 1, 3 and 5 or positions 2 and 4. Slabs of fuel in positions 1 and 5 were more reactive than the same slabs of fuel in positions 2 and 4, because of the poison separator plates. None of the shipments utilized all of the positions at one time. The one-dimensional neutron

diffusion program FOG-3 was used for determining which slots to fill and evaluation of all of the loadings. The calculated k_{eff} s were in the 0.8 to 0.96 range. However, very conservative assumptions were used, e.g., no burn-up, uniform loading in the slabs equal to the highest loaded fuel element, and slab thickness equal to the maximum diameter of any of the fuel elements. Check calculations on critical arrays of cylindrical fuel elements showed that slab thicknesses based on an equivalent cross sectional areas agreed with the experimental results. Also, nuclear constants were for homogeneous fuel cells with no disadvantage factors considered.

Paducah Depleted Uranium Casks

The Paducah depleted uranium cask is proposed for the shipment of Spert IIIC fuel to SRP. This fuel is in the form of 4.8 percent enriched oxide pins 0.466 inch OD. The cask is modified by the insertion of a cadmium poisoned box lattice with the pins stacked in contact in the boxes. A sketch of the lattice is shown in Figure 9. The first proposal was to fill only the five rectangular boxes. This was mocked up for a PDQ calculation in XY geometry. With the fuel assumed to be at near optimum spacing (0.5 inch E to E) and the cask flooded, the k_{eff} obtained was 0.63. The k_{eff} was only 0.32 for a close-packed assembly. A problem was also run in RZ geometry, where the lattice was replaced by a cadmium liner against the inside walls and the fuel filled the cask. With this arrangement, there is more fuel and the cadmium is much less effective. The homogenized fuel corresponded to optimum spacing. This case gave a k_{eff} of 0.86.

KENO II was used when it was later decided to completely fill the cask with loose Spert IIIC pins. One problem used the generalized geometry version of the code with the fuel assumed to be close-packed and homogenized into each lattice box. The k_{eff} obtained was $0.68 \pm .01$. In a second problem, the mixed box version was used. The fuel pins were put in explicitly in a square lattice with a center-to-center spacing of 0.5 inches. This mockup required five box types and used a water reflector. The calculated k_{eff} was $0.846 \pm .011$. For this calculation, the Knight corrected U-238 cross-sections were used. An additional calculation shows water is a better reflector than depleted uranium.

Evaluation of the Pegase Poisoned Fuel Cask

The Pegase irradiated fuel shipping cask was designed for shipping 25 ETR-type fuel elements, each containing 380 grams U-235 before burnup. The cask contains a stainless steel axle and tube with five vertical equidistant grooves machined for receiving five radial boron stainless steel plates. Thus, five compartments are formed by the central tube, the inner shell and the five radial plates. Machined aluminum blocks, each with five lodgements for fuel elements, fit into the five compartments. A sketch showing the arrangement is shown in Figure 10. The criticality safety evaluation of this cask considered complete water flooded conditions, both with and without the 0.5 w/o natural boron in the stainless steel radial plates. These calculations were performed on the IBM-7044 with the PDQ-40 diffusion code in XY geometry. The calculated k_{eff} was 0.63 with the boron present and 0.67 for no boron present in the stainless steel plates. An additional calculation was performed in which the uranium loading in the fuel elements was doubled. This yielded a k_{eff} of 0.80 with boron present.

CONCLUSIONS

The results of the calculations show that each of the containers evaluated is safe under the most reactive credible condition. These containers all have a rather large factor of safety to allow for uncertainties in the evaluations. However, the methods have been thoroughly verified by comparisons with experiment, both by the present authors and many others. In other words, the procedures and methods used at the NRTS by Idaho Nuclear Corporation have been found to be adequate for criticality safety evaluations of various shipping containers. The effects of damaged containers was not discussed. However, it is felt that credible wreck damage would not compress the fuel arrays appreciably. Thus, the reactivity would remain essentially constant.

REFERENCES

1. G. E. Whitesides and N. F. Cross, KENO - A Multigroup Monte Carlo Criticality Program, USAEC Report CTC-5, Union Carbide Corp., Nuclear Division (1969).
2. G. E. Hansen and W. H. Roach, Six and Sixteen Group Sections for Fast and Intermediate Critical Assemblies, USAEC Report LAMS-2543, Los Alamos Scientific Laboratory (1960).
3. C. L. Beck, Letter: Input Description of SCAMP Codes, Private Communication, Idaho Nuclear Corporation.
4. F. R. Mynatt, DOT - Two Dimensional Discrete Ordinates Transport Code, USAEC Report CCC-89, Union Carbide Corp., Nuclear Division.
5. R. L. Curtis, et al, PHROG - A Fortran IV Program to Generate Fast Neutron Spectra and Average Multi-Group Constants, USAEC Report IN 1435, Idaho Nuclear Corporation (1971).
6. R. L. Curtis and R. A. Grimesey, INCITE - A Fortran IV Program to Generate Thermal Neutron Spectra and Multi-Group Constants Using Arbitrary Scattering Kernels, USAEC Report IN 1062, Idaho Nuclear Corporation (1967).
7. W. R. Cadwell, PDQ-7 Reference Manual, USAEC Report WAPD-TM-678, Bettis Atomic Power Laboratory (1967).
8. H. P. Flatt, The FOG One-Dimensional Neutron Diffusion Equation Codes, USAEC Report NAA-SR-6104, Atomics International, Division of North American Aviation Corporation (1961).
9. E. C. Crume, et al, Evaluation of a Shipping Container for Enriched Uranium Metal and Dry Compounds, USAEC Report Y-KC-106, Union Carbide Corporation, Nuclear Division (1967).
10. W. H. Piper and J. W. Langhaar, 70 Ton Shipping Cask for the Savannah River Plant - Design and Fabrication, USAEC Report DP-357, E. I. du Pont de Nemours & Co., Atomic Energy Division (1959).

INC-D-12468

Figure 1. ETR Shipping Box.

INC-A-18848

Figure 2. ETR Shipping Box PDQ-7 Quadrant Layout.

INC-A-18858

Figure 3. Y-12 Model FD Shipping Container.

INC-C-18857

Figure 6. ETR Transfer Cask.

INC-C-18860

Figure 7. JMTR Fuel Cask.

INC-B-18859

Figure 8. Savannah River 70-Ton Cask.

INC-D-18856

Figure 9. Paducah Depleted Uranium Cask.

INC-C-18855

Figure 10. Pegase Shipping Cask.

ATMX RAILCAR - NUCLEAR SAFETY EVALUATION

J. D. McCarthy

ABSTRACT

A Nuclear Safety Analysis of the ATMX-600 Series Railcar shows that this railcar meets the requirements for Class I Shipments, as specified in the Appendix to AEC Chapter 0529. This railcar is used for shipping waste material mixed with fissile material.

INTRODUCTION

The ATMX-600 Series Railcar, a double-walled steel car, was originally built to transport explosive ordinance. For a complete description of this railcar, refer to RFP-1411¹.

It was necessary to provide a Nuclear Safety evaluation prior to the use of this car for the transport of radioactive waste material. This evaluation gives the limiting criteria for Class I shipments as specified in the Appendix to AEC Chapter 0529.

The criteria specify the size of waste containers as a function of the amount of fissile material in combination with non-fissile material permitted in the container.

PACKAGING CONFIGURATION

The data pertinent to a Nuclear Safety evaluation of this railcar are:

ATMX-600 Series Railcar

The outer structure is an ATMX-600 Series Railcar (Figure 1).
The useable inner dimensions are:

length - 50 feet 0 inches
width - 9 feet 4 inches
height - 9 feet 2 inches

Waste Containers

The containers (Figure 2) are:

1. Specification ICC-17H 30-gallon containers² having the following dimensions:

minimum inner diameter - 18.187 inches
minimum inner height - 27.219 inches
minimum thickness - 0.0428 inches

and/or

2. Specification ICC-17H 55-gallon containers³ having the following dimensions:

minimum inner diameter - 22.437 inches
minimum inner height - 32.719 inches
minimum thickness - 0.0428 inches

Contents of the Waste Containers

The fissile material used in this evaluation is specified below. These numbers were chosen because they are a practical limit and an amount which can be shown to be acceptable from a criticality aspect.

1. A maximum of 100 g Pu-239 mixed with non-fissile material (such as graphite, paper, plastic, etc.) in the ICC-17H 30-gallon container.

and/or

2. A maximum of 200 g Pu-239 mixed with non-fissile material (such as graphite, paper, plastic, etc.) in the ICC-17H 55-gallon container.

CLASS I ANALYSIS

Normal Conditions of Transport

The requirements for Class I Shipments are specified in the Appendix to AEC Chapter 0529. These regulations specify the normal and accident conditions that must be applied to ICC-17H 30-gallon containers and the ICC-17H 55-gallon container with the restriction that these containers are placed in the ATMX-600 Series Railcar.

Under normal conditions, the nuclear safety requirements for Class I shipments is that any number of containers is sub-critical in any arrangement with optimum moderation. Interpreting the phrase "any arrangement" to mean "any arrangement under normal handling conditions", this requirement is met. The following evaluation shows that an infinite number of these ICC-17H 30-gallon containers and/or ICC-17H 55-gallon containers are sub-critical in an infinite planar array of ATMX-600 Series Railcars.

Assumptions used in the Analysis

For this analysis, the containers were assumed to be undamaged and in a fully reflected array with optimum moderation.

In the calculations, the material in the containers was considered to be uniformly distributed through the containers.

Analysis

Previous calculations⁴ have shown that the most reactive configuration for this type of waste is a composition of 200 g Pu-239 and graphite per container. The addition of water to a graphite-Pu system results in a less reactive system. The amount of graphite in a drum must be limited to a maximum of 200 pounds of graphite since calculations show that 300 pounds of graphite will result in a more reactive system. The 200 pound graphite limit is also compatible with that used at Rocky Flats.

Using Pu and 200 pounds of graphite as the composition of the material in the 30-gallon and 55-gallon containers, a series of

calculations* were run. These calculations determined the critical height of an infinite number of cylinder cells. The DTF calculations⁵ were done on the CDC-3800 with sixteen group cross sections⁶ and S_4 approximation.

The configuration for the cell calculations was an infinite planar array of cylinders (Figure 3). The composition of the material in the containers and the dimensions for the arrangement of the containers are:

ICC-17H 30-Gallon Containers

Material: 100 g Pu-239 and 200 pounds of graphite per container

Inner Diameter: 18.18 inches

Gap Between Containers: 0.629 inches

Container Wall Thickness: 0.0378 inches

ICC-17H 55-Gallon Containers

Material: 200 g Pu-239 and 200 pounds of graphite per container

Inner Diameter: 22.43 inches

Gap Between Containers: 0.629 inches

Container Wall Thickness: 0.0428 inches

The results of the DTF-II calculations (after adjustment for reflection on the top and bottom of the array) indicate that the critical height of an infinite reflected slab of 30-gallon containers in the configuration specified is 121.2 inches and the critical height of an infinite reflected slab of 55-gallon containers in a similar configuration specified is 166.8 inches.

Since the useable height of the ATMX-600 series railcar is 110 inches, an infinite number of these 30-gallon and/or 55-gallon containers, packaged as specified above, is sub-critical in an infinite number of fully reflected railcars.

Accident Conditions for Transport

Another of the requirements for Class I shipments, as specified in the Appendix to AEC Chapter 0529, is that two hundred

---

* Calculations done by D. C. Coonfield, L. E. Jackson, Rocky Flats Division, The Dow Chemical Company, 1968.

fifty (250) of these ICC-17H 30-gallon and/or 55-gallon containers must be sub-critical in any arrangement with full reflection on all sides and optimum moderation. The condition of each container is to be that which would result if each container was subjected to the hypothetical accident conditions specified in Annex 2 of the Appendix to AEC Chapter 0529.

Assumptions used in the Analysis

The containers were assumed to have the same spacing in the array as the undamaged containers and the material in the containers was assumed to be uniformly distributed through the containers.

The containers were assumed to be in an array with full reflection and optimum moderation.

Analysis

The 30-gallon and 55-gallon containers were assumed to have the same spacing in the array as the undamaged containers since the package-to-package spacing will not change to any large extent under accident conditions⁷ as defined in the Appendix to AEC Chapter 0529.

The contents of the 30-gallon containers were assumed to be 100 g of Pu-239 and 200 pounds of graphite per container and the contents of the 55-gallon containers were assumed to be 200 g Pu-239 and 200 pounds of graphite per container. These containers have optimum moderation, as explained above, provided the graphite was limited to a maximum of 200 pounds.

The reflected critical infinite slab thickness, calculated for 30-gallon containers, was converted to a reflected cubic array by equating the Buckling equations.

Infinite Slab:

$$B_s^2 = \left(\frac{-}{T + 2\delta} \right)^2$$

where: T = infinite critical slab thickness
δ = extrapolation length

Cubic:

$$B_c^c = 3 \left(\frac{-}{h + 2\delta} \right)^2$$

where: h = length of edge

δ = extrapolation length

The extrapolation length (δ) used for the slab was 2.47 inches⁸ and for the cubic array was 2.23 inches⁸.

For the 30-gallon containers, the calculated value of T was 121.2 inches. This gives a value of h equal to 213.76 inches.

The number of 30-gallon containers on an edge of the base of the cubic array is obtained by dividing the length of the base by 18.88 inches (the inner diameter of the container plus two wall thicknesses plus the air gap) and the number of containers high is obtained by dividing the height of the array by 27.22 inches (the inner height of the container).

The critical number of 30-gallon containers in a cubic array is:

$$\begin{aligned} N_c &= (\text{number on edge})^c (\text{number high}) \\ &= (11.32)^c (7.85) \\ &= 1005 \end{aligned}$$

This shows that two hundred fifty (250) ICC-17H 30-gallon containers are sub-critical in a cubic array with full reflection and optimum moderation.

This same type of analysis, when applied to the 55-gallon containers, gives a critical number of one thousand four hundred thirty one (1431) ICC-17H 55-gallon containers, loaded as specified above.

This shows that two hundred fifty (250) ICC-17H 55-gallon containers are sub-critical in a cubic array with full reflection and optimum moderation.

Crates containing contaminated equipment, etc., will be limited to 5 g of Pu-239 per cub foot. This system is a less reactive system than has been shown for the graphite-Pu system of 30-gallon containers and 55-gallon containers.

Therefore, these 30-gallon containers, 55-gallon containers, and crates meet the requirements as specified in Section II.H for

Class I shipments, provided these drums are packaged in the ATMX-600 series railcar.

SUMMARY

This evaluation has shown that this method of transporting fissile waste material is acceptable from a criticality aspect.

It has been shown that for normal conditions of transport, an infinite number of 30-gallon containers, 55-gallon containers, and crates are sub-critical in an infinite planar array of ATMX-600 railcars with full reflection and optimum moderation.

The analysis for accident conditions of transport has shown that at least two hundred and fifty (250) of these 30-gallon containers and 55-gallon containers are sub-critical in a cubic array with full reflection.

This evaluation has been shown to be based on the waste being distributed among:

ICC-17H 30-gallon containers containing a maximum of 100 g Pu-239 and non-fissile material (with the further limitation that the non-fissile material does not contain more than 200 pounds of graphite per container).

and/or

ICC-17H 55-gallon containers containing a maximum of 200 g Pu-239 and non-fissile material (with the further limitation that the non-fissile material does not contain more than 200 pounds of graphite per container).

and/or

Crates containing contaminated equipment, etc., with a limitation of a maximum of 5 g Pu-239 per cubic foot.

The ATMX-600 series railcar as a means to transport radioactive and/or fissile material has been issued DOT SP 5948 which incorporates the limitations specified above.

REFERENCES

1. F. E. Adcock, ATMX-600 Series Railcar for Radioactive Waste Shipments, RFP-1411, The Dow Chemical Company, Rocky Flats Division, Golden, Colorado, January 7, 1970.
2. Steel Shipping Container Institute, American Standard Specifications for Metal Drums, MH 2.13-1960, page 6, American Institute, New York, 1960.
3. Steel Shipping Container Institute, Specifications for Metal Drums and Pails, MH 2.5-1965, page 10, American Standards Institute, New York, 1965.
4. Internal Report, The Dow Chemical Company, Rocky Flats Division, Golden, Colorado, 1968.
5. B. G. Carlson, W. Guber, M. Shapiro, and W. J. Worlton, DTF Users Manual, UNC Phys/Math-3321, United Nuclear Corporation, White Plains, New York, Volume I, Nov. 1963, Volume 2, May, 1964.
6. G. E. Hansen and W. H. Roach, Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies, LAMS-2543, University of California, Los Alamos Scientific Laboratory, Los Alamos, New Mexico, December, 1960.
7. F. E. Adcock, Internal Report, The Dow Chemical Company, Rocky Flats Division, Golden, Colorado, December 4, 1967.
8. H. C. Paxton, Criticality Control in Operations with Fissile Material, Los Alamos Scientific Laboratory of the University of California, Los Alamos, New Mexico, 1966.

Figure 1 ATMX-600 Series Railcar

Figure 2 Waste Containers

Figure 3 Arrangement Used in Calculations

STUDY ON FIRE RESISTIVITY FOR IRRADIATED
FUEL SHIPPING CASKS

T. Moriya
H. Shimada

ABSTRACT

The purpose of this study is to assess the fire resistivity of spent fuel casks. Heat durability experiments were made in a furnace for two cask models of different dimensions. It was found that when the experimental results were analysed with an Analog computer, the air gap between the lead shield and the outer shell affects largely the heat resistivity of casks. The analysis in which the heat generation from spent fuel is taken into consideration suggests that the prototype casks are fire-resistive enough for their lead to be kept unmelted against the 800°C fire of a period of 30 min.

INTRODUCTION

Casks used to transport radio active spent fuel require high fire resistivity to cope with possible fire accidents with which they would meet. The assessment of the heat resistivity of casks of different dimensions were performed by experiments and theoretical analysis.

DISCRIPTION OF CASKS AND FURNACE

Casks refered to in this paper are spent fuel shipping containers which are cylindrical and composed fo three layers, an outer shell of steel, a lead shield and an inner shell of stainless steel. Details of two different scale cask models used in our tests as well as those of prototype casks are presented in Table 1. A tunnel shaped furnace the inside dimension of which is 1.6 m x 1.6 m x 1.6 m was used for the tests. In addition to the two front and rear sliding doors, the furnace has two bottoms separable from it, each of which has four wheels under it and can be moved on two paralled rails. One of the two bottoms, which are called trucks hereafter, is used for preheating period and the other for experiment period. When a test is started, the truck on which a specimen is mounted replaces the other one which

has served as the bottoms of the furnace during the preheating period.

Table 11 Dimensions of Casks

		Prototype Cask No. 0	Cask Model No. 1 No. 2	
Outer Diameter	D (mm)	1600	800	400
Inner Diameter	d (mm)	1000	500	74
Outer Shell	t_1 (mm)	38	19	10
Lead	t_2 (mm)	250	125	150
Inner Shell	t_3 (mm)	12	6	3
Length	L (mm)	5200	800	600
Weight	W (kg)	80000	3000	636

FIRE TESTS

The top and bottom of a cask model on the truck outside are covered with asbestos heat insulator and firebrick for the purpose of heating the model only from its side. Chromel-alumel thermocouples are set at various points in the furnace, the outer shell, the lead layer and the cavity center for measuring their temperatures. The locations of the thermocouples for No. 1 and No. 2 models are shown in Fig. 1. When the inside temperature of the furnace is raised to approximately 900°C, a cask model on the truck is conveyed into the furnace by an electric motor after the other truck is pushed out. The cask model is pulled out after a heating period of 30 min. at 800°C and is cooled to the ambient temperature. During the experiment, temperatures of various points in the furnace and the model are recorded continuously.

RESULTS

The experimental results for No. 1 and No. 2 cask models are shown in Fig. 2 and 3 respectively. The temperature distribution in No. 1 cask

model after a heating period of 30 min. and that in No. 2 cask model after a heating period of 20 min. are shown in Fig. 4 and 5 respectively. Those figures clearly show that in every experiment the outer shell temperature is much lower than the inner lead temperature. It is considered that the temperature discrepancy between the outer shell and the inner lead resulted from the very thin air gap between them. As shown in Fig. 2 and 3, though the outer shell temperature began to drop immediately after the cask model was pulled out of the furnace, the inner shell temperature still continued to rise to approximately 260°C from 140-200°C during the cooling period after a heating period of 30 min. Therefore, whether the lead would be melt or not cannot be discussed without paying attention to the lead temperature during the cooling period.

ANALYSIS

The heat resistivity of casks must be evaluated by the radiation shielding ability and tightness after the tests. It is generally considered if the lead was melted. So, the temperature of lead can be used as a yard stick of the heat resistivity of casks. Only for one dimensional heating from radial direction (see Fig. 6), heat conservation differential equations given below are solved with the Analog computer.

$$\frac{\lambda_1 S_1}{\Delta r_1} (\theta_1 - \theta_2) - \frac{\lambda_1 S_3}{\Delta r_1} (\theta_2 - \theta_3) = M_{13} \frac{d\theta_2}{dt} \quad (1)$$

$$\frac{\lambda_2 S_4}{\Delta r_2} (\theta_4 - \theta_5) - \frac{\lambda_2 S_6}{\Delta r_2} (\theta_5 - \theta_6) = M_{46} \frac{d\theta_5}{dt} \quad (2)$$

$$\frac{\lambda_2 S_6}{\Delta r_2} (\theta_6 - \theta_7) - \frac{\lambda_2 S_8}{\Delta r_2} (\theta_7 - \theta_8) = M_{68} \frac{d\theta_7}{dt} \quad (3)$$

$$\frac{\lambda_2 S_8}{\Delta r_2} (\theta_8 - \theta_9) - \frac{\lambda_2 S_{10}}{\Delta r_2} (\theta_9 - \theta_{10}) = M_{810} \frac{d\theta_9}{dt} \quad (4)$$

$$\frac{\lambda_3 S_{10}}{\Delta r_3} (\theta_{10} - \theta_{11}) - \frac{\lambda_3 S_{12}}{\Delta r_3} (\theta_{11} - \theta_{12}) = M_{1012} \frac{d\theta_{11}}{dt} \quad (5)$$

$$\frac{\lambda_4 S_{12}}{\Delta r_4} (\theta_{12} - \theta_{13}) = M_{1214} \frac{d\theta_{13}}{dt} \quad (6)$$

$$\alpha_1 (\theta_0 - \theta_1) = \frac{\lambda_1}{\Delta r_1} (\theta_1 - \theta_2) \quad (7)$$

$$\frac{\lambda_1}{\Delta r_1} (\theta_2 - \theta_3) = \alpha_2 (\theta_3 - \theta_4) \quad (8)$$

$$\alpha_2 (\theta_3 - \theta_4) = \frac{\lambda_2}{\Delta r_2} (\theta_4 - \theta_5) \quad (9)$$

$$(\theta_5 - \theta_6) = (\theta_6 - \theta_7) \quad (10)$$

$$(\theta_7 - \theta_8) = (\theta_8 - \theta_9) \quad (11)$$

$$\frac{\lambda_2}{\Delta r_2} (\theta_9 - \theta_{10}) = \frac{\lambda_3}{\Delta r_3} (\theta_{10} - \theta_{11}) \quad (12)$$

$$\frac{\lambda_3}{\Delta r_3} (\theta_{11} - \theta_{12}) = \frac{\lambda_4}{\Delta r_4} (\theta_{12} - \theta_{13}) \quad (13)$$

where

λ : thermal conductivity of materials

M : heat capacity of each element.

S : surface area

α_1 : total heat transfer coefficient at surface S_1

α_2 : thermal conductance of gap

θ_0 : fire temperature

$\theta_1 \sim \theta_{14}$: temperature in cask

The thermal conductance α_2 of the air gap between the outer shell and the lead shield was taken into consideration in the analysis. The total heat transfer coefficient at the surface of the outer shell α_1 is the sum of the radiation heat transfer coefficient α_r and the convection heat transfer coefficient α_c which is constant.

The total heat transfer coefficient α_1 is given by

$$\alpha_1 = \sigma \epsilon F \frac{(T_0^4 - T_1^4)}{T_0 - T_1} + \alpha_c \quad (\text{kcal/m}^2 \text{h}^\circ\text{C}) \quad (14)$$

where

- T_0 : fire temperature °K
 T_1 : temperature of surface of cask °K
 σ : Stefan-Boltzmann constant
 F : geometric shape-emissivity factor

The outer shell temperature θ_2 and the lead shield temperature, θ_5 , θ_7 and θ_9 of each of No. 1 and No. 2 cask models were computed. The results of the computation is as follows.

The experimental value agreed considerably well with the computed value when the thermal conductance of the air gap for No. 1 and No. 2 models was given as 40 and 55 kcal/m² h°C respectively. The thermal conductance of the air gap was assumed to be constant regardless of the temperature.

The computed temperature-time curves of No. 1 and No. 2 cask models are shown in Fig. 2 and 3 in broken lines respectively. The size of the gap estimated from the value of d_2 based on the thermal conductance of air at 350°C is 1.03mm for No. 1 cask model and 0.75 for No. 2 cask model. The gap which is formed by contraction of lead when each cask model is moulded was estimated by calculation to be 0.94mm for No. 1 cask model 0.79mm for No. 2 cask model. The gap calculated based on the value of d_2 was in a near agreement with that from the calculation of contraction. Therefore, it may be possible to know the extent of gap by calculation when materials and dimensions are known.

d_2 of prototype casks can be obtained as 22 kcal/m² h°C by the estimation from the gap size calculated from the contraction.

Spent fuel contained in casks generates some heat as it decays. The rate of heat generation from the fuel is negligibly small compared with that of the heat transfer to casks from fires. In the computer analysis, as the initial cask temperature depending on the decay of the fuel, 120°C was used each for No. 1 cask and prototype cask as initial value, though the term of the heat generation rate from the fuel was not added to the

equation. The cask temperature, 120°C was estimated from the results of experiments conducted concerning heat extraction from casks. The temperature time curves of No. 1 caks from the computer analysis is shown in Fig. 7. In these curves, melting of lead is not taken into consideration. According to Fig. 7, the temperature of lead rises to 315°C in 30 min. and exceeds its melting point during the cooling period. It means that the lead begins to melt during the cooling period after the heating period of 30 min..

As for the prototype cask, temperature-time relations at various points of the cask against the standard fire (800°C , 30 min.) were obtained from the theoretical computation using some values of a_2 not less than $22 \text{ kcal/m}^2 \text{ h}^{\circ}\text{C}$ mentioned above.

The temperature-time curves of the outer shell and of the lead surface obtained from the computation are shown in Fig. 8. According to Fig. 8, the temperature of outer shell rises to approximately $570\text{--}610^{\circ}\text{C}$, but the maximum temperature of the lead, which varies depending on the heat conductance of the air gap, does not go up to over 260°C which is lower than the melting point of lead, 327°C .

It can be considered that this kind of prototype cask is safe enough from the standard fire.

Fig. 1 Cross-Sections and Thermocouple Locations of No. 1 and No. 2 Cask Models

Fig. 2 Temperature vs. Time Curves of No. 1 Cask Model

Fig. 3 Temperature vs. Time Curves of No. 2 Cask Model

Fig. 4 Temperature Distribution Curve of No. 1 Cask Model at 30 min after Start of Fire Test.

Fig. 5 Temperature Distribution Curve of No. 2 Cask Model at 20 min after Start of Fire Test.

Fig. 6 Illustration of a Cask Element Divided in to Several Parts for Analysis.

Fig. 7 Temperature vs. Time Cruves of No. 1 Cask Model for Standard Fire.

Fig. 8 Temperature vs. Time Cruves of the Outer Shell and the Lead Surface of Prototype Cask for Standard Fire.

HEAT TRANSFER CALCULATIONS FOR SPENT FUEL DURING
HYPOTHETICAL ACCIDENT CONDITIONS

C. J. Anderson

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

A THERMAL TEST EVALUATION METHOD FOR LEAD & URANIUM SHIELDED CASKS

J. C. Glynn
R. H. Odegarden

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

X SHIPPING CONTAINER DESIGN FOR FIRE ACCIDENT PROTECTION,
MINIMUM RESTRICTION TO DECAY HEAT REMOVAL

A. W. Serkiz

E. C. Lusk

BATTELLE
Columbus Laboratories

ABSTRACT

Fire survival requirements impose various constraints on the design of nuclear material shipping container designs. The result has been, in many cases, a bulky package which is difficult to handle, load or unload, and quite susceptible to environmental conditions. Casks which must ship spent fuel assemblies having a high level of decay heat are presently considered to be a major design problem. However, the small casks (which often have a negligible amount of decay heat) must also survive the fire environment and not lose their shielding capability as specified by the AEC-DML regulations. Methods used in the past have included wood overcoats, concrete or water impregnated materials, and more recently, organic materials such as used in the Paper Tiger concept. This paper outlines several design techniques which have evolved over a number of years at Battelle-Columbus and which are based on the optimal use of various cask components to serve a multiple function (i.e., structural and thermal protection). Several of the more obvious techniques are the use of external heat rejection fins, multiple or laminated structural shells, and water-filled outer shells which also serve as neutron shields. This paper presents several cask concepts and the calculated thermal response of these cask systems when they were subjected to the hypothetical fire specified in the AEC 10-CFR-Part 71 regulations.

INTRODUCTION

Of the accidents postulated in the AEC regulations 10-CFR-Part 71, the thirty minute, 1475 F industrial fire poses the most severe and thermal environment that the cask will experience. Exposure of the shipping container in this environment could result in the melting and potential loss of part of the biological shield, as well as possible overtemperaturing of the internal contents, particularly of the fuel assembly claddings with the subsequent release of fission products. Since the industrial fires postulated are defined as an equivalent thermal radiative source having an effective emissivity of 0.9, utilization of cask design concepts, which would effectively block this radiant heat input, would enhance container survival. However, indiscriminate utilization of radiation shields has the obvious drawback of resulting in too high a thermal impedance.

Over the past ten to twelve years, Battelle-Columbus cask designers have evolved a variety of cask designs and, in general, drawn on the philosophy that various cask components can serve a multiple function. Illustrative of this approach is the use of decontamination shells (which are placed on the exterior of the cask surface to ease cleaning) to serve as a fire buffer. Cooling fins, which are required to dissipate high decay heats and also to reduce g-loads at impact can also serve as fire buffers because of their thermal decoupling characteristics. The more recent cask designs which require a water jacket on the exterior of the cask in a hybrid neutron shielding concept provide water which can be allowed to boil, and therefore, absorb the thermal input from the fire. In all these cases, we have taken advantage of components that are required for other functions to naturally impede the thermal pulse from the hypothetical fire. The next section will illustrate three typical systems and the calculated thermal response when the cask system is subjected to 10-CFR-Part 71 regulations.

Typical Cask Systems and Calculated Results

Three types of cask designs will be described next, and can be categorized as follows:

- ° Thermally decoupled
- ° Fin shielded
- ° Water shielded.

These categories are generally descriptive, and will be detailed next.

Thermally Decoupled Cask Systems

This type of cask is generally one where the decay heat load is small, and is typically a small cask such as the BCL-3 cask shown in Figure 1. On the other hand, these casks can go up to the 63,000 lb class as evidenced by the Peach Bottom No. 1 cask which has an external envelope of 173 inches long x 42.6 inches in diameter. As the above title implies, the "thermally decoupled" cask makes use of a secondary exterior thin-walled shell (typically stainless steel for corrosion and decontamination reasons) which is separated from the primary "outer shell" with a $1/8$ in. $\pm$ $1/16$ in. air gap thickness. The air gap thickness is normally kept small to provide for some thermal coupling to the exterior. Separation of outer plate from the primary cask shell is accomplished by means of a myriad of "weld spots". The outer shell is also seam welded at several axial positions for structural coupling.

A typical cask fire analysis is illustrated in Figure 2, and was submitted as part of the Peach Bottom No. 1 Safety Analysis Report. The thermal decoupling is quite evident as evidenced by the temperature differences between the fire buffer shell and the outer liner. Zero lead melt is predicted and the analysis techniques were adjudged adequate to justify fire survival. It should also be noted that this "cover" method is particularly adaptable to upgrading cask systems operating under the "grandfather clause", as was done with the ATCOR cask several years ago. Figure 3 illustrates the predicted thermal history of various components of the ATCOR cask. The thermal decoupling is again evident.

Fin Shielded Cask Systems

An illustration of a cask system relying on the heat rejection fins to effectively buffer the fire pulse is shown in Figure 4. This is the NFS-100 cask which was designed and licensed by Battelle-Columbus for Scientific Advances, Inc., who in turn was under contract to Nuclear Fuel Services, Inc. The heat rejection fins that were designed to dump waste decay heat during normal operation, also act as radiation shields. This can be deduced by examining the fin cavity region and its interaction with the radiant thermal source as shown in Figure 5. This figure also contains the geometric view factors for the fins and the cylindrical shell. The heat transfer which would occur can be expressed by the Stefan-Boltzmann equation results in Equation (1). It is evident that the geometric view factor area product is the key parameter.

$$q_{\text{net}} = (F_{\text{fins}} \cdot A_{\text{fins}} + F_{\text{cyl}} \cdot A_{\text{cyl}}) (\epsilon_{\text{surface}} \sigma_j T_{\text{surface}}^4 - \epsilon_{\text{flame}} \sigma_j T_{\text{flame}}^4) \quad (1)$$

Of interest, is the summation of the FA products divided by the total area, or for the NFS-100 cask.

$$\frac{\Sigma \text{FA}}{\Sigma A} = 0.20 \quad (2)$$

This ratio is in effect, the thermal radiation blocking afforded by the fins during the 30-minute fire. A 80 percent reduction is significant. Therefore, these same fins which are effective as heat dissipators during normal shipment, can impede the fire pulse.

The "blocking action" afforded by the fin cavity is shown in Figure 6 which contains calculated temperature histories for the NFS-100 cask system analyzed with and without fins when exposed to the hypothetical fire. The temperature response of nodes 17 and 18 (the fin material) illustrates the thermal gradient which can be sustained by the fin during the duration of the fire without undue conduction of the heat into the cask shield region. Utilization of the fins in the analysis results in predicting a zero lead melt while the simplified approach still utilized by some analysts (which discards the thermal shielding afforded by the fins) results in a prediction of 3/4" lead melt for a total lead thickness of 8.75 inches. The "blocking" effectiveness of the fins is self evident. This "thermal blocking" is collaborated by the data presented in Lusk's⁽¹⁾ paper describing the remelting of the NFS cask lead volume which took 17 hours to effect.

Water Shielded Cask Systems

The high burnup levels (35,000 - 50,000 MWD/MT) projected for PWR and BWR fuel assemblies, have necessitated the incorporation of neutron shielding in the next generation shipping casks. Our studies on behalf of the Whitehead & Kales Company have revealed that a minimum weight cask system results from use of water both within the cask cavity, and external to the lead gamma shield. This has been termed a hybrid (or split) shield concept and requires approximately six inches of water which is contained within an external water jacket. Pressure relief valves will permit venting pressure at a predetermined level to prevent jacket rupture, while accommodating a large heat input by means of phase change.

A calculated fire transient for a water-shielded system is shown in Figure 7. These calculations represent the anticipated thermal response for a two-element PWR truck-type cask. Since this cask has a rectangular inner cavity, three thermal histories are shown to illustrate the corner-side thermal coupling which is often discussed. The effectiveness of the fins is again evident with the water effectively isothermalizing the system. Less than three percent of the water was calculated to vaporize at 350 F (the pressure valve setting chosen). Loss of more than a small volume of the water due to the structural accidents postulated is minimized by compartmentalizing the outer water jacket as well as the use of a supplemental crash system. The thermal model was composed of 53 nodes and provided for natural circulation of the water.

ANALYTICAL TOOLS EMPLOYED

Although a variety of generalized heat-transfer codes (i.e., THT-D⁽²⁾, TRUMP⁽³⁾, CINDA-3G⁽⁴⁾, TIGER, etc.) exist throughout the country, our heat transfer analyses of cask systems have been carried out with the THT-D code. The THT-D code is a generalized computer program capable of analyzing multidimensional transient heat-transfer problems for both transient and steady-state conditions. Temperature solutions are obtained by iterative solution of simultaneous heat-balance equations derived within the program by finite-difference techniques. The use of simultaneous equations (the implicit method of formulating heat balances) precludes any stability limitations on time increments and permits a direct steady-state solution at any stage of the calculation, including solutions which serve as initial conditions for a transient. Convergence of a solution is controlled by the preset tolerances on either temperature tolerance, or heat balance, or both. The key features of the program are its:

- (1) Capability to use temperature-dependent material properties in calculations
- (2) Capacity to handle time-variant boundary conditions
- (3) Ability to calculate node-to-node heat transfer by the simultaneous mechanisms of conduction, convection, and radiation including internal heat generation
- (4) Ability to accommodate melting of nodes by means of a latent heat option for isothermal phase change.

Based on long-term utilization (i.e., 15 years) by various companies and a demonstrated correlation with actual heat transfer systems, THT-D has been accepted by the AEC-DML as an acceptable analytical tool. We have (at BCL) run our own comparative analyses to determine the adequacy of this computer code.

ACKNOWLEDGMENTS

Acknowledgment is extended to all those sponsors who have engaged Battelle's Columbus Laboratories for the design and licensing of shipping containers over the past 12 - 15 years, and in particular, to the Whitehead & Kales Company, Scientific Advances, Inc., Nuclear Fuel Services, Inc., and the ATCOR Company whose sponsored cask design activities have been used for illustrative purposes within this paper.

REFERENCES

- (1) E. C. Lusk, Lead Remelt Experience with 60 Ton Cask, to be published in the proceedings for the Third International Symposium, Packaging and Transportation of Radioactive Materials (August, 1971).
- (2) Skirvin, S.C., "User's Manual for the THT-D Comput Program (Transient Heat Transfer - Version D)", GE-NMPO Report P.O. No. 036-926052-T0602, (June 23, 1966).
- (3) Edwards, A.L., "TRUMP: A Computer Program for Transient and Steady-State Temperature Distributions in Multidimensional Systems", LRL-University of California, Report No. UCRL-14754, Rev. II, (July 1, 1969).
- (4) Connor, R.J., Kannady, R.E., Jr., and Almanza, J.E., "Adaptation of Chrysler Improved Numerical Differencing Analyzer to CDC 6000 Series Computers", Martin Marietta Corporation, Report No. M-68-22, (November, 1968).

Permit No. -SP-5950

Modes of Transportation - all except by water

Maximum Thermal Decay Heat = 200 watts

* Maximum Fissile Contents:

Fissile Class I = 100 gms U-235 equiv. mass

Fissile Class III = 2,000 gms U-235 equiv. mass

Lid Weight = 185 lbs

Total Cask Weight = 2,500 lbs (1,136 kgs)

Floor Loading = 428 lbs/sq ft (195 kg/sq ft)

Cask Diameter = 18.50 inches O.D.

Cask Height = 25.94 inches overall

Base Plate = 24.00" x 24.00"

(2) 1/2" diameter eye bolts required to remove lid

Non "Special Form" type B and large quantity material
require inner can - per drawing BCL-3-38

* Refer to DOT-SP-5950 permit.

Fig. 1. Battelle Memorial Institute BCL-3 Series, Radioactive Material Shipping Container

Fig. 2. Temperatures Versus Time for W&K PB No. 1 Cask System (During and After Hypothetical Fire Accident)

Fig. 3. Temperature History of the Cask Cylindrical Region for the ATCOR Cask

Fig. 4. NFS-M100 Cask System

Surface (1)= fin surface
 Surface (2)= cylinder
 Surface (∞)= external environment

$$F_{G1-\infty} = 0.17$$

$$F_{G2-\infty} = 0.18$$

$$F_{G1-1} = 0.69$$

$$F_{G2-2} = 0$$

$$F_{G1-2} = 0.14$$

$$F_{G2-1} = 0.42 \times 2^*$$

$$\Sigma_{F_G} = 1.00$$

$$\Sigma_{F_G} = 1.00$$

* Note that the cylinder sees two fins simultaneously

Fig. 5. Geometric View Factors for a Typical Fin-Cylinder Cavity

Fig. 6. Computed Temperature Histories for the NFS-100 Cask System

Fig. 7. Computed Temperature Histories for a Two-Element PWR Cask

EVALUATION OF PLUTONIUM PRODUCT FORM ON SAFETY DURING TRANSPORTATION

L. M. Knights

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

X DROP TESTS OF SHIPPING CONTAINERS FOR
PWR TYPE NEW FUELS

Y. Seki	S. Aoki
S. Nakata	S. Shimamura
Y. Kanae	T. Koguchi

ABSTRACT

For the purpose of evaluating shock absorbing characteristics and structural integrity, a series of drop tests was conducted on two sizes of prototype shipping container for light water reactor new fuel assemblies. The No. 1 prototype container, which was designed to ship large PWR fuel assemblies, was drop tested from the height of up to 90 cm onto an essentially unyielding horizontal surface installed at the Government Mechanical Laboratory of Japan, while the No. 2 prototype, which was designed to ship small fuel, was drop tested from the height of up to 1.2 m and 9 m. Decelerations at the fuel receptacle and the container shell, displacement of the fuel receptacle, and dimensional changes of shell were recorded and evaluated.

INTRODUCTION

In the design of shipping containers for new fuel assemblies, the following two aspects are to be taken into account;

- (1) Constructional integrity to meet the requirements of Safety Regulations for shipping container concerned.
- (2) Protection of fuel assembly from shocks and vibrations encountered during handling and shipping.

The constructional integrity of the container will be more important and particularly because the shipping container for Pu-bearing UO_2 new fuel

assemblies for light water reactor falls in the classification of the type B packaging (or the large quantity packaging). From the point of view of nuclear safety, the container shell must have excellent shock absorbing characteristics in order to prevent fuel rods from failure at the time of 9 m drop, on the other hand it is required to have a rigidity to keep reasonable spacings between the fuel assemblies in damaged containers to avoid criticality.

In order to design effective shock absorbing suspension system, its transmissibility, T , must be small at the input frequency range. It can be more clearly understood by the following equations for forced vibration of a damped single degree of freedom system;

$$T = \frac{1 + (2\zeta \frac{\omega}{\omega_n})^2}{\sqrt{(1 - (\frac{\omega}{\omega_n})^2)^2 + (2\zeta \frac{\omega}{\omega_n})^2}} \quad (1)$$

and

$$A = A_0 T \quad (2)$$

where, T = transmissibility

ω = input frequency

ω_n = undamped natural frequency of system

ζ = damping ratio

A = resultant system acceleration

A_0 = input acceleration

To make the value of T small, ω_n must be small in compare with ω . However in practice, ω_n must be so determined as to deviate the frequency range of 2 ~ 4 HZ, which normally encounters at vehicle deck during transportation.

DESCRIPTION OF PROTOTYPE CONTAINERS

Taking the above mentioned factors into account, prototype containers for transporting new fuel assemblies were designed and fabricated. Principal

particulars of the prototypes are shown in Table 1, while schematic drawing is shown in Figure 1. The prototype containers tested consist basically of a split type horizontal cylindrical shell of steel plate in which a fuel assembly (or two assemblies) is accommodated on a receptacle with shock absorbing means provided. Two sizes of production prototype, No. 1 and No. 2 containers, were loaded with dummy loads simulating the weight of fuel assemblies and instrumented with accelerometers.

Table 1. Principal Particulars for Prototype Containers

Items		No. 1 Container	No. 2 Container
Dimension	Overall length	4,350 mm	1,860 mm
	Overall height	1,200 mm	670 mm
	Overall breadth	1,140 mm	610 mm
	Thickness of shell plate	2.3 mm	1.0 mm
Weight	Container Shell	580 kg	140 kg
	Fuel and receptacle	1,560 kg	125 kg
	Package weight	2,140 kg	265 kg
No. of fuel ass'y accommodated		2	1
Mass ratio of suspension system and shell		2.67	0.89 ^e
Designed natural frequency of suspension system		4.2 HZ	4.5 HZ

TEST FACILITY AND EQUIPMENT

Drop Test Facility

A drop test facility installed at the Government Mechanical Laboratory of Japan built specially for drop testing spent fuel shipping cask models was employed.¹ It essentially consists of a steel tower with a lifting device of 10 ton - capacity and a target having an essentially unyielding horizontal surface as required by U.S. AEC and IAEA regulations.

Instrumentation and Data Processing

Strain gage and piezoelectric type accelerometers were fixed at the fuel receptacle and the container shell. Output signals from the accelerometers were amplified and recorded on a magnetic tape. The data stored were then filtered and reproduced in visible form by means of a recording oscillograph. A part of the data on the magnetic tape was recorded on continuous loops of magnetic tapes and analysed to yield power spectral density by means of a tracking analyser. For the detection of displacement of the fuel receptacle at the time of impact, simple displacement monitors were installed at the bottom of the container shell. Dynamic behaviors of the container were recorded by 16 mm cinecamera and high speed cinecamera.

DROP TESTS

1.2 m Drop Tests

The No. 1 container was flat, corner and edgewise rotational drop tested up to the height of 90 cm, while the No. 2 container was flat drop tested up to 120 cm. Deceleration peaks measured at the shell and the fuel receptacle are plotted in Figure 2. Deceleration ratio of the receptacle and the shell, which is considered to be the transmissibility, T , of the receptacle is also plotted in Figure 2. Deceleration data recorded on continuous loops of magnetic tape were analysed and power spectral density diagrams were obtained. Typical examples of PSD diagram of the No. 2 containers are shown in Figure 3. Displacement of the fuel receptacle due to the drop shock is plotted in Figure 4. Dimensional measurements of the shell before and after the drop tests have shown that there existed no appreciable deformations, and clamping bolts at shell flanges remained sound.

9 m Drop Test

The No. 2 container was so drop tested as the cover of the container flatly strikes the target. It comply with the requirements of the regulations

for 9 m (30 ft) drop test since the maximum damage is expected when the cover flatly strikes the target. It is because the fuel spacings in the damaged containers are of primary importance for the nuclear criticality assessment, rather than the leak tightness of the container. Deceleration peak as high as 390 G was recorded at the fuel receptacle. In Figure 5 and 6 results of dimensional measurement on the deformed container are shown. From above figures, it is clear that the container lost the volume considerably, but collapsed areas were confined to roughly one-half of the periphery of the cover. Cover clamp bolts remained sound, but leak-tightness was lost at flanges and some broken areas of the cover.

DISCUSSIONS AND RESULTS

- (1) Let W_i be the weight of fuel and receptacle (suspension system) and W_o be that of shell. When the container drops from the height, H , then the potential energy possessed by the container, E , is

$$E = (W_i + W_o) H \quad (3)$$

which is entirely transformed into kinetic energy, E_k , at the impact. Neglecting the losses, E equals E_k . From the deceleration-time-histories it is found that the deceleration peak of the fuel receptacle delayed certain amount of time from that of the shell. It implies that at the instant of impact, the container receives the kinetic energy equivalent to $E = W_o H$, rather than $E = (W_i + W_o) H$. And certain time later, the container shell receives the compressive force transformed from $E_i = W_i H$ through the suspension springs. As a result, damage to the container will be greatly reduced than in the case of container which has no elastic suspension system.

- (2) Fuel receptacle suspension system showed superior shock isolation characteristics and demonstrated a correctness of design.

- (3) Use of elastic suspension system reduced shocks to the shell also, thus minimizing the damage of the shell.
- (4) Relatively less rigid shell design greatly reduced shocks to the fuel receptacle at the 9 m drop test. It contributed much to the protection of fuel rods from failure.
- (5) Collapse pattern and amount of collapse at the container shell may be affected by the relation between modulus of section of circumferential ribs and the radius of the shell.

ACKNOWLEDGEMENTS

Thanks are due to Dr. Muramatsu who helped us in the work of high-speed cinematography.

REFERENCES

1. S. Aoki, IAEA/SM-147/3 Experimental Studies on the Spent Fuel Shipping Cask in Japan, Proc. IAEA Seminar on Test Requirements for Packaging for the Transport of Radioactive Materials, Feb. 1971.

Fig. 1. Schematic Drawing of No. 1 and No. 2 Containers.

Fig. 2. Decelerations Measured at Fuel Receptacle and Shell VS. Drop Height, and Transmissibility VS. Drop Height.

Fig. 3. Power Spectral Density Diagram of Deceleration at 120 cm Drop Test for No. 2 Container.

Fig. 4. Displacement of Receptacle VS. Drop Height
Plotting for No. 1 and No. 2 Containers.

— NO.1 RIB
 - - NO.2 "
 - · - NO.3 "
 - · - NO.4 "

Fig. 5. Plastic Deformation of Circumferential Ribs after the Crash.

Fig. 6. Plastic Deformation of Shell Flanges after the Crash.

1 SHIPPING PACKAGE FOR
UNIRRADIATED POWER REACTOR FUEL ASSEMBLIES

R. E. Kropp

ABSTRACT

A package for the shipment of unirradiated power reactor fuel assemblies was designed, tested and then used during the past four years. This package is identified as model UNC 2800 and was issued DOT Special Permit No. 5419. Evaluations were performed using criteria set forth in the AEC and DOT Regulations for fissile material packages. Nuclear safety analyses were performed using empirical techniques. Although less exacting than computer analyses, these techniques yielded conservative results. Operating needs resulted in modification of the loading of this package. Nuclear safety analyses associated with this modification were performed using reactor design computer codes.

INTRODUCTION

A shipping package for unirradiated, low enrichment uranium, power reactor fuel assemblies was designed and tested in 1967. It was developed to provide a single package type for the shipment of Boiling Water Reactor (BWR) or Pressurized Water Reactor (PWR) fuel assemblies at enrichments up to 2.34% and 3.5% respectively. Operational requirements included ease in loading and unloading plus a replaceable assembly holder. Modes of transport were to be motor vehicle and rail. Plans called for a ten-year life span for this package prior to replacement by advanced design models.

The package consists of a U-shaped strongback fuel assembly support 9 inches high by 10 inches wide by 192 inches long constructed of 7-gage steel and also having adjustable end clamps and cross support brackets. The fuel assembly support is shock mounted to an outer container by shear mounts. The outer container is a 12-gage steel cylinder 36 inches I.D. by 207 inches long with a flange closure, skids, stacking brackets and roll rings. The package is identified as model UNC-2800 and issued DOT Special Permit No. 5419.

DESIGN AND TESTING

Basic Design Criteria

Original design requirements were that individual packages hold either two BWR fuel assemblies or one PWR fuel assembly and that packages could be shipped in truck load quantities while meeting the criteria for Fissile Class II and III shipments. Structural requirements were established such that the container would meet the U. S. Atomic Energy Commission's hypothetical accident conditions and the U. S. Department of Transportation's Special Test Criteria for Type B Packages. The package was required to hold a 1,510 pound load (net weight). After construction, the empty package weighed 2,340 pounds (tare weight) resulting in a gross weight of 3,850 pounds.

Structural Evaluation

Test Description

The steel inner container was loaded as shown on Figure 1 with metal dummy fuel assemblies to simulate the weight of the SNM bearing material for which this package would be used. The inner container and the outer container were then sealed in the same manner as that for shipments. The loaded package first was dropped 40 inches upon a 6-inch diameter by 8-inch high steel spike as shown on Figure 2. The container was dropped on the bottom center of a welded seam. A 30 foot free fall test was next conducted in the facilities of a neighboring scrap yard. The package was raised by a crane to a 30 foot

height at approximately a 45° angle from the end as shown on Figure 4. The height was ascertained by a premeasured weighted cord hanging from the container. A quick release mechanism was utilized to unlatch the package which fell perfectly at approximately a 45° angle landing on the bottom corner of the container. A second 30 foot drop test was then made. The package was oriented to fall flat upon one flanged side where the top and bottom shells were joined as shown on Figure 7. The drop test was perfectly executed and the package fell horizontally on the intended flange.

Thermal and water immersion tests were not performed. These tests were omitted since: (1) only materials which will not thermally decompose would be shipped in these containers, and (2) the nuclear safety of individual packages and arrays was to be evaluated assuming the in-leakage of water.

Test Results

After the puncture test, the package experienced a three inch deflection in the impact area, deforming uniformly up to the central roll rings. There was no evidence of any damage to the welded joint. The results of this test are best shown on Figure 3. The corner free fall test caused significant crumpling at the impact end of the package. The end roll ring deformed first and then broke off the package. The shell itself on the corner was crumpled approximately five inches. The deformation did not extend significantly back to the package shell. There was some indication, due to bumps in the outer shell, that some movement of the strongback brackets had taken place. The results of this test are shown on Figures 5 and 6. The side free fall test resulted in mostly superficial damage. The fork truck lifting brackets which are welded to the outside of the package for lifting and stacking purposes were crushed. But there was no indication of damage to the flanged joint except the one end where the flange was dented slightly and the corner bolt sheared off. There was additional indication that movement of the internal strongback brackets had taken place. Results of this test are shown on Figure 8.

Upon removing the top cover it was observed that the dummy fuel assemblies were in place in the strongback channel. All internal clamps holding the dummy fuel assemblies down were in place. There had been approximately a 3/4-inch shift of the end brackets securing the ends of the dummy fuel assemblies at the end where the package fell during the 45° angle test. There was evidence of damage to the outer shell at the joints where the gusset brackets of the strongback had moved laterally into the shell. This lateral movement is made possible by the shock mount installation. The shock mount brackets on both the strongback and the shell were significantly deformed and some of the bolts securing the shock mounts to the shell brackets had sheared off. The shock mount, however, had returned the strongback to the approximate center position of the shell. There was a two inch lateral displacement of the strongback toward the flange side that impacted, and the end opposite the 45° impact point experienced a two inch downward drop. The effects of the tests on the inside of the container are best shown in Figures 9 and 10. It appears that the dummy fuel assemblies after undergoing two 30 foot drop tests experienced two inch lateral and vertical displacements within the 36 inch diameter shell and that the external deformation was so limited as to prevent any other packages from approaching the fuel location closer than would occur with an undamaged package array.

Nuclear Safety Considerations

Original nuclear criticality safety evaluations were performed using the Density Analog Method.¹ Optimum water moderation in the interstices of the fuel assemblies was assumed under both normal and accident conditions. Although fuel assemblies are shipped dry, this assumption yielded conservative analytical results. Also, the neutron absorption effects of the steel in the package were neglected. Results of this analysis are shown on Table 1.

Based on the size and weight, 40 packages were selected for Fissile Class II Shipments. This resulted in a 1.0 Transport Index based on the "40 unit" rule which was changed to a 1.3 Transport Index upon adoption of the "50 unit" rule.

Table 1. Nuclear Safety Results Based on Density Analog Calculations

Assembly Type	Transport Conditions	Moderation Ratio (H/U-235)	Number of Containers Critical	Allowable Fissile Class II	Allowable Fissile Class III
BWR	Normal	165	4320	864	2160
BWR	Accident	165	2459	1229	2458
PWR	Normal	121	1555	311	778
PWR	Accident	121	932	466	931

RECENT MODIFICATIONS

In 1970, it was desired to increase individual container loadings from two to four BWR fuel assemblies to reduce shipping costs. Also, transportation modes needed to be extended to include aircraft and vessel. Additional structural evaluation was not required since the proposed load (net weight) did not exceed that originally tested. Revised nuclear criticality safety evaluations were performed using reactor design computer codes. This was done to factor in the neutron absorption effects of the steel in the package. Individual containers and arrays of them are not considered hydrogen moderated and, therefore, are subcritical. Under accident conditions, water inleakage is possible. Accident condition analysis was performed using a combination Transport-Diffusion Theory Model. This model uses the LOCALUX-3² Code to determine the diffusion coefficient and macroscopic cross sections for the fuel-moderator-inner container region and within the outer container region. Diffusion coefficients and macroscopic cross sections for the regions between the inner and outer container and outside the outer container were calculated using the FORM³ and TEMPEST⁴ Codes. Array infinite multiplication factors and critical buckling were then obtained using the AIM-6 Diffusion Code (an

extension of AIM-5⁵). The results shown on Table 2 indicate that the infinite multiplication factor is less than unity under all cases. The maximum infinite multiplication factor occurs with optimum water moderation in the interstices of the fuel assemblies and no water between inner and outer portions of the container or between individual packages. Increasing water density between inner and outer container and increasing water thickness between individual packages results in decreased infinite multiplication factors.

Since the infinite multiplication factor is less than unity for all conditions, this package was shown to meet the requirements for Fissile Class I for BWR fuel assemblies.

SUMMARY

Four years operational experience with this package has been good. It has proved to be the versatile package it was designed to be. Anticipated future modifications include the shipment of longer assemblies and assemblies with increased enrichments. The use of advanced computer codes with Monte Carlo considerations also should provide more accurate assessments of the nuclear safety of individual packages as well as arrays of them.

ACKNOWLEDGEMENT

The assistance of Mr. E. Fass of the Research and Engineering Center for his aid in obtaining computer analysis is gratefully acknowledged.

Table 2. K_{OO} Vs. Amount of Water Between Inner and Outer Container
and Water Outside the Outer Container

Amount of Water Between Inner and Outer Container (%)	Amount of Water Outside The Outer Container (cm)	K_{OO}
0.0	0.0	0.818
	1.0	0.744
	2.0	0.713
	3.0	0.695
	4.0	0.684
1.6	0.0	0.805
	1.0	0.741
	2.0	0.711
	3.0	0.694
3.22	0.0	0.788
	1.0	0.733
	2.0	0.708
	3.0	0.693
5.0	0.0	0.770
	1.0	0.726
	2.0	0.705
	3.0	0.692
10.0	0.0	0.736
	1.0	0.711
	2.0	0.696
	3.0	0.687
100.0	4.0	0.770 ^a

^aNote that flux is assumed to be zero at the boundary.

REFERENCES

1. H. C. Paxton, Criticality Control in Operations with Fissile Material, LA-3366 (January 1966)
2. S. Kellman and R. R. Calabro, LOCALUX-3 -- A Depletion Program for Multi-zoned Reactor Assemblies, UNC-5230 (July 1969)
3. D. J. McGoff, Form - A Fourier Transform Fast Spectrum Code for the IBM-709, NAA-SR-Memo-5766 (September 1960)
4. R. H. Shudde and J. Dyer, TEMPEST - II -- A Neutron Thermalization Code, AMTD-111 (June 1962)
5. H. P. Flatt and D. C. Baller, AIM-5 -- A Multigroup One Dimensional Diffusion Theory Code, NAA-SR-4694

FIGURE 1 - Container Open showing Dummy Lead - CONTAINER UNC-2800

FIGURE 2 - Container at Elevation Prior to Puncture Test

FIGURE 3 - Deformation After Puncture Test

FIGURE 4 - Container at Elevation Prior to Corner Free Fall Test

FIGURE 5 - Deformation Due to Corner Free Fall Test

FIGURE 6 - Details of Deformation Due to Corner Free Fall Test

FIGURE 7 - Container at Elevation Prior to Side Free Fall test.

FIGURE 8 - Deformation Due to Side Free Fall Test

**FIGURE 9 - Container Open Showing Internal Deformation
Due to All Tests**

**FIGURE 10 - Details of Internal Deformation Due
to All Tests**

A SIMPLE SHIPPING SYSTEM FOR POWER GRADE PLUTONIUM OXIDE

R. E. Giebel
R. G. Leebl

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

TIEDOWN SYSTEMS FOR CASKS

E. E. Lewallen

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

COMPACT METALLIC IMPACT LIMITERS FOR SHIPPING CONTAINERS

R. J. Burian

E. C. Lusk

BATTELLE
Columbus Laboratories

ABSTRACT

Two types of all-metal impact limiters have been developed to protect radioactive shipping containers from damage due to accident conditions such as those outlined in the AEC 10-CFR-Part 71 Regulations. One type is designed to protect the shipping container from blows directly on the end or on an edge at the critical oblique angle. This type of impact limiter is comprised of a cluster of metal tubes or a welded metallic honeycomb within a specific array. The kinetic energy of the container is absorbed by plastic strain of the tubes or honeycomb structure, which in turn protects the cask from deformation and excessive forces which could damage the container or its contents. This type of impact structure is fully enclosed in a metallic shell (normally stainless steel) for corrosion reasons and easing decontamination. The development of the design is detailed below and the mechanics of applying the design equations are also illustrated.

The second type of limiter is designed to protect the shipping container from impact on the side. This design incorporates metallic tubes, partially filled with water, attached to the surface of the shipping container so that the tube axis is parallel to the container axis. Upon impact the tubes are deformed causing the pressure in the tubes to increase. At a pre-designed pressure, relief ports open and the water is expelled through orifices. The kinetic energy of the cask is absorbed not only by plastic deformation of the tube but also by the inertia and viscous losses in the water as it is expelled. The water also serves as a neutron moderator for containers which are used to ship neutron emitting materials.

INTRODUCTION

The primary concern in the shipment of any radioactive material is nuclear safety. To assure a high degree of safety, the Atomic Energy Commission in the 10-CFR-Part 71 Regulations has outlined a group of hypothetical accident conditions which a shipping container must survive while maintaining adequate shielding, containment, and subcritical conditions. The 10-CFR-Part 71 Regulations require a shipping container withstand a hypothetical accident in which the cask free falls 30 feet onto a hard, unyielding surface without failure of the "containment" seal, without change in the container's criticality conditions, and without loss of shielding beyond defined limits. It is often necessary also to limit the impact forces to prevent failure of the fuel-pin cladding or damage to failed fuel canisters within the cask.

Three orientation attitudes lead to the greatest probability of severe damage or cask deformation: (1) end-on impact, (2) flat-side impact, and (3) oblique impact on an end edge at such an angle that the line of reaction of the impacting force passes directly through the cask's center of gravity (Figure 1). If it can be shown that a container is capable of surviving the 30-foot free-fall condition in these critical attitudes, the likelihood is high that the container will be able to survive impact in intermediate attitudes.

To preclude excessive deformation or unacceptably high impact forces, the container is often covered or surrounded with cushioning material. The diversity of irradiated materials and equipment and unirradiated fissile materials leads to shipping containers of vastly different design and weight; thus, the energy absorbers are often very different. This paper is confined to two impact absorbers designed and conceived at Battelle-Columbus for use with the heavier containers (casks) for the shipment of radioactive power-reactor fuels.

Many types of impact absorbers have been used with shipping casks. Some earlier designs of pipe and/or I-beam cages supported the cask in the center. Some designs incorporated rubber mountings at strategic locations, other designs utilized massive fins on the side and end of the cask to absorb the impact energy by deformation of the fin, thereby protecting the cask itself. Other designs utilized wooden overcoats to absorb the energy. Many of these impact absorbers were heavy and bulky, increasing the envelope volume as much as an order of magnitude over the volume of the cask itself. Others interfered with removal of heat from the cask.

During the past year, Battelle-Columbus has been developing designs for impact absorbers that overcome many of the shortcomings of presently employed absorbers. This paper discusses two of these designs, one of which has been found acceptable by the Department of Materials Licensing for use in the Peach Bottom No. 1 cask. This absorber (a tube cluster) is best suited for protection of the cask from end-on and corner impacts. The second design, which is still being studied, is best suited for protection of the cask from impact on the side. The development of both of these designs was undertaken for the Whitehead & Kales Company in Detroit, Michigan.

Tube-Cluster Impact Limiter

The tube-cluster impact-limiter design was evolved to protect the cask from the end-on, and critical-angle edge-impact conditions (Figures 1a and 1c). This impact limiter is composed of metallic tubes (or pipes) joined together in a group or cluster (Figure 2). The tube cluster is fitted between end plates but need not be joined to them. For purposes of tube containment, cleanliness, and ease of decontamination, a thin metallic wrapper surrounds the cluster and is welded to the two end plates. The impact limiter is attached to the cask so that one end plate rests directly on the flat end surface of the cask. A severe blow on the end (or at an oblique angle), such as the cask would experience in a 30-foot free fall, will collapse the tube cluster thus absorbing the kinetic energy of the cask through plastic strain of the tubes. In this way the impact forces are held to a tolerable level and the cask is not deformed.

Design Theory

Specific energy, the parameter used to evaluate the worth of a crushable energy-absorbing device, is the ratio of the maximum energy the structure can absorb to the weight of the structure, or

$$E_{sp} = \frac{E_{ab}}{W} = \frac{E_{ab}}{\rho_u Ah} \quad , \quad (1)$$

Where

E_{sp} = specific energy,
 E_{ab} = maximum energy absorbed,
 ρ_u = unit density of undeformed structure,
 A = gross area of structure, and
 h = undeformed height of structure.

The absorbed energy is equivalent to the work required to deform the structure,

$$E_{ab} = \frac{\sigma_{mc} A d}{\rho_u A h} = \frac{\sigma_{mc} d}{P_u h} \quad , \quad (2)$$

where

σ_{mc} = mean crushing stress,
 A = area upon which the crushing stress acts, and
 d = deformed distance.

Therefore,

$$E_{sp} = \frac{\sigma_{mc} A d}{P_u A h} = \frac{\sigma_{mc} \eta}{P_u} \quad . \quad (3)$$

The term η (the ratio of the deformed distance to the height of the undeformed structure), often called the thickness efficiency or stroke efficiency, is a measure of the distance the structure can deform before "bottoming" occurs.

In 1964, McFarland* investigated the energy-absorption properties of the three types of all-metallic structures shown in Figure 3. His studies involved both static and dynamic loading (impact velocities up to 50 fps). The typical collapse mode of the structures was the formation of convolutions as shown in Figure 4. The energy absorbed during the collapse of the structures correlated with a rigid-plastic mode of strain response.

---

* McFarland, R. K., Jr., "The Development of Metal Honeycomb Energy - Absorbing Elements", Jet Propulsion Laboratory, Pasadena, California, Technical Report No. 32-639 (July 24, 1964).

Utilizing McFarlands investigation we developed the following expressions for the mean crushing strength of the three types of structures:

$$\text{Closed-packed tubes: } \sigma_{mc}_{cp} = 55.13 \sigma_y \left(\frac{t}{s}\right)^2 \quad (4)$$

$$\text{Loose-packed tubes: } \sigma_{mc}_{lp} = 47.74 \sigma_y \left(\frac{t}{s}\right)^2 \quad (5)$$

$$\text{Hexagonal cell: } \sigma_{mc}_{hex} = 40.53 \sigma_y \left(\frac{t}{s}\right)^2, \quad (6)$$

where

σ_y = yield stress of the material of construction, and
 $\left(\frac{t}{s}\right)$ = ratio of thickness to the characteristic dimension,
 Figure 3.

The "thickness efficiency" of three structures, being the same within 5 percent, can be expressed as

$$\eta = 88.13 - 329 \left(\frac{t}{s}\right) \quad (7)$$

The unit weight is a function only of the geometry of the structure and is defined for the three configurations in Figure 3.

In the actual mechanics of limiter design, additional relationships must be considered. The weight of the impact-limiter tubes can be expressed as the ratio of the kinetic energy of the cask to the specific energy of the limiter. Then

$$w_{tube} = \frac{E_h}{E_{sp}} = \frac{WH}{E_{sp}}, \quad (8)$$

where

W = the cask static weight, and

H = the fall height.

The limiter tube weight can also be expressed as

$$w_{tube} = Ah_{\phi_u}, \quad (9)$$

where

A = gross area of the limiter,
h = limiter height, and
 ρ_u = unit weight.

Combining Equations (3), (8), and (9) and solving for the limiter height yields

$$h = \frac{E_h}{\sigma_{mc} \eta A} = \frac{WH}{\sigma_m \eta A} \quad (10)$$

This is the minimum limiter height required to prevent bottoming of the limiter before all the kinetic energy is absorbed.

Assuming then that the mean crushing stress is constant during the limiter collapse, the impact force in G units can be expressed as

$$g = \frac{F}{W} = \frac{\sigma_{mc} A}{W} \quad (11)$$

These relations then permit the design of an impact limiter for virtually any weight vessel and any desired impact force.

Mechanics of Design

Equations (4), (5), and (6), show that the close-packed tube configuration results in the stiffest structure followed, in order, by the loose packed array and the hexagonal-cell structure. The hexagonal cell structure, provides more versatile design since a cell size can be chosen that will give a limiter having the exact crushing stress desired. These structures probably would have to be fabricated for each application as they are not normally "shelf" items. We found it to be more cost effective to utilize commercially available tubing. Commercial tubing is restricted to specific wall thicknesses and tube diameters, however, and the designer must compromise on a structure having a mean crushing stress reasonably close to that desired, unless special tube sizes can be obtained.

In tube selection, curves such as those shown in Figures 5 and 6 for a loose-packed array can be used. From these curves, the required value of t/s can be determined for a known cask weight and gross area available for the impact limiters, and a standard tube having a t/s value acceptably close can be used. Once a tube size is selected on the basis of Figure 5, the total weight of the tubes in the impact limiter can be selected from curves

similar to those shown in Figure 6 for a loose-packed array. The weight of end plates, side shell, and any attachments for handling, etc., must be added to get the total weight of the impact limiter. The minimum required height of impact limiter can be obtained directly from Equation (10).

The metallic tube-cluster impact limiter can also be designed to protect the cask for the end-edge (corner) impact condition. Many casks have a large aspect ratio (length to diameter ratio). Thus, in corner impact where the line of action passes through the cask's center of gravity, the critical cask impact angle (Figure 1b) is only 15 to 25 degrees from the vertical. Since the impact limiter is designed primarily for axial impact loading, it will perform less efficiently for corner or oblique loading. Therefore, it was assumed that the effective resistance and energy absorption of the impact limiter during oblique impact can be expressed as a function of the impacting angle, or

$$E'_{sp} = E_{sp} \cos \alpha \quad . \quad (12)$$

The specific energy, however, is directly proportional to the mean crushing stress of the impact limiter. Therefore,

$$\sigma'_{mc} = \sigma_{mc} \cos \alpha \quad , \quad (13)$$

where

σ'_{mc} = an effective mean crushing stress acting on the
area of the deformed impact limiter

σ_{mc} = mean crushing stress for axial loading.

From Figure 7, it is seen that as the cask deforms during impact, and the deformation distance (d) increases, the area (a_c) upon which the effective crushing stress (σ'_{mc}) acts also increases, and the decelerating force increases.

A computer program was written to evaluate the maximum impact load that the cask will experience in the corner-impact orientation. The design equations presented above were solved for small finite steps of limiter deformation, and the cask velocity was reduced to zero by successive iterations. A subsequent calculation was required to assure that, up to the maximum impact load, the limiter would not fail by shear due to the transverse component of the impact force.

For the oblique corner attitude, the effective crushing stress is never as great as that for direct end impact, and the resisting force is not constant since the contact area of the deformed volume increases as the deformation distance increases. Consequently, greater axial deformation of the limiter is required to absorb all of the kinetic energy. The weight of the tubes and wrapper is proportionally increased, although the weight of the end plates and other attachments is unchanged.

The tube-cluster limiter has an advantage not attributable to some impact absorbers presently in use. Its all-metal construction and the axial orientation of the tubes provide relatively good heat transfer during normal operation. However, it still has good thermal impedance to protect the cask from high heat fluxes of short duration such as the hypothetical fire conditions stipulated in the AEC 10-CFR-Part 71 regulations.

The impact limiters designed for the Peach Bottom No. 1 shipping cask are shown in Figure 8. The cask weighs 61,600 lb fully loaded. The impact limiters comprise approximately 225 tubes made of mechanical steel tubing having a yield stress of 60,000 psi. The 2-1/2-inch-OD x 0.095 inch thick wall tubes are 8.5 inches long. (For end-impact only, the tubes would need to be only 5-1/4 inches long.) They are welded together in a loose-packed array and sandwiched between two, 1/4-inch stainless steel plates to provide a bearing surface for the tubes. For ease of surface decontamination, the side of the limiter is enclosed with a 1/8-inch stainless steel wrapper welded to the top and bottom plates. A skirt (1 inch thick and 4 inches long) is welded to the bottom plate and fits over the end of the cask. It is designed to take the transverse load for the oblique impact condition. Four holes penetrate the top end plate and tube cluster of the limiter to permit bolting the limiter to the cask through the bottom end plate. The limiter weighs 630 lbs; and of this 385 lb. is attributed to the tubes.

This limiter design has many advantages over others that have been used. It is compact and easy to decontaminate. It provides complete protection for the end of the cask, thereby making it easy to demonstrate cask-closure integrity after an impact of the nature stipulated in the regulations. It is all metallic, thereby making it sturdy and capable of withstanding moderately "rough" handling. The all-metal construction also makes it relatively unaffected by neutron exposure. Its compactness enables constructing exposed surfaces of corrosion-resistant materials at reasonable cost. It completely covers the end of the cask, thereby protecting it from the high thermal flux incident during an accidental fire exposure. Finally, and of special concern to the designer, it can be quickly analyzed and the tradeoffs rapidly and easily evaluated.

The tubular structures are highly efficient and result in a compact limiter, but the fabrication requires time-consuming joining of many small pieces. Battelle is investigating the properties of an all-brazed honeycomb structure that is commercially available on a special-order basis and for which substantial crush-strength data are available.

Hydraulic-Tube Impact Limiter

The necessity of shipping "high burnup" and Pu recycle fuels in the near future has established the need for neutron shielding. Consequently, future-generation shipping casks will provide both gamma-ray and neutron shielding. Our preliminary designs for casks include a water jacket around the gamma shield. In Battelle's work for Whitehead & Kales, we examined the feasibility of making the water jacket double as an impact limiter for side-impact conditions. The basic concept is relatively simple - the kinetic energy is absorbed by throttling the water through a series of orifices, similar to the way in which hydraulic shock absorbers operate in automotive vehicles.

The design now being examined is in Figure 9. As illustrated, the water jacket consists of an annulus divided into several compartments by pipes or tubes extending the length of the cask. Each isolated compartment is filled with water, leaving a small air space for thermal expansion of the water and initial impact-load cushioning. The orifices fitted in the ends and along the pipe are covered and sealed with pop-off plugs. Upon side impact, one or more of the compartments will collapse, causing the pressure of these compartments to rise rapidly. The pop-off plugs will then open, permitting the water to eject through orifices and absorb the kinetic energy of the cask.

A computer program written to assist in the analyses takes into account the energy dissipated by ejection of the water through the orifices, the energy absorbed in the viscous and inertia forces developed in the fluid, and the energy absorbed in deforming the metal pipes.

Parametric studies designed for better understanding the system and to assist in planning an experimental program are continuing. Preliminary results are enlightening, and modifications of the designs have been indicated. It was hoped that one orifice at each end of each compartment would be adequate. The preliminary analyses have revealed, however, that even with fully open ends the water could not be expelled fast enough to prevent the

tubes from rupturing. Consequently, it appears necessary to consider placing orifices at intervals along the pipe. Placement may be critical, to assure that the orifices will not be obstructed during the impact. Alternative designs include placing one or more smaller pipes in each compartment, and using a metallic honeycomb to absorb the energy after the pipe ruptures.

Although this design shows promise, additional analysis and experimentation must be performed to establish a practical design.

Fig. 1. Orientations of Cask at Impact for Which the Highest Probability of Damage to Cask and Contents Exists

Fig. 2. Tube Cluster Impact Limiter

Loose Packed
Tubes

$$\rho_u = \pi \rho \frac{t}{s}$$

Close Packed
Tubes

$$\rho_u = \frac{2\pi}{\sqrt{3}} \rho \frac{t}{s}$$

Hexagonal Cell
Honeycomb

$$\rho_u = \frac{8}{3} \rho \frac{t}{s}$$

ρ_u = Unit weight

ρ = Material specific weight

t = Wall thickness

s = Mean diameter or cell size

Fig. 3. Three Types of Tube Cluster Impact Limiters

Fig. 4. Typical Collapse Mode of Tube Cluster Impact Limiters

Fig. 5. Tube Cluster Impact Limiter Tube Size Selection Curves for Loose Packed Array

Fig. 6. Tube Cluster Impact Limiter Tube Weight Curves for Loose Packed Array

Fig. 7. Model of Tube Cluster Impact Limiter During Impact at an Oblique Angle

Fig. 8. Tube Cluster Impact Limiter Designed for Use on the Peach Bottom No. 1 Shipping Cask Before Attachment of Top End Plate

Fig. 9. Hydraulic-Tube Impact Limiter

SHOCK AND VIBRATION MEASUREMENTS DURING NORMAL
RAIL AND TRUCK TRANSPORT*

J. T. Foley
M. B. Gens

At the suggestion of Division of Construction, Headquarters AEC, the Union Carbide Corporation invited the Environmental Criteria Group of Sandia Laboratories to participate in a transportation test involving the transportation of a radioactive materials (RAM) cask from Paducah, Kentucky, to Oak Ridge, Tennessee, by railroad and the return to Paducah by truck. Sandia instrumented the vehicles and measured the shock and vibration inputs to the RAM cask. This paper reviews the conduct of the Sandia data experiments, describes the instrumentation, discusses the data reduction techniques used, and presents the resulting data in summary form. The summarized data provides a comparison of the intensity of the shock and vibration encountered in the rail and truck transport environments.

The main purpose of the Sandia participation was to obtain data to be used in applied research pointed towards developing descriptions of environmental inputs to cargo, and more specifically the following points:

1. To develop a more complete description of the rail transport dynamic environment than had theretofore been available.
2. To develop a description of in-transit rail transport shock as opposed to switching yard shock.
3. To provide additional information on the truck environment to supplement the results of prior work.

---

*This work was supported by the U. S. Atomic Energy Commission.

4. To provide a means of comparison between environmental levels within the two modes of transport.

The rail shipment covered approximately five-hundred miles of varied terrain.

The truck shipment was approximately three-hundred miles in length. A team of Sandia engineers and technicians planned the experiment and carried it out. Data reduction and interpretation was part of the Sandia contribution. The instrumentation system was operated remotely from the cabooses of the trains involved, but directly from the cab of the truck. Piezo-electric accelerometers were employed to measure shock and vibration.

Summaries of the reduced data as well as the methods of data reduction are discussed. Summaries of the reduced data are presented in a form which permits comparison of the environments produced by rail and truck modes of transportation at the "input to cargo" interface.

When the variable dynamic characteristics of general cargo are considered in conjunction with these environment comparisons, the differences in the two modes of transport are found not to be significant from a shock and vibration standpoint.

FOREWORD

The Oak Ridge Nuclear Safety Information Center Cask Designer's Guide (ORNL-NISC-68) indicates that no description of the vibration and shock normally incident to transport of radioactive materials is described in Annex I of AECM 0529.

This paper is the record of an attempt to quantify the nature and magnitude of the normal vibration and shock environment incident to rail and truck transport. The authors hope this paper will be useful as an estimate which can be used by designers of casks and tiedown systems, and by those who are attempting to establish test criteria for general cargo systems.

INTRODUCTION

In the spring of 1968, Sandia Laboratories Criteria and Heat Transfer Division 1543 participated in a program to measure the dynamic (shock and vibration) environment of a shipment which was being transported by rail from Paducah, Kentucky to Oak Ridge, Tennessee, and returned to Paducah via truck. The cargo was a laminated uranium Fuel Element Shipping Cask and Tiedown System which had been developed by the Nuclear Division of Union Carbide Corporation at Paducah.^(1, 2) A portion of the Union Carbide cask development effort required subjecting the cask tiedown system to shipment by rail and truck. It was suggested by the Division of Reactor Development and Construction, Headquarters, AEC, that the Environmental Criteria Group, Sandia Laboratories, had, in addition to a capability for making measurements of the dynamic shipping environment, a number of other, in-house uses for the type of data that could be obtained. As a result, an invitation to join in the test was issued by Union Carbide and accepted by Sandia.

Operation of an AEC/DOD Environmental Data Bank is a primary function of the Sandia Environmental Criteria Group.⁽³⁾ Data deposited in the Bank is obtained by various means including direct measurement. The work reported in this paper is the result of such a direct-measurement activity. In addition to the primary purpose of providing a description of the dynamic environment of the trip for the Data Bank, the following complementary objectives concerning the data were satisfied during the program.

1. Measurement of the rail and truck transport environment under identical loading conditions and with identical instrumentation.
2. Measurement of the rail transport environment using more precise techniques and sophisticated instrumentation than had been utilized in previous work.
3. Further measurement of the truck transport environment to supplement other truck environment measurement work.

TEST PLANS

Activity Assignments

Discussions with Union Carbide personnel resulted in a division of assignments in the general conduct of the transportation program. Union Carbide provided the following:

1. Cargo (cask and tiedown system).
2. Railcar and truck modified as necessary to accommodate cask and tiedown system.
3. Space to work and assistance to Sandia personnel in setting up instrumentation.
4. Arrangement with the carriers for cooperation as required to carry out the joint test program.
5. Arrangements for on-board transportation of Sandia personnel during the actual testing.
6. The packaging and shipment of instrumentation to Albuquerque at the conclusion of the program.

Sandia Laboratories: (1) Provided, installed, maintained, and operated environment-measuring instrumentation, and (2) reduced, interpreted, and reported on data obtained. Each organization separately funded the costs of its activities and personnel.

Preliminary Joint Plans

A preliminary joint plan was developed during early discussions. The first portion of the shipment would be by rail. The railcar would be a flat car belonging to Union Carbide. The cargo system would be loaded and the environment measuring instrumentation mounted, checked, and calibrated at the Union Carbide plant in Paducah, Kentucky. Movement from that point to railroad switching yards would be unattended. Shock/vibration environment encountered within switching yards, including

normal switching of the car, would be recorded. Movement from Paducah to Louisville, Kentucky would be accompanied by Sandia personnel and recordings made of shock/vibration produced by notable events enroute. Movement on the railroad from Louisville to Corbin, Kentucky, and from Corbin to Knoxville, Tennessee, would be accompanied by Sandia personnel and additional environment recordings made. Movement from Knoxville to Oak Ridge, Tennessee, would be unaccompanied.

Upon arrival at Oak Ridge, the car would be routed to the Union Carbide Y-12 plant, the cargo removed from the car and reinstalled on a flat-bed trailer. Environment measuring instrumentation used on the railroad car would also be transferred to the trailer and the cargo hauled over the road back to the Union Carbide plant at Paducah. Sandia personnel would accompany this shipment and record shock/vibration produced during truck transport events.

Sandia Test Details

Guidelines

Sandia activity in conjunction with the Union Carbide shipment consisted of determination of the kinds of events which should be measured, determining the shock and vibration parameters which we needed to measure in order to make a description of the transportation environment more complete and utilization of the capabilities of our data recording equipment to its fullest extent during the actual shipment. Several guiderules were established in the planning:

- a. Measurements taken would provide a statistical sample of shock/vibration produced by various events in each mode of transport.
  1. The entire rail trip route would be investigated before making the shipment to determine the events to be sampled in transit.
  2. Truck events measured would be according to a list of events developed in prior truck tests.

- b. As many locations on the vehicle and cargo as possible would be instrumented but the key locations evaluated would be those which described the "input to cargo" interface of the vehicle/cargo systems.

Event Sampling Determinations

The preliminary survey of the proposed rail route showed six events were significant from a shock/vibration standpoint. The events checklist used in the instrumented test was as follows:

1. Run-in/run-out shock
2. Crossing bridges
3. Traversing tunnels
4. Crossing of switches
5. Crossing of other tracks
6. Switching yard activity

In a prior, more extensive truck test program, ⁽⁴⁾ seventeen events, characterized for the most part as road types were determined to be significant from a shock or vibration standpoint. The events checklist used in this test, based upon this prior work, was as follows:

1. Accidents/Incidents
2. Unpaved Road
3. Hills (climbing)
4. Hills (descending)
5. Paved Detours
6. Overpasses
7. Concrete Interstate Roads
8. Parking Lot Maneuvers
9. Asphalt Interstate Road

10. Bumps
11. Blacktop Over Concrete, Two-Lane
12. Railroad Crossings
13. Bridges
14. Concrete Two-Lane Road
15. Blacktop Two-Lane Road

Instrumentation

The rail car used was USAX38655. The loaded car is shown in Figure 1.

The cask was positioned at the center of the car and tied down by tie rods through the floor to the car frame and to plates which had been welded to the car frame. These are shown in Figure 2. The instrumentation was placed on the aft end while the forward end of the car was occupied by auxiliary equipment relating to the cargo.

The transducers were located in positions considered to give a complete picture of the dynamic environment.

The accelerometers were located on the forward tie down plate. Five accelerometers and a pressure transducer were mounted to monitor the cask and its skid. The pressure transducer is shown in Figure 3 as is the triaxial accelerometer group on the skid floor. A vertical accelerometer was mounted on each trunnion plate as may be seen in Figure 3.

Upon reaching the terminus of the rail transport demonstration test at Oak Ridge, Tennessee, the Union Carbide cask was transferred to a flatbed tractor-trailer rig for the demonstration test return trip by truck to Paducah, Kentucky.

Figure 4 shows the completely instrumented truck. The battery power supply and recording system are located ahead of the cask. The remote control lines to the truck cab were led from the recorder into the window of the truck cab. Note the location of a large C-clamp, mounted on the fifth wheel support of the trailer, near the spare tire.

This clamp mounted a triaxial accelerometer cluster at this point. This cluster was used by the instrumentation operator in the truck cab to estimate visually the accelerations being encountered in transit.

Figure 5 shows the equipment available to the instrumentation operator in the truck cab. From left to right are:

1. Monitoring device for the accelerometers near the fifth wheel.
2. Voice microphone input to tape recorder on truck.
3. Control box for recorder system on truck.
4. Hand-held tape recorder for additional commentary, and
5. Checklist of events to be sampled.

Performance of Environment Measurement Tests

Rail Tests

The rail car containing the Union Carbide cask tiedown system and Sandia instrumentation was moved to the switching yards of the first railroad to be used. Sandia engineers met the car at the switching yards, where switching shock measurements were taken.

Through the cooperation of railroad personnel, a number of events were recorded. The procedure used was for the Sandia engineer observing the events being performed to activate the recorders by radio signal just prior to impact, and to place a code number on the tape after the event, also by radio control. At the same time, another Sandia engineer made a running commentary about the event on a hand-carried tape recorder, and also recorded the event code number. This made possible the correlation of test data with event data during data reduction efforts.

After switching yard activity was completed, the instrumented car was spotted on the trip train immediately ahead of the working caboose. The Sandia test group was housed in an additional caboose immediately ahead of the test car. The test group included the Project Test Engineer and an instrumentation technician, both from Sandia, and a travelling engineer from the participating railroad. This engineer was very well acquainted with the route. His assistance was invaluable, particularly during the night, in pinpointing locations where data measurements were to be taken.

The car was attached to three different trains during the trip. The length of the trains varied from 65 to 120 cars. On one leg of the trip, the terrain was such that run-ins were essentially absent. On other legs, even though the trains were handled with great skill, run-ins were not eliminated completely. We found that many normal events encountered in rail transport tend to occur regardless of operator skill, due to such things as train condition, train length, roadbed conditions, and the topography. The route used covered approximately 500 miles.

Truck Tests

Following installation of the cask and instrumentation on the truck, the load was returned by highway to Paducah, a distance of approximately 300 miles.

One incident measured during this truck trip was an outgrowth of tornados and heavy rains in the area during the trip. Streams and rivers were running quite high and at one location, a stream had overrun its banks at a bridge crossing. The approach to the bridge was flooded to a depth of several feet, for a distance of 50-75 yards.

Upon entering this flooded roadway, a definite "floating" effect was experienced and the side pressure of the flood waters tended to force the truck off the road bed. (An overturned tractor-trailer van which had been caught in the flood crest some time earlier could be seen about 100 yards downstream.)

From a shock/vibration standpoint, the data sample obtained during this event indicated it had no shock/vibration significance. (It might have, however, had the vehicle overturned.)

With regard to bumps encountered and crossing railroad tracks, since this test was to duplicate "normal" events, bumpy areas were traversed at slow speeds, and railroad tracks were crossed slowly after coming to a full stop. Upon completion of the normal trip, however, some special high speed runs over bumps and railroad crossings were made.

Upon arrival at the Union Carbide plant at Paducah, the cask/tiedown system was examined by Union Carbide personnel, internal cask pressure was checked, and Union Carbide determined their normal transit test had been successful.

Data Reduction

Data reduction was accomplished in five steps:

1. Composite oscillograph playback
2. Bandpass oscillograph record playback
3. Calculation of shock spectra from acceleration time histories
4. VIBRAN reduction
5. VAIL combinations

The composite playback is a real-time oscillograph analog record of the data. Playback was through a 2 kc low pass filter. The playback was examined to determine whether data had been obtained, and which data sections could be considered for further detailed reduction. The second step was to take the sections where more detailed data reduction was warranted and display them on a bandpass oscillograph playback record. Examination of bandpass records permitted the choice of still shorter time segments containing the most representative acceleration amplitudes for detailed computer analysis. Certain specific events were reduced in shock response spectrum form. These were events in which a very short burst of shock-like high amplitude was found. The other events

were reduced in VIBRAN form. This VIBRAN operation resulted in a statistical presentation of acceleration amplitudes that were present within certain frequency bands (see Figure 6). Each number shown within a frequency band represents the percentage of acceleration peaks which reached the level shown on the ordinate. The program additionally computes the RMS acceleration amplitude for each frequency band, computes the overall (Broad Band) RMS acceleration amplitude and reports the total number of peak acceleration amplitudes that were counted in the sample. As a next step, the VIBRAN records for each event sampled were combined in the VAIL program. This computer program permitted the combination of all VIBRAN events in a summary form containing the same type of information as the individual VIBRAN events records.

For this paper, the VAIL program summarized the data samples obtained from each event at the "input" to cargo location on the vehicles for each of the three axes of the vehicle. This reduces all the vibration data taken in these tests to six statistical data presentations of the vibration environment experienced in the Union Carbide shipping program.

Summary of Results

General

A brief synopsis of the measurement program results was given by Clifford ⁽²⁾ in his paper at the Second International Symposium. The data herein is an amplification which provides more detailed descriptions of the environments.

Shock

Rail Test Results. -- Rail transit imposed shocks upon the cargo from various events. The most severe shocks were found during switching and coupling activities associated with shunting cars in yards and making up trains. The shocks measured during this test (on the order of 2-5 mph)

have a very high probability of occurrence (0.95); they should be considered as happening during every rail journey. Shocks measured during this test are shown in shock response spectrum envelopes in Figure 7. Shock response spectra are, in essence, predictions of the reaction of cargo systems having various natural frequencies to the input accelerations or shocks which are being considered. In Figure 7 for instance, the lowermost rail spectrum (crossing railroad tracks and switches) indicates that a cargo system with a natural frequency of 10 Hz will respond with a maximum 1 g peak acceleration to this type of event. However, if the natural frequency of a cargo system is 100 Hz, a response approaching 6 g's may be expected. The spectrum was reduced from data taken during the crossing of railroad tracks that were laid at 90° to those on which the test car was running. Speed was approximately 45 mph. The upper rail spectrum was obtained during a normal speed coupling event with the test car coupled to the engine. The car then was coupled to another car at a speed estimated at 3 mph.

While in motion, the responses of the car to shocks induced by run-in, run-out, crossing other tracks and switches show the following characteristics:

1. Regardless of the original direction of the input shock, in the lower frequencies (1-3 Hz) the response amplitude is higher in the transverse axis than in the other axes.
2. Regardless of the original direction of the input shock, in the frequencies above 30 Hz the response amplitude is higher in the vertical axis than in the other axes.

Truck Test Results. --The events which were found to produce the highest shock damage potential in the truck transport test were encounters with bumps, dips, or potholes and crossing railroad tracks at high speeds.

In general, bumps and potholes appear to produce the most severe overall conditions for cargo systems in truck transport, but the recurrent shock event, crossing railroad tracks, appears to have a higher damage potential for cargo having natural frequencies above 15 Hz than below 15 Hz. In all events measured, the vertical axis of the truck produced the highest damage potential amplitudes.

The composite truck cargo damage potential spectra for shock are shown as Figure 8.

Vibration

Rail Tests. --The vibration records obtained in the rail portion of the test tended to reinforce the results of previous work. Earlier work with less sophisticated equipment has shown few vibration amplitudes higher than 1 g. The present work had the same results. However, in the two frequency bands in which the highest amplitudes are found, the 0-5 Hz and 5-10 Hz bands, the data cannot be considered in the same manner as the data above 10 Hz. An analysis of the amplitude distribution within these low-frequency bands shows no resemblance to vibration-like distributions. It appears that a substantial number of the peaks in these bands are the result of recurring transient impulses rather than steady state vibration. If the frequency bands from 0-10 Hz are disregarded, the vibration levels are, without exception under 0.72 g in amplitude.

The data obtained were combined into summaries by program VAIL. When combined, the total number of peaks counted in each frequency band ranged from 2500 in the low frequency band to more than 75,000 in the high-frequency band. As a result, the amplitude distribution within each band may be treated statistically. Analysis of the distributions, except in the lowest two bands indicates that the distribution resembles a Gaussian one.

The 99 percent level was calculated for each frequency band. The result is displayed in Figure 9. It is noteworthy that in all bands, the amplitudes of the peaks in the vertical axis were higher than those in the other axes. In general, the transverse axis shows higher levels than the longitudinal, although in the frequency bands between 20 Hz and 60 Hz this general order is reversed.

Truck Tests. --Vibration data for the truck portion of the test was treated in the same manner. Here again, the lower (0-10 Hz) frequency bands show a high proportion of intermittent, shock-like inputs, but the rest of the data has a distribution which resembles Gaussian random.

Figure 10 is a representation of the 99 percent level of the vibration measured. The vertical axis once again predominates. In general, the longitudinal axis is second in amplitude, although in the bands between 10 and 45 Hz, the transverse axis shows higher values. In the 60 to 87 Hz band, the longitudinal and transverse axes are equal in amplitude.

Comparisons of Truck vs Rail Transport

The comparisons that we made indicate that while the dynamic environments of rail and truck transport in this test differ in precise detail, the overall environment input to cargo differs little.

Figure 11 compares two typical shock spectra envelopes. During transit, the rail shock switching spectrum indicates there is a lower shock response damage potential for cargo having natural frequencies of 3-25 Hz than that generated by the truck when it is in transit. At 25 Hz, however, the rail shock response damage potential spectrum becomes higher than the truck spectrum and maintains that position for the rest of the record. The events from which these spectra were derived are normal and may be encountered often in the course of a trip.

If one considers, that for the most part, massive cargo such as RAM containers with their tiedowns tend to have natural frequencies from 3-60 Hz, neither mode offered a distinct advantage from a shock standpoint in this test.

Figures 12 through 14 are comparisons of rail and truck vibration in each of three axes. In both modes, the vertical axis predominates. It is evident that in many frequency bands, there is little difference. It should be reiterated that the data below 10 Hz reflect, to a large degree, recurrent, shock-like impulses for these modes of transport rather than vibration. In this area, the shock spectra referred to above provide the better comparison.

In the transverse and longitudinal axes, again there is little to choose between rail and truck transport. The levels in general are quite low, rarely higher than 0.1 g.

Conclusions

The shock and vibration environment produced in rail and truck modes of transport on an identical cargo/tiedown system were measured and quantified. Comparisons of data for each mode in this test indicate no distinct difference was offered by either mode with regard to normal environment intensities encountered.

REFERENCES

1. Stitt, D. H. and Smith, V. A., "Demonstration Fuel Element Shipping Cask from Laminated Uranium Metal-Design and Fabrication," Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, 1968, Union Carbide Corporation and USAEC, pp. 50-1520.
2. Clifford, C. B., "Demonstration Fuel Element Shipping Cask from Laminated Uranium Metal-Testing Program," Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, 1969, Union Carbide Corporation and USAEC, pp. 521-556.
3. Gens, Mark B., "The Environmental Operations Analysis Function," 1967 Proceedings, Institute of Environmental Sciences, 13th Annual Technical Meeting, Vol. I, pp. 29-38.
4. Foley, J. T., The Environment Experienced by Cargo in a Flatbed Tractor-Trailer Combination, SC-RR-66-677, Sandia Laboratories, Albuquerque, New Mexico, December 1966.
5. Gens, M. B., The Rail Transport Environment. The Journal of Environmental Sciences, 70 July/Aug., pp 14-20.
6. Foley, J. T., Gens, M. B., Shock and Vibration Measurements During Normal Rail and Truck Transport, SC-M-71-0241, Sandia Laboratories, Albuquerque, New Mexico, September 1971.

D70-1482

Figure 1. Rail Car - Instrumented

D70-1480

Figure 2. Tiedown Plate - With Instrumentation - Rail Car

D70-1478

Figure 3. Cargo Instrumentation - Rail Car

D71-3137

Figure 4. Tractor-Trailer, Instrumented

D71-3138

Figure 5. Tractor Cab - With Instrumentation

LOC-FWD TIE PLATE										AXIS=VERT		OVERALL RMS		0.15
ACCELERATION 0 TO PEAK G	PEAK DISTRIBUTION IN PERCENT													
1.400														
1.000														
0.720														
0.520														
0.370														
0.270	4.76													
0.190	1.90 15.87 0.32 0.23 0.24													
0.140	2.75 16.78 4.65 1.39 0.55													
0.100	4.65 18.59 13.41 3.01 1.03 0.21 3.24													
0.072	0.51	12.05	14.06	20.19	6.72	3.17	2.91	27.18	0.09					
0.052	0.00 0.51 3.33	12.05	13.15	17.77	13.79	9.26	5.29	36.96	0.15					
0.037	3.00 0.51	22.22	26.85	7.71	11.43	20.05	14.33	13.90	19.91	6.33				
0.027	15.00 2.02	27.78	13.53	3.85	11.63	17.50	17.97	15.27	6.17	12.13				
0.019	20.00 22.73	19.26	12.05	2.04	7.59	14.83	17.34	23.71	3.71	12.19				
0.014	20.00 25.25	11.11	5.07	1.36	5.17	9.73	12.99	15.91	1.29	14.24				
0.010	19.00 29.29	5.19	3.38	0.91	2.26	6.84	10.77	10.41	0.63	16.50				
0.000	17.00 19.19	11.11	5.71	0.91	5.17	5.91	12.35	12.79	0.90	38.36				
FREQUENCY BAND=CPS	0 5	5 10	10 20	20 30	30 45	45 60	60 87	87 125	125 175	175 250	250 350	350 475	475 625	
RMS LEVEL	0.03	0.02	0.02	0.05	0.10	0.05	0.04	0.03	0.02	0.05	0.01			
TOTAL PEAKS	100	198	270	473	441	619	863	1263	1892	2561	3272			

Figure 6. Typical VIBRAN Record

Figure 7. Shock Spectra Rail Envelope Comparison

Figure 8. Shock Response Spectra Envelopes, Truck Transport Events, High and Low Encounter Speeds, All Three Axes

Figure 9. Rail Vibration Summary - 99 Percent Level

Figure 10. Truck Vibration Summary - 99 Percent Level

Figure 11. Shock Response Spectra Comparison - Normal Transportation Events - Envelopes of Three Axes

Figure 12. Truck-Rail Vibration Summary Longitudinal Axis

Figure 13. Truck-Rail Vibration Summary Transverse Axis

Figure 14. Truck-Rail Vibration Summary Vertical Axis

UPGRADING OF PM-3A CASKS

R. A. Scaggs

ABSTRACT

Four casks were fabricated nine years ago for shipment of spent PM-1 and PM-3A fuel cores from small power reactors at Sundance, Wyoming and McMurdo Sound, Antarctica. The casks have remained in service for the latter reactor. Compliance with present U.S.* and International Atomic Energy Agency regulations for shipment was considered doubtful because of possible failure of the closure and possible loss of molten lead under hypothetical accident conditions. Upgrading of two of these casks was undertaken at the request of the AEC. The casks were restored to useful service by extending the protective shroud, installing an insulating plate under the cask, modifying the tie-down devices and lifting trunnions, and modifying the cask seal.

As part of the program to show compliance with shipping regulations, impact and leak tests were performed on a 500-lb, 1/4-scale test model. Calculated temperatures during the hypothetical fire indicated that the protective shroud will prevent melting of the lead shielding. A nuclear safety study showed that the shipment would comply as Fissile Class I, provided no water leaked into the casks. Leak tests on the model after the drop tests demonstrated the integrity of the seal.

After all modifications, the casks were hydrostatically tested at 30 psig and pneumatically leak tested at 20 psig, well above the maximum expected pressure 7 psig. Heat dissipation tests were then performed.

* Includes regulations from both the Department of Transportation and the Atomic Energy Commission.

INTRODUCTION

In 1962, four shipping casks were built and tested by the Martin Company for the purpose of shipping spent PM-1 and PM-3A fuel from Air Force and Naval reactor sites to reprocessing facilities in the United States.¹ Later, the United States and the International Atomic Energy Agency (IAEA) regulations for shipping fissile material were revised, so it became necessary to review the cask design with respect to the newer requirements. Preliminary analysis indicated that modifications to the casks would be necessary for compliance with the regulations, and that suitable modifications were feasible provided a small reduction of heat dissipation capacity and a weight increase of perhaps 1000 pounds were acceptable. It was considered that without modification, there was significant probability that under hypothetical accident conditions the closure would leak and that weld failure would occur either from impact or from pressure of molten lead. The purpose of this paper is to discuss the modifications and tests that were made to bring the casks into compliance with the current U. S. and IAEA regulations.

DESCRIPTION OF FUEL SHIPMENT

Spent PM-3A reactor cores are to be shipped dry from the power plant at McMurdo Sound, Antarctica, to the reprocessing site in the United States. The cask will be loaded underwater, lifted from the water, drained, and allowed to approach thermal equilibrium before sealing. The fuel cores have the following characteristics:

Type II - enrichment 93%, cermet, stainless steel clad pin type, 31.0 kg U-235; irradiated up to 24 MW·Yr; cooled 1 year, minimum; total weight 1600 pounds

Type IV - enrichment 9.6%, stainless steel clad pin type, 57.3 kg U-235; irradiated up to 45 MW·Yr; cooled 1 year, minimum; total weight 3500 pounds.

Shipments in the cask, after modifications, were to constitute a large quantity of Group II material (Fissile Class I, as defined by IAEA) and were to comply with applicable sections of U. S. and IAEA regulations.

DESCRIPTION OF CASK BEFORE MODIFICATIONS

The cask assembly consists of the following major components: a lead-shielded cask, 7-7/8-inch nominal thickness, (Fig. 1) into which the spent core is placed; a steel crash frame surrounding the cask to provide protection in the event of an accidental free fall; a shroud around the side; and an aluminum shipping pallet. The cavity walls consist of 1/4-inch thick stainless steel. Six stainless steel support blocks are welded to the base of the cavity to provide support for the spent core assembly. The outer cask walls consist of 1/2-inch thick stainless steel to which are welded about 100 vertical 3-inch-deep fins on approximately 1-inch centers around the circumference of the cask for heat transfer purposes. The cask cover consists of 8-1/4-inch thick lead encased in 1/4-inch thick stainless steel around the seals and bottom and 3/4-inch thick stainless steel on top. The crash frame over the cask cover is bolted to four H-beams located symmetrically around the cask circumference. Also bolted to the vertical H-beams is a 1-inch-thick shroud divided into four segments. The cask, crash frame, and shroud are mounted on a shipping pallet.

Some of the cask components were modified as discussed later. Originally, the cask, including its contents and pallet, weighed 35,500 pounds. Tie-down points located between pairs of extra heavy and enlarged fins permitted the cask to be secured both to its shipping pallet and the transport vehicle floor. A pair of trunnions facilitated handling. The cask, including the gasketed seal, was designed to withstand 60 psig. It was not vented directly to the atmosphere during normal transit; however, a relief valve on the cask was set at 30 psig, well above the maximum expected pressure.

CASK MODIFICATIONS

With respect to the more recent shipping regulations, several suspected deficiencies existed in the cask design which necessitated correction before compliance could be shown. At the time the casks were fabricated there was no requirement for maintaining fabrication records; consequently, weld inspection records were unavailable to provide assurance that the welds were of a quality that would withstand stresses generated during a 30-foot drop or from the pressure of molten lead during the hypothetical fire. Failure of a weld would have permitted loss of at least a portion of the lead shielding, and it could not be shown that the radiation dose rate after such an incident would still be within regulatory requirements.

A second suspected deficiency was the inability to maintain a leaktight closure either during the hypothetical free fall because of forces on the lid or during the hypothetical fire because of excessive temperature of the neoprene O-ring.

To provide adequate assurance of compliance with regulations, the following modifications (Fig. 2) were made to two casks. The modifications added 1800 pounds to the original weight of each.

Protection Against Fire

The existing shroud was extended upward by 15 inches, an insulating plate was installed on the bottom of the cask, and openings around the shroud and crash frame were covered with louvers to protect the cask from direct thermal radiation in the hypothetical fire.

Lid Fastening and Closure

The number of lid bolts was doubled (from 12 to 24) to reduce the potential lid distortion during a 30-foot drop. The existing neoprene O-ring seal was replaced by a "Gaskoseal" [Parker Seal Co. trademark] unit which accommodates more lateral motion between lid and body; a component part of this unit is a "Viton" fluoroelastomer [Du Pont trademark] seal which has a higher service temperature than does neoprene.

Tie-down

Brackets for attachment of wire rope were installed on the crash frame above the cask. The brackets are of sufficient size and number to meet the 2-5-10 g-force requirements of the regulations (Fig. 3).

Trunnions

The existing 2-inch diameter trunnions were replaced with 3-inch diameter trunnions, reinforced to prevent serious damage to the inner shell of the cask in the event of a 30-foot drop on a trunnion.

Stress analysis calculations were made on the lifting and tie-down devices after modification to show compliance with regulations. Stresses in the modified lifting trunnion due to three times the cask weight were found to be less than the yield strength. For the tie-down devices, no stress greater than yield would result from a static force with longitudinal, transverse, and vertical components of 10-g, 5-g, and 2-g, respectively.

TESTING

30-Foot Drop Test

Regulations specify that the package must withstand a 30-foot drop on an essentially unyielding surface without significant reduction in shielding or excessive loss of its radioactive contents, and it must remain subcritical. In order to prevent melting and possible escape of lead, it was considered necessary that the protective thermal shroud, including the new extension discussed above, stay intact during impact. A nuclear criticality study of PM-3A cores II and IV in the cask indicated that Fissile Class I criteria are satisfied for both types of cores provided they are shipped dry and provided the cask shroud is in place. Consequently, a drop test was performed to show that the modified design met these criteria.

A 500-lb, 1/4-scale model of the cask was built for this purpose. The model (Figs. 4 and 5) was a replica of the actual cask except that:

- o Cooling fins on the cask shell and various louvers around the thermal shroud were deleted, thereby subjecting the model to more severe damage than would otherwise have occurred.
- o Trunnions and tie-down devices were deleted. Calculations indicated that impact upon a trunnion would not puncture the outer shell.
- o The shipping pallet was deleted. For the actual cask, the protrusion of the shipping pallet relative to the cask protects the cask from impact at its most vulnerable areas, i.e., the cylindrical shroud plates and the bottom plate.
- o Cask contents were simulated by a solid, cylindrical weight having 1/32-inch clearances radially and longitudinally.
- o It was considered appropriate for leak testing purposes to use a full-scale cross section of the gasket in the model.
- o The upper shroud extension was a solid piece instead of two plates separated by an air gap.
- o The shroud was fabricated from four pieces instead of eight. The greater weight and rigidity of the shroud arrangement for the model were considered to be conservative, in that they would tend to make the model more susceptible to failure.

The initial drop on the top corner of the crash frame resulted in negligible damage and the cover remained sealed. To obtain maximum impact energy, the next drop was at an impact angle such that the center of gravity of the package was directly above the impact point on the bottom plate. Portions of the bottom plate and the bottom flange of the cask were permanently deformed. Several bottom-plate attachment bolts were broken, but the bottom plate remained in place; thus, the package survived the drop in a manner which satisfied the regulations.

Two drop tests on the side of the cask showed that further modifications were necessary to keep the thermal shroud and bottom plate intact. The shroud was then reinforced with wire rope and flexible clip plates and the diameter of the bottom plate was reduced. In the second of these side drops the shroud remained intact but the bottom plate came off due to failure of the attachment bolts; therefore, for the actual casks the ratio of bolt strength to plate strength was increased by more than 12 times. Based upon a stress analysis, it was concluded that the modified bottom plate would remain attached to the cask after being subjected to the impact of a 30-foot drop. Upon completion of the drop tests, immersion of the cask model in water and pressurization

with air to 30 psig for 30 minutes showed that the seals had not failed upon impact. It was concluded that no water would have entered the real cask.

Thermal Analyses

The casks had been fabricated about eight years before the present regulations came into effect. It was not considered feasible to demonstrate that weld failure and loss of lead would not occur from impact or from expansion of lead upon melting. Many of the modifications mentioned above were made to shade the cask from direct thermal radiation (and thus prevent melting of lead in the fire) while still permitting air circulation and removal of heat in normal transport.

Demonstration of compliance with requirements relating to loss of shielding, loss of coolant, and escape of radioactive material under hypothetical fire conditions was based on calculations which show that: (1) the lead remains well below its melting point, (2) the lid closure does not reach the maximum acceptable operating temperature for the "Viton" seal, and (3) the maximum pressure will be well under the leak-test pressure.

To determine the maximum temperatures in the cask during the hypothetical thermal accident, a thermal analysis was performed by Battelle Memorial Institute by means of the THT-D heat transfer code. It was assumed that:

- o The cask contents produce decay heat of 2.5 kw.
- o Before the fire the cask system is at a steady state corresponding to ambient air of 100°F, with normal circulation of air and some chimney effect in the annular space between the cask wall and the external shroud.
- o During the 30-minute fire all heat input is radiant heat to the exposed external surface, with no chimney effect by with convective as well as radiant heat transfer between the shroud and the cask wall.
- o After the fire cooling takes place by normal processes of radiation, convection, and conduction, with chimney effect and with 100°F ambient air. (The effect of the absence of chimney effect during the cooling period is also discussed.)

Some approximations and simplifications were incorporated in the mathematical model to permit computer analysis with a reasonable number of nodes, and to accommodate certain restrictions of the code. The maximum calculated temperature for any point in the lead during or after the fire is 341°F, occurring at the bottom outside corner of the lead region. The average lead temperature is in the range of 220 to 240°F at the end of the fire, and reaches a maximum of about 300°F one hour after the fire. The melting point of lead is 621°F. The maximum temperature of the "Gaskoseal" assembly is about 300°F which is well below the recommended maximum continuous temperature of 400°F for "Viton."

Although the regulations require accounting for only radiant heat input, chimney effect could be important in an actual fire when the cask is surrounded by a thermal radiation shield. In this case, as an added safety feature, intumescent paint (swells with heating) was added to the bottom louvers of the shroud to substantially eliminate chimney effect during and after the fire.

Heat dissipation tests were performed on both shrouded casks using dummy spent fuel loads (2.5 kw heat input) to show how the modifications changed the heat dissipation characteristics of the casks. Thermocouples measured the temperature drop across the cask wall as thermal equilibrium was established. These results, aside from the primary purpose of assuring that the cask would dissipate the required amount of heat during transport, provided a basis for slight adjustment of the previous computer calculations of temperatures both in normal transport and during the hypothetical fire. No significant difference exists between the two casks, and the maximum temperature difference between their cavity surfaces and ambient air was 88°F for an internal heat load of 2.5 kw, assuming no external heat source. The 28°F excess over the temperature estimated in the computer study was concluded to be not significant, and was attributed partly to air-flow restrictions of the cask modifications and to thermal resistance at the lead-steel interfaces.

Based on results of the calculations described above and the heat dissipation tests, it was concluded that exposure of the cask to the hypothetical fire produces no results which violate the applicable regulations.

Leak Testing

After all modifications were made, the casks were hydrostatically tested at 30 psig and pneumatically leak tested at 20 psig, well above the maximum expected pressure of 7 psig during the hypothetical thermal accident. The sensitivity of the leak test was considered to be about 1×10^{-4} atm cc/sec.

DISPOSITION

Test results were assembled in a Safety Summary Report² which was submitted as part of the application for a shipping license, and the licensed casks were returned to the Naval Facilities Engineering Command for use.

REFERENCES

1. W. P. Haass (Ed.), Hazards Summary Evaluation - Spent Fuel Shipping Cask for PM-1 and PM-3A Nuclear Power Plants, The Martin Company, Nuclear Division, Baltimore, Md., MND-M-1856 (May 1961).
2. J. W. Langhaar and R. A. Scaggs, Safety Summary Report - PM-3A-M Cask, F. I. du Pont de Nemours & Co., DPSPU 71-124-3 (June 1971).

FIGURE 1. PM-3A CASK BEFORE MODIFICATIONS. (Not shown crash frame, thermal shroud, shipping pallet.)

FIGURE 2. PM-3A CASK ASSEMBLY AFTER MODIFICATIONS

FIGURE 3.

FIGURE 4.

FIGURE 5.

FUTURE TRENDS IN AUTOMATED FREIGHT HANDLING SYSTEMS

J. C. Wolff

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

QUALITY CONTROL OF SHIPPING CASK FABRICATION

Allen Short

ABSTRACT

Vitro Hanford Engineering Services recently redesigned and inspected the modification of a cask (HAPO-1C-1) for shipping cesium-137. Safety requirements involving structural and containment integrity were understandably high. To achieve this conformance to stringent safety criteria, a carefully devised and administered quality-control program was prepared and implemented.

Because of the complex structural configuration of the cask, a detailed fabrication plan was prepared to insure that each component was correctly installed in the proper sequence. This fabrication plan provided the framework for the ensuing quality-control program. Every requirement for conformance to codes, specifications, and drawings was carefully listed for each step in the preparation of every component and for the subsequent assembly of components to complete the fabrication. Thus, a "control" document evolved that was completely comprehensive and chronological in arrangement.

An adequate force of carefully instructed and experienced inspectors maintained constant surveillance over every facet of fabrication and assembly.

Accidents involving shipping containers or casks filled with radioactive material are certainly to be avoided, especially situations where the containment integrity of casks are breached. Happily, breaching of such containers has not occurred in the US.

Stringent safety factors are incorporated into shipping containers to positively insure their integrity if exposed to the violent forces of a transportation accident. One effective and direct method of exercising quality control over container fabrication to assure safety in case of such an accident is the subject of this paper.

Hanford Engineering Services (HES) recently accepted the responsibility for the design of an insert to be used in conjunction with an existing isotopes shipping cask. The new insert was to be used for cesium-137 shipment. The insert as positioned in the cask is shown in Figures 1 and 2.

Seven drawings were required for the detail design of the insert; it consisted of 229 parts and was basically fabricated of AISI Type 304L stainless steel. The predominant assembly method was inert-gas, shielded tungsten arc welding. Removable covers and blind flanges were held in place by threaded members. Safe shielding of the highly radioactive cesium-137 was achieved using 5192 lbs of poured lead.

Required quality levels for fabrication had been established in the cask insert design drawings and referenced specifications. Determining the verification of highest fabrication quality levels become the concern of HES inspection personnel.

During design of the insert (See Figure 3), it became apparent that a combined quality control and assembly sequence program was imperative. A system of quality control "checks" arranged in assembly sequence plus 100 percent inspection surveillance was settled on as the most effective approach.

The contracting organization, for whom the cask insert was fabricated, required a final document at the completion of the job attesting to the complete job conformance. A "check-list" type of quality-control document filled this requirement.

The fabricating contractor elected to work on the insert on a 24-hour per day basis to meet extremely tight completion schedules. Several inspectors were thus required to maintain the degree of surveillance that was considered necessary. The use of more than one inspector introduced the necessity of standardizing the inspection and quality control. It also required that there be an uninterrupted transition at shift changes, and that each inspector establish and support his own individual responsibility.

The objectives of the inspection procedure were to:

- Provide an inspection instrument that would insure the attainment of the highest quality levels in the completed fabrication.
- Emphasize through high specificity of detail the detection and prevention of defects and deviations as soon as possible in the fabrication process, thus minimizing corrective efforts.

- Provide permanent documentation of the high level of quality control exercised during the fabrication of the shipping cask insert.

A study of the fabrication specification from the perspective of the cask insert resulted in the arrangement of a "pre-fabrication" check list. It was mainly a brief outline form of the specification requirements that required constant observance.

The specification requirements for the cask insert were not uncommonly severe. There were no unusually difficult fabrication problems anticipated, and non-destructive testing was limited to dye penetrant examination of the initial root weld pass. All other inspection was visual. It was strongly felt that with visual examination being the predominant inspection implement to be used that a check list should be written in enough detail that every advantage of such inspection could be exploited.

As part of the design package, an assembly-sequence document (HWS-8417) was provided. It described an orderly sequence for assembling the components. This "assembly sequence" served as the foundation upon which the check-lists of the quality-control procedure were structured. Also, a "pre-fabrication" check list was developed (See Attachments A-1 and 2). This list is basically a brief outline form of specifications.

A thorough study of the cask insert design was a prerequisite to starting the check list. Next a drawing was selected representing the assembly of a segment of the insert. A reference to the Assembly Sequence indicated the first part for which an individual check list should be prepared. An entirely comprehensive evaluation was then made of every possible activity that might be involved in the acceptance, preparation, treatment, and integration into a larger component of that one particular part. Those that were applicable were listed in the proper order and in the same detail that were to later mark its union with another part.

Some representative steps are shown in Attachment B. Eighty-one such steps were included in the quality-control procedure. Some of them contained as many as 33 items.

When all the inspection requirements for one part were listed, the Assembly Sequence was consulted for the next part to be considered. Then, the entire process was repeated until a check list had been developed for all the parts; and all the parts had been integrated into the completed insert. There were some instances where the check lists added additional steps to those listed

in the assembly sequence where it was felt necessary. In a few isolated cases, there were design errors that were detected as a result of the infinite detail involved in creating the check lists. In addition to the check lists, a drilling plan was created which located, dimensioned, and identified all the holes to be drilled in the upper, 1-1/2 in.-thick circular plate. It was felt that centralizing all the drilling requirements was justified by the resulting simplification.

When the check lists were completed, they were immediately sent to the principal individuals in the purchasing organization for their review and approval. In a few cases, there were revisions and corrections that were suggested and made. As soon as the check lists were approved, it was then possible to consider the beginning of fabrication.

Several meetings were arranged between the inspection personnel and representatives of the fabricator. Discussions were held regarding the extent of documentation the fabricator was expected to provide and the best methods of handling it.

Welding procedures that had previously been qualified were selected to represent welding that would be done on all types of joint geometries involved in the fabrication. Those procedures were examined by the inspection personnel and, if found acceptable, were retained to become a part of a permanent file. Specification requirements allowed the optional use of three welding processes, but virtually all the welding was done with the gas tungsten arc (TIG) process.

Some weldors who were scheduled to work on the cask insert had been qualified previously. Their qualification records were submitted to the inspectors for review. Other weldors were tested at the time of fabrication and their records submitted for review.

Material certifications for all new materials were carefully checked, the specific material identified with the correct material certifications, and the documents filed. Each material segment was examined to verify that its identity in relation to material certification was maintained. Material preparation was monitored carefully to assure the use of proper equipment such as surface grinders that had not been previously used on carbon steel. Cleanliness was emphasized, and machined and threaded parts received protection as soon as possible after preparation. The exposed edges of stainless steel, when cut by either the flux-injection method or plasma arc, were carefully ground back to expose clean base metal. Prepared weld joints were carefully checked for

conformance to the appropriate welding procedures and protective measures provided until the part was ready for fabrication.

The fabricator personnel had been given copies of the quality-control procedure for their use if they elected to use it as a guideline. Because the procedure was more specifically detailed than the assembly sequence (HWS-8417), it proved helpful as a fabrication aid. On a few occasions, it was found advisable to deviate slightly from the assembly sequence and chronology of the quality-control procedure. However, both documents were flexible enough to allow the changes to be properly noted without impairing their effectiveness. Changes were reviewed and approved.

Fabrication and assembly continued without interruption (three shifts each day, five days each week) with constant inspection surveillance exercised on all shifts. The fabricator had scheduled the work so that, when problems were encountered, the craftsmen's efforts could be shifted to other areas without lost time. The use of the quality-control procedure greatly simplified the surveillance and documentation of such work interruptions. The transfer of inspection activities from one area to another, or from one inspector to another, was performed effortlessly and without loss of continuity or application. With fabricator craftsmen and inspectors virtually working from the same document, their efforts were concurrent and harmonious.

The few problems mentioned above that were encountered were quickly referred to the project engineer for resolution. His solutions were quickly conveyed back to the fabricator and inspector. The changes were documented in the quality-control procedure and the required adjustments made in the check lists. Daily visits by the inspection supervisor verified the entry of the changes and their correctness. Changes were noted on the drawings for future "as-building"; these created only a minor problem, there being only eleven changes in the entire insert.

The resin-containment annulus of the insert was tested for containment integrity by pressurizing to 3 psig after the introduction into the annulus of a halogen gas mixture. A halogen detector (See Attachment C), sensitive to 1 part halogen gas in 250,000, was used to detect the presence of any leaks. None were found. The thermowells and cooling coils were pressurized with air and tested for leaks by painting all potential leak areas with liquid soap. No leaks were detected. All the above-mentioned tests were noted in the quality-control procedure in the appropriate places for permanent documentation. Testing of the insert and cask for radioactive shielding capabilities had been assigned previously to another organization.

Throughout the entire program, fabrication and inspection continued smoothly with few interruptions. The nature of the quality-control procedure required that fabrication, inspection, and documentation remain concurrent. Many advantages were realized as a result. There were no forgotten or overlooked items to require work retrogressions for installation. There were no frantic "last minute" recapitulations to develop a "punch-list" of neglected items. When the assembly was finished, the job was completed in its entirety with all documentation intact and in good order. There were no "loose ends."

The final document stating that "inspection of the fabrication of the HAPO-1C-1 shipping cask was done in strict accordance with the subject Quality Control procedure or as noted therein," was signed by the three participating inspectors and the supervisor (See Attachment D).

In summation, this quality-control procedure:

- Outlines the inspection activities and creates a coherent and sequential structure for the entire inspection effort.
- Provides a completely and comprehensively detailed check-off list that eliminates forgotten and overlooked items.
- Simplifies the termination and resumption of inspection activities and the transfer of responsibility from one inspector to another.
- Serves as a final check for drawing accuracy and eliminates delays and work retrogressions caused by drawing errors.
- Minimizes office record keeping time for the inspector and allows maximum time on the job site.
- Eliminates the uncertainty of the "punch-list" system.
- Simplifies documentation responsibilities.

The shipping cask described here continues to be routinely used for transporting cesium-137 across country by commercial carriers.

1 - Shirring Cast (Top View), Covers Removed

2 - Shipping Cask (Top View), Covers in Place

LIST OF ATTACHMENTS

- A - Pre-Fabrication Check List
- B - Check List for Drawing H-2-34553
- C - Halogen Testing of HAPC-1C-1 Shipping Cask
- D - Quality Control Procedure for Fabrication of HAPC-1C-1 Shipping Cask

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

Attachment A-1

PRE-FABRICATION CHECK LIST

- I. Weld procedure qualifications
  - A. Each weld joint of different geometry requires a separate procedure qualification (reference HWS-8820, sec. 6.0).
  - B. All procedures must be approved before production welding is initiated.
- II. Welder performance qualifications
  - A. Performance qualifications of all welders working on this project shall be submitted to the Title III inspector.
- III. Weld filler metal
  - A. Must conform to HWS-8067-A, except, no corrosion requirements (ref. letter, Hammond/Kirkpatrick, June 22, 1970).
  - B. Must be certified to ASTM A371-62T.
  - C. TIG welding filler rod to be identified per HWS-8067-S.
  - D. GMA welding wire to be 0.70 to 0.90 silicon.
- IV. Tungsten electrodes
  - A. Must conform to ASTM-B-297 Classification E.W. Th 2.
- V. Shielding gas
  - A. Welding grade - 99.99% pure
  - B. Argon dew point - minus 60 F
  - Mixed gases dew point - minus 50F
- VI. Grinding wheels
  - A. Control to be rigidly exercised to insure use of grinding wheels that are new or have not been used previously on carbon steel.
- VII. Wire brushes
  - A. Control to be rigidly exercised to insure use of only stainless steel brushes never previously used on materials other than stainless steel.
- VIII. Attachments
  - A. All temporary attachments, clamps, wedges, slips, welded to or mechanically fastened in contact with stainless steel components of the cask, are to be stainless steel.
- IX. Cleanliness
  - A. Base metal in and near weld area to be clean.
  - B. Weld filler metal to be kept clean.
  - C. All pipes and interior surfaces to be cleaned internally before assembly.
  - D. All pipe ends to be kept covered at all times.

SHIPPING CASK
HAPO-IC-1
W.O. #CE-0597

- X. Machined surfaces
  - A. All threaded and machined surfaces must be carefully protected from damage at all times.
  - B. Bolt and dowel holes must be kept covered at all times to prevent entry of any extraneous material.
- XI. Arc scars
  - A. Striking bars are to be provided for GTA process in lieu of arc decay devices.
  - B. Accidental arc scars must be prevented.
- XII. Residue from dye penetrant testing must be completely removed prior to welding.

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
Sheet #1

(1) Weld Pt. 15 to Pt. 3

(a)	Verify approved welding procedure for this joint.	<u>OK</u>
(b)	Verify welder performance qualification for this joint.	<u>OK</u>
(c)	Verify material identification.	<u>OK</u>
(d)	Verify that Pt. 15 is 1" long radius, sched. 80.	<u>OK</u>
(e)	Verify cleanliness of Pt. 15 and area of attachment to Pt. 3.	<u>OK</u>
(f)	Verify purge capabilities for tack welding.	<u>OK</u>
(g)	Check tack welds for oxidation and cracks.	<u>OK</u>
(h)	Check upper face of Pt. 15 and verify that it is square with longitudinal axis of Pt. 3.	<u>OK</u>
(i)	Verify that joint geometry conforms to applicable weld procedure.	<u>OK</u>
(j)	Dye check root pass per HWS-8811.	<u>OK</u>
(k)	Check inner surface of root pass and verify complete penetration.	<u>OK</u>
(l)	Check during and after completion of welding for:	

- 1 Arc scars on adjoining surfaces
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

plm-7-8-70
R.C

(2) Install Pts. 17 and 10 into Pt. 3.

(a)	Check Pt. 17 for signs of physical damage.	<u>OK</u>
(b)	Verify that Pts. 10 have been prepared correctly to match pitch of coil at 60° intervals without distortion.	<u>OK</u>
(c)	If coil is installed from bottom check carefully that deflection of outlet tube does not result in flattening, buckling, or crimping.	<u>OK</u>
(d)	If coil is installed from top, check carefully that screwing past Pt. 15 does not result in scratching, gouging, bending, kinking, or depressing any portion of the coil.	<u>NOT DONE</u>
(e)	State which method was used and fill in the appropriate space.	
(f)	Check to see that coil (Pt. 17) is placed in final position as Pts. 10 are inserted.	<u>OK</u>
(g)	Check coil for any damage from step (f).	<u>OK</u>
(h)	As Pts. 10 are placed in final position verify that coils (Pt. 17) are not deflected from required dimensions.	<u>OK</u>
(i)	Verify that the lower ends of Pts. 10 are flush with the lower edge of Pt. 3.	<u>OK</u>
(j)	Verify that welding Pts. 10 to Pt. 3 is according to approved procedure and drawing.	<u>OK</u>
(k)	Check tack welds for cracks.	<u>OK</u>
(l)	Dye check root pass per HWS-8811.	<u>OK</u>

(m) Check final welds for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Excessive concavity ✓
- 4 Excessive convexity ✓
- 5 Lack of fusion ✓
- 6 Pinholes ✓
- 7 Uniformity ✓
- 8 Undercut ✓
- 9 Weld craters. ✓
- 10 Cracks ✓

OK p Han

(3) Assemble lower portion of outlet tube below Pt. 15 (Det. V) Weld outlet tube into Pt. 3.

- (a) Verify approved weld procedure for this joint. OK
- (b) Verify welder performance qualification for this joint. OK
- (c) Verify material identification. OK
- (d) Verify 1" sched. 80 pipe and fittings. OK
- (e) Verify that two 90° elbows (Pt. 15) have been trimmed to provide required angles and dimensions. OK
- (f) Verify all dimensions and angles as noted on ref. dwg. OK
- (g) Check cleanliness of all components. OK
- (h) Check purge capabilities for tack welding. OK
- (i) Verify that joint geometry conforms to applicable weld procedure. OK
- (j) Check tack welds for oxidation and cracks. OK
- (k) Dye check root pass per HWS-8811. OK
- (l) Check during and after welding for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

p Han

- (m) Check insertion of outlet tube through Pt. 3 and verify that there is no interference between outlet tube and inner coils. OK
- (n) Check weld area for cleanliness. OK
- (o) Check tack welds for cracks. OK
- (p) Dye check root pass per HWS-8811. OK
- (q) Check final weld for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
Sheet #1

- 6 Lack of fusion
- 7 Pinholes
- 8 Uniformity
- 9 Undercut
- 10 Weld craters.

plm
7-8-70

(4) Weld inner shell bottom plate (Pt. 11) to Pt. 3.

- (a) Verify that joint geometry conforms to applicable weld procedure. OK
- (b) Verify approved weld procedure for this joints. OK
- (c) Verify weldor performance qualification for this joint. OK
- (d) Verify material identification. OK
- (e) Verify purging capabilities for inner surface of root pass. OK
- (f) Verify cleanliness of weld areas. OK
- (g) Check tack welds for cracks and oxidation. OK
- (h) Because of inaccessibility of root side of this weld, it is required that the inspector monitor the welding process in its entirety.
- (i) Dye check root pass per HWS 8811. OK
- (j) Note number of weld passes visually monitored during welding process and any remarks.

- 1st pass
- 2nd pass
- 3rd pass
- 4th pass
- 5th pass
- 6th pass

OK
OK
OK
~~X~~
~~X~~
~~X~~

(k) Check between weld passes and after completion of weld for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-9-70
plm

- (1) Verify that weld was ground flush with surface of adjoining plate. WELD FLUSH - NOT GROUND.

(5) Fabricate lower distribution ring (Det. IV) and insert through Pt. 9 (lower screen).

- (a) Verify approved weld procedure for this joint. OK
- (b) Verify weldor performance qualification for this joint. OK
- (c) Verify material identification. OK
- (d) Verify 1/2" Sch. 80 pipe and fittings. OK
- (e) Verify that bending of pipe to required radius did not cause flattening, wrinkling, buckling, or any other physical damage. OK

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
Sheet #1

(f)	Verify correct diameter for ring.	<u>OK</u>
(g)	Verify cleanliness of weld areas.	<u>OK</u>
(h)	Check purge capabilities for tack welding.	<u>OK</u>
(i)	Verify that joint geometry conforms to applicable weld procedure.	<u>OK</u>
(j)	Check tack welds for oxidation and cracks.	<u>OK</u>
(k)	Dye check root pass per HWS-8811.	<u>OK</u>
(l)	Check during and after welding for:	
	1 Arc scars on adjoining surfaces ✓	
	2 Cold lap ✓	
	3 Cracks ✓	
	4 Excessive crown ✓	
	5 Excessive width ✓	
	6 Lack of fusion ✓	
	7 Pinholes ✓	
	8 Uniformity ✓	
	9 Undercut ✓	
	10 Weld craters. ✓	<u>OK</u>
(m)	Verify that layout for drilling holes is correct (see Sec. C-C.)	<u>OK</u>
(n)	Verify that holes are .062 diameter, countersunk to remove burrs.	<u>OK</u>
(o)	Verify that all extraneous matter has been removed from interior of pipe after drilling, and that pipe interior has been cleaned per HWS-8820, sec. 13.0.	<u>OK</u>
(p)	Cover end of pipe riser from coil and insert through screen.	<u>OK</u>
(6)	Weld elbow (Pt. 14) to protruding end of outlet tube (Detail V), and weld lower distribution ring to elbow.	
(a)	Verify approved weld procedure for this joint.	<u>OK</u>
(b)	Verify weldor performance qualification for this joint.	<u>OK</u>
(c)	Verify material identification.	<u>OK</u>
(d)	Verify 1/2" 90° elbow, sch. 80.	<u>OK</u>
(e)	Check cleanliness of weld areas.	<u>OK</u>
(f)	Check purge capabilities for tack welding.	<u>OK</u>
(g)	Verify that joint geometry conforms to applicable weld procedure.	<u>OK</u>
(h)	Check tack welds for oxidation and cracks.	<u>OK</u>
(i)	Dye check root pass per HWS-8811.	<u>OK</u>
(j)	Check during and after welding for:	
	1 Arc scars on adjoining surfaces ✓	
	2 Cold lap ✓	
	3 Cracks ✓	
	4 Excessive crown ✓	
	5 Excessive width ✓	
	6 Lack of fusion ✓	
	7 Pinholes ✓	
	8 Uniformity ✓	
	9 Undercut ✓	
	10 Weld craters. ✓	<u>7.9.70</u> <u>plan</u>

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
Sheet #1

(7) Weld screen part 9 to inner shell part 3 and weld to distributor ring riser.

- (a) Verify approved welding procedure for this joint.
- (b) Verify weldor performance qualification for this joint.
- (c) Check fit, both shells, before proceeding.
- (d) Verify cleanliness of weld area.
- (e) Check tack welds for oxidation and cracks.
- (f) Check screen for level and location. Ascertain that screen is 90° to shell surface and that 50 mesh screen is face up.
- (g) Dye check root pass per HWS-8811.
- (h) Check during and after completion of welds for:

OK
OK
OK
OK
OK
OK
OK

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-9-10

plm
RC

(8) Weld part 5 (outer shell) to part 9.

- (a) Verify 1/8" minimum clearance between Pts. 11 and 24.
- (b) Verify approved welding procedure for this joint.
- (c) Verify weldor performance qualification for this joint.
- (d) Verify cleanliness of weld area.
- (e) Check tack welds for oxidation and cracks.
- (f) Check screen location, even fit, proper location in outer shell, and that outer shell is properly located to inner shell on marked axis.
- (g) Dye check root pass per HWS-8811.
- (h) Check during and after completion of welds for:

OK
OK
OK
OK
OK
OK
OK

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

plm

(9) Weld part 8 (upper screen) to inner shell (part 3) and outer shell (part 5).

- (a) Verify approved welding procedure for this joint.
- (b) Verify weldor performance qualification for this joint.
- (c) Verify cleanliness of weld area.
- (d) Check screen flatness, 50 mesh side down and properly located with outer lower angle base 1-3/16" below top of outer shell.

OK
OK
OK
OK

- (e) Check tack welds for oxidation and cracks. OK
- (f) Check sufficient room available for 1/2" sch. 80 pipe between screen and bottom of top ring (.84"). OK
- (g) Dye check root pass per HWS-8811. OK
- (h) Check during and after welding for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-10-70

plm
RC

- (10) Weld upper distribution ring (part 21) to part 2 (top ring.)

- (a) Verify approved welding procedure for this joint. OK
- (b) Verify weldor qualification for this joint. OK
- (c) Verify material identification. OK
- (d) Verify cleanliness of weld area. OK
- (e) Check tack welds for oxidation and cracks. OK
- (f) Check position and that ring is concentric. OK
- (g) Dye check root pass per HWS-8811. OK
- (h) Check during and after welding for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-10-70

plm

- (11) Weld top ring (part 2) to part 3 and 5 (inner and outer shell.)

- (a) Verify approved welding procedure for these joints. OK
- (b) Verify weldor performance qualification for these joints. OK
- (c) Check joints for proper joint preparation and fit-up cleanliness. OK
- (d) Check tack welds for oxidation and cracks. OK
- (e) Dye check root weld per HWS-8811. OK
- (f) Check during and after weld for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
Sheet #1

Attachment B-7

- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-13-70
plm

(12) Weld spider (parts 25 and 28) to top ring (2) and support bracket (10).

- (a) Verify approved weld procedure for this joint.
- (b) Verify weldor performance qualification for this joint.
- (c) Verify material identification.
- (d) Check cleanliness of areas to be welded.
- (e) Check tack welds for oxidation and cracks.
- (f) Check geometry; verify spider and support brackets (10) are in correct position.
- (g) Dye check root pass per HWS-8811.
- (h) Check during and after welding for:

OK
OK
OK
OK
OK
OK
OK
OK

- 1 Arc scars on adjoining surfaces,
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-17-70
plm
R.C.

(13) Weld bottom plate (part 24) to outer shell (part 5).

- (a) Verify approved weld procedure for this joint.
- (b) Verify weldor performance qualification for this joint.
- (c) Verify bottom plate material identification.
- (d) Verify cleanliness of area of attachment.
- (e) Verify purge capabilities for tack welding.
- (f) Check tack welds for oxidation and cracks.
- (g) Dye check root pass per HWS-8811.
- (h) Check during and after welding for:

OK
OK
OK
OK
OK
OK
OK

- 1 Arc scars ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive crown ✓
- 5 Excessive width ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-18-70
plm

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
Sheet #1

(14) Fabricate resin inlet tube (Detail I) and install.

(a)	Verify approved weld procedure for this fabrication.	<u>OK</u>
(b)	Verify weldor performance qualification for this fabrication.	<u>OK</u>
(c)	Verify material identification: Pipe is 1" sched. 80.	<u>OK</u>
(d)	Check cleanliness of Pts. 6 areas of attachment.	<u>OK</u>
(e)	Verify purge capabilities for tack welding.	<u>OK</u>
(f)	Check tack welds for oxidation and cracks.	<u>OK</u>
(g)	Check geometry of Detail I and tube being fabricated agree.	<u>OK</u>
(h)	Dye check root pass per HWS-8811.	<u>OK</u>
(i)	Check during and after completion of welding for:	

<u>1</u>	Arc scars on adjoining surfaces ✓
<u>2</u>	Cold lap ✓
<u>3</u>	Cracks ✓
<u>4</u>	Excessive concavity ✓
<u>5</u>	Excessive convexity ✓
<u>6</u>	Lack of fusion ✓
<u>7</u>	Pinholes ✓
<u>8</u>	Uniformity ✓
<u>9</u>	Undercut ✓
<u>10</u>	Weld craters. ✓

John

(15) Fabricate upper portion of outlet tube (Detail V) and weld to lower portion (installed in step 3).

(a)	Verify approved welding procedure for these joints.	<u>OK</u>
(b)	Verify weldor performance qualification for these joints.	<u>OK</u>
(c)	Verify material identification.	<u>OK</u>
(d)	Verify that material is sched. 80 and elbows 1", long radius.	<u>OK</u>
(e)	Verify cleanliness of parts to be joined.	<u>OK</u>
(f)	Verify purge capabilities for tack welding.	<u>OK</u>
(g)	Check tack welds for oxidation and cracks.	<u>OK</u>
(h)	Verify geometry of Detail V and fabrication of tubing agree.	<u>OK</u>
(i)	Dye check root pass per HWS-8811.	<u>OK</u>
(j)	Check during and after completion of welding for:	

<u>1</u>	Arc scars on adjoining surfaces ✓
<u>2</u>	Cold lap ✓
<u>3</u>	Cracks ✓
<u>4</u>	Excessive concavity ✓
<u>5</u>	Excessive convexity ✓
<u>6</u>	Lack of fusion ✓
<u>7</u>	Pinholes ✓
<u>8</u>	Uniformity ✓
<u>9</u>	Undercut ✓
<u>10</u>	Weld craters. ✓

John

(16) Fabricate inlet tubing Detail III, position and weld to upper distributing ring.

- (a) Verify approved welding procedure for these joints.
- (b) Verify weldor performance qualification for these joints.
- (c) Verify material identification.
- (d) Verify that material is 1/4" and 1" Sch. 80 pipe.
- (e) Verify cleanliness of joints to be welded.
- (f) Verify purge capabilities for tack welding.
- (g) Check geometry of Detail III; verify it corresponds with tubing.
- (h) Dye check pass per HWS-8811.
- (i) Check during and after completion of welding for:

OK
OK
OK
OK
OK
OK
OK
OK

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity -
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters, ✓

pdfm

(17) Weld thermocouple (13) well into top ring (2).

- (a) Verify approved welding procedure for this joint.
- (b) Verify weldor performance qualification for this joints.
- (c) Verify cleanliness of parts to be joined.
- (d) Check tack welds, if used, for oxidation and cracks.
- (e) Dye check root pass per HWS-8811.
- (f) Check during and after completion of welding for:

OK
OK
OK
OK
OK

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

pdfm

(18) Install outer cooling coil and weld upper tube into position.

- (a) Verify approved welding procedure for these joints.
- (b) Verify weldor performance qualifications for these joints.
- (c) Verify cleanliness of all parts to be joined.
- (d) Coil spacers (#23) installed. Welds checked for:

OK
OK
OK

- 1 Oxidation ✓
- 2 Cracks ✓
- 3 Scars on adjoining surfaces ✓
- 4 Lack of fusion ✓
- 5 Pinholes ✓

- 6 Uniformity ✓
- 7 Undercut ✓
- 8 Weld craters. ✓

OK JTB

- (e) Coil is installed evenly, with equidistant spacing throughout.
- (f) Verify area to be welded is clean.
- (g) Coil length above top ring (2) checked to ascertain required length.
- (h) Check tacks on top ring to tubing for oxidation and cracks.
- (i) Dye check root pass per HWS-8811.
- (j) Check during and after completion of welding for:

OK
OK

OK
OK
OK

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

*Coil tested 7-17-70
15th AIR + 50AP*

JTB

- (19) Form portion of coil marked "at assy" and weld inner and outer portions together.

- (a) Verify approved welding procedure for these joints.
- (b) Verify welder performance qualification for these joints.
- (c) Verify cleanliness of all parts to be joined.
- (d) Verify coils are properly aligned, spacing is proper.
- (e) Check forming of tubing that bends are free of flattening, buckling, or crimping; that formed tube 17 terminates 10-5/8" above the upper plate and is positioned as per assembly plan.
- (f) Check purge capabilities for tack welding.
- (g) Check tack welds for oxidation and cracks.
- (h) Dye check root pass per HWS-8811.
- (i) Check during and after completion of welding for:

OK
OK
OK
OK

OK
OK
OK
OK

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excess concavity ✓
- 5 Excess convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters ✓

7-17-70
JTB

(20) Weld coil support studs (#7) to outer shell (#5).

- (a) Verify approved welding procedure and method for this operation. OK
- (b) Verify weldor performance qualification for this method. OK
- (c) Verify material identification of ~~studs~~. SUPPORTS OK AS REVISED
- (d) ~~Verify that studs are type N.B.L., 5/16 dia. X 3/4" long.~~ OK AS REVISED
- (e) Check coil spacing satisfactory, check layout for points to be welded to see that they are clean and properly prepared.
- (f) Check ~~stud~~ welds for: Used 3/16" X 3/16" X 3/4" LG. 304L - SST.
  - 1 ~~Excessive flashing~~
  - 2 Lack of fusion OK
  - 3 ~~Spatter~~
  - 4 Undercut. OK

(g) Verify that all ^{SUPPORTS} ~~studs~~ are fair and true to plate surface. OK AS REV.
phm

(21) Thermocouple well (#12) and collar (#18), weld in cover (#2).

- (a) Verify approved welding procedure for these joints. OK
- (b) Verify weldor performance qualification for these joints. OK
- (c) Verify material identification. OK
- (d) Verify that tube is 1/4 O.D. X .065 wall, bent per dwg. H-2-34553, sheet 2/2, Detail VII, and ~~collar~~ is 5/8 O.D. X 5/16 I.D. X 1/8" thick ASTM-A240, 304-L-SST. COLLAR NOT USED OK AS REV.
- (e) Check tack welds for oxidation and cracks. OK
- (f) Verify geometry of tube is per assembly plan and elevation. OK
- (g) Dye check root pass per HWS-8811. OK
- (h) Check during and after completion of welding for:

- 1 Arc scars on adjoining surfaces ✓
- 2 Cold lap ✓
- 3 Cracks ✓
- 4 Excessive concavity ✓
- 5 Excessive convexity ✓
- 6 Lack of fusion ✓
- 7 Pinholes ✓
- 8 Uniformity ✓
- 9 Undercut ✓
- 10 Weld craters. ✓

7-18-70
phm

(22) Weld cask filler top (#19) to top ring (#2).

- (a) Verify approved welding procedure for these joints. OK
- (b) Verify weldor performance qualification for these joints. OK
- (c) Verify material identification part 19. OK
- (d) Verify location part 19 to part 2 per plan and rupture disk requirements. OK RSD.
- (e) Check fit up and cleanliness at weld joint. OK
- (f) Check tack welds for oxidation and cracks. OK
- (g) Dye check root pass per HWS-8811 OK
- (h) Check during and after completion of the weld for:

SHIPPING CASK
HAPO IC-1
W.O. #CE-0597

H-2-34553
sheet #1

Attachment B-12

- (h) 1 Arc scars on adjoining surfaces ✓
2 Cold lap ✓
3 Cracks ✓
4 Excessive concavity ✓
5 Excessive convexity ✓
6 Lack of fusion ✓
7 Pinholes ✓
8 Uniformity ✓
9 Undercut ✓
10 Weld craters. ✓

OK
RSP,
7-18-70

NOTES

1. HOLES "A" ARE .281 DIA. FOR THERMOCOUPLE.
2. "B" IS 1.315" FOR 1" PIPE. VERIFY O.D. PIECE OF PIPE TO BE USED PRIOR TO DRILLING.
3. "C" IS .572 DIA. THRU $1\frac{3}{16}$ " C'SK.
4. "D" $\frac{3}{16}$ DIA HOLE
5. "E" IS FOR 2x $1\frac{7}{32}$ IN. HOLES, COOLANT INLET AND OUTLET.
6. "F" ON CENTERPOINT 2.375" DIA. HOLE FOR NOM. 2" PIPE. VERIFY O.D. OF PIPE TO BE USED PRIOR TO DRILLING.
7. "G" ARE SPRUE & VENT HOLES, 2" DIA. THE $1\frac{1}{2}$ " FROM THE X-X' AXIS IS APPROXIMATE.
9. "H" IS A $1\frac{1}{32}$ " DIA. HOLE THRU LARGER FLANGE CUTS & C'SK. SEE SECTION "D-D", DWG. H-2-34589.

DRILLING DETAILS, PART 2,
DWG H-2-34589. DET SH-T. 1/2

10. "J" REPRESENTS 9 STUD HOLES, DRILL 2.4 AND TAP $\frac{3}{4}$ -10 UNC-2B. SEE DETAIL BELOW FOR LOCATING HOLES.
11. "K" REPRESENTS 3 DOWEL HOLES, DRILL $\frac{.56}{.58}$ IN DIA. BY $\frac{3}{8}$ " DEEP. SEE DETAIL BELOW FOR LOCATING.

"H" HOLE "K", DOWEL ON $1\frac{1}{2}$ " R., FOR CONNECTION LEFT OF Y-Y' AXIS AS SHOWN ABOVE 15° COUNTER CLOCKWISE, 180° AXIS; FOR CONN. $\frac{1}{4}$ " RIGHT OF Y-Y' AXIS, DOWEL R = $1\frac{1}{2}$ " 15° COUNTER CLOCKWISE FROM 90° AXIS; AND FOR CONN 5" RIGHT OF Y-Y' AXIS, DOWEL IS 15° COUNTER CLOCKWISE OF 0° AXIS

DRILLING DETAILS
PART #2, DWG H-2-34589
DETAIL SHT 2/2

July 23, 1970

Halogen Testing of HAP0-IC-1 Shipping Cask

Testing of the resin-containment annulus of the subject shipping cask was accomplished as follows:

All openings to the annulus were closed except the resin pipe. A halogen family gas mixture was then introduced into the vessel through that pipe, and the pipe closed and the internal atmosphere pressurized. All weld areas involved in resin containment were checked for leaks.

The detector was checked for sensitivity before and after testing the vessel, and was found to be operating properly at all times.

Gas used: Methylene Chloride/Ethylene Dichloride Mixture

Pressure: 3 psig

Results: No leaks

Equipment: General Electric - Type H-2 Halogen Detector
General Electric - Halogen Leak Detector Control Unit
Model - 5797934G1

A handwritten signature in cursive script, likely of the person who performed the testing or verified the results.

Q U A L I T Y   C O N T R O L   P R O C E D U R E
FOR
FABRICATION
OF
HAPO-IC-1
SHIPPING CASK

Inspection of the fabrication of the HAPO-IC-1
Shipping Cask was done in strict accordance
with the above-noted Quality Control Procedure,
or as noted within.

Shipped from J. A. Jones' shops 8-5-70.

<u>C. B. Cloud</u>	Inspector/Vitro/HES
<u>Paul H. McIntosh</u>	Inspector/Vitro/HES
<u>R. L. Smith</u>	Inspector/Vitro/HES
<u>Allen Short</u>	Supervisor/Vitro/HES

Date Aug. 10, 1970

3 - Drawing H-2-34550, Shipping Cask HAPQ-1C-1

DESIGN, FABRICATION, AND PERFORMANCE TESTING OF MODEL 160 SHIPPING CASK

C. M. Kershner

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

URANIUM CASK FOR SHIPMENT OF
FORT ST. VRAIN HTGR SPENT FUEL

C. R. Davis

The Fort St. Vrain fuel shipping cask was the first uranium-shielded containment vessel to be licensed by the USAEC for commercial transport operations. The uranium shield is fabricated of cast cylinders encased in stainless steel. The use of uranium permits the highest possible ratio of HTGR spent fuel element weight to cask weight for legal-weight truck transportation. This report presents the significant design features and engineering analyses used to satisfy the federal regulations for normal transportation and for hypothetical accident conditions.

INTRODUCTION

The Fort St. Vrain fuel shipping cask FSV-1 is designed for shipping six spent fuel elements at one time from the Fort St. Vrain high-temperature gas-cooled reactor (HTGR) to a nuclear fuel reprocessing plant.¹ The spent fuel elements are hexagonal graphite blocks containing fissile material (primarily U-233 and U-235) and fertile material (primarily thorium) (Fig. 1). The cask may also be used for the handling and transportation of radioactive reprocessed fuel.

The FSV-1 shipping cask was designed to satisfy the following general requirements:

1. All regulations of federal and state government agencies.
2. Dry loading and unloading of HTGR fuel elements by remotely operated machinery.
3. Maximum ratio of spent fuel weight to gross shipping weight limited by highway restrictions of 73,280 lb gross vehicle weight for legal-weight truck transportation.

FSV-1 SHIPPING CASK DESIGN

The FSV-1 cask employs cast depleted uranium cylinders for primary shielding in order to obtain the minimum size and shipping weight for the six HTGR fuel

LC65895

Fig. 1. HTGR Fuel Element

elements being transported. The unloaded weight of the FSV-1 has been calculated to be 43,000 lb. An equivalent weight cask made of lead shielding would carry only four HTGR fuel elements, and a cask of solid steel would transport only three*.

A basic design philosophy of the FSV-1 cask is that of "double containment." The six fuel elements are packaged in a spent fuel shipping container of stainless steel, which is sealed with a bolt-on lid containing both elastomer and metal O-rings. This container fits snugly in the shipping cask, which is sealed by a cover that also contains elastomer and metal O-rings (Fig. 2). Structural and heat transfer analyses indicate that there is no release of radioactive gas or particulates from the fuel shipping container during or after any of the hypothetical accidents described in Appendix B of 10-CFR-71.

The Fort St. Vrain shipping cask is a long, cylindrical vessel with a cavity 17-3/4 in. in diameter by 192-1/2 in. long and an outer body diameter of 28 in. The head diameter is 31 in. and the overall cask length is 208 in. (Fig. 3). The fuel shipping container cavity is 16-5/8 in. in diameter by 187-5/8 in. long, and the container body is 1/2 in. thick. The cask cover includes a depleted uranium disk 16 in. in diameter by 2-1/4 in. thick for shielding, and the cask bottom is 9-in.-thick solid stainless steel. Located in the bottom of the cask is a tube connection for helium, used in testing the seals, and a removable center plug (Fig. 4). The plug is removed for decontamination of the cask and for transfer of a fuel shipping container from the cask through the wall into the Gulf General Atomic (GGA) hot cell.

Shielding

Dose rates for gamma radiation were calculated for several points along the body and at the ends of the FSV-1 cask surface. Since the cask will be transported by a "sole use" vehicle with a barrier enclosing the cask on all sides, dose rates were also calculated at distances 6 ft from the barrier and at the rear outer surface of the truck cab.

Radiation sources for fuel that had decayed for 100 days were used as input for the calculations and are listed in Table 1. The source represents the highest fission product inventory a fuel element will have accumulated after schedule burnup in the equilibrium core. The activity was assumed to be uniformly

---

*By reducing the annual number of spent fuel shipments by one-third or one-half, the greater cost of depleted uranium (over lead or steel) is recovered in 2 to 3 years by reduced freight costs.

24 1 1/4-7 UNC CAP SCREWS = 3156
ALLOY STEEL-CHROME PLATED
WITH 1 1/4-7 HELIX-COIL INSERT

Fig. 3. FSV-1 Container and Cask Assembly

Fig. 4. FSV-1 Container and Cask, Bottom Section

Table 1. Source Data for One HTGR Fuel Element From Fort St. Vrain
Equilibrium Core with 100-Day Decay Storage

<u>Isotope</u>	<u>Curies</u>
Y-91	1.19 (+4)
Zr-95	1.36 (+4)
Nb-95m	2.7 (+2)
Nb-95	1.36 (+4)
Ru-103	2.7 (+3)
Rh-106	1.60 (+3)
Ba-137m	2.08 (+3)
Ba-140	1.86 (+2)
La-140	1.86 (+2)
Ce-144	2.52 (+4)
Pr-144	2.52 (+4)

distributed in each fuel element. The Gulf General Atomic (GGA) PATH code was used to calculate dose rates, which were then adjusted for buildup factors in a laminated shield.

It should be noted that the transplutonium isotopes Cm-242 and Cm-244, which are present in LWR spent fuel,² are not listed in Table 1 since they are almost nonexistent in HTGR fuel; therefore, neutron shielding is not required for the FSV-1 shipping cask. For a nominal shield thickness of 3-1/2 in. of depleted uranium plus 2-1/8 in. of stainless steel, dose rates of 6 to 7 mr/hr were calculated 6 ft from the barrier that surrounds the cask during transport. The dose rate on the rear outside of the truck cab was less than 2 mr/hr.

Heat Transfer Analysis

Owing to residual isotope activity, there is a continuing "after-heat" (which decays with time) in the HTGR fuel elements being shipped in the FSV-1 cask. This heat generation is predictable and has been calculated to be 2,322 Btu/hr per fuel element at 100 days decay time. Of this heat generation, 88% is realized within the fuel element and the remaining 12% is generated within the first inch of the surrounding shielding.

Steady-state and thermal analyses were conducted on thermal models of each end of the cask. The models extended far enough down the length of the cask to eliminate any end effects.³ No unexpected results were obtained from the analysis of the relatively simple cask bottom; temperatures of the cask head are summarized in Table 2. The following three conditions were used to define the ambient conditions for the analyses:

1. Maximum temperature day. This condition assumes an ambient temperature of 130°F around all parts of the cask. Solar heating has a daily average effective intensity of 96 Btu/hr-ft², excluding the effect of the personnel shield. Fuel heat generation simulating 100 and 200 days from reactor shutdown was used.
2. Minimum temperature day. This condition is for a -40°F ambient condition with no solar heating. Fuel heating of 100 days was used for comparison with Case 1.
3. Fire accident. The fire accident assumes a 1475°F fire completely surrounding the cask. A high surface heat of 300 Btu/hr-ft²-°F was set to simulate the worst possible condition of hot, blowing gases; this condition lasts for 30 minutes, after which the cask is returned to the condition of Case 1, the 130°F day.

Table 2. Summary of the Thermal Analysis

	Temperatures (°F)					Fire Acci- dent Max
	130° Ambient			-40° Ambient		
	100-Day Heat		200-Day Heat	100-Day Heat		
	Max Gaps	Min Gaps	Max Gaps	Max Gap	Min Gap	
Case seal	141	137	133	-14	-15	1220
Container seal	155	146	137	3	0	655
Fuel surface, max	278	266	200	153	143	284
Fuel centerline, max	284	272	203	160	149	291
Container wall, max	174	159	147	28	14	502
Cask inner liner, max	161	149	141	13	2	715
Shielding, center	152	146	137	3	0	829
Cask outer shell, max	146	144	134	-4	-3	1397

Temperatures were calculated by a nominal finite-difference method using GGA digital computer code RAT (acronym for radial-axial temperatures). RAT is applicable to calculating transient temperatures in a two-dimensional network of points. It may also be used to obtain steady-state solutions by extending a transient calculation to the point where time dependence of results becomes negligible. The network is specified by establishing two sets of grid lines parallel to the coordinate axis in one of the following three systems: orthogonal (X-Y), cylindrical (R-Z), or circular (R- θ).

Because of manufacturing tolerances, it is very unlikely that all parts of the fuel container and cask would ever be manufactured with all-maximum or all-minimum gaps. Nevertheless, thermal analyses were made using both extreme cases to illustrate the maximum range of temperatures that might be encountered. The centerline fuel temperature was calculated to increase from 272°F for minimum gaps to 284°F for maximum gaps, while the outer cask surface was the same, 145°F, for the 130°F ambient day.

In the thermal analyses, the minimum and maximum radial dimensions of the fuel container and cask components (inner liner, uranium shield, and outer shell) were used with appropriate coefficients of thermal expansion; thus, the correct gaps were recalculated as the cask changed in temperature. The net effect of this process was to decrease the gap sizes under normal conditions when the container and inner liner are both hotter than the outer shell. In the fire accident, however, the outer shell and shield will expand away from the inner liner, increasing the resistance of the gaps and retarding the heat flow into the fuel.

Structural Details

The use of uranium as a shielding material in place of lead in the design of the FSV-1 casks provided the opportunity for overall weight savings. In contrast to lead, it was possible to benefit from the structural strength of the uranium by correctly evaluating its energy-absorbing ability. Uranium does not permit the bulk slump and gross deformation common to lead-filled casks on the standard 30-ft free-fall drop test; however, there is definite absorption by a uranium-filled cask of comparatively large percentages of the kinetic energy generated.

Conventional stress analysis procedures are largely related to maximum developed stress levels in a static structure. The specification of a simple height-of-drop condition requires, therefore, some acceptable computational

conversion relating dynamic data to maximum stress values. It is neither necessary nor justifiable to engage in sophisticated approaches to impact analysis based on wave phenomena, especially for the relatively low velocity of 44 ft/sec involved. A simple and most satisfactory conversion formula given by authorities Goldsmith⁴ and Cristescu⁵ relates maximum stress to impact velocity and the compressive stress-strain curve of the material:

$$\rho v^2 = \epsilon_1 (p_1 - p_y)$$

where $\rho = \frac{1b/in^3}{386}$,

$$v^2 = (44 \text{ ft/sec} \times 12)^2 = 278,784 \text{ (in./sec)}^2,$$

ϵ_1 = maximum strain corresponding to p_1 ,

p_1 = initial, and maximum, stress corresponding to moment of impact, and

p_y = static compressive yield point (by 0.002 in./in. strain method).

Knowing the compressive stress-strain curve, it is a simple matter to find a point whose ϵ_1 and p_1 values satisfy the equation. Tensile curves are not satisfactory data, as they frequently differ from compression curves, especially for uranium.

Values of p_1 (maximum compressive stress) are calculated to be 90,000 psi for uranium⁶ and 53,800 psi for Type 302 stainless steel.⁴ It is notable that these are not ultimate compressive stresses, indicating that, at the velocity of impact involved, the full value of the kinetic energy per cubic inch of completely displaced material has not been reached. The phenomenon of impact is thus more like a finite time plastic deformation of a relatively large volume of material suffering only partial compression.

The stress values given above, being in the plastic zone, are distinctly greater than the static yield properties of the two materials. The obvious result is the development of correspondingly higher "g" values to be applied in the static stress analysis of the structures. This basis for analysis is used for the FSV-1 cask and is genuinely conservative.

Plastic deformation is presumed to cease when the peak stress in compression is developed. This allows acceptable configuration distortion calculations of corner impacts. In this respect, the reduced distortion associated with greater maximum stress is of benefit in preventing gross warpage of the vital sealing surfaces of the spent full container and shipping cask.

Structurally, the use of uranium provides a material which, unlike lead, not only supports itself but is also able to add strength and rigidity to the inner liner and outer shell of the cask.

Criticality

The criticality evaluation used the following assumptions:

1. The fuel elements contain the most reactive fresh fuel composition anticipated for fuel shipment, i.e., about 1.4 kg of 93.5% enriched uranium and 11.3 kg of thorium per element, or a maximum of 8.4 kg of uranium and 68 kg of thorium in the spent fuel container. The total amounts of all materials present in a fuel element as well as the homogenized atom densities (over a fuel element) are summarized in Table 3.
2. The presence of burnable poison or other neutron absorbing material, other than U-235, U-238, thorium, silicon, and graphite, is neglected.
3. The fuel is at room temperature.
4. All fission products are neglected.

These assumptions are all conservative. In general, spent fuel elements will contain considerably less fissile material. Also, all other fresh fuel element types are less reactive due to their lower uranium content and/or higher thorium content.

Two geometrical models were used to evaluate the criticality situation for the shipping cask.⁶ The one-dimensional model assumes an infinitely long cylinder. This model is adequate as long as it can be assumed that the fuel is well contained within the fuel elements.

A two-dimensional geometry model was used for the maximum criticality situation due to accident conditions, which include block breakage, internal flooding, and an accumulation of fuel particles at the bottom of the cask.

Fabrication

All steel materials for the spent fuel container and FSV-1 cask were purchased to commercial specifications such as ASTM, ASME, AISI, etc. Chemical and/or physical test reports were provided for the materials of all critical parts. Welding was performed in accordance with Section IX of the ASME Code. Detailed manufacturing procedures were prepared, outlining step-by-step

sequences for all fabrication, machining, and assembly operations. Inspection points were integrated into these procedures to provide 100% in-process and final inspection.¹

Table 3. Most Reactive Fuel Element Material Contents

	Total Amount (kg)	Homogenized Atom Density (atoms/b-cm)
Th-232	11.25	3.28×10^{-4}
U-235	1.31	3.76×10^{-5}
U-238	0.081	2.22×10^{-6}
Silicon	4.63	1.12×10^{-3}
Carbon	111.30	6.27×10^{-2}

Five uranium cylinders about 40 in. long were welded together by a proprietary method developed by the National Lead Company (NLC) to obtain a one-piece shield. The shield was inspected for integrity by gamma probing. The uranium cylinder was then sprayed completely with copper to prevent corrosion and uranium-steel eutectic formation, after which the uranium shield was assembled over the cask inner liner.

The cask inner liner was welded to the cask base prior to assembling the uranium shield cylinder in place. After the uranium cylinder was fitted to the inner liner, the outer shell was placed in position and welded to the cask base. Last to be assembled was the cask head, which was first welded to the inner liner and then to the outer shell. All gaps and voids between the steel parts and the uranium were evacuated and backfilled with helium before completion of the final welding. After welding, final machining of the cask head was accomplished to provide proper fitting of the cask cover.

All fabrication of the FSV-1 cask and containers was accomplished in accordance with the ASME Code, Section III, Class B vessels. The manufacture of the cask was followed by in-process inspections performed by personnel of the GGA Quality Assurance Department. In addition, an authorized code inspector performed all the inspections required by Article 15 of the ASME Code. Copies of material certifications, dimensional records, inspection reports, nondestructive test reports, and the manufacturers' process sheets, with the code inspector's concurrence indicated at appropriate checkpoints, have been maintained and are available for future reviews.

Prior to delivery of the spent fuel container and shipping cask, a checkout test of the seals was accomplished by pressurizing the cask cavity with helium to 50 psig and then holding this pressure without decay for 24 hr. Shielding integrity was checked by a gamma probe and scan over 100% of the cask surface, and heat transfer was checked by dissipating known quantities of electrical energy inside the fuel container and monitoring the surface temperatures of the cask inner liner and outer shell by thermocouples.

SUMMARY

The Fort St. Vrain fuel shipping cask satisfies all requirements of the USAEC and the U. S. Department of Transportation. Because depleted uranium is used for shielding, the cask can transport the maximum possible number of HTGR fuel elements by legal-weight truck. The cask is designed for dry loading by remotely operated machinery at the Fort St. Vrain reactor.

The FSV-1 cask design employs the basic philosophy of double containment, with the inner spent fuel container providing the primary containment by the use of both elastomer and metal O-ring seals. In case of a fire accident, the uranium shield does not melt, and the shield and steel inner liner and outer shell expand in such a way as to allow a minimum temperature rise of the fuel elements. While the energy absorption of uranium is not as great as lead, it does not deform as does lead when subjected to impact. In addition, its strength in compression is much greater than that of stainless steel. All materials used in the fabrication of the FSV-1 cask were closely controlled and tested to current commercial specifications. The manufacture was followed by in-process inspections performed by an ASME Code inspector, by GGA Quality Assurance inspectors, and by NLC Quality Control personnel.

ACKNOWLEDGMENT

The author wishes to express his appreciation to Messrs. W. C. Finch, R. F. Hinz, and W. A. Simon of Gulf General Atomic, and to Mr. Harold A. Backus of National Lead Company for their contributions and assistance in the preparation of this paper. This work was supported by the U. S. Atomic Energy Commission under Contract AT(04-3)-633.

REFERENCES

1. C. R. Davis et al. (GGA) and D. C. Cocciolone et al. (NLC), "Final Design Report for Fort St. Vrain Fuel Shipping Cask," GADR-55, April 17, 1969.*
  2. H. S. Bailey, R. N. Evatt, C. P. Ruiz, and G. L. Gyorey, "Neutron Shielding Problems in High Burnup Thermal Reactor Fuel Shipping," Transactions, 1969 Winter Meeting, American Nuclear Society, pp. 448-9.
  3. C. R. Davis et al. (GGA) and H. A. Backus et al. (NLC), "Final Design Report for Fort St. Vrain Fuel Shipping Cask, Supplement A," GADR-55, November 15, 1969.*
  4. W. Goldsmith, Impact, Edward Arnold, Ltd, London, 1960.
  5. N. Cristescu, Dynamic Plasticity, Interscience, New York, 1968.
  6. C. R. Davis et al. (GGA) and H. A. Backus et al. (NLC), "Final Design Report for Fort St. Vrain Fuel Shipping Cask, Supplement B," GADR-55, November 15, 1969.*
-

*These documents supported the application for AEC License for the Fort St. Vrain fuel shipping cask. They are on file in the AEC Public Information Center, 1717 H Street N.W., Washington, D.C., under Docket No. 70-1165.

SHIPPING CASKS FOR SPENT FUEL WITH HIGH BURN-UP

F. Schmiedel

R. Dietrich

This paper was not available at the time of printing. It has subsequently been issued as BNWL-SA-3906, available from NTIS, 5285 Port Royal Road, Springfield, Virginia 22151.

X "GENERAL DESCRIPTION OF EXPECTED SIZE AND CAPABILITY OF
1100 MWe REACTOR SPENT FUEL SHIPPING CASKS"

J.P. Malone

ABSTRACT

A general consideration of the design parameters for 1100 MWe reactor spent fuel shipping casks permits reasonable estimates to be made of the expected size and capabilities of future spent fuel casks for 1100 MWe reactors.

The allowable weight of the cask, the required length, and the shielding thickness are factors used to determine the expected cavity diameter. The cavity diameter and the heat transfer requirements are then employed to estimate the spent fuel capacity of the cask.

INTRODUCTION

Three modes of transportation are commonly associated with the shipment of spent fuel from nuclear reactors. They are truck, rail, and water. Truck transportation is feasible at all reactors while rail and/or water transportation are not always available. For this reason motor freight has become the most common method of spent fuel transportation. It is for this reason, also, that the electric utilities show more than a passing interest in the ability of motor freight to fulfill the spent fuel shipping requirements of the 1100 MWe light water reactors in an economical manner. The purpose of this presentation is to provide a general description of the expected size and capability of 1100 MWe reactor spent fuel shipping casks for highway use.

Cask Dimensions

By considering the design parameters, the expected size and capability of the future spent fuel casks for servicing the 1100 MWe reactors can be deduced. The first parameter to be considered is the weight. A practical limit of gross vehicle weight for road transportation is about 95,000 pounds. The weight of the tractor-trailer combination must be subtracted from the gross vehicle weight in order to determine the allowable weight of the cask and its contents. Tractor-trailer combinations weighing 20,000 pounds can be expected to be available. This is a substantial improvement over the 30-35,000 pound tractor-trailer combinations formerly required to transport a large payload. This weight reduction is largely the result of employing lightweight-highstrength materials in the construction of the trailer thereby reducing its weight to the 7,000 pound range. The 95,000 pound limit less 20,000 pounds for the tractor-trailer leaves 75,000 pounds for the cask and its contents.

Since the weight of the cask and its contents has been established the length and shielding thickness parameters may be examined in order to determine the maximum permissible cask dimensions. The heat transfer requirements of the cask must be kept in mind and will be considered in a subsequent section. The cask length will be established first.

The 1100 MWe boiling water reactor fuel is typically 171 inches long while the pressurized water reactor fuel is 166 inches in overall length. About 6 inches of cavity length are required for a shock absorber to prevent damage to the fuel during transport. Considering the boiling water reactor fuel, a cask intended for both BWR and PWR fuel should have a cavity length of approximately 177 inches.

Shielding Thickness

The next design parameter is the shielding thickness required to reduce the dose rate to an acceptable level. For estimation purposes the shielding thickness may be considered to be insensitive to number of elements in the cavity, that is, essentially a constant. In reality the required shield thickness will change

only slightly as the number of elements in the cavity changes. The thickness of lead required to reduce the dose rate from one assembly to an acceptable level will not double if it is desired to shield two assemblies. Perhaps an additional 1/2 inch of lead may be required in order to accomplish this. Preliminary calculations using the SDC code¹ were performed in order to determine the approximate thickness of lead equivalent shielding with a 1 inch inner steel shell which is necessary to reduce the gamma radiation dose rate to an acceptable level for transportation. The PHOEBE code² was used to obtain the gamma ray source strength used in the SDC calculations. An approximate operating history of a typical 1100 MWe reactor and a 150 day cooling off period were used in the PHOEBE calculation of the gamma ray source strength. The results of the calculations indicate that about 8 1/2 to 9 1/2 inches of lead equivalent shielding are required with a 1 inch inner steel shell to reduce the gamma ray dose rate to about 5 mr/hr at 8 feet from the side of the cask.

Unlike prior generation reactors the radiation dose from the 1100 MWe spent fuel is due to both neutrons and gamma rays. The significant neutron activity in 1100 MWe reactor spent fuel will be the result of the presence of transuranium elements heretofore unencountered in spent light water reactor fuel. Certain isotopes of these elements present neutron shielding problems that were not present in spent fuel from prior generation light water reactors.³ The presence of isotopes of curium is the result of attaining the high fuel exposures (burnup) that are the goal of the 1100 MWe reactors. The formation of one atom of ²⁴²Cm requires four neutron captures, while ²⁴⁴Cm requires six neutron captures. Therefore, the concentrations of these isotopes will depend, roughly on the fuel exposure to the 4th and 6th power until the concentrations approach equilibrium.⁴ These isotopes produce neutrons by spontaneous fission and (α ,N) reactions with the oxygen in the UO₂ fuel.

As indicated above, the 8 1/2 to 9 1/2 inches of lead equivalent shielding will reduce the gamma ray dose rate to about 5 mr/hr at 8 feet from the side of the cask. The balance of the allowable 10 mr/hr dose rate will be due to neutron activity. Calculations have shown that for initial cycle, UO₂, 1100 MWe reactor fuel no special neutron shielding, such as a water gap in

the lead, will be required.⁵ The above considerations indicate that the expected shielding configuration should be about a 1 inch steel inner shell, 9 inches of lead and a 1 1/2 inch thick steel outer shell.

Cask Diameter.--Knowing the weight of the cask and its contents, the length of the cask, and establishing the shield thickness then fixes the diameter of the cask cavity. A 23 inch diameter cavity is determined from this approach. This cavity size can accommodate 3 PWR assemblies or 7 BWR assemblies.

Heat Removal

Unlike the shielding thickness, the heat removal problem is compounded in direct proportion to the number of spent fuel assemblies in the cavity. The goal with respect to heat transfer is to remove the decay heat so that the temperatures of the fuel and the cask are maintained at a reasonable level. The decay heat load from 3 PWR assemblies is about 33KW while 7 BWR assemblies have a decay heat load of about 21KW. The use of aluminum fins on the outer surface of the cask seems attractive from a weight saving point of view. Aluminum fins could permit the dissipation of the decay heat without assistance from a forced cooling system. Current methods employed in joining dissimilar metals (such as aluminum and steel) result in strong joints with excellent heat transfer characteristics.

SUMMARY

Based on the above development, the casks used for highway transportation of 1100 MWe light water reactor spent fuel can be expected to be about 16 feet in overall length, weigh approximately 70,000 pounds, have a 23 inch cavity diameter and an outside diameter of about 5 feet including the fins. Such a cask could physically accommodate 3 PWR assemblies or 7 BWR assemblies and dissipate the corresponding decay heat after reasonable cooling time without the need for forced cooling.

REFERENCES

1. E.D. Arnold, B.F. Maskewitz, SDC, A Shielding-Design Calculation Code For Fuel-Handling Facilities, ORNL-3041 (March, 1966).
2. E.D. Arnold, PHOEBE - A Code For Calculating Beta and Gamma Activity and Spectra For ^{235}U Fission Products, ORNL-3931 (July, 1966).
3. J.D. Rollins, R.E. Best, "Neutron Shielding Problems Associated With the Shipment of High-Burnup LWR Fuel", Transactions of the American Nuclear Society, volume 12, no. 2, Nov. 1969.
4. H.S. Bailey, et.al., "Neutron Shielding Problems in High Burnup Thermal Reactor Fuel Shipping", Transactions of the American Nuclear Society, volume 12, no. 2, Nov. 1969.
5. E.D. Arnold, H.F. Soard, "Neutron Shielding Requirements for LMFBR and LWR Fuel Shipping Casks", Transactions of the American Nuclear Society, volume 13, no. 1, June-July, 1970.

CONTRIBUTION TO THE DESIGN OF TYPE B PACKAGES
FOR HIGH RESIDUAL POWER IRRADIATED FUELS ELEMENTS

Henri BRUEL
Camille BOCHARD

ABSTRACT

The cost of the fuel cycle and especially of the irradiated fuels transportation, deserves of the specialists attention, taking into account the large number of power stations being presently erected.

However, the LWR fuels characteristics entail new technical problems concerning the thermal evacuation and the neutron shielding of the casks.

This article summarizes the solutions praised by the ROBATEL SLPI Company, casks designers and manufacturers, to solve these problems while stressing the economical and safety aspects.

INTRODUCTION

For several years the Nuclear Equipment Department of the ROBATEL SLPI Co. has been devoted itself to study and to manufacture shielded packagings for radioactive materials. In this purpose, our "Research and Development Group" has perfected new techniques enabling to follow the evolution of the Régulations on Radioactive materials transportation, on one hand, and to meet the new problems set by the power reactors irradiated fuels elements transportation, on an industrial scale, on the other hand.

The first uncommon solution we have been using since 1964 for manufacturing type B casks consisted in incorporating between the lead shielding and the outer shell, a thin layer of an insulator, based on hydrous calcium sulfate. (U. S. Patent N° 3.466.662). With this technique, now called "Thermal switch" technique, we designed and built a lot of casks in perfect conformity with the mechanical and thermal norms on irradiated fuel elements transportation, as they have been carried on till now. To-day, 175 casks using this technique have already been manufactured in our workshops, the heaviest ones weighing 50 tons (used for Chinon and Saint-Laurent-des-Eaux E D F Power plants fuels) and releasing a 15 kW power. As a matter of fact, the hydrous calcium sulfate "thermal switch" secures a heat transfer of about 1,700 W/sq.m (540 BTU/h sq.ft)

STUDY OF TECHNICAL SOLUTIONS

1 - Present problem data

The economical considerations in connection with the power reactors development tend to impose a reduction of the dead times, that are storing and fuel transportation. This principle raises two new difficulties in solving the package problem :

- The increasing of irradiated fuel elements residual power, following to the combustion rates increase and to a short cooling time,
- The necessity of including, in addition to the classical protection against gamma rays, a neutron shielding, most often made of materials which are bad conductors of heat.

These conflicting problems have entailed to develop new perfectings, enabling to keep the packages at the highest level of safety.

The point was to design packages using materials which could in the whole serve the following fonctions :

- protection against gamma rays,
- protection against neutronic rays,
- dissipation of the thermal power emitted by the irradiated fuel elements,
- protection against the effects of the reglementary drop-tests,
- protection against the effects of the reglementary fire-tests.

We have also been guided towards the chosen materials by considerations less important, but not negligible, since they involve the economical aspects :

- weight of the package,
- easy use, i. e. manufacturing cost.

As a matter of fact, it stands to reason that a too important increase of the packages cost, consequently of the transportation cost, reduces as much the advantage to get relatively short cooling times.

2 - Gamma shielding

The materials more often used to make gamma protection are steel, lead or depleted uranium. We thought over these three possibilities, our workshops having the experience of making protections with these three metals.

From the economical point of view, the two main criteria are the material price ~~after~~ working it and setting, and the total weight of the package

Fig. 1 - Relative fluctuations of the weight of a LWR package, according to the type of the gamma protection chosen.

during transportation. As an example, the curves of Fig. 1 show the fluctuations of the LWR package weight, according to the gamma protection chosen, in relation to the corresponding package weight with a lead gamma protection. The given values correspond to gamma protections equivalent to 200 mm of lead, for gamma rays of 2 MeV.

It is to note that the percentage of weight increase is obviously more important as the package is small. Also for a 32 tons package carried by road, uranium enables to reduce the weight by 12 % compared with lead and, on the reverse, a steel package is 22 % heavier compared with a lead package.

For heavy packages carried by railways that we designed (85 tons) there are not so many differences, for uranium enables to decrease the weight only by 6 %, and steel increasing it by 12 %.

Those figures compared to the three protection types real costs have led us to choose lead. As a matter of fact, uranium high cost and its difficult processing are far from balancing the advantages of weight it gives ; on the reverse, the noticeable loss in weight due to steel is not balanced by a sensible manufacturing cost decrease.

3 - Neutron shielding

3 - 1. Material. - - In 1967, together with the French Commission to the Atomic Energy (C.E.A.), we perfected a material absorbing neutrons (U.S. Patent Pending N° 716 330) mainly composed of polyethylene, boron and hydrous calcium sulfate.

This material, particularly well adapted for the cask problem, shows the following advantages :

- neutron efficiency equivalent to that of massive polyethylene.
- easy processing enabling to fit the most complicated forms, hence a very interesting price.
- swelling when setting and good mechanical holding, which secures the gamma shielding a perfect conservation with respect to the steel resisting structures.
- interesting thermal insulation in case of an accidental fire.

The reglementary drop and fire tests performed on a small gamma neutrons model have shown that the neutron absorbing compound was burnt only on 10 mm thick and that the temperature of the lead shielding did not exceed 101° C (214° F).

3 - 2. Heat transfer. - - So as to get the minimum dead weight, this light material absorbing neutrons has been placed outside the gamma shielding. But as its good insulating properties did not enable to dissipate satisfactorily the heat spread by the fuel elements, we have placed good conductor metal structures inside this neutron absorbing protection.

Forms and spots have been judiciously chosen, which enables us to reach apparent thermal conduction coefficients for the whole protection around 36 kcal/h m °C (24 BTU/hr sq.ft. (°F/ft)).

4 - Thermal switch

Regarding the protection against fire, we keep on using our thermal switch technique, which shows many advantages. The aims of such a process are, let us remind of it :

- a) To make the transfer of heat towards outside easier, by placing between the gamma shielding and the outer steel shell, a product securing an excellent contact between these two parts (avoiding the air gaps) and presenting a good conduction coefficient.
- b) In case of fire, to block the heat penetration towards the inside of the package, in order to avoid a too important rise in temperature.
- c) After the fire, to ensure the evacuation of the heat spread by the transported products, and therefore the material must keep an acceptable coefficient of conduction after accident.

These purposes, apparently incompatible, are reached with our new "thermal switch" made from alumina (U.S. Patent Pending N° 59 037). We call it "high flux thermal switch", as it allows a heat transfer of about 4,500 W/sq.m (1,430 BTU/hr.sq.ft.).

The "high flux thermal switch" is essentially composed of alumina trihydrate and hydrated iron oxides ; it is in a dry state, but yet contains a large quantity of water under combined form (conduction coefficient around 2,5 kcal/h m °C ; 1.7 BTU/hr. sq.ft. (°F/ft)). The releasing temperatures being relatively high, this thermal switch may be used under normal conditions at a temperature of 160° C (320° F).

The swelling occurring when the material hardens ensures an excellent contact between the various shieldings.

In case of fire, the water contained in the material steams off through safety outlets, what limitates the rise in temperature. Even when it is free from its water, this material keeps a perfect mechanical holding and a conduction coef-

- Test in natural convection -

- Test in forced convection -

Fig. 2 - THERMAL MODEL

ficient high enough to ensure the heat transfer, after the fire, under normal safety conditions. This conduction coefficient is about 1.5 kcal/h m°C (1 BTU/hr.sq.ft. (°F/ft)).

5 - Forced cooling and ejectable envelope

Most of the forced cooling systems are doubtful owing to their vulnerability in case of accident. We think that this situation increases heavily the cost of the fuel cycle because it would be necessary either to increase the cooling time of the fuel elements, or to cut down their quantities per package, and, therefore, the ratio effective load/total transported load.

That is the reason why we have designed a forced cooling system which facilitates the handlings and the transportation of large packages spreading an important thermal power, while granting the transportations safety under any circumstances.

This system called "ejectable envelope" (U.S. Patent Pending) is composed of :

- a) a fan driven either by an electric motor, or by an autonomous unit with, in any case, the usual means to prevent eventual break-down (reserve motor, sound or light giving signals),
- b) an envelope covering the package, under which the air is blown by the fan, in order to increase the efficiency of the cooling elements such as fins or preferably studs.

In case the air circulation is stopped, whatever the cause may be, and supposing that all the safety devices of the ventilation unit did not work, the removable envelope is ejected and the package is cooled down by natural convection in the open air, without infringing the package conformity with the reglementary tests.

Thus, this forced cooling system allows - under normal conditions - to dispose of a relatively cold package that facilitates the handling and reduces the two following dead times :

- The time for reaching thermal equilibrium before shipment of the package,
- The required cooling time before immersing the package into the pool for unloading.

6 - Test model

On Fig. 2 is shown the photograph of a model used for thermal tests which enabled us to check the values defined during the preliminary tests and also the calculation assumptions leading us to design the real packages.

Fig. 3 - Repartition of the temperature in the thermal model.

This model represents a package section and includes an inner cavity, a gamma lead shielding, the high flux thermal switch, a neutron shielding, and, on the outer shell, 1,460 cooling studs 10 mm in diameter, 100 mm long.

As a matter of fact, we have preferred studs to fins for the three following reasons :

- a) the use of studs allows a weight reduction of 30 % with respect to the weight of the fins for the same cooling efficiency,
- b) far better efficiency by forced ventilation,
- c) possibility to ensure the proper cooling of the package, whatever its position may be, horizontal or vertical.

The second photograph of Fig. 2 shows the model equipped with its cooling envelope, the air being blown at one end and discharged when hot at the other end.

The thermal tests results are given by curves on Figure 3.

THE NEW GENERATION OF PACKAGES

Most of the economical studies dealing with irradiated fuels transportation prove the advantage of using large casks for reducing shipping and handlings costs (for instance, the article published by "Nuclear News" on July 1970 and written by C. W. Smith and W.B. Bigge).

This is true for fast reactors as well as for light Water Reactors (see economical studies made by M. M. LABROUSSE, French Atomic Energy Commission).

We only give hereunder two representative figures :

- a package of 32 tons contains about 900 kg of uranium,
- a package of 85 tons contains about 4,100 kg of uranium.

These figures show that the performances of the larger package are superior by 70 % to those of 32 tons package. To this advantage are to be added the lower tariffs and the better administrative facilities for railway transportation with respect to overweight truck shipment.

That is the reason why we have designed, by application of the above described techniques, a 85 tons package which will permit the transportation of fuel elements under economic conditions.

The dimensions of this package are the following :

- . internal diameter : 1,100 mm
- . internal height : 4,550 mm

Fig. 4 - Repartition of the temperatures in a 85 tons package of 100 kW.

- . over-all diameter : 2,300 mm
- . over-all length : 5,800 mm

The thermal principal characteristics of such a package, for a surrounding temperature of 38° C (100 F), are :

- Capacity
  - . 9 PWR elements or 20 BWR elements
  - . weight of transported uranium : 4,100 kg
  - . total power : 100 kW
- Ventilation : 20,000 m³/h (26,000 cubic yards/hr)
- Package outer shell temperature : 64° C (147° F)
- Inner temperature : 146° C (295° F)
- Relative pressure : 3.5 bars (50 p.s.i.)

On Fig. 4, we have summarized the temperatures distribution in the thickness of the various materials composing the package in the following cases :

- a) curve (a) shows the equilibrium temperatures under normal conditions, i. e. with forced cooling.
- b) in case the forced ventilation is stopped, the temperature of the package outer shell progressively rises up and reaches, after two hours, the removable envelope ejection temperature (140° C) (284° F) ; then, the inner temperature is 157° C (315° F) (curve b).
- c) after ejection, the temperatures aim towards equilibrium temperatures by natural cooling (curve c). Under these conditions, the package outer shell is at 114° C (237° F), the insulator maximum temperature is 154° C (310° F), the inner temperature 196° C (385° F) and the steam pressure about 15 bars (213 p.s.i.).
- d) Under the conditions of the reglementary fire, the temperatures evolution has been calculated by the graphical method of Binder and Schmidt.

We have taken an absorption coefficient equal to 1 to take into account the studs being, what practically amounts to double the thermal flux penetrating into the package, compared with the regular prescriptions.

Curves (d₁), (d₂) and (d₃) respectively show the temperatures reached inside the package 30 minutes after the starting of the fire (extinction), then 8 minutes and, at last, 30 minutes after extinction of the fire. It can be seen that the inner temperature does not rise above the equilibrium temperature by natural cooling, i. e. 196° C (385° F).

CONCLUSION

Our purpose was to demonstrate that, thanks to the above described solutions, the package users and carriers will find, in the near future, perfectly adapted packages, which will lead to more economical fuel cycle.

This is possible thanks to the improvements brought to the package materials performances, for, only a few years ago, the technical solutions allowing the conception of large capacity packagings did not appear clearly at all.

It is also to be noted that these characteristics are not obtained to the detriment of safety, which remained our main worry all along our researches.

In our newly erected plant of GENAS, near LYONS (France), we have moreover provided for installations, testing pool and lifting capacities up to 120 tons. Which enable us, not only to manufacture the packages designed by our Engineering Department, but also to test them carefully before delivery under close to normal operation.

AN LMFBR FUEL SHIPPING CASK CONCEPT AND OBSERVATIONS
PERTAINING TO HAZARD SURVIVAL POTENTIAL*

A. R. Irvine and L. B. Shappert

Oak Ridge National Laboratory
Oak Ridge, Tennessee

ABSTRACT

Shipment of irradiated fuel from liquid-metal-cooled fast breeder reactors (LMFBR) is expected to be accomplished after a short decay period. The existence of gross amounts of dispersible radioiodine and radioactive rare gases at short decay times makes the ability of the shipping cask to survive accidental hazards a matter of prime concern.

The favored design for an LMFBR fuel shipping cask employs steel for both gamma radiation shielding and the cask structure. Heat is transported within the sealed fuel cavity by thermal convection in a sodium bath that surrounds multiple fuel subassemblies. Cask seals are designed to minimize the probability of damage due to impact, and to be capable of leak-free performance at elevated temperature and pressure. The fuel subassemblies are supported within the fuel cavity by means of a rotatable magazine such that they can be placed in and removed from the cask through a relatively few small ports. Cask temperature under normal conditions is maintained, by a special cooling system, at a moderate level in order to optimize cask material strength and ductility, as well as to increase the quantity of heat required to elevate the cask temperature to the maximum permissible level. Upon loss of secondary cooling, the cask temperature would rise to a sufficient level to dissipate the decay heat to the atmosphere. Additional protection against damage due to impact is provided by an array of energy-absorbing buffers located in the vicinity of the seals and at other strategic points.

Results of impact tests with various scale model casks, heat transport experiments with liquid coolants, and calculations of thermal transients indicate that a rail shipping system which can resist credible impact and fire hazards can be designed and constructed at reasonable cost.

---

* Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

INTRODUCTION

Shipment of spent fuel from liquid-metal-cooled fast breeder reactors (LMFBR's) to central reprocessing plants at relatively short times after removal from the reactor will be economically desirable when processes have been developed to accommodate short-cooled fuel, a substantial number of LMFBR's are in operation, and plutonium is not in oversupply. Development work is proceeding with some success on processes to accommodate short-decay fuels,¹ and a successful demonstration of a large-scale LMFBR can be expected to bring about the latter two conditions.

LMFBR spent fuel at short decay periods will contain gross amounts of radioiodine and radioactive rare gases in a dispersible form. Therefore, the container in which these materials are shipped must be such that the probability of dispersal of these materials is minimal under both normal and accident conditions. This paper is concerned with the potential for hazard survival of a cask designed specifically for LMFBR fuel transport by rail. The hazards of greatest concern are impact and fire.

CASK CONCEPT

A concept of a cask for LMFBR fuel shipment is shown in Fig. 1. The principal features are as follows:

1. Steel structure and gamma shielding.
2. Small seals (as compared to cask diameter) isolated from cask structure.
3. Fuel supported by a rugged structure to maintain its position under impact conditions.
4. Liquid primary coolant plus a volume-expansion system that will limit rate of deceleration (and hydrostatic pressure) of coolant.
5. Energy absorbers placed around vulnerable parts to reduce probability of damage.
6. Auxiliary cooling system for maintaining normal temperature of cask at relatively low temperature.
7. Thermal shields for reduction of heat uptake during a fire.

Fig. 1. An LMFBR Shipping Cask Concept.

HAZARD SURVIVAL POTENTIAL

Impact tests have been made with models of a cask similar to that described in Fig. 1. These have shown no substantial damage to containment capability of the models when subjected to a 30-ft free-fall impact against an unyielding surface.²

A total of 17 different model casks of five different scales were tested. All except one of these model casks were unprotected by any form of energy absorber. Four of the units were equipped with seals which were mechanically isolated from the cask structure. The damage to the unprotected cask structures was of a superficial nature, with no perceptible change in the dimensions of the fuel cavity. Of the four units equipped with seals, the initial negligible seal leak rates were increased somewhat after the impact tests. However, there was no indication of damage to the sealing surface. Each of these four units was impacted several times; in one case the unit was impacted twice on the same point. In no instance did unacceptable damage occur.

Tests were made with an energy absorber which demonstrated the capacity of a properly designed energy absorber to almost completely prevent cask structure deformation. A 1/8-scale model cask was dropped nine times, with various sized tube-in-tube energy absorbers attached to the impacting surface, without causing any deformation of the cask structure beyond rounding of a sharp edge.

Only cask models having relatively small diameter seals have been tested. Seals of full cask diameter, in addition to being more difficult to seal to a minimal leak rate, would be expected to be more difficult to protect from impact damage. To permit use of small-diameter seals, the fuel support system must be capable of motion to render a number of fuel receptacles accessible to one opening. This fuel support system must be designed to hold each subassembly securely and to prevent their being more closely assembled. Further, the support system must be open enough to allow relatively free circulation of coolant for uniform temperature distribution throughout the cask fuel cavity. No device for this purpose has been designed, constructed, or tested, but no reason is immediately apparent to indicate that the requirements cannot be met.

A liquid coolant, in particular sodium, has been demonstrated to be exceedingly effective in distributing heat throughout the cask fuel cavity. However, free space must be provided within the fuel cavity to allow expansion of liquid coolant when temperature increases. This expansion space can also serve to minimize hydrostatic pressures developed when the cask is decelerated. An unbuffered steel cask striking an unyielding object flat

on an end would come to rest in about 1 in. A properly designed expansion space could increase this stopping distance for the liquid coolant by a factor of > 10 , thus reducing the hydrostatic pressure to easily manageable proportions. Although this pressure will be of short duration, the seal and cask structure will be capable of withstanding such pressure continuously.

As all fires are of finite duration, low rates of rise in cask temperature due to exposure to fire are desirable to increase the probability that the fire will be extinguished before a damaging condition is reached. Using an auxiliary cooling system to maintain the cask at temperatures only moderately above ambient will be beneficial in this respect as well as in making cask handling easier at the reprocessing site. Use of steel shielding, instead of lead or uranium, will increase the heat capacity of the cask as a result of both increased mass and the approximately fourfold greater heat capacity of iron over lead and uranium. Figure 2 shows calculated temperatures³ for casks shielded with uranium and steel when exposed to 30-min and 4-hr fires. The higher heat capacity of steel can be seen to be a definite advantage in a longer fire. Other means may be employed to either increase the cask heat capacity or reduce its uptake of heat from a fire, or both. The cask neutron shield might be water that would boil away to absorb much heat. Also, insulation conceivably could be placed around the cask, which would remove itself when the temperature of the cask reaches the maximum acceptable level. Figure 3 shows the calculated change in temperature of a steel-shielded cask³ when it is insulated in such a way that the heat uptake from a fire is decreased to 10% of the "normal" amount. The temperature of this cask can be seen to be at acceptable levels even after 10 hr of exposure to a 1475° fire.

CONCLUSIONS

The question of hazard survival is ultimately related to the magnitude of the hazard. Very heavy casks capable of surviving a 30-ft free-fall impact against an unyielding surface would appear to have a high probability of surviving any impact that might occur in the course of surface transport. Speeds are encountered in surface transport in excess of that attained in a 30-ft free fall, but surfaces which are unyielding to a 50-ton-or-more object are very scarce indeed. A maximum credible duration and intensity for a fire is more difficult to estimate. As has been observed on many occasions, some finite amount of time is required to respond to an emergency. However, it appears that the inherent capability of a steel-shielded cask may, without assistance, provide sufficient time for

Fig. 2. Temperature Transients for Uranium- and Iron-Shielded Casks Exposed to 30-min and 4-hr Fires. Cask ID, 48 in. Internal heat generation, 50,000 Btu/hr-ft cask length.

Fig. 3. Temperature Transient for an Iron-Shielded Cask Exposed to 1475°F Fire While Insulated to Limit Heat Uptake from Fire to 10% of the Normal Amount. Cask ID, 48 in.; cask OD, 84 in.; internal heat generation, 50,000 Btu/hr-ft cask length.

response. The addition of other devices to retard the temperature rise of a cask exposed to fire can allow even longer times to gain control over an accidental fire. It is concluded that adequate protection against containment failure due to impact and fire can be provided for a rail shipment of short-decayed LMFBR fuel when the development work now in progress is completed.

REFERENCES

1. W. E. Unger, R. L. Cox, W. S. Groenier, A. R. Irvine, and C. D. Watson, "The Adaptation of Present Nuclear Fuel Reprocessing Technology for the Reprocessing of LMFBR Fuels," Paper 67c, presented at AIChE Sixty-Third Annual Meeting, Chicago, Illinois, November 29 - December 3, 1970.
2. D. E. Ferguson and Staff, Chem. Technol. Div. Ann. Progr. Rept. for Period Ending March 31, 1971, ORNL-4682 (July 1971).
3. R. L. Cox, ORNL, personal communication, 1971.

AIR CUSHION SUPPORT FOR HIGHWAY TRANSPORT OF HEAVY LOADS

W. H. Trask

ABSTRACT

Air cushion technology is defined and related to applications in highway hauling. Equipment design requirements are discussed. Load spreading as a highway "preserving" technique is introduced with consideration for bridge capacities. The technology and economics of cushion support applied to spent fuel hauling are briefly considered. The possible effects of cushion vehicles on highway regulations are predicted. And, finally, the role of highway commissions in developing cushion support technology is suggested.

INTRODUCTION

Air Cushion Technology

Air Cushion Technology may be defined as that branch of motion engineering devoted to the support of gravity and inertial loads with low pressure air (less than 10 psig) acting over significantly large areas. A typical air cushion device, whether it be a seagoing hovercraft, warehouse pallet mover, or highway vehicle, employs a low pressure, high flow air source such as a centrifugal fan to pressurize a plenum chamber formed between the load and ground. Air flow is restricted to reasonable levels by the ground-chamber interface, usually an air gap of controlled dimension. The load floats slightly above the ground on a cushion of air making it quite mobile. Very large loads can be supported at pressures of a few pounds per square inch.

Air cushion technology began with waterborne craft as a means for

side-stepping hydrodynamic hull resistance at high speeds. Large ships of conventional design are essentially limited to speeds under 50 knots by the rapid growth of power requirements beyond this limit. Speeds of 70 knots, however, are readily obtainable when a craft rides on air several inches above the water. This fact forms the basis for the British-developed SRN4, a hovercraft with a normal operating weight of 165 tons (50 ton payload) and capable of speeds up to 77 knots. This craft absorbs 13,600 horsepower at a cruise speed of 70 knots with about 40 percent of this power going for vertical lift. Cushion pressure is 50 lbs/sq ft.¹ Ignoring the substantial cost of development, this vehicle is proving economically feasible in channel ferry usage.² Other designs, such as a 100-ton naval craft under development in this country, will ride a "captured air bubble" at speeds in excess of 92 mph. Bubble pressures between 95 and 110 psf are anticipated. Two versions of this craft to be tested during the fall of 1971 will develop between 12,000 and 13,000 horsepower.³

The British have also pioneered the use of very high cushion pressures, up to 5 psig and beyond, for transporting heavy electrical transformers over marginal roadways and bridges. One transporter with a "cushion lift" capacity of 100 tons has been successfully operated.⁴ Air cushion pallets and dollies are now commonplace.⁵ French-built high speed trains run along a single inverted "T" concrete "rail" supported by 1/2" thick cushions of air. Amphibious trucks and boats of various designs use the air cushion principle.⁶ Oil drilling platforms weighing 100 tons and more will employ air cushion support for mobility during exploratory drilling in arctic

regions.⁷

As air cushion technology continues to develop, the economics becomes more attractive. Proper equipment design and shrewd applications engineering are today putting air cushion hardware on a money-making basis. Continuing developments in materials and fabrication techniques will see a significant increase over the next five years in dry land applications of air cushion support systems.

Application to Land Vehicles

Technology is ready now to produce air cushion support systems for land transportation equipment, particularly as applied to highway vehicles. Many hauling situations are amenable to air cushion support. Movement of construction machinery, power plant components, spent nuclear fuel, steel and other bulk materials could be accomplished with less wear and tear on highway structures with air cushion transporters.

The initial applications should come in highly specialized situations where large expensive loads must be moved by highway. Short skirts attached to lowboy trailers will be able to cushion 50 to 60% of the gross load at 2-3 psig. In some instances vehicles will be massive and move slowly, from 5-10 mph. Low speed movement is very compatible with cushion support. Other vehicles, transporting nuclear fuel casks, for instance, will move at highway speeds of 45-55 mph and demand much more of the skirt/seal system. As sealing systems are perfected, cushion support will be applicable to conventional trailers with three to four feet of ground clearance. Packaged cushion systems will eventually be available to fit almost all vehicle sizes and applications. Road seal technology will improve, keeping power plant

requirements small. User experience fed back to suppliers will spur applications and improvements in hardware and materials.

EQUIPMENT REQUIREMENTS

Four basic hardware "units" are required to apply air cushion technology to highway transporters. These are:

- . Skirts
- . Road Seal
- . Blowers
- . Control Systems

Skirts

This element forms the cushion chamber sidewalls which, together with the load itself and ground, constitute a semi-pressure tight plenum of air. Skirts function as a pressure barrier and as a suspension system. Skirting a highway trailer would require tailoring the design and materials to the specialized dynamics and pressure requirements of these vehicles. For example, a standard 40-ton lowboy trailer could be equipped to contain 2 psi of air pressure between the trailer well and ground. The skirt should be rigid enough to support the internal pressure while maintaining the resiliency needed to maintain contact with the roadway as the trailer undergoes vertical motion.

Since the lower periphery of the skirt which contacts the road presents a particular problem it has thus been separated out and designated as the "Road Seal."

Road Seal

Approximately the lower three inches of skirt length are designated as the "road seal." This element will be detachable from the skirt because it will undergo wearout. A good pressure seal between skirt and road is necessary to keep air flow requirements from becoming excessive. For example, a standard 40-ton lowboy with skirts pressurized at 2 psi will support 58,000 lbs on air. Total air flow allowable through the skirt-road seal periphery of 66 feet may be calculated assuming the on-board blower power is limited to 100 horsepower. A maximum leakage rate of 105 CFM/lineal foot of road seal results. This is the maximum leakage rate that can occur under dynamic conditions if the cushion pressure is to be held at 2 psi. A minimum service lifetime for the detachable road seal may also be estimated by considering a typical hauling situation. At 45 mph a large tractor-trailer combination might cover 400 miles during a 12-hour driving day. Total or partial seal replacement after a single such day would probably be required from initially available seal materials. Changeout would of necessity be simplified so that the driver could replace the entire seal in less than one hour.

Blower

This unit, including a power source, supplies air pressure to the support cushion. A 100 hp, diesel-powered package delivering 7000 CFM of air at 2 psig would weigh approximately 3500 lbs and require about 30 sq ft of area for mounting. Several fans are available commercially that would be suitable. A belt drive would be used to allow optimization of available space and engine RPM. Blower and engine would be shock mounted to isolate

the equipment from road vibration. A blower intake silencer would be required to suppress noise. The blower-engine combination can be considered essentially state-of-the-art requiring only minor modifications to insure its roadability.

Control Systems

These systems would function to maintain a near constant cushion pressure over the support area during perturbations induced by road-vehicle interactions. Such systems would require feedback control for effective dynamic regulations. The state-of-the-art in this area is not well defined and would be a major part of any development program. The problem is aggravated by limitations on the air volumes that can be contained beneath a lowboy trailer.

Practical control of cushion pressure could be accomplished using a combination of several different hardware systems. Partitioning of the cushion volume, automatic flow control, accumulator systems and plenum venting devices are some possible approaches. Overall cushion pressure, as measured near the cavity centerline, for example, could be the regulatory signal governing inlet air flow from the main fan. Excess fan air, about 20 percent, would be vented during normal cushion operation but could, if required, be diverted partially or totally into the cushion area.

LOAD SPREADING AND HIGHWAY STRENGTH

The effect of air cushion load spreading on highway systems can be subdivided for clarity into the effect on 1) pavement structure and 2) on bridge structures.

Pavement

Intuitively at least, load spreading should have a beneficial effect on highway durability. Among the many stresses highways must withstand, including freezing and thawing, ground motion and thermal expansion, are those imposed directly by vehicular traffic. Truck tires inflated with 75-85 psi of air pressure load the highway surface accordingly. Multiple tires and tandem axle arrangements tend toward load distribution but usually do not include a significant reduction in tire pressure. Except for special hauling situations such as those requiring the one time movement of hundreds of tons, simply adding additional axles and wheels is impractical economically and to some extent technologically. Load distribution between axles is difficult to maintain. Most states have limited permissible gross loadings regardless of how many axles are used to subdivide the load, at least for repetitive haulings where highway lifetime becomes a factor. The use of trailers with 20, 30 or even 40 wheels even if permitted would be expensive terms of tire maintenance alone.

Air cushion load spreading, for example, can support a 60,000 lb load over an area of 300 sq ft at a 2 psi ground pressure. This is the approximate area contained beneath the well section of a 50-ton lowboy trailer. It is obvious at least that the highway degradation factor associated with a uniform 2 psi loading is significantly less than that associated with even the multiple pad loadings at 75-85 psi produced by tandem axle trailers.

Bridges

Overstressing of bridge structures and overpasses is a source of major concern to state highway commissions. Where overstressing is requested on a

repetitive basis, highway commissions find it difficult to justify the issuance of permits. At least one industry at present requires the repetitive shipment of overweight loads by highway. Spent nuclear fuel is hauled on a daily basis from power-producing reactors to special reprocessing plants. Very heavy containers are required, limiting payload fractions to less than three percent. Cushion support of such excess tonnage, while of significant benefit to states in protecting highway systems, would be of minimal value in the bridge crossing situation. Cushion support does distribute the load and reduces dynamic transients. But basically, bridges and overpasses used by haulers in moving spent nuclear fuel, for example, will require upgrading to the point where repetitive crossings at the maximum expected gross loadings will be feasible. The cost of bridge refurbishment where necessary should be assumed by the primary user. In the nuclear fuel case this would be the utility company owning or operating the generating stations. Ways in which such expenses may be written off or even balanced out involve use by the utilities of pavement-saving cushion support equipment in return for major concessions from the highway commissions regarding maximum allowable gross, divisible loads, and permissible hauling times.

CUSHION APPLICATION TO HAULING SPENT NUCLEAR FUEL

Cask Weight

Second generation spent fuel elements will require a cask weighing up to 80,000 lbs for highway shipment to the reprocessor. This will result in a gross vehicle weight of 115,000 lbs. The 80,000 lb cask will carry two PWR units or four BWR units for a total payload of slightly under 2,000 lbs. A cask weighing 130,000 lbs would carry four PWR or eight BWR units with a

payload of nearly 4,000 lbs.⁸ This jump is highly significant in the economic context of spent fuel transport. However, the 180,000 lb vehicle gross required to haul such a cask is, under current considerations, unacceptable. If cushion support were employed to bear 80,000 lbs of the total gross at 2.5 psig ground pressure, axle loadings would be correspondingly reduced to below the 17,000 lb level for a six axle unit. Significantly, the payload fraction of the 65-ton cask is 50 percent greater than for the 40-ton cask. More important, however, is the fact that air cushion support, once initiated by nuclear fuel haulers, would soon be ready for use in other transport situations which are currently overstressing highways in terms of axle loads. The widespread application of cushion support particularly during the spring thaw season could materially reduce highway surface degradation per ton mile of travel and thus enhance highway lifetime. The total economic payoff would benefit utilities, truckers and highway systems alike. In return for substantial increases in the maximum gross loading permitted on a repetitive basis, utilities could 1) refurbish bridges used for spent fuel hauling and 2) underwrite the expense, together with transport companies, of developing operational air cushion trailers. Highway commissions could stimulate this development by easing time of day and divisible load restrictions now in effect, using as their justification the road saving effect of widespread cushion transportation.

EFFECT ON HIGHWAY HAULING REGULATIONS

Gross Loads

As mentioned above, utilities, truckers and fuel reproprocessors would all like to see highway commissions permit higher gross loads on a repetitive

basis. Air cushion support allows increasing gross weights with no increase in axle loadings. As an example, a vehicle grossing 200,000 lbs supports 100,000 lbs on air at 3 psig over an area measuring 8' x 40'. The axle supported "gross" is 100,000 lbs which divided among six axles gives approximately 17,000 lbs per axle. Axle loadings are seen as the significant factor in highway wearout unless of course argument can be made for the effect of 3 psig loadings, regardless of what the total gross may be.

Divisible Loads

Many states require that overweight loads, if divisible, be subdivided down to the smallest, single, non-divisible element. This regulation discriminates against spent fuel hauling while serving no useful purpose. Its modification would, however, represent a very major concession to one hauler by highway commissions and would, if made, require strong justification. As mentioned before, utilities, in funding the refurbishment of state and federal bridge crossings and in actively promoting the application of air cushion support, would be providing states with a reasonable and very significant justification for major regulatory concessions.

Repetitive Hauling

Repetitive hauling on a 24-hour basis will be needed by nuclear reprocessors to do their job economically. Repetitive hauling in the context of spent fuel movement means, of course, a closely controlled, highly supervised operation which occurs on a routine basis. Safety is the key consideration followed closely by control and enforcement. Highway commissions may not feel that such a concession could be properly supervised.

Radioactive materials, however, under transit are the subject of much concern to several agencies heavily charged with responsibility in this area. Utilities themselves along with the transporter could bear a larger share of the burden by nature of their involvement in bridge reconstruction and their investment in air cushion support equipment. With a heavy financial interest in the successful outcome of round-the-clock repetitive hauling, utilities and transporters alike could be expected to provide the self-regulation and control necessary to insure that transport was done in a safe and orderly manner at all times. Utilities, in particular, would become partners with the states by this involvement in preserving critical highway routings and in demonstrating the advantages of air cushion support.

ROLE OF STATE HIGHWAY COMMISSIONS

The basic function of highway commissions is to preserve and protect the state's investment in highways while providing equal access to the system for all segments of commerce desiring to use highway transportation. Highway vehicular traffic as a complex systems problem demands stringent regulation for two basic reasons: First, the rights of each user must be protected from infringement by others; and second, regulation is required to optimize highway availability and thus maximize the movement of goods to and from the marketplace. Obviously, allowing permissible gross loadings to increase significantly would benefit many segments of the hauling industry including firms hauling spent nuclear fuel. Such benefits would be short lived indeed if carefully chosen routes were disrupted by repeated overloadings. Whereas tires, bearings and even entire vehicles may easily be replaced when worn out, highways are, in today's economy, essentially irreplaceable. Electric

utilities, in forecasting 30 and 40 year service lifetimes for power reactors, must also consider the lifetimes of roads over which spent fuel from those reactors will move. Preservation of highways is fully as requisite to the function of utilities as to the states themselves. Because utilities stand to realize a significant economic gain from permitted increases in gross weight limits, it is utilities who should lead out in preserving the integrity of existing highway systems. Utilities can do this in two ways, given proper incentive by highways commissions. They can 1) renovate bridges and 2) underwrite air cushion hardware development. Concessions should favor the use of air cushion transportation as a road saving device. Taxation could also be levied in such a manner as to encourage investments by truckers in air cushion load spreading equipment. Axle weights should be re-emphasized while gross loadings are de-emphasized. Where bridges are involved, the cost for retrofit to handle increased gross weights should be assigned to the user. Highway commissions represented by AASHO should play a commanding role in providing the proper inducements for haulers to use air cushion hauling equipment. Centered around the air cushion principle and its application will be found a blend of participation by users and regulators alike. The rapid dispersal of air cushion technology throughout the heavy hauling industry will mean less dollars for road repairs and more profit for transportation companies and all those served by the trucking industry.

REFERENCES

1. T. Ford, Hovering Craft & Hydrofoil, 6(6) (March 1967)
2. L. Boddington, Hovering Craft & Hydrofoil, 8(7) (April 1969)
3. J. L. Decker, J. Astronaut. & Aeronaut., 8(6) (June 1970)
4. T. Ford, Hovering Craft & Hydrofoil, 7(10) (July 1968)
5. Anon., Design News, 25 (21), p. 62 (Oct. 1970)
6. Various authors, Hovering Craft & Hydrofoil, 10(1) (Oct. 1970)
7. L. Sheffer, Ocean Oil, Weekly Report 5(16) (Jan. 18, 1971)
8. C. Wesley Smith, Proceedings, SINB Conference-Transportation of Nuclear Spent Fuels, (Feb. 1970) p. 63

COMPUTER MOVIES FOR SIMULATION
OF MECHANICAL TESTS

L. H. Baker
B. J. Donham
W. S. Gregory
E. K. Tucker

ABSTRACT

Evaluation of results produced by digital computers need not be a tedious process of number evaluation. By means of the basic models used in simulating the effects of the 30-ft fall and the fire test of shielded shipping containers, computers can be programmed to produce a visual record of the results of the calculation in the form of geometric images. Viewed in rapid succession these images form a motion picture of the modeled process.

A number of different schemes may be employed to depict the numerous variables involved. Whenever possible one should present the various phenomena with generally accepted representations. Color, for example, may be used to represent temperature, while shape changes may be directly represented by a corresponding geometric presentation.

Along with the many benefits brought about by the speed and versatility of the digital computer there exist some inevitable disadvantages due to the very element of speed. Whereas the engineer using longhand calculations aided by mechanical calculators and slide rules can grind out a few well-thought-out solutions to some design problem in a work day, he can utilize a digital computer and in a similar time period produce literally thousands of potential problem solutions resulting from the variation of important parameters.

Problems involving optimum solutions often may be handled by the computer itself, but some classes of problems, e.g., time history calculations, must be examined by the engineer at many points in time. He is thus faced with the problem of examining many numbers and analyzing many data in order to evaluate the overall results obtained from the solution.

These data are generally far too valuable an analysis tool to give up in favor of fewer longhand calculations, so the best solution of this dilemma will

be obtained if one has a better method of result scanning. Programming ingenuity can provide valuable assistance in presenting numerical data, but the simplest and most understandable form of presentation is motion pictures.

Motion pictures, as certain computer graphics are called when time is one of the plotted variables, can be prepared by the computer to show the interrelationship among the various phenomena as calculated by the programmed digital computer. One of the simplest and best understood forms of animation occurs when one must display time as one of the several variables. This occurs, for example, when shielded shipping containers are evaluated for the mechanical tests of the 30-ft drop and the fire test. The results of these two tests are made to order for presentation in a computer movie.

It is of prime importance to any form of presentation of results that the results be calculated through the use of models that accurately represent the interrelationship of the important variables under consideration. For this reason the models used for the generation of the data shown in the movies will be covered in some detail.

The dynamic analysis model is a simple undamped, close-coupled, multi-mass scalar spring system which may have non-linear coupling characteristics. Figure 1 is a typical model showing a technique for simulating a cylindrical container oriented to impact in an end-on configuration.

Spring characteristics for the mass coupling must be determined for each segment of the container. In general the strength of lead has been shown¹ to be reasonably well predicted by the equations shown in Fig. 1. Contact areas may be calculated by considering that horizontal planes within the container remain horizontal and that the individual finite volumes remain unchanged. Fins providing significant impact protection may be represented by introducing a force displacement variable comparable to the action of a fin when subjected to large axial strains².

Corner drops may be modeled in much the same manner. Figure 2 shows a sketch of the possible configuration resulting from making the container into a series of discrete masses. Some error is introduced by the fact that all of the mass lumps do not have their center of gravity located along that of the total container. This effectively introduces a rotational component to each mass. Neglecting such components should introduce only a minor error, provided the container is not of unusually large diameter. Elementary masses of the corner drop do not have the traditional cylindrical shape of the preceding model. This, however, should not cause any significant problems since the established equations of strength are independent of cross-sectional shape. There are some considerations of shape that must be included because the element volume must be

While the ability of any model to accurately predict the behavior of the prototype is unquestionably the ultimate goal, it is for this report not the final objective. Although these models and the solution technique are believed to yield excellent results, many other techniques have been used to represent similar systems quite successfully. The objective here is to show the results obtained from one such technique in a unique and easily understandable manner. Solutions from this method of analysis consist of arrays of times and corresponding mass positions. Display and evaluation of these solutions will be covered in the latter portion of this report.

Heat transfer models have likewise become very popular and quite numerous in the past few years, but they, like the dynamic analysis techniques, are difficult and time consuming to analyze without some thoughtful presentation of results.

Results presented in this report have been obtained from a suitably reliable three-dimensional finite element heat transfer code that calculates spatial and transient effects of the standard hypothetical 30-minute-duration fire acting upon a lead-shielded shipping container. One unique feature of the presentation is the ability of the analysis to consider a change of state of said lead to molten or its replacement by air if the molten lead is assumed to escape through a breach in the steel jacket. Edges, corners, and fin effects may be readily modeled because of the versatility of the three-dimensional analysis.

The computational model used in the heat transfer code assumes that a cross-section of the specific configuration (body) of interest is divided into a system of small cells. Figure 4 shows one-quarter of the cylinder modeled in this analysis. There are 2600 cells in this model, describing three different materials. Although finned surfaces can be modeled, it was assumed for this report that the outside surface of the cylinder was a smooth steel jacket.

The size, orientation, material composition, and modes of heat transfer for each individual cell and interrelationships between cells are prescribed in the program by a general heat balance equation.

$$\dot{Q}_{\text{stored}} = \dot{Q}_{\text{cond}} + \dot{Q}_{\text{conv}} + \dot{Q}_{\text{rad}} + \dot{Q}_{\text{gen}} \quad (3)$$

The conduction equation is

$$\dot{Q}_{\text{cond}} = \frac{(T_i - T_j)}{R_{i-j}} \quad (4)$$

where

$\dot{Q}_{\text{cond}}$ = conductive heat exchange between cells i and j, btu/hr,

T_i = temperature of cell i, °R,

T_j = temperature of cell j, °R,

R_{i-j} = thermal conduction resistance, °R-btu/hr.

Fig. 4. Heat Transfer Model Container.

The radiation received on the outside of the container can be expressed as follows

$$\dot{Q}_{\text{rad}} = \sigma_{\text{sb}} \text{Abs } A_i [(T_f^4) (E_f) - T_i^4] \quad (5)$$

where

$\dot{Q}_{\text{rad}}$ = radiation heat exchange between outside cell i and the flame temperature, btu/hr,

σ_{sb} = Stefan-Boltzman constant = 0.1714×10^{-8} , btu/hr-ft²-°R⁴,

Abs = absorbance of the cask = 0.8,

A_i = area of cell i, ft²,

T_f = flame temperature of the standard fire = 1935°R,

E_f = flame emissivity = 0.9 .

The internal radiation exchange, calculated if a loss of lead is assumed, can be expressed as

$$\dot{Q}_{\text{rad}} = \sigma_{\text{sb}} A_i F_{i-j} (T_i^4 - T_j^4) \quad (6)$$

where

F_{i-j} = view factor.

The radioactive source in the shipping container may impart heat to the materials surrounding the radioactive source. This heat generation term may be expressed as

$$\dot{Q}_{\text{gen}} = V_i \dot{Q}_{\text{gen}_i} \quad (7)$$

where

$\dot{Q}_{\text{gen}}$ = generation of heat within cell i, btu/hr,

$\dot{Q}_{\text{gen}_i}$ = strength of the source of generation in cell i, btu/hr-ft³,

V_i = volume of cell i, ft³.

Free and/or forced convection can also occur on the outside of the container and also internally if air or a liquid coolant is present. The general form for convective heat transfer between cells is

$$\dot{Q}_{\text{conv}} = h_{i-j} A_{i-j} (T_i - T_j) \quad (8)$$

where

$\dot{Q}_{\text{conv}}$ = convective heat exchange, btu/hr,

h_{i-j} = convective heat-transfer coefficient, btu/hr-ft²-°R.

Combining the appropriate modes of heat transfer for each cell allows one to write an energy balance equation for a cell as

$$\begin{aligned} & [T_i (t+\Delta t) - T_i (t)] C_i = \\ & \Delta t [\dot{Q}_{\text{cond}} (i,j) + \dot{Q}_{\text{conv}} (i,j) + \dot{Q}_{\text{rad}} (i,j) + \dot{Q}_{\text{gen}} (i,j)] \end{aligned} \quad (9)$$

where

C_i = heat capacity of cell i, btu/°R,

t = time interval, hr,

$T_i(t+\Delta t)$ = temperature of cell i after time interval Δt , $^{\circ}\text{R}$.

The above equation can be used to solve for the future temperature of cell i after time Δt . Many techniques are available for solving a set of simultaneous temperature equations for each cell. The extrapolated Liebmann technique^{4,5,6} was used to obtain the results presented here.

Once again as in the case of the dynamic analysis model the accuracy of simulation of the prototype was not evaluated since the major objective was to develop and present movie methods of data evaluation.

Data presentation by computer movies is not a theoretically difficult task. Neither is it a trivial task when one considers the requirement of machine and programming effort. Since this effort is at neither extreme it is probably a medium of data presentation that will assume great importance in the future. It indeed enjoys a high level of utilization at the Los Alamos Scientific Laboratory, where film use approaches 350 miles per year.

Motion pictures are produced by the computer as a result of causing a sequence of images to be plotted on a cathode ray tube and then photographed, sometimes through a color filter, upon the appropriate size film. The instructions necessary to cause the desired image to appear on the cathode ray tube are prepared by the computer programmer. Details of the plotting instructions may vary somewhat from one computer installation to another but they are usually rather simple to use.

Displaying the data from the two basically different calculations previously described may best be done if we attempt to display a series of pictures which look as nearly as possible like the actual prototype test. In the case of the dynamic analysis this is an easily obtainable goal. One needs merely to draw a graphical representation of the container, at regular time intervals, based on the data obtained from the calculations.

All elementary masses used in the dynamic analysis are considered to be right circular cylinders whose volumes are equal to that of a hypothetical mass cut from the cylinder by two planes parallel to the assumed unyielding surface. If these masses are small enough the cylindrical representation will very closely resemble the geometry one wishes to project. Accepting the small error of lateral motion caused by using the cylindrical model one can then draw the cylinders for each time step for which height calculations are made. Figure 5 shows a typical example of how the results of these cylinders may be shown. Note that further refinements can be made to the display by smoothing the intersection between hypothetical cylinders as in Figure 5d. Also, it must be borne in mind that one has great flexibility in displaying the lateral motion. Exaggerated lateral deformation may be plotted because this display is for visual effect only

$T = 000. \mu \text{ SEC.}$
**CALCULATIONAL
 MODEL**
 (a.)

$T = 1037 \mu \text{ SEC.}$
**PLOTTED
 REPRESENTATION**
 (b.)

**DISTORTED LATERAL
 MOTION**
 (c.)

**BOUNDARY SMOOTHING
 TECHNIQUE**
 (d.)

Fig. 5. Typical Results from Computer Plotted Graph for Dynamic Model Display.

and does not affect the basic accuracy of the calculated vertical displacement. Figure 5c shows the results of such a distorted plot. Although the distortion shown gives a somewhat unrealistic view of the results it does give a closer look at the actual distortion pattern.

Color film, responsive to colors ranging from red to blue, may be used to improve the overall appearance of computer motion pictures, but it would be near-sighted of the user to make this the sole reason for use of color film. Although this black and white publication cannot show how colors are used to clarify the plotted image, the original container configuration may be shown in a contrasting color to the final deformed shape.

Displaying a geometrical model is a very straightforward process and is a natural method of presenting a geometrical time dependent phenomenon, but one is faced with a different problem when one wishes to show the simulated results of a 30-minute fire test. Logically a temperature phenomenon may be represented by a color phenomenon, while prototype spacial coordinates or geometry may still be represented by a geometric pattern for the model display. Time relationships should be proportional to movie viewing time but need not be on the same scale.

In this temperature display, color film plays a very important role in projecting information, although other techniques such as isothermal lines or a subgeometrical representation may be used when color is not available. Figure 6 shows the basic display capabilities of the computer. Isothermal lines have been substituted for color for the purpose of this report.

Time required for the production of a complete set of simulations such as those described in this report would be a few man-hours while the computer time and film costs would probably run two or three times that amount. This represents a total cost which cannot even be compared to the costs in labor and material required to conduct full scale tests. If sufficient data are available to prove the constants assumed for the mathematical models used, a remarkable saving may be realized while valuable and easily understood results are obtained. Unquestionably a thousand pictures shown as a movie will be worth a thousand-thousand numbers.

Fig. 6. Typical Results from Computer Plotted Graph for Temperature Display.

REFERENCES

1. B. J. Donham, A Mathematical Model for Prediction of Maximum Damage to Shielded Shipping Containers, Presented at IAEA International Seminar on Test Requirements for Packaging for the Transport of Radioactive Materials, Vienna, Austria, February 8-12, 1971.
2. F. C. Davis, Structural Analysis of Shipping Casks, Vol. 9, Energy Absorption Capabilities of Plastically Deformed Struts Under Specified Impact Loading Conditions, Oak Ridge National Laboratory, ORNL-TM-1312, Vol. 9.
3. N. M. Newmark, A Method of Computation for Structural Dynamics, Proceedings of the American Society of Civil Engineers, Vol. 85, EM3 (July 1959).
4. R. F. Thomas and M.D.J. MacRoberts, RATH Thermal Analysis Programs, LA-3264-MS, Los Alamos Scientific Laboratory, Los Alamos, New Mexico (1965).
5. M. N. Ozisik, Boundary Value Problems of Heat Conduction, International Textbook Company, p. 407 (1968).
6. B. T. Chao and A. M. Clausing, Numerical Methods in Heat Transfer, Advanced Heat Transfer, p. 189 (1969).

THE FOLLOWING PAPERS WERE NOT
PRESENTED AT THE SYMPOSIUM

/ HEAT TRANSFER ANALYSIS OF A CESIUM-137 SHIPPING CASK

G. R. Bray
J. J. Darby
C. E. Gruhl

ABSTRACT

Shipment of cesium-137 from Richland, Washington, to Oak Ridge, Tennessee, required the design of a new shipping cask. Design factors were based on the structural integrity of the cask and the maximum volume of product that could be contained per unit of product heat generated under varying conditions.

Conventional calculations were used in the cask's design, and then a computer program was developed to confirm them. In addition, steady-state temperatures were taken with various cask loadings.

INTRODUCTION

The fission product cesium-137 is one of the valuable (commercial and experimental) by-product isotopes produced in Hanford's plutonium-production reactors. This radioactive product must be shipped from Hanford to Oak Ridge in special containers designed to exacting standards; regulations of both the Department of Transportation and the AEC influenced this design.

Atlantic Richfield Hanford Company (ARHCO) requested that Vitro Hanford Engineering Services perform the necessary cask design. It entailed the design of a special insert for an existing cask's (HAPO-1B-3) outer shell (Figure 1). The cask's new insert (HAPO-1C-1) is shown in Figure 2.

The 1C-1 cask was designed to contain 3.86 ft³ of Zeolon 900 Resin* to absorb, by ion exchange, a 361 kilocurie loading of cesium-137 (Figure 3). The k factor (thermal conductivity) for the Zeolon Resin was given as 0.13 Btu/hr/ft²/F by

* T. H. Norton Co., Trade Name

the manufacturer. The potential heat loading was set at 5,920 Btu/hr by ARHCO. Maximum allowable resin temperature was determined to be 382 F with the resin bed being purged with helium.

CALCULATIONS

Calculations were first made using General Electric Design Data¹ to arrive at temperatures of various surfaces throughout the cask, and then the ORNL-TM-2410 formula² was applied to determine the cask's outer temperature.

The maximum allowable center-line temperature taken through a cross-section of the annulus was the determining factor for the annulus dimensions. An ambient temperature of 130 F and a maximum resin bed temperature of 382 F established a ΔT of 252 F.

Using the ORNL formula (Eq. 1), calculations were performed to determine the minimum ΔT between the cask surface and still air that would dissipate the heat load (5,920 Btu/hr) given in the criteria. The third trial indicated that 35 F difference would suffice, i.e., a cask surface temperature of 165 F. These calculations did not encompass solar loading since the cask is enclosed in a buffering structure during transport (See Figure 4).

$$Q_{\text{Total}} = h_c A_c (T_s - T_a) + 0.173 \overline{F_{12}} A_r \left[\left(\frac{T_s + 460}{100} \right)^4 - \left(\frac{T_a + 460}{100} \right)^4 \right] \quad (1)$$

where

Q_{Total} = Total heat transferred, Btu/hr

h_c = the convective heat transfer coefficient, Btu/hr ft² °F

A_c = effective convective surface area, ft²

A_r = effective radiative surface area, ft²

T_s = cask surface temperature, °F

T_a = ambient temperature, °F

$\overline{F_{12}}$ = the "gray-body" shape factor

The Design Data formula (Sect. G502.4, p 11)¹ is used to determine temperatures at the extreme point in location from the temperature given.

For a hollow cylinder of several layers

$$t_{\max} = t_s + \frac{q}{2\pi c} \left[\frac{\log e \frac{rb}{ra}}{k_a} + \frac{\log e \frac{rc}{rb}}{k_b} + \dots \right] \quad (2)$$

in which

$t_{\max}$ = temperature at point in resin bed, °C

t_s = input at inner surface, °C

q = heat flow, watts

c = linear dimension, inches

$r_{a,b,c}$ = radii of varying layers, inches

k = thermal conductivity, $\frac{\text{watts/sq in.}}{\text{deg C/in.}}$

This calculation gave a temperature at the surfaces resin bed of 175 F. Further calculations using GE Design Data¹ were made to approximate the maximum temperature of the resin bed itself.

$$t_{\max} = \frac{q^m}{4k} \left[r_2^2 - r_n^2 - r_n^2 \log e \frac{r_2^2}{r_n^2} \right] \quad (3)$$

in which

q^m = heat generation per unit volume, watt/cu in.

k = thermal conductivity, $\frac{\text{watts/sq in.}}{\text{deg C/in.}}$

r_2 = outside radius of cylinder, inches

r_1 = inside radius of cylinder, inches

$r_n = \begin{cases} \text{if there is no outward heat flow, } r_n = r_2 \\ \text{if there is no inward heat flow, } r_n = r_1 \end{cases}$

For these calculations r_n was taken to equal r .

Computed temperature of the resin bed by formula Eq. (3) was 360 F, a temperature below that given by the criteria as maximum.

COMPUTER MODEL

While the cask was being fabricated, a computer model program was run by Mechanics Research Inc.,³ of Tacoma, Washington.

The computer program showed a maximum resin bed temperature of 279.5 F under summer steady state conditions indicating that a larger volume space for cesium could safely have been designed into the cask.

STEADY-STATE TESTS

Tests made by the customer in October 1970 to determine actual steady state temperatures were made in a test cell under the following conditions:

Pressure (ab) (inches Hg)	Gas	Resin Bed	Temperature, F			Cell
			Pb	Top	Side	
18	Air	335	126	93	111	70
28	He	293	125	93	109	73

These measurements were taken with the cask in the 212-B cell at Hanford's 200 East Area. The cask was exposed to approximately 4600 scfm ventilation air down over and past its top. All temperatures were taken with iron-constantan thermocouples which demonstrated ± 1 F accuracy in an ice bath. The estimated isotropic heat load in the cask was 6720 Btu/hr. This value should be within $\pm 5\%$.

The increased heat loading was the result of achieving a cesium loading in excess of that given in the criteria, 430 kilocuries vs. 361 kilocuries.

Comparing these test results to the original manual computations indicates that the formulae are accurate and conservative.

The detailed heat transfer analysis produced on the computer showed a higher heat transfer rate (loss) than was evidenced in the loading test, considering the amount of cooling air passing over the cask. There should be some discrepancy because of the difference in value assigned to the resin thermal conductivity (0.13 from criteria vs. 0.25 from computer references).

ACCIDENT ANALYSIS

The steady-state temperatures were assumed to be in effect at the time of design accident. Considering the 48-hour buried condition with no external heat loss, calculations were made to determine the length of time for cask material temperature equalization (heat sink characteristics) and then the temperature that would result at the end of the period (48 hours). Calculations indicated a total average temperature of 351 F. The computer results show a resin bed temperature of 341 F.

References:

1. General Electric Design Data (Section G502.4). October 1963.
2. Irradiated Fuel Shipping Cask Design Guide, ORNL-TM-2410.
3. Report C2399, ARHCO Contract SA-125. Mechanics Research, Inc., Tacoma, Washington.

Figure 1. HAP0-1B-3 Cask

SECTION A-A (ROTATED 90° CW)

NOTE: OUTER CASK AND SEALING BAND
NOT SHOWN THIS VIEW

DESIGN DATA

- 1 3.86 CU FT OF ZEOLON RESIN
- 2 361 KC. CESIUM 137
- 3 HEAT LOAD 5920 BTU HR
- 4 RESIN BED HEAT LOAD 2663 BTU HR
5. MAXIMUM RESIN TEMPERATURE 382°F WHEN RESIN
BED IS PURGED WITH HELIUM

Figure 2. Insert for Cask HAPO-1C-1

1C-1 ZEOLITE BED

LOADING CYCLE

Figure 3. Cask Insert Resin Bed

Figure 4. Lowering Cask into Buffering Structure

A STUDY OF THE EFFECT OF IMPACT AND OF THE USE OF A
TOROIDAL SHELL AS AN ENERGY ABSORBER ON A
CYLINDRICAL CONTAINER OF LEAD AND STEEL

L. R. Shobe

B. R. Dewey

ABSTRACT

Test specimens of 0.167 scale model of a full size container were dropped from 30 ft to determine their behavior during impact, their behavior during rebound, and the nature of the deformed regions. The severe damage experienced by the cylindrical specimen casks and the limited ability to absorb energy without failure lead to the development of a steel toroidal shell attached in such a way as to give maximum protection in corner impact. An approximate analysis of the energy absorption of such a shell and experimental verification shows that the stress state in the toroidal shell increases the shell's energy absorption by orders of magnitude over simple rings. A drop test with a toroidal shell attached to a scale cask substantiated the efficacy of this approach - a 14-lb model shell completely protected a 233-lb model cask. This represents a small percentage increase in weight for a major increase in protection against impact.

INTRODUCTION

Increasing use of radioactive materials makes it imperative that safer methods of transporting such materials be developed. In view of the catastrophic results of structural failures of containers, all possible accident conditions must be anticipated. AEC standards list hypothetical accident conditions: (1) free drop, (2) puncture, (3) thermal, and (4) water immersion. This investigation examined the free drop accident condition -- a free drop through 30 feet onto a flat essentially unyielding horizontal surface striking the surface in a position for which maximum damage is expected. The position for which

maximum damage is expected in a cylindrical container is that of impact on a corner. This accident condition was studied experimentally and a method for designing a necessary energy absorption device external to the container was formulated.

It is to be emphasized that the free drop condition which causes the maximum damage is one where the corner of the container strikes the unyielding surface with the mass center of the container directly above the corner. When the container is so oriented as it strikes the unyielding surface, it is hypothesized that all of the kinetic energy of the container is dissipated as plastic deformation of one spot on the corner. On the other hand, when a container falls such that the mass center is not above the corner that strikes the horizontal surface, some of the kinetic energy of translation is converted to kinetic energy of rotation. Thus several spots on the container ultimately impact with the horizontal surface and the plastic deformation is not confined to one location.

MODEL CASK-DESIGN I

The first specimen fabricated was a 0.167 scale model of a full size container designed by Shadid¹ in an analytical study of the effect of impact on cylindrical containers of lead and steel construction. The design of the full size container made economical use of materials and funds; and, consistent with good engineering practice, all details were kept as simple as possible. The design featured welded construction of carbon steel plate; the fasteners were also of carbon steel.

All details were maintained as accurately as possible in the scale model, and resulted in a model whose length and diameter were 14 in. and 7 in., respectively. With simulated contents it weighed 155 lb. The carbon steel components of this model cask are shown in Figure 1. During the development of this model by Welch², it became apparent that certain compromises, particularly with regard to welding, sealing, and fasteners, would have to be made in its design.

To overcome the difficulties involved in welding the thin materials in the model, the flange assemblies were machined from solid pieces of thick sheet steel. This eliminated all welds with the exception of those of the outer and inner shells to the flange assemblies, and was felt to give a model whose behavior would be quite similar to that of the full size container.

Metal gaskets were selected for sealing because of their greater likelihood of satisfying the hypothetical thermal accident condition. The design of the full size container indicated a soft aluminum gasket having a thickness

Fig. 1. Liner, Jacket, Flanges, and Closures of Model Cask I.

of 0.125 in. Application of a scale factor of 0.167 resulted in a thickness which presented formidable fabrication problems. A thickness of 0.050 in. was therefore specified, but on attempting to seal and pressurize the model, it was found that pressure could not be maintained. The fibrous gasket then substituted functioned properly and maintained a pressure of 50 psi. Subsequent examination of the gasket faying surfaces revealed that they were as much as 0.020 in. out of plane. Machining the faying surfaces at the end of the container opposite the end impacted to within 0.001 in. of plane permitted the soft aluminum gasket to function properly.

Scaled versions of the full size 0.750 in. diameter with 10 threads per inch fasteners were not commercially available in carbon steel. As a consequence, concessions were made in both material properties and size of the fasteners.

Six tests utilized this specimen. Initially the specimen was instrumented with electrical strain gages and an accelerometer, but the data obtained therefrom were not satisfactory. High speed motion pictures were made of all tests.

During the first drop from 9 ft. 5 in. onto a corner, the specimen sustained damage indicating that the full scale container would not be satisfactory: The seal and fasteners failed, and pressure was lost. The five subsequent tests, even though the model was modified, served to indicate that this design concept was not acceptable. Thus it became apparent that additional work would be required to develop a cylindrical container which would safely withstand impact.

MODEL CASK-DESIGN II

Because of the severe damage² experienced by the cylindrical specimen described above, it appeared that it would be appropriate either (1) to separate the sealing function from the radioactive shielding function, or (2) to provide a device whose sole function is to absorb the kinetic energy of impact without transmitting high impact loads to the shielding and sealing regions of a container. Adopting the first course of action led to the development of a two closure cask (1) in which the outer closure was designed to absorb most of the energy and the inner closure was expected to maintain the seal. The carbon steel components of this model cask are shown in Figure 2.

Fig. 2. Liner, Jacket, Flanges, and Closures of Model Cask II.

Subsequent testing of this model by dropping it 30 feet resulted in a loss of the seal. Because it was felt to offer more promise of success, the merits of an external energy absorption device were investigated and consideration of a design change wherein a greater clearance might have been provided between the two closures was discontinued.

AN EXTERNAL ENERGY ABSORBER-THE TOROIDAL SHELL

It seems clear that some sort of external energy absorber must be provided to protect such casks in the event of an accident in shipment. Such an energy absorber should be economical, should require no special maintenance, and should effectively protect the cask from impact in all possible directions, but needs only to be a one-time device. A hollow toroidal steel shell fastened on the ends of the shipping container is such an energy absorbing device. Since this sort of external energy absorber need only be attached to casks during shipment, and since only a small fraction of the total number of casks in the country will be in transit at any one time, this sort of concept certainly seems to be more economical than any one which would involve designing new containers and declaring the old ones obsolete. The utilization of the plastic deformation of ductile steel is one of the most efficient means of energy absorption. The geometry of the toroid offers ease in manufacturing, fairly efficient utilization of the material in absorbing energy, protection from all directions with the maximum protection where it is needed most, and suitability for rational design.

In analyzing the energy absorption in a toroidal shell, it is necessary to deal with large deformations in the plastic range; and, as a consequence, the elastic contributions to strain energy are negligible. A biaxial state of stress is assumed in the deformed shells, whereby the two principal directions used give the "ring" contribution and the "dilatation" contribution to the strain energy absorbed. In Fig. 3, these principal directions are denoted I and II respectively; the strain energy due to stresses in the third principal direction (III) is negligible.

For I and II as principal directions, the energy absorption U can be symbolically indicated as

$$U = \int_V \sigma_I \epsilon_I dV + \int_V \sigma_{II} \epsilon_{II} dV \quad (1)$$

where V is the material volume and σ and ϵ are stress and strain.

Fig. 3. Orientation of Axes

Energy Absorption in Ring

With reference to the cross sectional ring element of Figure 4, the shaded areas denote regions where the material has become fully plastic. For simplicity σ_I of Eq. (1) is assumed constant and equal to the yield stress σ_{ys} .

At points A and C in the ring, it is assumed that a plastic hinge is formed whereby the energy absorbed in each plastic hinge U_A is given by

$$U_A = M_p \phi, \quad (2)$$

where the M_p is the fully-plastic bending moment. A refined expression for U_A has been developed by Estep² which considers the extent of a plastic zone at A and shows the simpler Eq. (2) underestimates the actual energy absorption.

In the flattened region at the top of the toroid, it may be shown² from the geometry of deflection that the energy absorbed is

$$U_B = \frac{1}{2} \sigma_{ys} t^2 \frac{L}{r} = 2M_p \frac{L}{r} = M_p \phi, \quad (3)$$

since the inextensibility of the ring center line gives $2L = r\phi$. Hence by Eqs. (2) and (3), the total energy absorbed in the entire ring is

$$U_I = 4M_p \phi. \quad (4)$$

Fig. 4. Ring Element of Idealized Deformed Toroid

The angle ϕ is related to the deflection δ of the ring, as shown in Fig. 4 by

$$\delta = 2r \sin(\phi/2) . \quad (5)$$

The fully plastic bending moment M_p for non-strain-hardening materials is given⁴ by

$$M_p = \sigma_{ys} t^2/4 . \quad (6)$$

Hence, the energy absorbed by the ring follows from Eqs. (4), (5), and (6), and is

$$U_I = 4\pi R t^2 \sigma_{ys} \sin^{-1}(\delta/2r) . \quad (7)$$

Static and dynamic tests performed by Estep³ on mild steel rings from 2 in. to 8 in. in nominal diameter and with various wall thicknesses indicate that the energy absorption predicted by Eq. (7) is about 20% too low. One cause of this under-estimation is neglect of strain hardening. Estep's more refined analysis of strain energy in the plastic hinge (regions A and C of Fig. 4) is not presented here because of space limitations. Furthermore, these

tests showed that essentially the same energy is absorbed by the rings in the static tests as is absorbed in a test where a weight is dropped on a ring from 30 ft. Hence, for toroidal shell calculations, the energy U_I as given in Eq. (7) is increased by a factor of 1.2.

Energy Absorption by Dilatation

The strain energy contribution of "dilatation" (direction II of Fig. 3) is 2 to 5 times that of the "ring" energy. This effect may be visualized by reference to Fig. 4. Before deformation (dotted outline), point C is at a distance $(R + r)$ from the axis of revolution. After deformation (solid outline), point C is somewhat further from the axis. Hence the circumference of the circle of radius $(R + r)$ has been extended or "dilatated."

Rhoades⁵ et al have developed a numerical procedure for computing this dilatation strain energy. Basically, the deformed cross section is assumed symmetric; one quarter of the cross section is divided into n elemental segments. The strain energy of each elemental segment is formulated in terms of deformation, and the contributions of the n segments are summed. The result of summation for $4n$ segments is of the form

$$U_{II} = 2 \sum_{i=1}^n \sigma_{i,i+1} \bar{\epsilon}_{i,i+1} V_{i,i+1} \quad (8)$$

where the interested reader should consult the reference (5) for details on the computation of the terms within the summation of Eq. (8).

The total energy absorbed by the toroidal shell is the sum $1.2(U_I + U_{II})$ from Eqs. (7) and (8), with the 1.2 correction factor previously mentioned. Static tests run on toroidal shells show that such predictions are within a range of -29% to 16% in error for a wide range of deflections and shell sizes.

In the case that a toroidal shell is not symmetrically deformed, a correction procedure has been developed⁵. A "worst case" for non-symmetric loading is taken as that orientation where the load is applied by two horizontal surfaces acting as shown on the torus of Fig. 5(a). The energy absorbed is a minimum where $\psi = \frac{\pi}{2}$ (Fig. 5(a)) and a maximum when $\psi = 0$, the case of symmetric loading previously developed. When ψ takes on intermediate values, linear interpolation is used. The minimum energy absorption is related to the maximum energy absorption $(U_I + U_{II})$ by

$$U^{\min} = U_I \frac{2X}{\pi} \int_0^X \cos \left(\frac{\alpha x}{2X} \right) dx + U_{II} \frac{2X}{\pi} \quad (9)$$

Fig. 5. Geometric Model for Correction Procedure

where the notations are defined in Fig. 5(b). Tests performed by Rhoades^b show the correction procedure predicts energy absorptions with a range of -19% to +29%.

Proof Test of Model Cask with Toroidal Shell

On the basis of the foregoing theory, a steel torus welded from four ells of 3 in. long radius, light wall pipe was fitted to the untested end of the model cask-design II. Theoretically, this torus, symmetrically loaded should absorb more than 12,000 ft-lb of energy which is the sum of 2000 ft-lb "ring" energy and 10,000 ft-lb "dilatation" energy. With a corner drop, the theoretical energy was about 5800 ft-lb. Experimental data indicated these calculated values were 18% too low. For the 233-lb model cask that was dropped from 30 ft, about 7000 ft-lb needed to be absorbed. Hence, it appeared that the toroidal shell was capable of absorbing nearly all the required energy. This assembly was allowed to fall freely from a height of 31 ft. 6 in. with the mass center directly above the point of impact, as shown in Fig. 6. High speed motion pictures of the event indicate that virtually all of the kinetic energy was absorbed by the torus.

The photograph, Fig. 7, shows that the left-hand end of the cask, the end which was impacted on toroidal shell, is virtually undamaged. The right-hand end shows the damage sustained in a similar drop without the toroidal shell protection. The kinetic energy of the assembly was sacrificially absorbed by the toroidal shell, shown in Fig. 8. It is noted that the shell could have absorbed more energy than it did here, and the inherent energy absorption of the cask itself provides an extra factor of safety.

Fig. 6. Drop Configuration with Torus Attached

Fig. 7. Comparison of Protection from Torus (Left-End)
vs. no Torus (Right-End)

Fig. 8. Deformed Torus from Free-Drop Test

SUMMARY

The University of Tennessee tests on model casks show that failure of unsupported end closures is inevitable for the "worse-case" 30 ft. free drop--where the mass center is directly above the point of impact. All other orientations of drop simply produce less damage.

To protect against closure failure, the add-on toroidal shell is an efficient sacrificial energy absorber. An approximate analysis to estimate the energy-absorbing capability aids in the selection of the appropriate size torus for specific applications.

ACKNOWLEDGEMENT

This paper is based on the Master's thesis research done in the Department of Engineering Mechanics, University of Tennessee, Knoxville, by C. T. Welch, F. E. Shadid, A. Estep, and G. L. Rhoades, Jr.; this work is gratefully acknowledged. Also, the major portion of the work was supported by the Oak Ridge National Laboratory through contract.

REFERENCES

1. F. E. Shadid, Analytical Investigation of Procedures Proposed for the Design of a Shipping Cask for Radioactive Materials, Unpublished Master's Thesis, University of Tennessee, Knoxville, Dec., 1967.
2. C. T. Welch, An Experimental Investigation of the Effect of Impact on a Cylindrical Container of Lead and Steel Construction, Unpublished Master's Thesis, University of Tennessee, Knoxville, Dec., 1967.
3. A. Estep, L. R. Shobe, and B. R. Dewey, Energy Absorption Parameters in Steel Rings, University of Tennessee, Knoxville, Report EM 69-1, Mar., 1969.
4. S. P. Timoshenko, Strength of Materials, Part II, Advanced Theory and Problems, Third Ed., Van Nostrand, 1956.
5. G. L. Rhoades, Jr., B. R. Dewey, and L. R. Shobe, Energy Absorption in Toroidal Shells Undergoing Large Deformations, University of Tennessee, Knoxville, Report EM 70-1, Jan., 1970.