Patram '80

6th International Symposium Packaging and Transportation of Radioactive Materials

6

November 10-14, 1980 – Berlin (West) Federal Republic of Germany

DO NOT MICROFILM COVER

INTERNATIONAL CONGRESS CENTER BERLIN (WEST)

organized by Bundesanstalt für Materialprüfung (BAM) and COC-Kongressorganisation GmbH

II

Organization

Bundesanstalt für Materialprüfung (BAM)
COC-Kongressorganisation GmbH

The Honorary Committee

Prof. Dr. G.W. Becker, Chairman
Präsident der Bundesanstalt für Materialprüfung (BAM), FR Germany

Dr. S. Eklund, Director General, International Atomic Energy Agency, Austria

Prof. Dr. Ing. D. Kind, Präsident der Physikalisch-Technischen Bundesanstalt, FR Germany

S. Meyers, Program Director, Department of Energy, USA

Dipl.-Ing. W. Sahl, Ministerialdirektor beim Bundesministerium des Innern, FR Germany

Dr. G.H. Scheuten, Vorsitzender des Wirtschaftsverbandes Kernbrennstoff-Kreislauf e.V., FR Germany

J. Schlegel, Senatsdirektor beim Senator für Wirtschaft und Verkehr, Senat von Berlin

The Scientific Committee

Dipl.-Ing B. Schulz-Forberg, Chairman Bundesanstalt f. Materialprüfung (BAM) FR Germany

R.F. Barker, IAEA, Austria

M. Blechschmidt, FR Germany

Dr. R. Christ, FR Germany

A. Cricchio, CEC, Belgium

P.I. Ek, Sweden

S. Fukuda, Japan

E.P. Goldfinch, United Kingdom

M.R. Grenier, France

Dr. K. Janberg, FR Germany

Dr. T. Musialowicz, Poland

A.H. Partridge, United Kingdom

S. Piermattei, Italy

R.B. Pope, USA

R.R. Rawl, USA

Dr. G. Wehner, FR Germany

Dr. K. Werner, FR Germany

The International Organizing Committee

Prof. Dr. R. Neider, Chairman Bundesanstalt f. Materialprüfung (BAM) FR Germany

R.F. Barker, IAEA, Austria

J.T. Daniels, United Kingdom

E.C. Hardin, USA

Dr. W. Kolb, FR Germany

L.B. Shappert, USA

Dr. Y. Sousselier, France

Dr. G. Wehner, FR Germany

Dr. K. Werner, FR Germany

Symposium Secretariat

Dipl.-Ing. H.W. Hübner, Miss F.E. Kirk, U. Dachroth Bundesanstalt f. Materialprüfung (BAM) FR Germany

W. Syborg, COC-Kongress-organisation GmbH, FR Germany

Sponsors

American Nuclear Society, USA

Bleiwerk Goslar KG, Goslar

Bundesministerium des Innern, Bonn

Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen mbH, Hannover

Gesellschaft für Nuklear-Service mbH, Essen

Kerntechnische Gesellschaft, Bonn

Kraftwerk Union AG, Frankfurt/Main

Reederei und Spedition „Braunkohle", Wesseling

Rheinisch-Westfälische Elektrizitätsgesellschaft (RWE), Essen

Sandia Laboratories, Albuquerque/USA

Der Senat von Berlin

Transnuklear GmbH, Hanau

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.

CONF-801115--Vol.2

6th INTERNATIONAL SYMPOSIUM
PACKAGING AND TRANSPORTATION
OF RADIOACTIVE MATERIALS

PATRAM '80

MASTER

PROCEEDINGS

CONF-801115--Vol.2

DE86 012664

VOLUME II

INTERNATIONAL CONGRESS CENTER

BERLIN (WEST) NOVEMBER 10-14, 1980

Editor: H.W. Hübner, BAM Berlin

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

FOREWORD

The Sixth International Symposium on the Packaging and Transportation of Radioactive Materials was held November 10—-14, 1980 in Berlin (West), Federal Republic of Germany, the first in this series of symposia to be held outside the United States of America. It was an honour for the "Bundesanstalt für Materialprüfung (BAM)" to act as organiser.

PATRAM '80 gave an international scientific and technical circle of experts from research institutes, firms and governmental institutions the opportunity to exchange their experience in transporting nuclear fuel and other radioactive materials together with new developments in packaging and testing techniques as well as the use of spent fuel casks for intermediate storage purposes. In addition, we also tried to make those efforts made in the area of transportation more understandable to the public.

In these proceedings you will find 175 papers, the dinner speech, the speeches made at the public evening, the panel discussion and of course the discussions following the presentations themselves.'

PATRAM '80 was considered an important step forward in the international exchange of ideas, also in view of the 1983 revision of the IAEA Regulations. A total of 498 participants registered for the symposium from 21 countries.

B. Schulz-Forberg
Chairman
Scientific Committee

R. Neider
Chairman
International Organizing Committee

TABLE OF CONTENTS

Paper No.	Title, Author(s)	Page

VOLUME I

Plenary Session
(Chairmen: B. Schulz-Forberg, BAM, FR Germany / W.A. Brobst, TTE, USA)

002 Safety in Transportation of Radioactive Materials
R. F. Barker 1

[illegible]14 Evolution of the Concept of Risk in Radioactive Materials Transport
Y. Sousselier, G. Cohendy, J. Hamard 9

099 Safety Philosophy and Acceptable Risk
G. Wehner 15

047 Recent Radiological Protection Considerations in the Transport of Radioactive Materials
K. B. Shaw, S. A. Beach 31

190 Transportation of Nuclear Materials: The Nuclear Focus of the 80's
S. Meyers 40

Session I — Regulations, Licensing and Standards
(Chairmen: A.H. Partridge, DOE, United Kingdom / K.E. Wieser, BAM, FR Germany)

070 Unilateral and Multilateral Approvals
Y. Sousselier, M. Grenier 43

005 An Alternative Appoach to the A_1/A_2 System for Determining Package Contents Limits and Permitted Releases of Radioactivity from Transport Packages
H. F. Macdonald, E. P. Goldfinch 48

200 Experience in the NRC Program for Inspection and Enforcement of Nuclear Waste Transportation in the USA
A. W. Grella 57

041 Radiation Level for Low Specific Activity Materials in Compact Stacks
U. Lauterbach 66

136 Low Specific Activity Materials — A New Look at an Old Concept
R. R. Rawl 72

[illegible]2 Design Rules for Containment Systems for Nucelar Spent Fuel and High-Level Waste Transport Packagings
R. E. Nickell, K. Goldmann, R. H. Jones 82

071 Transport of Radioactive Materials by Post. A New Approach for Rule-Making
K. Ulbak, A. Leinio, H. Aamlid, T. Bennerstedt 87

103 A Modal Study of Transportation Safety
W. R. Lahs 90

Paper No.	Title, Author(s)	Page
	Session II — LMFBR Systems Concepts	
	(Chairmen: L.B. Shappert, ORNL, USA / H. Keese, TNF, FR Germany)	
080	Design of a Transport Packaging for Plutonium Containing Unirradiated FBR Fuel Elements M. Conti, G. Forasassi, G. Tomassetti	99
051	Design of a Container for Fresh LMFBR Fuel Assemblies Type SNR 300 H. Geiger, K. Schneider, U. Kunzendorf	105
205	T-3, A Packaging and Transportation System for Irradiated US Breeder Reactor Experiments L. J. Hansen, J. D. Berg, J. D. Berger, E. T. Weber, J. J. Krupar	11[illegible]
059	Cask Design for Spent LMFBR Fuel Elements Type SNR 300 R. Christ, H. Diem	120
034	A UK Feasibility Study for an LMFBR Spent Fuel Transport Flask A. R. Baker, A. Emmerton, J. O. Pounder	126
039	Status of LMFBR Spent Fuel Cask Development in Japan Y. Ishida, S. Kikuchi, K. Konno	135
063	Special Spent Fuel Assemblies Packaging Method for the Creys-Malville Power Plant J. Geffroy, M. Dobremelle, J. J. Joly, MM Moulin, Vivien, Lemercier, Costa, Michoux, Portal	143
120	Assessment of LMFBR Spent Fuel Shipping Cask Concepts for the CRBRP and the US Conceptual Design Study R. B. Pope, J. M. Ortman, R. G. Eakes, W. B. Leisher, S. A. Dupree	152
	Session III — Risk/Safety Assessment I	
	(Chairmen: Y. Sousselier, CEA/CEN, France / K. Schneider, TNF, FR Germany)	
116	RADTRAN II — A Computerized Model for Risk Analysis of Transportation of Radioactive Material J. M. *Taylor*, S. L. *Daniel*, B. E. *Biringer*	161
094	Radioactive Waste Transportation Risk U. Tveten	168
192	Tansportation Risks in the US Nuclear Fuel Cycle R. E. Rhoads, W. B. Andrews	
004	Individual and Collective Doses Arising in the Transport of Irradiated Nuclear Fuels H. F. Macdonald	189
158	Relative Consequences of Transporting Hazardous Materials R. T. Reese, R. R. Fullwood, W. R. Rhyne, J. A. Simmons	197

Paper No.	Title, Author(s)	Page
193	Application of ALARA Principles to Shipment of Spent Nuclear Fuel D. W. Murphy, J. Greenborg, L. W. Brackenbush, R. A. Burnett, J. R. Lewis, W. B. Andrews	206
152	An ALARA Assessment of Spent Fuel and Nuclear Waste Transportation Systems S. H. Sutherland	211
081	Assessment of Risks Involved in Handling and Transportation of Spent Fuel from Loviisa Nuclear Power Station H. T. Kaikkonen, E. Rämö, S. Vilkamo, S. Vuori	219
196	A Method to Develop Data Supporting Consequence Analyses of Transporting Nuclear Materials in the United States R. T. Reese, R. P. Sandoval	227
	Session IV — Systems and Package Design	
	(Chairmen: S. Williamson, BNFL, United Kingdom / H. Baatz, GNS, FR Germany)	
217	CASTOR Casks: A New System for Transport and Storage of Spent Fuel Elements H. Baatz, K. Janberg, D. Rittscher	235
061	Cast Ferritic Stainless Steel Cask for the Shipment of Irradiated Fuel Elements H. Diem, R. Laug, A. Lührmann	240
060	Suitability of a Special High-Density Conrete as Material for Spent Fuel Shipping and Storage Casks R. Laug, A. Lührmann	249
208	HLDW Transportation System Design Basis and Concepts N. W. Johanson, R. M. Burgoyne, M. R. Hoehne	255
140	TRU Waste Transportation Package Development R. G. Eakes, G. H. Lamoreaux, L. E. Romesberg, S. H. Sutherland, T. A. Duffey	267
019	A New Concept of Packaging, Transport and Disposal of Intermediate Level Waste in the United Kingdom B. Wright, R. G. Freathy, M. S. T. Price	274
033	Concepts for Packaging Radioactive Waste Generated by Decommissioning of Nuclear Power Reactors H.-J. Engelmann, V. Duill, R. Köster, P. Petrasch	283
125	Development of a Versatile Department of Transportation Specification 7A Metal Waste Container for Interim and Terminal Storage/Burial R. K. Blauvelt, D. A. Edling, J. W. Doty, R. C. Humphrey	291
121	Developmental Testing of Partially Volatile Neutron Shields for High-Performance Shipping Casks R. B. Pope, G. C. Allen, H. J. Rack, B. J. Joseph, S. A. Dupree	299

Paper No.	Title, Author(s)	Page
Session V — US Institutional Issues		
(Chairmen: C.E. MacDonald, NRC, USA / R.F. Barker, IAEA, Austria)		
159	Insitutional Issues Affecting Transportation of Nuclear Materials R. T. Reese, R. E. Luna	309
160	Federal Legal Constraints on State and Local Regulation of Radioactive Materials Transportation R. T. Reese, F. A. Morris, B. W. Welles	316
224	US Intergovernmental and State Responsibilities and Carrier Participants in the Routing of Nuclear Materials J. DiStefano	32
131	Assessment of Institutional and Technical Measures to Reduce the Risk of Potential Accidents and Sabotage of Nuclear Spent Fuel during Transportation J. M. Elliott, K. E. L. Ross	334
175	The Oak Ridge Spent Fuel Logistics Model D. S. Joy, B. J. Hudson	344
172	*Motor Carrier Routing Using the COMPU.MAP Program* R. D. Seagren, B. J. Hudson, D. S. Joy	351
173	Predicting Routes of Radioactive Wastes Moved on the US Railroad System E. L. Hillsman, P. E. Johnson, B. E. Peterson	359
114	An Economic Analysis of the Transport of Radioactive Materials in LWR and LMFBR Fuel Cycles in the United States S. A. Dupree, R. B. Pope, L. C. O'Malley, D. A. Hankins	367
146	US Cask Requirements and Industry Capability Survey G. C. Allen, R. B. Pope, J. W. Cashwell, D. S. Joy, L. B. Shappert, W. R. Menchen	374
Session VI — Risk/Safety Assessment II		
(Chairmen: J.M. Rolland, AEC, Australia / G. Wehner, BMI, FR Germany)		
053	Transport of Spent Light Water Reactor Fuel by Goods Train — Estimation of Potential Release in the Event of Accidents W. Anspach, K. Schneider, A. Baran	383
095	Safety Analysis on Transportation of Radioactive Materials by Truck in Japan T. Nagakura, H. Takano, S. Takagi, Y. Maki, H. Shikata, Y. Ichikawa, S. Fukuda	392
054	Environmental Impact of Plutonium Transport in the Event of Accidents within the European Community — Estimation of Potential Releases K. Schneider, W. Anspach	401
040	Risk Assessment for the Transportation of PuO_2, UF_6 and Spent Fuel on Roads in the Federal Republic of Germany S. Hartwig, D. Flothmann, W. Heudorfer, E. Krieger, G. Roos, G. Schnatz, G. Würdemann	410

Paper No.	Title, Author(s)	Page
202	Risk Assessment Applied to Transportation of Tritiated Heavy Water G. J. Dicke	419
027	Radioactive Materials Packages and the British Railway Accident Environment A. R. Taig	428

Session VII — Leakage, Leakrate and Seals
(Chairmen: C.B.G. Taylor, The Radiochemical Centre, United Kingdom / R. Trapp, BAM, FR Germany)

[illegible]7	Importance of Leak Flow Mode Assumptions for Interpretation of Package Leaktightness Tests and Influence of Test Conditions on Conclusions B. P. Kirchner, P. T. Blum	437
089	Leak Testing and Activity Leakage Rate Evaluation — Practical Experiences, First Approaches to some Systemization and Outstanding Problems H. Kowalewsky	446
006	An Assessment of the Containment of the UK 250 Litre Plutonium Nitrate Package R. S. Butterfield	454
065	Leakage of Radioactive Powders from Containers W. D. Curren, R. D. Bond	463
088	Investigation of Gas Leakage from Sealing Constructions at Containments for Radioactive Materials K. Heumos, H. Kowalewsky, H.-P. Weise	472
126	Vacuum Drying and Leakage Testing of Irradiated Fuel Packagings K. Goldmann, H. Bernard	479
180	Pressurized Powder Releases through Micro-Openings in Faulted Containers J. Mishima, S. L. Sutter, P. C. Owzarski, L. C. Schwendiman	488

Poster Session A
(Chairman: M. Bauschke, BAM, FR Germany)

001	Conceptual Design of a ^{85}Kr Transportation System H. Brücher, D. Niephaus, O. Nommensen	495
008	A Handy Calculation Method of Transport Index and Radiation Spectrum outside of Cask H. Yamakoshi, M. Nakata, K. Ueki, A. Sekiguchi	501
010	Impact Test of Truck with Nuclear Fuel Shipping Cask S. Kikuchi, M. Kubo	512
015	Studies of Physical Properties of Plutonium Dioxide Powders C. Cuillerdier, M. Germain	517

Paper No.	Title, Author(s)	Page
021	Implications of the New Limits for Intake (ICRP 26 and ICRP 30) on the A_1 and A_2 Values F. Lange, W. Müller	526
026	TRIP — A Computer Program Using Detailed Route Modelling to Assess Comparative Risks from Overland Transport Operations C. A. Harter, R. J. Williams, P. M. Quirk, A. C. Mackenzie, B. Y. Underwood	531
031	Experimental and Calculational Analysis of Radiation Dose in a Spent Fuel Shipping Cask A. Sekiguchi, M. Nakazawa, K. Takeuchi, H. Yamakoshi, K. Ueki, T. Yamada, E. Kudo, S. Fukuda	5[illegible]
043	Confirmation Tests of PWR Surveillance Capsule Shipping Container (Comparative Study of Analyses and Tests by IAEA Regulations) Y. Fujise, N. Tomita, M. Ohashi, J. Imamasa, M. Irino	550
077	Radiation Doses from Air Transport of Radioactive Materials in Italy S. Piermattei, I. De Marco, S. Mancioppi, G. Scarpa, M. Moscati	558
128	Puncture Resistance of Type B Transport Systems H. J. Rack, M. C. Cheresh	568
151	Performance Testing of Large Metallic Seals W. B. Leisher, A. A. Trujillo	575
Session VIII — Operations and Systems Experience I (Chairmen: A.W. Grella, NRC, USA / H. Geiser, GNS, FR Germany)		
035	Experience of European LWR Irradiated Fuel Transport H. W. Curtis	583
085	Current Status of Nuclear Materials Transportation in Japan S. Aoki, S. Fukuda, M. Nomura, A. Yuki, T. Takeda	589
133	Management of Radioactive Materials Transportation Safety for the US Department of Energy — Oak Ridge Operations Office W. A. Pryor	597
216	Experience of a Manufacturer in Quality Assurance H. Bruel	605
046	Twelve Years Experience with the Transport of Radioactive Wastes in a Salt Mine O. Opp, K. Thielemann, H. Kleimann	613
207	Experience with Dry Cask Shipping of HTGR Spent Fuel R. M. Burgoyne, R. L. Moore, J. W. Gahm	625
171	Cask-Facility Interfaces in Spent Fuel Shipping C. C. Hoffmann, J. M. Viebrock	636

Paper No.	Title, Author(s)	Page
056	Experience in the International Shipment of Irradiated MTR Fuel Elements by Truck, Ship and Aircraft G. Armbrust, T. Staake	642
225	Airborne Contamination Released during Underwater Unloading of a Failed PWR Spent Fuel Assembly R. W. Klingensmith	646

Session IX — Manufacturing Processes and Materials
(Chairmen: A. Onodera, Hitachi Ltd., Japan / F. Schilling, Siempelkamp AG, FR Germany)

226	Fracture Mechanical Properties in Heavy Sections of Ferritic Nodular Iron Castings J. M. Motz	655
127	Post Test Evaluation of a Fire Tested Rail Spent Fuel Cask H. J. Rack, H. R. Yoshimura	666
203	Cask Manufacturing Methods and Quality Assurance Problems J. H. Riddle, H. D. Cross, A. A. Trujillo, R. B. Pope, H. J. Rack	673
219	Production of CASTOR Casks for the Transportation and Storage of Spent Fuel Element Bundles — Management and Quality Assurance D. Methling, F. J. Treffner, F. Schilling	682
130	Development of a Non-Volatile Boron Carbide-Copper Cermet Neutron Shield for High Performance Shipping Casks J. E. Smugeresky, H. J. Rack, G. B. Brassell	687

Session X — Quality Assurance and Maintenance
(Chairmen: D.S. Joy, ORNL, USA / P. Schmidt, TÜV, FR Germany)

100	Setting-up of Quality Assurance in France G. Cohendy, P. Warniez	697
086	The Quality Assurance Program in Small Series Construction of UF_6 Packages and the Role of Competent Authority C. Faloci, A. Orsini, G. Sofo	703
084	Quality Assurance Measures for RAM-Packagings in the Federal Republic of Germany B. Droste, H. W. Hübner, K. E. Wieser	708
093	Servicing and Maintenance of Uranium Hexafluoride (UF_6) Transport Containers R. Peclier	715

Paper No.	Title, Author(s)	Page

VOLUME II

Session XI — Safeguards-Related Problems
(Chairmen: R.T. Reese, Sandia Labs., USA / L. Sütterlin, GRS, FR Germany)

183	Environmental Impacts of the Transportation of Radioactive Materials in Urban Areas N. C. Finley, J. M. Taylor, S. L. Daniel, D. M. Ericson	723
113	Transport of Radionuclides in Urban Environs N. A. Eisenberg	7
048	Protection and Security of Transports of Special Nuclear Materials (Including Measures for Transportation by Rail) U. Alter, D. Jungclaus	737
057	International Shipping Experience with Special Nuclear Material Demonstrated on a Specific Transport between the Federal Republic of Germany and Japan R. Schüler, J. Muskovszky	742
076	Security Measures for the Transportation of Radioactive Materials on the Road P. Erlenwein	747
122	It's a Mad, Mad, Mad World J. Mangusi	754
162	Fuel Response to Severe Transportation Accidents M. E. Balmert, K. D. Kok, P. Baybutt, S. Raghuram, L. M. Lowry, A. R. Rosenfield	762
214	Transportation of Nuclear Material by Sea and Inland Vessels Considered under Security Aspects G. Baier	771
215	Transport of Nuclear Material by Air Carriers V. Behrendt	778

Session XII — Operations and Systems Experience II
(Chairmen: P.T. Blum, TNP, France / R. Christ, TNF, FR Germany)

197	Experience in Shipping a Radioactively Contaminated PWR Steam Generator by Ocean Barge J. M. Taylor, V. F. FitzPatrick	7
052	Packaging and Transport of Bulky Waste from Nuclear Power Plants H.-G. Knackstedt, R. Christ, K. Schneider	800
181	Rockwell International's Packaging and Shipping Program at the Rocky Flats Plant J. D. McCarthy, D. M. Krieg	806

Paper No.	Title, Author(s)	Page
187	A Review of US Accident/Incident Experience Involving the Transportation of Radioactive Material (RAM) 1971-1980 J. D. McClure, E. L. Emerson	811
038	Transport of Radioactive Materials. A Review of the Damage to Packages from the Radiochemical Centre during Transport D. H. Lohmann	818
018	The Performance of Type A Packaging under Air Crash and Fire Accident Conditions A. Hadjiantoniou, J. Armiriotis, A. Zannos	826
012	On the Domestic Transportation by "Hinoura Maru" the Ship for the Exclusive Use of Spent Nuclear Fuel K. Kage, K. Kuwashima, H. Kajishima	833
016	Fabrication Experience of HZ-75T Spent Fuel Shipping Cask N. Niomura, A. Onodera, A. Nishikawa, R. Asano, M. Yamashita, S. Ozaki, Y. Hara	842
066	Contamination Studies on Pond-Loaded Flasks W. D. Curren, R. D. Bond	850
	Session XIII — Neutronics and Criticality	
	(Chairmen: F. Marcus, NKA Risö, Denmark / W. Thomas, GRS, FR Germany)	
029	The Use of the Monte Carlo Neutronics Code MONK for the Criticality Safety Assessment of Transport Packages G. Walker	857
042	The handling of Criticality Problems during the Approval of Fissile Class Package Design by the Physikalisch-Technische Bundesanstalt (PTB) H.-H. Schweer	866
069	Validation of the APOLLO-MORET Neutronic Codes on Critical Experimental Configurations Simulating the Shipping Casks for Light Water Fuels D. Haon, J. C. Manaranche, D. Mangin, L. Maubert, G. Poullot	872
068	Neutron Isolation of Packagings with a Borated Compound usually Used in France D. Haon, L. Maubert, C. Bochard	881
016	The Dose Rate Survey of Spent Fuel Packages P. T. Blum, P. Morin, S. Lazarevitch	887
	Session XIV — Plutonium Systems	
	(Chairmen: E. Tomachevsky, CEA/CEN, France / W.K. Müller, GRS, FR Germany)	
144	Air Transport Package for Plutonium Safeguards Samples SAND 80-1735 J. A. Andersen	897

Paper No.	Title, Author(s)	Page
024	Analysis of the Accident Environment and Container Response for the Air Transport of Small Type B Packages M. L. Brown	905
025	Summary of the Risk Assessment Made of the Transport of Plutonium Nitrate J. C. Chicken	914
050	EuPuC II — A Transport Package for Large Quantities of Plutonium Oxide Raw Materials I. Lafontaine, J. Draulans, G. Evrard, J. P. Fabry, L. U. B. Bredael	921
	Session XV — Intermediate Storage in Casks (Chairmen: J.T. Daniels, UKAEA, United Kingdom / M. Blechschmidt, PTB, FR Germany)	
055	Storage of Spent Fuel in Transport/Storage Casks H. Keese, W. Anspach, R. Christ, E. Schlich	931
036	Long-Term Storage Behaviour of Spent LWR Fuel M. Peehs, G. Kaspar, W. Jung, F. Schlemmer	939
221	1500 t-U Spent Fuel Element Storage Plant Ahaus H. Malmström, G. Schroeder, H. Klöpper, A. Lührmann	955
082	Type B Casks for the Intermediate Storage of Spent Fuel Elements in the Federal Republic of Germany J.-P. Masslowski, R. Trapp	961
	Session XVI — Operations and Systems Planning (Chairmen: S.F. Johnson, AB Atomenergi, Sweden / K. Janberg, GNS, FR Germany)	
189	The US Department of Energy's Nuclear Materials Transportation Technology Development Program R. B. Chitwood	967
169	Assuring People Safety in Handling Radioactive Packages C. B. Killian, R. Devlin	977
073	Centralized Management of Radioactive Materials Packagings J. Fradin	982
211	Program for the Transportation of Irradiated CANDU Fuel R. Kortright, S. J. Naqvi, D. W. Souther, J. F. Tanaka, R. W. Barnes, M. R. Matthias, P. K. M. Rao	9
153	Evaluation of Dry versus Wet Unloading of Spent Nuclear Fuel Shipping Casks G. C. Allen, R. W. Lambert, D. J. Larkin	1001
011	Sea Transport of Spent Fuel B. Gustafsson, T. Milchert	1008
155	Subseabed Disposal Transportation System G. C. Allen, M. E. Vernon, D. R. Anderson	1020

Paper No.	Title, Author(s)	Page
	Session XVII — Institutional Issues	
	(Chairmen: H.J. Rack, Sandia Labs., USA / W. Kolb, PTB, FR Germany)	
166	The Archipelago of Nuclear Material Export Licensing D. W. Harmon	1027
007	Administrative Procedures Concerning the Transport of Radioactive Materials S. Piermattei, H. Nacfaire, M. Roberti, C. Faloci	1033
185	Nuclear Material Shipment Study E. W. Shepherd	1041
184	The System of Insurance and Indemnity Coverage for Public Liability Associated with Nuclear Materials in the Course of Transportation in the United States O. F. Brown	1050
058	Liability and Insurance during International Transports of Nuclear Fuel U. E. Timm	1059
174	Transportation of Radioactive Materials: Legislative and Regulatory Information System C. S. Fore, M. M. Heiskell	1067
	Session XVIII — Structural and Thermal Evaluation I	
	(Chairmen: S. Aoki, Tokyo Institute of Technology, Japan / K.E. Wieser, BAM, FR Germany)	
092	Resistance to Worse-than-Prescribed Fire Conditions G. Cohendy, V. Stadnikoff, B. Duret	1073
147	Modelling of Pool Fire Environments Using Experimental Results of a Two-Hour Test of a Railcar/Cask System J. E. Hamann, D. E. Klein, R. B. Pope, H. R. Yoshimura	1081
135	Modelling the Normal Thermal Environment W. H. Lake	1090
030	Transport Fire Hazard and the Calculations of Package Response R. L. D. Young	1098
111	Validity of Scaling Shipping Casks Experiencing Large Deformations R. J. Burian	1105
095	Experience with Fuel Damage from Abnormal Conditions in Handling and Transport W. J. Bailey, D. C. Langstaff	1113
078	Calculations and Experimental Testing of a New Conception Impact Limiter for Irradiated Fuel Shipping Cask A. Olivieri, G. Pochini	1121

Paper No.	Title, Author(s)	Page
074	Test for Demonstrating the Reliability of Containers Used for (Transporting) Spent Fuel — Results of 9-Meter Drop Impact Tests Conducted on 50-Ton Class Spent Fuel Transporting Containers T. Nagakura, H. Takano, K. Sato, Y. Maeno, H. Shiojiri, N. Kishi, S. Aoyagi	1138

Poster Session B
(Chairman: M. Bauschke, BAM, FR Germany)

017	Mock-Up Test on Plutonium Nitrate Liquid Packaging T. Iwasaki, N. Niomura, A. Onodera, N. Mori, M. Furudera, R. Asano, T. Ebisu, I. Kawahara	11
023	Impact Resistance of Shipping Containers for Radioactive Materials under Free Fall Test T. Sasaki, Y. Kanae, S. Shimamura	1155
032	A Simple Calculation Technique Using the Solid-Angle Method for Radiation Dose Rate Distributions around Large Packages A. Sekiguchi, M. Nakazawa, K. Ueki, H. Yamakoshi, K. Takeuchi, Y. Kanai, T. Yamada, E. Kudo, S. Fukuda	1163
044	Shield Design for a New Generation of Shipping Casks W. Denk	1172
067	Thermal Tests Conception of Type B(U) Packages for Radioactive Liquids J. Chatelet, R. Seran, R. Leblais	1176
087	Acceleration and Strain Measurements Performed on Casks for Radioactive Materials under Impact Conditions K. E. Wieser, B. Schulz-Forberg, M. Jais	1183
108	Special Form Encapsulation for Radioactive Material Shipments from Oak Ridge National Laboratory R. W. Schaich	1192
134	Containment Seal Reliability Testing W. H. Lake	1199
138	Evaluation of Maximum Impact Damages on Type B(U) Shipping Packages W. Gibson, J. Kotler	1205
141	Contact-Handled Transuranic Transportation System Structural Analysis G. H. Lamoreaux, L. E. Romesberg, S.H. Sutherland, T. A. Duffey	12
142	One-Dimensional Thermal Response Modeling of a Transuranic Foamed Overpack System to a Fire K. E. Suchsland, K. C. Kwong, E. F. Fretter, R. D. Boyd, I. Auerbach, H. R. Yoshimura	1222
149	Extended Testing of a Modified 18B Plutonium Nitrate Shipping Container H. R. Yoshimura, R. B. Pope, W. B. Leisher, B. J. Joseph, B. Schulz-Forberg, H. W. Hübner	1230

Paper No.	Title, Author(s)	Page
176	*Fleet Servicing Facilities for Testing and Maintaining Rail and Truck Spent Fuel Transport Systems* C. D. Watson, B. J. Hudson, M. K. Preston, D. A. Keith, P. N. McCreery, W. Knox, E. M. Esterling, A. S. Lamprey, G. Wiedemann	1238

Session XIX — Extended Testing I
(Chairmen: G. Cohendy, CEA/CEN, France / V. Behrendt, Dornier System, FR Germany)

150	Relative Response of Type B Packagings to Regulatory and other Impact Test Environments J. D. McClure, H. R. Yoshimura, R. B. Pope, R. M. Jefferson, R. D. Seagren, L. B. Shappert	1247
090	Classification and Safety Margins of RAM Packagings — Results of a Research Programme H. W. Hübner, B. Schulz-Forberg	1254
009	Analysis of Hypothetical Crash of a Nuclear Fuel Transport Vehicle S. Kikuchi, M. Kubo	1259
139	Analysis and Model Testing of Super TigerType B Packaging in Accident Environments H. R. Yoshimura, L. E. Romesberg, R. A. May, B. J. Joseph	1267
148	Thermal Analysis of HNPF Spent Fuel Shipping Container in Torch Environments P. E. Eggers, H. L. Crawford, A. A. Trujillo, R. B. Pope, H. R. Yoshimura	1275

Session XX — Structural and Thermal Evaluation II
(Chairmen: G. Jack, AECB, Canada / H.-J. Engelmann, KfK, FR Germany)

210	CANDU Irradiated Fuel Transportation System — Dynamics Analysis T. Loewen, J. W. Forest, M. A. Elbestawi	1283
064	Measurement of Vibrational Effects on an Assembly of PWR Fuel Elements during Handling and Transport by Road and Rail J. P. Prulhière, F. Israel	1292
117	Dynamic Analysis to Establish Normal Shock and Vibration of Radioactive Material Shipping Packages S. R. Fields	1302
[illegible]1	Recent Developments in Puncture Analysis Techniques of Nuclear Shipping Container Components R. C. Shieh	1311
143	Analytical and Empirical Evaluation of Low-Level Waste Drum Response to Accident Environments, R. A. May, L. E. Romesberg, H. R. Yoshimura, W. E. Baker, J. C. Hokanson	1321

Paper No.	Title, Author(s)	Page
	Session XXI — Extended Testing II	
	(Chairmen: J.A. Andersen, Sandia Labs., USA / H. Kowalewsky, BAM, FR Germany)	
091	Air Accidents: Simulation Facilities B. Bazelaire	1329
220	Simulation of Aircraft Crashes on Spent Fuel Transport and Storage Casks F. Glaser	1338
062	Radioisotope Packages in Crush and Fire C. B. G. Taylor	1347
186	An Analysis of the Longitudinal Crush Environment for Small Radioactive Material Packages Involved in an Aircraft Crash J. D. McClure, W. F. Hartman	1355
199	Comparison of Potential Crush and Impact Environments C. M. Romander, J. D. Colton	1362
	Session XXII — Emergency Preparedness and Response	
	(Chairmen: R.R. Rawl, DOT, USA / P. Erlenwein, GRS, FR Germany)	
003	The Need for Emergency Preparedness in Transport of Radioactive Materials B. C. Bernardo	1371
145	Evaluating Dangerous Goods Emergency Response with Time/Loss Analyses E. T. Driver, L. Benner	1378
161	Emergency Response Capabilities Developed in the United States to Deal with Nuclear Materials Transportation Accidents S. G. Vandevender, R. T. Reese, A. H. Schilling	1387
079	Rescue Container to Accomodate Defective Transport Containers Filled with Uranium Hexafluoride H. Beyen	1395
168	TEAM — An Emergency Response Plan C. Brantley, E. DeMaria	1401
	PANEL SESSION	
	(Chairman: R. Neider, BAM, FR Germany)	
	Communication between Experts and the Public	1405
	PUBLIC EVENING	
	(Chairman: D. Aurich, BAM, FR Germany)	
	Does the Transportation of Radioactive Materials Nuclear Fuels via Public Transport Routes Put the Public at Risk?	
	Radiation Protection during the Transportation of Radioactive Materials R. Neider	1422
	The Transportation of Radioactive Materials in the Federal Republic of Germany B. Schulz-Forberg	1438
	DINNER SPEECH	
	The Nuclear Box W. A. Brobst	1455
	LIST OF AUTHORS	1465
	LIST OF PARTICIPANTS	1469

SESSION XI
SAFEGUARDS-RELATED PROBLEMS

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.

ENVIRONMENTAL IMPACTS OF THE TRANSPORTATION OF RADIOACTIVE MATERIALS IN URBAN AREAS

N. C. Finley, J. M. Taylor, S. L. Daniel, D. M. Ericson, Jr.
Sandia National Laboratories, Albuquerque, NM, USA

PURPOSE OF THIS ASSESSMENT

The Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes (NUREG 0170)[1] concluded that the risks associated with such transportation are low, although severe accidents in urban areas have the potential for large radiological and economic consequences. The present study[2] investigates radioactive material transport in urban areas and addresses the specific urban features which influence environmental impacts. These features include the geographic and demographic make-up, and vehicular, population and transportation patterns in the area.

Previous efforts have not identified a most important population exposure pathway or group. This assessment examines several pathways and a number of urban specific population groups to evaluate their relative significance. In addition, because different causative events contribute to the overall environmental impacts, this assessment addresses four of these: incident free transport, vehicular accidents, human errors, and sabotage or malevolent acts.

Not only does radioactive material transport produce radiological and economic consequences but also it can have social impacts. The objective of this study is to examine both the quantitative environmental impacts of radioactive material transport in urban areas and the more subjective social effects of this process. The social impacts assessment was performed by Battelle Human Affairs Research Centers, Seattle, Washington and their conclusions are only summarized here.

METHOD OF APPROACH

Using previous studies on radioactive material transport in the United States as a general foundation, we examined the urban area to determine which population groups were most likely to be exposed to materials in transit. After identifying these groups we constructed computer simulation models to assess the expected population exposure, from either incident free transport or vehicular accidents, to each group. The impacts from human errors or sabotage were examined using adaptations of the accident analyses.

A section of New York City encompassing portions of the boroughs of Manhattan, Brooklyn, and Queens was selected for the initial application of the computer models. Using the results of sensitivity analyses and insights gained from detailed examination of several classes of radioactive materials, the initial results were extended to a selected set of the top Standard Metropolitan Statistical Areas (SMSAs) in the United States. Twenty sites were selected, ranging from several of the densely populated areas on the eastern seaboard to the more spread out, but still heavily populated areas on the west coast.

In addition, several alternatives to current transport practice were examined. These include transport mode changes, increased accident damage resistance for shipping packages, rerouting of materials around the urban area, and a series of operational alternatives which would affect the population groups within the urban area most closely associated with the transport process (warehouse personnel, handlers, etc.).

The assessment contains detailed descriptions of the urban area studied, the simulation and health effects models used for evaluating radiological and economic impacts, and an evaluation of the social impacts of the transport of radioactive materials in urban areas. The material included in this assessment comes from a number of different sources, including the Task Group on Radioactive Material Transport in Urban Environs, established to assist the Sandia staff in developing the scope and details of the study. The results of the analyses are presented in terms of expected numbers of health effects per shipment year.

RESULTS

Incident Free Transport

For the selected New York City study area, the overall population dose from incident free transport of radioactive materials was found to be less than ten person rem per shipment year. This is equivalent to an expected number of latent cancer fatalities (LCF), for the total population at risk, of the order of 10^{-4} per shipment year. These estimates of LCF are a measure of the total expected additional cancers occurring over the entire period of time following a given year of shipping activity, not the number of cancers expected per year. The population groups receiving the majority of this dose were handlers, warehouse personnel, transport vehicle crew, pedestrians and people in vehicles. Other population groups were addressed although they did not contribute significantly (people in buildings, people in air terminals, etc.).

When the analysis was extended to other urban areas, the values for population dose ranged from 4 person rem ($\sim 10^{-4}$ LCF) per shipment year for the San Diego area, to a high of ~ 200 person rem ($\sim 10^{-3}$ LCF) for Newark. This range is not unexpected since there are widely different population and shipment patterns across the nation.

Vehicular Accidents

Radioactive materials are shipped in either nondispersible (or special) form (metals or doubly encapsulated sources) or in dispersible (or normal) form. Both types of materials were included in the shipment model developed for this analysis and required different methods for assessing the expected consequences of vehicular accidents involving them. When the results of the separate analyses were combined, the expected numbers of health effects per shipment year can be obtained. For the baseline accident analysis (limited New York City area) results are approximately 10^{-5} early morbidities, 10^{-3} latent cancer fatalities, and 10^{-3} genetic effects per shipment year. Economic risks from a year's shipment activity were calculated to be $\sim \$10^{6}$.

For other urban areas, the expected number of latent cancer fatalities per shipment year (shipment level for each urban area) ranges from $\sim 10^{-5}$ for San Diego to a high of ~ 0.2 for Chicago. These variations arise from several different sources including the population and shipment differences referred to

earlier. Here an additional factor is the specific character of the shipments in each urban area. Interpretation of the results of the generic extension must be made in view of the assumptions required for the analysis.

Human Errors or Deviations from Accepted Quality Assurance Practices

Human errors can occur at any stage during the transport sequence and pre-transport errors can cause impacts during transit. To evaluate these environmental impacts, the accident analysis model was altered with accident rates replaced by incident (or occurrence) rates for known incidents involving human errors. From this point the analysis paralleled that for vehicular accidents insofar as expected numbers of health effects per shipment year were calculated. In this case, no early effects were observed but approximately 10^{-3} latent cancer fatalities are expected per shipment year for the limited New York City area. Economic risk from human errors would total approximately $\$10^{4}$.

Extension to other urban areas resulted in a range of expected values from $\sim 10^{-6}$ latent cancer fatalities per shipment year for San Diego to ~ 0.4 latent cancer fatalities per shipment year for Chicago. This overall range is consistent with that for vehicular accidents.

Sabotage or Malevolent Acts

There are no data available with which to estimate the probability of occurrence of sabotage or malevolent acts. This precludes calculating risk values. Although a successful attack is considered unlikely, we have assumed that one had occurred and calculated the consequences in terms of expected numbers of health effects. Several different materials were examined in terms of the numbers of health effects to be expected given that the event occurred. Results of this analysis indicate that for a range of materials, expected numbers of early fatalities would be small, expected numbers of early morbidities would range from 1's to 100's and expected numbers of latent cancer fatalities would range from 1's to 1000's in the limited area considered. Economic effects are strongly dependent upon the amount of radioactive material released. Thus, there is unquestionably a wide range of environmental impacts potentially exceeding $1 billion for the largest radioactive material release.

Generic extension of the sabotage calculations is restricted by several factors. In general, it can be stated that the results for a particular event could be scaled, approximately linearly with population densities.[3] However, the scaling must be done with caution since there are several assumptions inherent in the calculation which affect the extrapolation.

Alternatives to Current Transport Practice

A number of alternatives to current transport practice are considered and the expected values for radiological consequences or risk were determined, by assuming that the alternative had been imposed for an entire year of shipment activity. In general, the effects of implementing these alternatives ranged from no change in the calculated risk to reduction in the expected numbers of early morbidities resulting from vehicular accidents. Decreases in the expected numbers of latent cancer fatalities by factors of two to five were also observed. In some cases, where implementation of the alternative involved change of shipment time, a range of values for expected radiological health effects resulted. In the sabotage cases,

a range of expected number of health effects was also observed, depending upon the particular alternative. For the alternative involving changes in transport mode, no change in the expected number of health effects was observed for the incident free and human errors situations although a significant increase in consequences from a sabotage event and a factor of two increase in the accident risk were observed. Rerouting materials outside the urban area resulted in decreased numbers of expected health effects from vehicular accidents, human errors and sabotage because of lower populations and vehicular traffic patterns in these areas. The treatment of rerouting involved comparisons of expected consequences of severe accidents, for a limited number of shipment types, thus no risk comparisons are made. The process of rerouting does not eliminate the environmental impacts but simply transfers them to other geographical areas.

Social Impacts of the Transport of Radioactive Materials in Urban Areas

The social impacts study performed by Battelle Human Affairs Research Centers,[4] concludes that transportation issues are a particularly visible component of the nuclear energy controversy. Although the number of transportation incidents which have occurred has been small and the consequences slight, the political and legal attention given transportation is likely to increase. These impacts may ultimately prove more significant in decisions regarding the transportation of radioactive materials than will strictly technical concerns.

REFERENCES

1. U. S. Nuclear Regualtory Commission, Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170, December 1977.

2. N. C. Finley, et al, Transportation of Radionuclides in Urban Environs: Draft Environmental Assessment, SAND79-0369, NUREG/CR-0743, July 1980.

3. D. M. Ericson, Jr., Observations on the Use of the Consequence Model CRAC for Small Non-Reactor Atmospheric Releases, SAND80-0366, June 1980.

4. C. Cluett, et al, Identification and Assessment of the Social Impacts of Transportation of Radioactive Materials in Urban Environments, SAND79-7032, NUREG/CR-0744, July 1980.

DISCUSSION

Question by R. Neider, FR Germany: You stated that the LCF's would decrease if the transportation routes went around the densely populated areas. But I think that in these cases the lengths of the transportation routes would increase and the probability of conventional accidents and deaths from these would increase too. Do they not compensate for the decrease of LCF's?

Answer: We have not evaluated this question directly but I believe your statement is correct. The increased travel distances resulting from re-routing materials around urban areas would contribute to possibly larger numbers of accidents and deaths from the transport itself, independent of the radioactive materials.

Question by B. W. Emmerson, United Kingdom: When estimating the economic risk associated with an accidental release of radioactive material, a number of factors must be taken into account. These will include the need for decontamination, for which costs in excess of $ 100 per m^2 may not be exceptional; and the third party liability claims arising from harm to persons for both mortalities and a possibly wide range of morbidities. The costs associated with these latter claims can be very high, particularly if, as in the USA, compensation claims can be accompanied by a large claim for punitive damages. Could you elaborate on the factors you took into account in your study for estimating the potential economic risk?

Answer: The factors considered in estimating potential economic risk include: costs for emergency response to the vehicular accidents, survey costs, evacuation costs and the costs of permanent land use denial. This latter cost includes the security personnel required to control the area once the perimeter has been secured by fencing or other means. We do not evaluate the legal or governmental costs associated with the accidents since these are essentially impossible to quantify. We recognise, however, that these punitive damages, as you say, may far outweigh even the very high land use denial costs calculated in our study, for severe transportaiton accidents or sabotage.

The details of our economic impacts model are available in an appendix to NUREG/CR-0743, the Draft Environmental Assessment of Transportation of Radioactive Materials in Urban Environs.

Question by A. L. Kaplan USA: Have the magnitudes of the health effects from the transportation of radioactive materials in urban areas, as determined by your study, been compared to the magnitudes of those same effects normally occurring in the population groups from these urban areas?

Answer: No, we have not directly evaluated or compared the incremental increase in health effects arising from radioactive material transport in urban areas as compared with their normal occurrence rates in the population.

TRANSPORT OF RADIONUCLIDES IN URBAN ENVIRONS

NORMAN A. EISENBERG
OFFICE OF STANDARDS DEVELOPMENT
U.S. NUCLEAR REGULATORY COMMISSION

PURPOSE AND SCOPE

The Nuclear Regulatory Commission (NRC) has been preparing a generic environmental impact statement (GEIS) on the transportation of radioactive material near, in, and through a large densely populated area in connection with a re-evaluation of present regulations regarding transportation of radioactive material. This effort is intended to: (1) determine the impacts, (2) identify regulatory alternatives and their costs and benefits, and (3) decide whether regulatory changes should be pursued further. This effort is not intended by itself to evaluate any specific rulemaking action or other regulatory activity. If the study indicates that additional regulatory actions may be appropriate, these will be pursued by specific studies of those actions.

Other generic transportation studies have treated urban areas in a general manner without detailed consideration of the special features of cities (Ref. 1) or have only considered certain types of shipments (Ref. 2). This generic environmental impact statement considers such unique facets of the urban setting as:

- high population densities;
- shielding effects of buildings under normal and accident conditions;
- the effect of local meteorology on accident consequences;
- the convergence of transportation routes in cities;
- diurnal variations in population;
- micrometeorological effects in urban areas;
- heavy pedestrian traffic.

Emphasis is placed on radiological health effects, but all environmental impacts, both radiological and nonradiological, are assessed. The study considers shipments through or near cities of all types, except weapons, weapon components, and shipments on military vehicles.

METHOD

The NRC staff selected Sandia Laboratories, Albuquerque, New Mexico, to perform an environmental assessment, upon which the GEIS is largely based. To assist in the development of an environmental assessment, Sandia Labs formed an Ad Hoc Task Group to supply comments, recommendations, and information. The Task Group meetings provided a forum for the interchange of ideas and information, and because the meetings were public, provided a vehicle for early public input. The broadly based Task Group is comprised of individuals from Federal, State, and local government as well as persons associated with industrial, academic, and environmental activist interests. A total of five Task Group meetings were held in various urban areas in the U.S. Table 1 lists dates and locations of these meetings. In addition to using the Task Group meetings to inform the public and to involve the public in the study, a Working Draft Assessment, SAND 77-1927 (Ref. 3), was issued in June 1978 to give the public advance knowledge of models and results, while work was still in

TABLE 1. Meetings of the Ad Hoc Task Group

Date	Location
September 20, 1976	New York, New York
November 16-17, 1976	Arlington, Virginia
March 24-30, 1977	Baltimore, Maryland
July 13-14, 1977	Houston, Texas
July 24-25, 1978	New York, New York

progress. As a result of Task Group and public input, the original scope of the environmental assessment has been broadened to include: a methodology decisively generic for all cities rather than concentrating mainly on New York City as originally envisioned, consideration of social impacts, an intensive treatment of sabotage and security in cities, and detailed consideration of the impacts arising from human error. The manner in which the study has been conducted, with a concerted effort to be open, to involve the public, and to obtain early input from the public and concerned parties, has hopefully yielded a more comprehensive, more accurate, better understood environmental study.

The GEIS is an evaluative document. It assembles and coordinates the technical input from the three principal supporting documents, critically evaluates their methods and results, and modifies their results when, in the judgment of the NRC staff, such modification is warranted. Finally, it identifies those aspects of the findings that are significant to regulatory decisionmaking. First, independent technical assessments of environmental impact were developed under contract to NRC. Second, the NRC staff has proceeded to independently evaluate these findings. The staff generally has accepted the findings of the independent technical studies. However, in some instances, a different view with some impact on regulatory thinking is held by the staff. The primary technical data basis was NUREG/CR-0743 (Ref. 4), the Draft Environmental Assessment (DEA). Two subcontractors to Sandia Labs produced two studies of social impacts. A team from Rice University and the University of Texas (Chad Gordon, principal author) authored NUREG/CR-0742 (Ref. 5). The Human Affairs Research Center, Battelle Memorial Institute, authored NUREG/CR-0744 (Ref. 6). This trio of documents, with most emphasis placed on NUREG/CR-0743, comprises the bulk of the technical basis for the GEIS.

For purposes of analysis, four causative event categories are used: incident-free transport and the three nonroutine conditions of transport, (1) vehicular accidents, (2) human error, and (3) malevolent acts. Under each causative event category the following radiological health impacts are quantitatively estimated: (1) early fatalities, (2) early morbidities, (3) latent cancer fatalities, and (4) genetic effects. Also, direct economic impacts are estimated. For incident-free transport, vehicular accidents, and human error, radiological risk is estimated. In addition, for vehicular accidents and malevolent acts, the consequences of severe incidents are estimated. Direct nonradiological impacts, e.g., impacts of exclusive use vehicles, are estimated. Indirect, i.e., social, impacts are difficult to quantify, but are given substantial consideration. They include psychological, sociological, organizational, legal, and political impacts.

Primary emphasis in the DEA is on presentation of impact estimates for a limited study area in New York City based on detailed modeling and data; however, approximate estimates for other cities (including the entirety of New York City) using less detailed data and models (obtained from the sensitivity analysis) provide an important perspective on how typical the limited study area estimate is and what the range of impacts for cities might be. It is believed that this detailed treatment of a part of New York City, coupled with an extensive sensitivity analysis, provides sufficient generic treatment in order to make regulatory decisions.

As shown schematically in Figure 1, the shipment of radioactive materials in urban environs interacts with the affected environment, namely people and property, to produce environmental impacts. The models developed to estimate the impacts depend on the causative event category, the population subset or property affected, and the shipment characteristics. The urban features are parameters in the various models. For example, the model developed to estimate impacts on people in buildings from incident-free transport uses information about the radioactive shipments (type of radiation, source strength, frequency, distance traveled, route), about the population affected (number of people in buildings, location), and about the urban features (average height of buildings, average shielding ability, i.e., wall material and thickness, average number of buildings per length of shipment travel). The model estimates the dose to people in buildings by processing the cited information. The resulting health effects are estimated using an additional model of radiation-dose-generated health effects. The impacts from each type of shipment are summed to obtain the overall impact; impacts are selectively summed to obtain impacts from shipping in each end-use category. Since each type of shipment requires a slightly different calculation using the same basic set of models, a computer is used to keep track of the many variables and parameters and to perform the necessary summing up of impacts.

The models developed to estimate impacts are quite general. When the data describing any particular circumstance are entered into the appropriate models, the impact estimate for that circumstance is obtained. Thus the impacts from a given shipment, a given city, and a given route within that city, may all be obtained. Total impacts of a given class are estimated by summing impacts over the appropriate variable, e.g., summing over shipments. The goal of this study, however, was to assess impacts generically, not for a specific set of data. In previous studies, the NRC and its predecessor, the Atomic Energy Commission (AEC), used the term "generic impact" to mean impacts typical of an activity or facility of a given class (e.g., a light water reactor), although a given facility may have special features that make its impacts atypical. In this study, the goal was to assess the impacts due to radioactive material transport typical of a large city.

A sensitivity analysis was used to obtain a generic assessment of impacts from the general models and site-specific data (Figure 2). The sensitivity analysis is described in detail in Appendix D to the DEA (Ref. 4). The sensitivity analysis is performed by (1) establishing an estimate of impacts using site-specific data (data for a portion of New York City was used) and the general models - this is designated the "base case," (2) changing the value of various input variables by moderate amounts and noting the resultant change in impacts estimated, (3) comparing the perturbed impact estimates to the base case estimates to deduce which variables are most important, and (4) developing simplified expressions (equations or models) for impacts in terms of the most significant input variables.

The sensitivity analysis takes a "black-box approach" to the environmental impact process. A set of input variables are related to a set of output variables by performing numerical experiments on a computerized set of models. The sensitivity analysis approach has the advantage of being simpler and less costly to use and

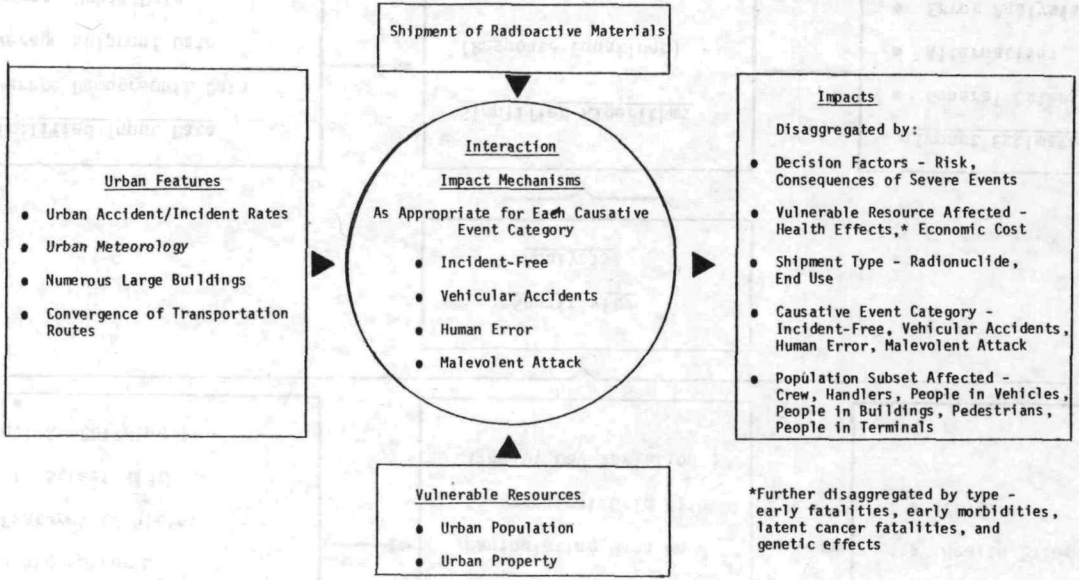


FIGURE 1: ENVIRONMENTAL IMPACT PROCESS FOR TRANSPORT OF RADIONUCLIDES IN URBAN ENVIRONS

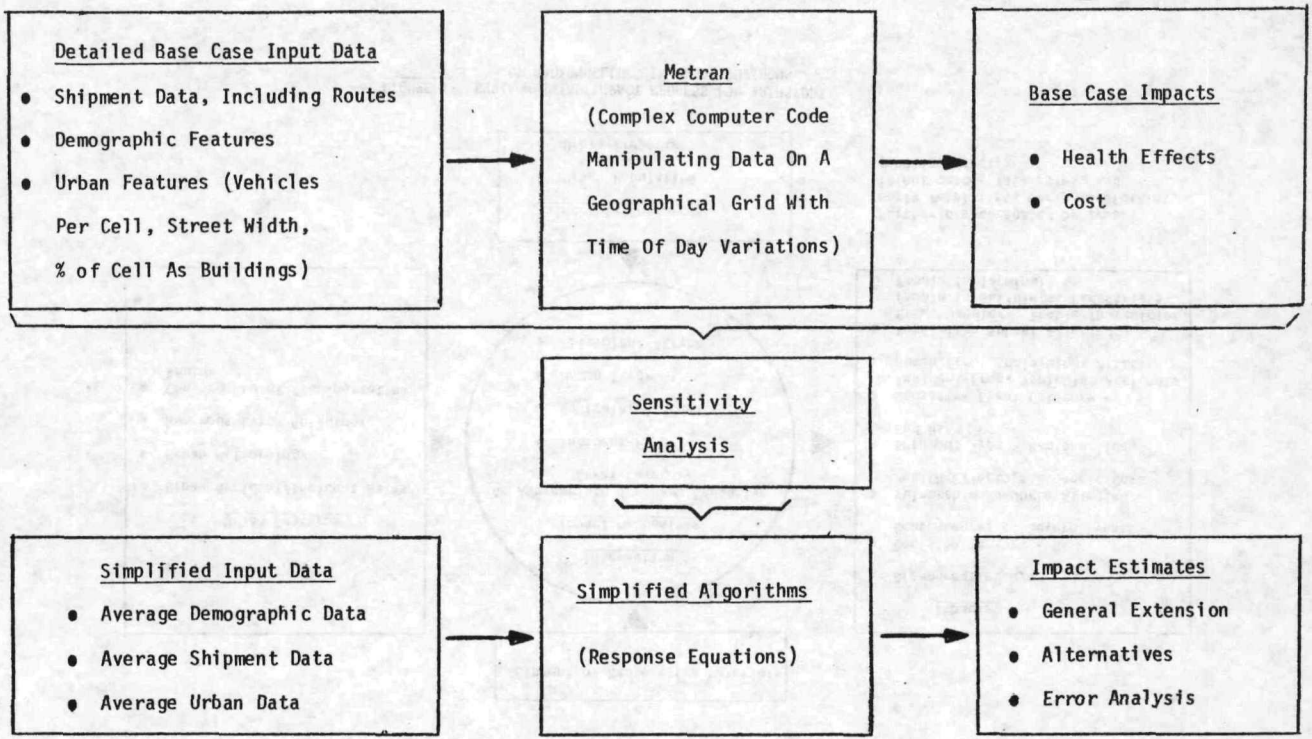


FIGURE 2: ANALYSIS METHODOLOGY USED IN THE DRAFT ENVIRONMENTAL ASSESSMENT

to gather input data; however, some precision is lost and analysis based on fundamentals of physics and biology is replaced by a phenomenological analysis.

In the DEA, the sensitivity analysis is the primary tool used to obtain results in three important areas: (1) alternatives; (2) generic extension, i.e., estimates of impacts in cities other than New York; (3) error analysis, i.e., estimates of the error in the impact estimates caused by error in the input data.

The sensitivity analysis is used as a bridge to go from the base case impact estimate for New York City to a new set of impact estimates resulting from various regulatory changes as they would be felt in New York City and to another new set of impact estimates for other cities under current regulatory practices. The DEA does not look at the combined perturbations of different cities and different regulations since the uncertainties in data and modeling would cause too small a degree of confidence in estimates for such conditions.

The success of the sensitivity analysis is not uniform over the four causative event categories. Comprehensive and fairly accurate treatment was achieved for incident-free impacts; moderate success was achieved with vehicular accidents and human error; but virtually no success was achieved in using the sensitivity analysis for impacts from malevolent attack.

FINDINGS

In most cases the estimates of impacts in this paper are larger than those of the DEA. The reasons for increasing the DEA estimates include: (1) not employing dose effectiveness factors for low dose rate radiation and (2) representing the city by a larger geographical area.

Radiation exposure of transport workers and of members of the general public will result from normal permissible radiation on the outside of packages in incident-free transport. This population exposure is estimated at 58 person-rem for New York City, producing no early fatalities or morbidities, but on a statistical basis may produce 0.007 latent cancer fatalities and 0.01 genetic effects resulting from the shipping activity of a single year. Compared to health effects from other causes, these effects from incident-free transport are considered to be small.

Radiation exposure of the public may occur in the vicinity of transportation accidents that involve radioactive material shipments. Looking at a broad spectrum of potential accidents and considering the probability and consequences of releases of radioactive material in those accidents, the average radiological risk is estimated to be small. Expected values of health effects resulting from the shipping activity of radioactive materials for a single year are 0.91 latent cancer fatalities, 0.4 genetic effects, 2.8×10^{-3} early morbidities, and no early fatalities for New York City. The estimated value of economic risk appears to be large per year of shipping activity, but is currently being re-evaluated. In the event of a low probability, very severe accident involving truck transport of spent reactor fuel, consequences are estimated to be 11 latent cancer fatalities, no early morbidities, no early fatalities, and $\$4 \times 10^{9}$ in decontamination costs. The probability of occurrence of such a very severe accident is estimated to be 1.0×10^{-9} per shipment or 1.5×10^{-8} per year (assuming 12 such shipments through the urban area per year) for a 63 km path through the densely populated region.

Excessive radiation exposure and release of radioactive material may result from human error and deviations from quality assurance procedures that affect packaging, labeling, handling, and stowage of radioactive materials in transport. The health effects resulting from human error are estimated to produce no short-term fatalities or morbidities, but on a statistical basis may produce 1.5×10^{-2}

latent cancer fatalities and 2.2×10^{-2} genetic effects resulting from the shipping activity of a single year.

Radiation exposure of the general public may occur as the result of a malevolent attack on a shipment of radioactive material. It does not appear possible to quantify the likelihood of a malevolent act, although such an act is perceived as very unlikely. Of course, given that a malevolent act is attempted, the degree to which the perpetrators' purposes are served or the amount of public harm resulting depends on a variety of difficult-to-quantify factors such as the skill of the attacker, his resources (technical, logistic, etc.), the response of the transporters, the response of authorities, the degree of inherent strength of the package, and the nature of the material transported. For shipments currently safeguarded by physical protection measures, the urban area appears to slightly enhance the level of protection offered by the safeguarding measures. At the time the DEA was begun spent fuel shipments were not safeguarded by special protective measures. Under those conditions it is estimated that a successful act of sabotage (perceived as difficult but possible to accomplish) using explosives on a truck shipment of spent fuel, might result in hundreds to thousands of latent cancer fatalities, tens to hundreds of early morbidities, a few of early fatalities, and several billion dollars in economic cost. In part because of these estimates of potential impacts, the NRC has instituted interim protective measures for certain spent fuel shipments.

Social impacts are very difficult to quantify. Some of the more significant qualitative findings are:

1. The general public is more concerned with the potential health and safety consequences of a tranportation incident than with the probability that such an incident will occur.
2. For a given magnitude of physical consequences, nuclear materials transportation has greater social impacts than hazardous materials transportation generally.
3. The potential for causing social impacts varies with causative event category. Malevolent act have the highest potential, vehicular accidents and human error have less potential, and incident-free transport has the least potential.
4. Many cities do not have their own radiological emergency response plans in place, and most State and Federal formal procedures do not differ with respect to the density of an area; thus, the adequacy of emergency response to transportation incidents in urban areas is questioned.

CONCLUSIONS AND RECOMMENDATIONS

Preliminary findings of this GEIS show that the health risk of transporting radioactive materials in cities is small. The economic risks are not necessarily small, but the estimates need to be refined. The economic consequences of a severe accident in a city could be large; the health consequences of such severe accidents, though serious, do not appear to be catastrophic. The risks from human error are estimated to be about the same as those from vehicular accidents. The successful sabotage of certain large shipments could produce significant health and economic consequences. The social impacts of radioactive material transport in cities are significant. Nonradiological impacts are not significant.

Because of the high consequences estimated from a successful sabotage attack on spent fuel, the NRC has instituted interim protective measures until additional research is completed. The NRC staff has not yet reached a consensus on recommendations; however, the staff is considering giving further study to the following subject areas that look promising for improving safety.

(1) Economic costs resulting from incidents involving radioactive materials in transit in cities and potential mitigating measures;
(2) Impacts due to human error;
(3) Making packages more accident resistant;
(4) Rerouting, in the context of alternative or synergistic safety improvements;
(5) Urban-specific features of radiological emergency response planning; and
(6) Contents released from packages subjected to sabotage.

REFERENCES

1. NUREG-0170. Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes. U.S. Nuclear Regulatory Commission, December 1977.

2. WASH-1238. Environmental Survey of Transportation of Radioactive Material to and from Nuclear Power Plants. U.S. Atomic Energy Commission, December 1972.

3. SAND 77-1927. The Transport of Radionuclides in Urban Environs: A Working Draft Assessment, Arthur R. DuCharme, et al., SANDIA LABORATORIES, May 1978.

4. NUREG/CR 0743. The Transport of Radionuclides in Urban Environs: Draft Environmental Assessment. Nancy C. Finley, et al., U.S. Nuclear Regulatory Commission, July 1980.

5. NUREG/CR 0742. Review and Integration of Existing Literature concerning Potential Social Impacts of Transportation of Radioactive Materials in Urban Areas, C. Cluett, et al., U.S. Nuclear Regulatory Commission, July 1980.

6. NUREG/CR 0744. Identification and Assessment of the Social Impacts of Transportation of Radioactive Materials in Urban Environments, Chad Gordon, et al., U.S. Nuclear Regulatory Commission, July 1980.

DISCUSSION

Question by C. B. G. Taylor, United Kingdom: Have you been able to compare the possibly severe consequences of a sabotage attack on a nuclear package with the consequences of attacks on other sensitive targets, for example large passenger aircraft?

Answer: The NRC staff has considered making such comparisons, but no published quantitative comparisons have yet been issued. I would like to point out that in my opinion such comparisons are not necessarily easy to make, because, although calculations of the public consequences of a successful sabotage attack can be compared in a rather straight forward fashion, comparisons of other potentially significant factors, such as the frequency of shipments, the quantities in a single shipment, the vulnerability of the shipping container to sabotage, the degree of concern of the public and government authorities regarding an act of sabotage directed at a given material, and attractiveness of such shipments to potential saboteurs, cannot be made in such a straightforward fashion. Further, the charter of the NRC is to protect public health and safety, regarding nuclear material. It is not clear to what degree the comparisons you speak of would be useful in fulfilling the NRC mission.

Question by K. B. Shaw, United Kingdom: Has the involvement of environmental groups in your discussions been fruitful? Have they been persuaded by technical data and have they produced their own studies?

Answer: Yes, I believe the involvement of persons associated with environmental activist groups in the Task Group meetings and the development of this study has been quite fruitful. First the Task Group meetings were working sessions that did not have the adversary nature of hearings or other formal proceedings. In that environment it was possible to engage in productive adversary proceedings. I believe that this dialogue has led to a greater understanding of different points of view by individuals on the various sides of this issue and in some cases has moderated certain points of view. By providing an early opportunity to express their views, to try to get their concerns addressed, and to follow the development of the study, thereby avoiding the undesirable and frequently objected to situation of being presented with a fait accompli. In my view this openness and accessibility has produced a more credible, better understood study. A number of environmental activists do appear to have been persuaded by technical information, if that information is presented dispassionately and originates from what are perceived to be credible sources (i.e. sources with no axe to grind). Some environmental activist groups have produced their own studies, but these are generally less comprehensive in scope and execution than the government sponsored studies and often are based on information published in government or industry reports.

PROTECTION AND SECURITY OF TRANSPORTS OF SPECIAL NUCLEAR MATERIALS (INCLUDING MEASURES FOR TRANSPORTATION BY RAIL)

by Dr. U. Alter and D. Jungclaus
Gesellschaft für Reaktorsicherheit, GRS, Cologne

In the Federal Republic of Germany, the "Act on the Peaceful Utilization of Atomic Energy and the Protection against its Hazards (Atomic Energy Act)" constitutes the legal basis for all regulatory measures concerning the protection and security of transports of radioactive materials. Sec. 4, para. (2), subpara. 5 of this Act requires the provision of the necessary protection of nuclear fuels in transit against disturbances and other interference by third parties.
The respective goal of protection and security is

- the protection of life, health and property against the hazards of nuclear energy and the adverse effects of ionizing radiation

In the Federal Republic of Germany, a noticeable increase in transports of nuclear fuels is expected during the next few years. This is why plans have been made to change the current practice by dividing responsibilities within the frame of protection and security between law enforcement agencies (police) and licensees.

The following division is considered desirable:

- the licensee will be responsible for the preventive security of its transports in accordance with the principle of originator responsibility;

- police forces will assume the protection of a transport in accordance with the principle of originator responsibility;

- police forces will assume the protection of a transport only in the event of actual danger.

The goals of protection and security for transports of nuclear fuels are:

- to prevent any diversion of fissile material for the construction of nuclear explosives;

- to prevent any release of radioactive materials on the spot or, following a diversion, at any other location.

The above-mentioned protection against disturbance by third parties is derived from the Atomic Energy Act of the Federal Republic of Germany.

These national requirements are supplemented by international recommendations such as

- INFCIRC 225 and

- the International Convention for the Physical Protection of Nuclear Materials.

The requirements arising out of the special conditions prevailing in the Federal Republic of Germany and the international recommendations and conventions were concretized in so-called catalogs of security measures.

These catalogs of security measures are an integral part of the national system of directions and regulations.

The following security philisophy was prepared with respect to the specific requirements contained in the catalogs of security measures:

- an attack should be indentified by reliable surveillance systems;

- the law enforcement agencies (police) shall be notified through redundant and reliable communication systems;

- the act should be delayed by means of barriers;

- protection of the transport will be assumed by the law enforcement agencies (police) in order to prevent the act.

To proceed along the lines of the above-mentioned security philosophy, both engineered measures and personnel as well as administrative and organizational measures are needed.

The different characteristics and requirements of the various types of transports have to be taken into account with respect to the transportation of nuclear fuels.

In principles, these various types include transport by road, rail, air and water.

As transports of nuclear fuels in the Federal Republic of Germany have so far been effected almost exclusively by road or rail, the following discussion will be restricted to transports by road and/or rail.

The different types of transport also involve different requirements with respect to both engineered measures and personnel, administrative and organizational measures which have therefore been laid down in special catalogs of security measures relating to the respective type of transport.

Detailed further requirements are contained in relevant specifications. Such specifications may e.g. deal with the requirements relating to the cask for nuclear fuels.

At present, work on the catalogs of security measures for transports by road and rail has advanced farthest. The catalog for road transport is being revised, whereas that for rail transport has been submitted as a draft version.

These catalogs of measures are intended to serve both regulatory bodies and applicants as a basis for the measures to be taken.

Under these aspects, the current catalog of measures is applied to transports of nuclear fuels by road.

Up to now, road transport has accounted for a greater amount of nuclear fuel shipments than rail transport. This has been due to the greater flexibility of trucks as compared with trains and to the relatively small weights - up to a maximum of 35 tons - of the fuel element transport casks for nuclear power plants of the class up to 1000 MW.

In the near future, however, greater numbers of transports of spent fuel elements have to be anticipated between 1300 MW nuclear power plants and the intermediate storage facilities to be established and/or the reprocessing plant at home or abroad. As fuel element casks of a total weight of 80 to 105 tons are provided for these transports, transport by rail will gain in importance.

In accordance with the respective catalog of measures to be applied, the transportation of nuclear fuels by rail and road requires surveillance measures ensuring the timely detection of possible attempts to divert or release nuclear fuels.

These surveillance measures include:

- surveillance of all transports by a transport control center;

- communication between the transport control center and the transport vehicle (and vice versa) at any time and from any place;

- times of transportation shall be as short as possible.

As far as transport by rail is concerned, it has been planned to transfer the functions of a transport control center to the district control office of the responsible directorate of the German Railroads.

The district control office is in charge of

- ensuring optimal railroad operation on the main lines of the railroad network; this is achieved by station masters at stations and signal towers along the lines who report the times of passage or stay of the individual trains to their district control office.

Thus, the district control office has a survey enabling it to provide for economical rail traffic involving high traveling speeds.

This means that when nuclear fuels are shipped by rail the district control office will have permanent information as to the section and/or station where the train under surveillance is.

For transports of nuclear fuels by road, a transport control center is being planned which is to be established and operated by the carriers.

Surveillance of the transport vehicle will be ensured by regular reports stating time and location of the vehicle via radio.

For the transportation of nuclear fuels by rail, the two-way railroad radio system of the German Railroads can be used as a means of communication. The technical characteristics of this two-way railroad radio system are as follows:

- Radio link characteristic from a transmitter/receiver mast along the line to the transmitter/receiver on the traction vehicle; otherwise, telephone lines to the district control office;

- Frequency of 470 mcps (corresponding to approx. 0.7 m wave length); as this frequency involves a quasi-optical propagation of the electromagnetic waves, antenna installations along the lines are provided at intervals of 2 - 10 km, depending on the prevailing topographic features.

  Tunnel sections are equipped with additional antennas.

- The radio communication will not be affected by the operation of electric traction vehicles.

For reasons of security, a communication system is required to provide communication with the transport control center from any place and at any time.

The two-way railroad system meets this requirement to a far-reaching degree, as its time and local availability is better than 95 %. In addition, it should be noted that a priority emergency call feature can be provided whose time and local availability can be assumed to be almost 100 %.

For transportation by rail, communication systems are being tested which will ensure a voice communication from the transport vehicle to the transport control center from any point along the route.

For both transport by rail and transport by road, the motorcar telephone system has been provided as a redundant communication system for the time being.

Short times of transport are desirable for all transports of nuclear fuels. In this context, the major aim is not an increase in maximum traveling speeds, but a minimization of operational stopping times such as they are needed to change an engine or the crew or escorts of a train.

In the case of rail transports, this means that special trains will have to be used which will run directly from a nuclear power plant siding to an intermediate storage facility, a reprocessing plant or a border station of the neighboring railroad administration thus avoiding the stays at switchyards which would be necessary for normal freight trains.

Thus, as in the case of door-to-door truck transports by road, short traveling times can also be reached by special trains on direct routes from consignors to consignees.

DISCUSSION

Question by R. Schüler, FR Germany: How can it be guaranteed that the communication system (relay stations) along railways in the FR Germany will not be rendered unoperational by sabotage during transportation?

Answer: The railway radio system can be made unoperational by external influence, e.g. by sabotage. Defects are signalled to the competent rail official. In case of defects, a redundant communication system must be available, e.g. a car telephone.

Question by K. Schneider, FR Germany: 1. You mentioned that you take into account economical considerations concerning rail transport. I think that this would be desirable for road transport too. But that is not at present the state of the art? 2. Will special trains be prescribed in the future and do they really enhance the security level?

Answer: 1. We do not take into account economical considerations, we only mentioned that the district control office has supervisory powers enabling it to provide economical rail traffic. 2. For the railway management the use of special trains will be the best way to optimise short travelling times.

Question by W. Heudorfer, FR Germany: Are two communication systems provided in the case of rail transport, i.e. radio system and car telephone system so that redundancy is provided?

Answer: Redundancy is assured by two communication systems prescribed in the security measures catalogue. For rail transport these are the train radio system and the car telephone.

INTERNATIONAL SHIPPING EXPERIENCE WITH SPECIAL NUCLEAR MATERIAL DEMONSTRATED ON A SPECIFIC TRANSPORT BETWEEN THE FEDERAL REPUBLIC OF GERMANY AND JAPAN

R. Schüler
J. Muskovszky

TRANSNUKLEAR GmbH , Hanau

One aspect of Transnuklear's worldwide activities is shipping special nuclear material (non irradiated high enriched uranium and plutonium). Because of this the Transnuklear company operates with a number of offices and correspondents etc. all over the world. This widespread organisation is necessary in order to be in a position to move these materials according to today's daily changing requirements, restrictions, local political aspects and physical protection measures.

It goes without saying that nobody can handle this job without knowing the international transport regulations such as Imco, ADR, IATA - RAR (Restricted Article Regulations) and IAEA regulations both the 1973 edition for Germany and Japan and the 1967 edition for the US, as required in our case. In addition to the above quite a number of restrictions are imposed by national laws and regulations such as customs laws in the countries of export, origin, transit and final destination plus the normal terms of an international business transaction, export license etc. Last but not least there is the Infcirc recommendation for physical protection for this kind of shipment and the resulting modifications to suit the needs of the countries involved.

The example chosen for this paper is based on a case where 45 kgs of high enriched uranium metal had to be shipped from Germany to Japan. On Sep. 6th, 1976 Transnuklear received the first information about this transport to Japan. 1976, 1977 and 1978 were needed to settle license problems and technical matters like purchase of the necessary containers.

At the initial stage of the project it was planned to do the job in the best way possible with a minimum of transfers: one shipment by truck from Hanau to the nearby Frankfurt airport and by air directly to Tokyo.

Due to immense difficulties between buyer, seller, intermediates, the Nuclear Regulatory Commision (NRC) requirements, the necessary "end use statement" and the M/B 10 form as well as physical protection the transport had to be delayed for month after month.

Finally at the beginning of 1979 Transnuklear received the o.k to proceed with the shipment.

The following facts then influenced the transport-schedules, routes etc.:

1.) air shipment from Frankfurt to Tokyo was not accepted by the Japanese authorities due to local problems at Narita airport Tokyo. All subsequent problems arose from this decision such as:

2.) Nine part shipments of 5 kgs uranium metal each instead of one shipment of 45 kgs due to security requirements imposed by Nuclear Regulatory Commision (NRC) regardless whether or not shipments crossed US territory on the way from the FRG to Japan.

3.) Neither the consignee nor the US authorities wanted shipments to go through communist countries.

4.) The consignee needed to have the material in Tokyo before 31.06.79 (the export license expiring date).

5.) The Nuclear Regulatory Commision (NRC) insisted that a second shipment can only leave after the first has arrived. Put more precisely this means shipping everything by sea the whole campaign would have lasted about 18 months based on the fact that all countries and ports in transit accept radioactive cargoes, the Suez Canal is open and ships are always in position. In other words entire shipments by sea are not feasible.

6.) Several US-airports were closed to high enriched uranium and plutonium shipments including New York and Chicago.

Therefore the first route had to be a combination of various transport systems and was planned as follows:

Frankfurt	- Anchorage	- Lufthansa airfreight
Anchorage	- Los Angeles	- Flying Tiger airfreight
Los Angeles	- Tokyo	- Ship (NYK-line, Japanese carrier)

Two shipments worked almost smoothly. The 3rd failed to leave Los Angeles on the scheduled ship due to the late arrival of the aircraft from Anchorage (bad weather). The ship left for Tokyo and the material had to be stored with a special permit. As the Japanese Ministry of Transport (MOT) requires 4 weeks prior notification of the shipment including information about the material, containers, ship and the captains name and his acceptance etc. the next ship could not take the cargo. The shipment had to wait for the next authorized ship. This automatically delayed the 4th shipment.

On preparing the 5th shipment, one of the 2 weekly "cargo only" flights to Anchorage was cancelled due to lack of fuel.

Other airlines flying directly to Tokyo rejected the cargo because according to the IATA-rules they had no cargo rights in Anchorage which means they are not allowed to handle cargo in Alaska en route from Europe to Japan. The shipping schedule had to be renewed. The 5th and 6th shipments arrived in Tokyo. Shortly after that all cargo flights to Anchorage were cancelled.

All investigations had to start afresh. We tried to fly the material directly to the US east coast from the Federal Republic of Germany - then by an internal American flight to the west coast and by sea from Los Angeles to Tokyo.

The only airport to accept material on the east coast which handled transatlantic cargo only flights from Germany was Boston, Mass. But there were no direct cargo flights from Boston to Los Angeles; only via Chicago but Chicago airport is closed to shipments of nuclear material (SNM).

It was theoretically possible, but how could we put the ends together? The Lufthansa cargo flight flies first New York and then to Boston, the truck needs four days from coast to coast, the vessel has a fixed schedule and will not wait for 5 kg shipments which no carrier is interested in transporting anyway.

The result: Lufthansa rerouted at extra cost the Boeing 747 (payload 100 t) for the 5 kgs to Boston first and then to New York. The truck then went coast to coast non stop (approx. 3 000 miles) in four days. The ship accepted the cargo and set sail from Los Angeles in time. 6th shipment left and arrived at Tokyo.

Shortly before the 7th shipment, Lufthansa cancelled the booking because of rerouting the Boeing 747 Boston - New York - Boston instead of New York - Boston. The crew claimed excess working hours which they are not allowed to do (30 min. in Boston). Fullstop.

The client is upset and nervous. Willing to pay for 2 cargo flights using a Boeing 707 (payload approx. 30 t) from Frankfurt to Los Angeles at 50 000 $ each.

We try to ship material to Los Angeles.

- via Canada - not possible
- via Mexico - not possible
- via Brazil - !!! not possible
- via Hong Kong by air and from Hong Kong to Tokyo by ship - not possible
- to go to Boston by sea from Bremerhaven. This would suit the 7th and 8th shipments but not the last (9th) if it is to arrive in Tokyo in time.

One last try with Lufthansa. The suggestion was made: would Lufthasa agree to Transnuklear hiring an extra crew to take over the Boeing 747 Jumbo at Boston airport and fly to New York and back to Boston to meet the working hours regulation? Lufthansa agreed, the client accepted the extra cost - which was in any case better than two charters for 2 shipments of 5 kgs to Los Angeles.

The 7th shipment runs "smoothly". The 8th shipment hit tremendous schedule problems. To catch the vessel in Los Angeles in time and to avoid storage during transit which was not allowed. The material had to leave on Friday from Frankfurt to Boston. Lufthansa accepted the booking and 5 minutes later cancelled it again. A much more important cargo had to go first (according to IATA regulations). What could be more important than a shipment of uranium metal? Live elephants from South Africa to Boston via Frankfurt - off goes the whole schedule. One last chance - a day later - Saturday, Lufthansa flew the same route but only to New York. The same procedure - would Lufthansa accept an extra stop at Boston and on what conditions. Result: Lufthansa accepted - the client accepted an off went the 8th shipment --- stop the aircraft operator is ill at Boston airport - no go? ridiculous - problem solved.

Off goes the 8th shipment - no! No police escort available for the transport between Hanau and Frankfurt airport due to a traffic accident. Problem settled - all regulations met.

The 9th shipment was also run the same way, problems? Of course not, just the same as the others!

You think the companies mentioned here were uncooperative, or the authorities problematical? - totally wrong. If it was not due to tremendous efforts by all parties involved and a good communication flow between the offices, the authorities and Transnuklear group nothing could have been moved. An enormous amount of all kinds of regulations make these transports "almost" impossible.

In isolated case - no! Possibly one of the most complicated but there are others with similar results. It becomes evident from this case that international shipments of special nuclear material become increasingly complicated - increase in safety and security? It is up to you to judge!

The only thing, and please allow to make this remark, which all parties and countries involved do still have in common is the use of the English language.

DISCUSSION

Question by B. W. Emmerson, United Kingdom: Can you say whether the difficulties you have encountered in the international transport of radioactive materials could have been reduced if all countries involved had been working to the same edition (presumably the 1973 edition) of the IAEA Transport Regulations?

It may be that a number of your difficulties have stemmed from the need to provide against potential third party liability claims associated with such shipments. Do you consider that there may be some substantial benefit in having an international third party liability convention for application in the field of transport of radioactive materials, on similar lines to the Paris Convention on Third Party Liability in the Field of Nuclear Energy?

Answer: Difficulties could have been reduced if all countries had been working to this same edition of the IAEA Transport Regulations. Problems did not arise from potential third party liability claims, but regarding your suggestion for a third party liability convention, yes there would be benefit from this if all countries which accepted the IAEA 1973 Regulations have an international third party liability without any supplementary national request. For example, the Paris convention is ratified by many countries but some of them require in any case national insurance company backing.

Question by C. B. G. Taylor, United Kingdom: You have described your problems very clearly but you have not suggested any solutions. What needs to be done, and who by?

Answer: We do not have any solutions.

Question by J. A. Andersen, USA: What package was the 5 kg uranium carried in, by air, from the FR Germany to Japan?

Answer: It was a USA type B container DOT-6M, 15 gal. size.

SECURITY MEASURES FOR THE TRANSPORTATION OF RADIOACTIVE MATERIALS ON THE ROAD

Dr. P. Erlenwein
Gesellschaft für Reaktorsicherheit (GRS) mbH
Glockengasse 2
5000 Köln 1

Pursuant to the German "Act on the Peaceful Utilization of Atomic Energy (Atomic Energy Act)", transports of radioactive materials have to be protected against disturbance and other interference by third parties.

The individual security measures to be taken and to ensure the required protection depend on the hazard potential of the materials transported. Transports will be classified into different security categories as a function of the size of the hazard potential involved.

Thus, the hazard potential of each transport has to be determined first. This determination of the hazard potential proceeds from the basis of the unsecured transport which is nevertheless provided with all necessary safety equipments.

The hazard potential is investigated under two separate and different aspects with a view to disturbance and other interference by third parties:

(1) The diversion of radioactive materials
(2) The release of radioactive materials on the spot.

The hazard potential with respect to diversion depends on the effects which can be achieved by an abuse of the material that has been diverted. Accordingly, the hazard potential is quantified, and the transport allocated to one of the security categories, under the aspect of diversion from the inventory of nuclear fuel. This classification is a function of the limits with respect to kind, amount and degree of enrichment of the nuclear fuel,and these limits, in turn, depend on the respective IAEA recommendations (Infcirc 225).

The hazard potential with respect to release depends on the possible effects of a release of radioactive materials of a transport to the environment as a result of an act of sabotage. The quantification of the hazard potential starts out from the identification of the sabotage scenario that may result in the maximum release of radioactive materials. This is followed by a calculation of the long-term and short-term radiation exposure of persons in the environment which may be caused by the dispersion of the released radioactive materials in the environment. Finally, the security category is determined on the basis of given limits.

Allocation of the transport will be to the highest security category determined under the aspects of diversion and release.

The requirements for the security measures to be applied within the individual categories are contained in a socalled catalog of security measures. The security requirements were selected in such a way that the effects of possible disturbances or other interference by third parties will be reduced to a measure that is comparable to that achieved by safety measures with respect to possible accidents in transit.

Considering these boundary conditions and the aims of protection termed

- prevention of the diversion of radioactive materials and
- prevention of release,

two requirements are bound to result with respect to engineered security measures which must be met by any type of transport that needs to be secured:

- the materials transported will have to be surrounded by barriers which will impede any potential attacker's access to the cargo;

- there must be a reliable surveillance and communication system that will ensure a sufficiently fast notification of the responsible agencies (e.g. transport control center) of any unauthorized interference with the transport.

The following is a description of engineered security requirements for such transports of nuclear fuels which belong to the highest security category.

Fig. 1 provides a survey of these requirements:

Major System Elements in Transportation Protection / Road

Transporter/ Container		Category I
	Propulsion	Protected
	Undercarriage	Protected
	Control cab	Special type
	Container/Casks	Special type
Communication Systems	Transport Control Center	Redundant communication systems
	Police office	Telephone
	Escorting Vehicle	e.g. Walkie-Talkies
Escorts	Crew	Several crew members Redundant communication systems
	Guards	Several guards Redundant communication systems
Transport Control Center	Communication Systems	Independant Telephone Connexions Radio Control Center
	Schedules and Lists	Transportation Schedule Alarm Schedule List of strong points

BARRIERS

The nuclear fuel to be transported must be provided with enclosures which, in their entirety, possess sufficient resistance (barrier function) against hidden or forcible attacks or attempts aiming at a diversion. These enclosures may be the container itself or suitable transport casks. In any case, the enclosures must be designed as mechanical barriers against chemical, thermal and mechanical devices as well as blasting charges. It must be ensured in this context that the time required to overcome these barriers (time of resistance) is longer than the time required to bring up security forces (delay time).

TRANSPORTER

The catalog of security measures requires the transporter to be designed and constructed in such a way that the essential equipment needed for protection of personnel and material is sufficiently protected against interference by third parties.

The driver's cab shall consist of two compartments: the cab as such for driver and co-driver and a separate and lockable compartment for the transport officer who has to escort the transport. The whole driver's cab shall be designed so as to be intrusion-retardant, bullet-retardant, fire-retardant and gas-tight. In addition, the vehicle shall be equipped with a feature enabling its complete immobilization in the event of an attack. It shall be impossible to lift this immobility during the delay time.

Any manipulation on security equipment shall cause an automatic alarm at the transport control center. This requirement alone justifies the necessity of a reliable communication and surveillance system.

SURVEILLANCE AND COMMUNICATION SYSTEMS

Apart from a particularly protected vehicle for escorts and cargo, the major components of a system for the protection of transports of radioactive materials are suitable equipments for the surveillance of the transport as well as for communication between escorts and transport control center.

These equipments are characterized by two functions:

(1) They must be capable of detecting and automatically reporting to the transport control center any disturbance or other interference by third parties; this is necessary as the lo of the escorts has to be taken into account.

(2) They must enable reliable communication of escorts (transporter and escorting vehicle) among themselves and between escorts and the transport control center and vice versa.

The capability to perform both functions must be ensured at any time and any location where the transporter may be.

This means that the system must be reliable and efficient over the entire territory of the Federal Republic of Germany.

A possibility of using existing facilities is the motorcar telephone.
The motorcar telephone enables communication between stationary or mobile subscribers in the entire territory of the Federal Republik of Germany, with the exception of just a few areas, and offers the same transmission band-with as the conventional telephone network. Thus, this system is capable of transmitting not only voice signals but also digitized signals such as they are required when using digital speech inverters or data telegrams. The relatively good coverage of the system is achieved by a great number of land and relay stations. The mobile party's connection with the telephone network is established by the responsible land station within the respective radio traffic area. However, if the mobile party leaves the area of a land station or if the connection is interrupted for more than 9 seconds, e.g. as a result of radio pockets, the connection has to be established anew.

To establish a connection between the telephone network and a mobile subscriber, the approximate position of the mobile subscriber must be known so that the land station can be dialed in whose range the mobile subscriber is. Otherwise, no connection will be established.

The system's advantage of relatively good coverage conflicts with a low time availability which depends on time and place, as the number of available channels is very limited. Unpredictable waiting times may occur particularly in densely populated areas during peak hours. Because of the equal treatment of all telephone subscribers, the available channels are distributed in accordance with the competition principle. Although priority connections or reservations of exclusive channels are feasible in the existing system as far as the engineering aspects are concerned, they are impossible from an organizational point of view, since the available channels are already overcrowded.

As the present number of channels is inadequate for all subscribers, the Post Office attempts to extend the existing network until the end of 1980 and intends to make available another 148 channels in an additional network by the end of 1984.

Of all the available mobile radio services, the motorcar telephone will involve minimum expenditures and best meet the requirement for mobile communication between a stationary transport control center and the transports under surveillance.

The integration of the system into the communication system would be a low-cost solution of the communication problem and could be realized at short notice under the following boundary conditions:

- allocation of connections which cannot be blocked by third parties;
- concession of a higher priority to the transport connection than to other subscribers or provision of exclusive channels for certain subscribers.

As such a solution will involve the exchange of security-related information through a public communication system, special measures must be taken to protect such information against evaluation by third parties. For radiotelephony connections, speech deception and/or speech inverter facilities may be used.

The transmission of information containing only status and position reports, i.e. routine information, might be effected by means of digitally coded short data telegrams. This method is characterized by the following advantages: the information content is hardly accessible to third parties; such reports can be transmitted automatically; and channel occupancy times are reduced considerably as compared with transmission by telephone. Under these conditions it may still be acceptable to share an exclusive channel with other parties for data transmission purposes. The additional effect that short channel occupancy times in the order of magnitude of seconds will reduce charges should not be neglected in this context.

Although the above-mentioned boundary conditions have not yet been realized, the motorcar telephone ist used at present as one of the communications systems between transporter and transport control center.

The establishment of a redundant communication system meeting the high standards of reliability and range is certainly a sophisticated project in terms of both time and money. The possibilities for the establishment of an independant radio system are presently under examination.

It seems to be favourable to establish a system which operates in the high-frequency-range. This would imply only a few additional relay stations because of the high range of such systems.

The dependance of the propagation of electromagnetic waves on the time of day requires several frequencies to ensure an optimum of operational reliability.

Such a system has already been established in the U.S. for the purpose of surveillance of transports of governmental radioactive material. This system is used for the transmission of data telegrams in normal operation mode, but is also capable of transmitting voice signals.

A further possibility of a communication system redundant to the motorcar telephone is the possibility of sharing the radio systems of the public utility companies. But to be able to meet the requirement for radio coverage of the entire territory of the Federal Republic of Germany, the radio coverage areas would have to be extended by the installation of additional relay stations.

DISCUSSION

Question by H.-U. Berger, FR Germany: What security measures are planned for other radioactive materials, e.g. for use in nuclear medicine?

Answer: Graded measures corresponding to the endangering potential which is lower than for nuclear fuel, establishment of a special security measures catalogue.

Question by A. R. Baker, United Kingdom: You did not distinguish between the security requirements of different types of radioactive material. Could you comment on whether the same standard of physical protection of the vehicle and so on should be required for spent as for fresh fuel? In particular, are high-specification lock systems required?

Answer: In the framework of categorisation both aspects (theft, release) and therefore also the different material properties were no longer taken into consideration. The measures for one category do not distinguish between radiated and non-irradiated material.

Comment by R. Schüler, FR Germany: Depending on experience with car telephones, it must be stated that an emergency call could take 30 minutes to send! I am sure that this will not be changed in the future.

Answer: The present relatively small availability, with regard to time and place, of the car telephone is well-known. Therefore, use of the car telephone is only a temporary solution under the present parameters.

Question by W. Heudorfer, FR Germany: You mentioned that, with regard to the delay time, it is required that the time taken for security forces to arrive on the scene should be small in relation to the time in which events occur. If security forces have to be called, this will not be possible in every case?

Answer: It must in every case be assured that the delay time is smaller than the resistance time. The availability of security forces must be assured by organisational and administrative agreements between the competent authorities. The relationship between both times influences the barrier measures.

IT'S A MAD, MAD, MAD WORLD

John Mangusi
Transnuclear, Inc.

Transnuclear, Inc. is a nuclear fuel cycle service company specializing, among other areas, in operations relating to all aspects of the transportation of radioactive materials, including spent fuel and strategic special nuclear materials.

Since the implementation on March 6, 1974 of 10 CFR 73 regulations for the protection of strategic special nuclear materials in transit, we have made numerous export and import shipments of highly enriched uranium, plutonium oxide and plutonium nitrate.

The majority of international shipments of strategic special nuclear materials are between the U.S. and Europe. The high unit value of these materials and the high degree of security required makes air shipment most desirable.

I presented a paper in 1975 at the Annual Meeting of the Institute of Nuclear Material Managers on this subject and, at that time, I thought the world of transportation safeguards was a mad world. Now, five years later, the transportation safeguards world has added a few more mad mads to it and now is definitely at least a mad, mad, mad world.

During the 1975 presentation I detailed several examples illustrating the conditions which prevailed at the time.

One of the first actions which affected our transports was the airline pilots embargo on hazardous materials. Pilots had experienced many cases of persons shipping hazardous materials under improper names to obtain lower freight rates. What is a pilot supposed to do when flying at 35,000 feet, a box allegedly containing assorted bolts starts to leak, fizz or smoke? Understandably, the pilots felt they had to take some action and did so by embargoing all hazardous materials for a period of time. This embargo was felt primarily on domestic U.S. shipments, as the foreign carriers and the international portion of the U.S. pilots did not react as strongly.

At that time I had spoken to a representative of the pilots' organization and was told that radioactive material shipments were not a major concern to the pilots since most radioactive material shippers had a very good record. However, in order to bring their point home, the pilots felt that an

overall embargo was necessary. At that time we replaced the domestic air legs of our HEU shipments with land routes.

In early 1975, Section 108 of the Transportation Safety Act was implemented limiting the carriage of radioactive materials on passenger carrying aircraft to those intended for use in, or incident to, research, medical diagnosis or treatment. This has been unofficially extended to cargo flights, and today a shipper may find that although the airline accepts the cargo for transport a particular pilot may refuse to fly the aircraft.

Accordingly, it is not uncommon to find that no precise booking will be given for the cargo and that the shipper will be told that his material will be shipped on a space available basis. This particular situation will vary from airline to airline, and again the situation is more applicable to domestic U.S. airlines rather than to foreign carriers or international flights of U.S. carriers.

One incident which occurred when we were using domestic air routes to connect with international routes for the transport of strategic special nuclear material involves the cancellation of a flight. The domestic flight was the only all-cargo flight scheduled that day to the ultimate destination. The aircraft which the airline intended to use was involved in an on-the-ground mishap which damaged one wing severely enough that the aircraft was placed out of service with no replacement available. The freight therefore had to remain at the airport in excess of 24 hours until the next flight the following day.

The regulations required preplanning to avoid in-transit storage in excess of 24 hours. Because the freight was held at the airport for longer than 25 hours, we had to explain to the authorities what we would do to preclude the same from happening again. We obviously had preplanned to avoid storage in excess of 24 hours as required by the regulations but, nevertheless, were treated as if it were our fault.

Because of the preplanning and scheduling which must go into one of these shipments it was felt that the shipment was more secure under armed guard in the security cage of the airline awaiting the next day's flight than it would have been if we hastily attempted to reschedule the move to another airport or to transport via land.

Since the presentation of the 1975 paper, the regulations and requirements have changed several times thereby generating more situations about which to tell stories. When we first began these shipments we were utilizing JFK airport in New York City as the exporting or importing terminal point for the international portion of HEU shipments. In order to comply with the New York City hazardous material regulations,

we were required to transit the New York City boundaries on our way to JFK between the hours of midnight and 6:00 a.m.

For the shipment I am about to discuss we had made our customary notifications to all the proper authorities and had made arrangements with the airline personnel to accept the shipment. As was customary, also at that time, I had planned to be at the airport prior to loading of the aircraft to witness loading and departure of the aircraft.

In this particular incident I was at home at about 11:30 in the evening when I received a phone call from the director of the New York City Radiation Council, who said that he had been given orders to go the George Washington Bridge to stop our shipment. I couldn't believe what I was hearing, and envisioned a very comical situation of this individual trying to stop a large tractor semi-trailer at the approach of the George Washington Bridge.

At any rate, this gentleman did go to the George Washington Bridge and planned to intercept the truck. When he got to the bridge, he was unable to convince the police officer on duty that this truck should be stopped. The police officer had his orders to permit this vehicle to go through.

The ironic part of this whole thing was that the New York City Radiation Office had earlier that day given the permission to the police department to allow this material to go through, so when the director tried to thwart the shipment he was unsuccessful in changing his own order, because the latest official order the police at the bridge had received was that the shipment could pass. The truck did go through with the Director following in his car.

I, later in the morning, met the director of the New York City Radiation Council at JFK airport and we stood side-by-side as the freight was unloaded from the truck, loaded aboard the aircraft and flown away.

Shortly thereafter the New York City authorities published their now infamous regulations which effectively amount to a ban of this type of material going through New York City. The regulation actually states that a Certificate of Emergency Transport must be obtained, and such certificates will be issued only for the most compelling reasons involving urgent public policy or national security interest transcending public health and safety concerns, and that economic consideration alone will be not acceptable for the justification for the issuance of such certificate. The New York City regulation went into effect in January of 1976 and while many, many claimed that it was illegal, the ruling effectively stood.

At that point we had two choices: to openly defy the New York regulation on the basis that the New York regulation was in violation of Federal rules and risk a chance of

getting arrested, or to switch our operations to another airport. At that time we chose to switch our operations to Chicago.

We performed many shipments out of Chicago after meeting with the proper authorities in setting up our guard requirements and communications with police. One of these sticks in my mind as particularly interesting.

The episode occurred in the airline cargo building while the drums were awaiting palletizing and loading aboard the aircraft. The regulations required continuous visual surveillance of the drums. We had the proper number of armed guards in attendance with radio contact to another guard stationed in the airport police office. The cargo storage room was a little larger than this room, and certainly the guards had no trouble seeing all parts of the room. At least one of the guards was stationed at an elevated point for better surveillance.

However, the inspectors who were there stated that the line of sight of the guards was obstructed by a pillar which was holding up the roof; and, therefore, continuous visual surveillance was not maintained. They hypothesized that someone could enter the building unnoticed, very carefully take one of these drums which weigh 300 lbs and slowly back it out of the pile while remaining completely behind this pillar such that the guards could never see him, stow this drum, return later and take it elsewhere. We maintained this scenario was absolutely ridiculous and lodged a rather severe protest with the authorities, which generated a well attended meeting in Chicago, in order to try to rectify this kind of thinking.

I believe as a result of this meeting, the interpretation of the phrase continous visual surveillance has been given a more realistic interpretation by the authorities. Also, at this same shipment, the various governmental agencies involved had a total of 18 people in attendance, all of whom had their identification checked by our guards.

We complained bitterly when we were taken to task because one governmental group complained that people went on the aircraft and were taking pictures aboard the aircraft without challenge from our guards. These people were representatives of one of the agencies and had been previously checked by our guards.

For a while after that we restricted attendance to three people whose names, serial numbers, etc. would be given to us ahead of time. There was no other way we could operate. We still maintain similar controls and the authorities have been very cooperative.

Because of the press coverage devoted to this episode, eventually the Chicago authorities banned shipments of HEU

via Chicago's O'Hare field. NRC had several meetings with the Chicago authorities to no avail. NRC then told us as far as they were concerned, we could ship via Chicago, and the Chicago authorities said if we did the driver would be arrested.

The authorities in Washington thought this would be a good way to test the rule. However, we could find no one who really cared to get himself arrested. In the interim, between the continuous visual surveillance episode and the closing of O'Hare, we were first requested to use military aircraft for these shipments.

The next incident that comes to mind was just prior to the first shipment that we were going to do utilizing the French airforce and flying out of Columbus, Ohio, to France. Obviously, since this was the first time, it took us several weeks to get everything worked out. It was about two days before the shipment was to take place that we received a call from NRC stating that they wished us to change the departure from Columbus back to Chicago. We told NRC that it took too long to set this thing up and that we could not just switch back to Chicago.

We requested to know the reason why they wanted us to go back to Chicago. NRC's reason for asking us to switch from Columbus back to Chicago was that since there had already been a fair amount of adverse publicity at Chicago and none yet out of the Columbus area they felt it safer from a publicity standpoint to depart from Chicago. We thought the whole thing was absolutely ridiculous, and I requested official confirmation of the request. Several hours later they called back to say we could proceed via Chicago.

The mad, mad, mad world of safeguards has now extended itself into the spent fuel shipping area primarily to thwart any attempts to hijack a spent fuel cask, transport it to a highly populated area and blow it up. Although the jury is still out as to the proper level of safeguards required for spent fuel shipments, NRC felt that it should institute requirements which have since, naturally, been upgraded.

At this time, Transnuclear was in the process of bringing into the United States irradiated fuel from research and test reactors in Europe. This fuel was entering an east coast port of the United States and being transported to Savannah River for reprocessing under DOE contracts.

NRC held a meeting in Richmond, Virginia, to discuss the potential increase in safeguards on spent fuel shipments. It was pointed out to us that some safeguards would have to be instituted while studies were being conducted to determine need and proper level; and that if the studies proved that safeguards weren't needed or didn't need to be so stringent, the requirements that were being placed upon us would be removed or be changed.

We mentioned, at that time, that it probably was not a good idea to put these new requirements in the form of regulations because once they were placed as regulations they would become etched in concrete and never get changed. It was NRC's intention, and they so stated at that time, that they would issue the requirements in a form less permanent than a regulation such that they could be more easily changed. We still had our doubts that any requirements would be lessened as time went on.

The safeguard requirements were published shortly thereafter as a regulation. As a matter of fact they were published very shortly after the TMI episode. We questioned this procedure ad asked why the change from the intended plan. We were informed that in order for the rule to come out in a form less permanent than a regulation, the signatures of three men who were at Three-Mile Island were required. Since they were unavailable and it was desired that the rules be implemented as soon as possible, the only way was to issue them as a regulation. We were very, very upset by this course of action. We feel that regardless of the results of any ongoing studies, the present level of safeguards will never be reduced.

During the meeting in Richmond, mention was made of requiring safeguards based on the population density. No numbers were given at that time as to the population density which would trigger the requirement for safeguards. Shortly thereafter the requirement was defined and boils down essentially to areas over 100,000 people. The port through which we were coming with the shipments fell into that category and, therefore, we then had to demonstrate that there was no practicable alternative to using that port or else we would have to switch to another place.

As we started to investigate other seaports and our investigation became known, the people in these areas raised objections to the shipments coming through their areas and they started passing restrictions and ordinances. We were ultimately able to demonstrate to NRC that we had no other practicable alternative to our present method and we were given permission to continue utilizing our original port.

As I reflect on this whole issue, it wasn't the safeguards per se which caused the problems but rather the definition of the urban area being 100,000 population. Maybe more time should have been given to setting the threshold level. Once it is out, how do you then redefine it and say, "Well I'm sorry I didn't mean that - the population trigger level is now 500,000." Very careful thought must be given before any of these rules come out. in this case I think the rule came out much, much too fast and it came out as a regulation.

Since then I believe the staff at NRC, and others have seen that this may not have been one of their best

judgments and have worked with us in trying to live within the requirements of this rule, but the regulation still stands.

The Department of Transportation in the U.S. has now proposed a highway routing rule which is supposed to eliminate the local jurisdictions from passing requirements against the transport of radioactive materials such as were being passed to keep us from changing our port of entry; but as I understand it, this rule does not automatically rescind all the local laws. It just provides the hammer for one who wishes to legally challenge any local law.

Back on HEU shipments for a minute. At one point Transnuclear wrote a letter trying to give these shipments to the Federal Government. Our basis was that military level security (which seemed to be what we were being led toward) could not be achieved in the private sector.

We suggested that the U.S. Government perform the HEU shipments from the U.S. to the foreign country, at random unannounced times, in large quantities. The material would then be stored in U.S. military installations overseas. When the export license is issued, the recipient of the material would use the license as his ticket to claim the material.

This suggestion was refused because apparently it violated some rules of the Department of Defense with regard to government involvement in commercial endeavors, and also the plan could compromise some of the security surrounding weapons storage and transport.

In closing I request - urge - beg our legislators and those involved in writing and promulgating our rules, requirements and regulations to very carefully assess the ramifications of a particular rule before acting. To have to back peddle at a later date and start patching up the rule contributes only to making a mad, mad, mad world turn into a mad, mad, mad, mad world.

DISCUSSION

Question by A. L. Kaplan, USA: What alternatives to the use of commercial carriers, in the manners that you have described, have you investigated, for the transportation of strategic special nuclear materials to or from foreign countries?

Answer: We have proposed that DOE perform the shipments at random unannounced times. These shipments would probably be only once or twice per year in rather large quantities. The material would then be stored at an overseas military base. When the overseas customer receives his export license, he would be able to claim his material at the US base by presenting his licence.

This proposal apparently is not acceptable to Department of Defense personnel due to possible compromise of weapons security procedures and use of US government personnel in private commercial enterprises.

FUEL RESPONSE TO SEVERE TRANSPORTATION ACCIDENTS

M. E. Balmert, K. D. Kok, P. Baybutt, S. Raghuram,
L. M. Lowry, and A. R. Rosenfield
BATTELLE'S COLUMBUS LABORATORIES
Columbus, Ohio U.S.A.

INTRODUCTION

Although the safety record of transportation of radioactive materials has been an impressive one, it is still important to determine the overall risk to the public from this portion of the nuclear fuel cycle. As the first step in defining risk, the consequences of transportation accidents must be determined. The response of shipping packages to normal and regulatory-defined accident conditions is comparatively well understood. What is missing is the response to more extreme conditions--those that establish a failure threshold and beyond.

The Transportation Technology Center of the U.S. Energy Department at Sandia Laboratories is supporting research to determine the consequences of spent fuel failure due to severe environments. Battelle's Columbus Laboratories (BCL) has been selected to quantify the response of spent light water reactor fuel to extreme accidents. This paper will present the activities of the past year, which consist of an investigation of fuel failure mechanisms and the development of a radionuclide transport model.

The overall interaction between the radioactive contents of shipping packages and the internal and external cask environments is illustrated in Figure 1. This diagram shows how the various initial conditions and component responses are factored into the transport code. The action of a severe external environment, coupled with the cask's initial state, produces a cask response. For example, a sudden deceleration of a cask may be severe enough to fail the primary seal, thus breaching containment. In response to this extreme external environment, an internal cask environment is produced. The radioactive contents (e.g., spent fuel) characterized by its own initial state, reacts to this condition to produce a response. An example of this response might be a deceleration of a spent fuel assembly severe enough to generate and release respirable particles into the cask containment.

The content's inventory, under the action of the internal environment, creates a source term which is input to the transport model. In Figure 1, this model has been defined as the TRACK computer program. This code, using as input the cask response, internal environment, and radionuclide source term, models the transport of the radionuclides released from the fuel, through the cask interior, and to the environment via the cask breach. At this point, an environmental consequence code, adapted to radioactive materials, can be used to predict the health effects of these events.

The lower half of Figure 1 highlights the area of this past year's activity: the response of spent fuel to an internal cask environment and the transport of fuel material through the cask to the external environment. The emphasis of our research has been on literature surveys and development of analytical models to

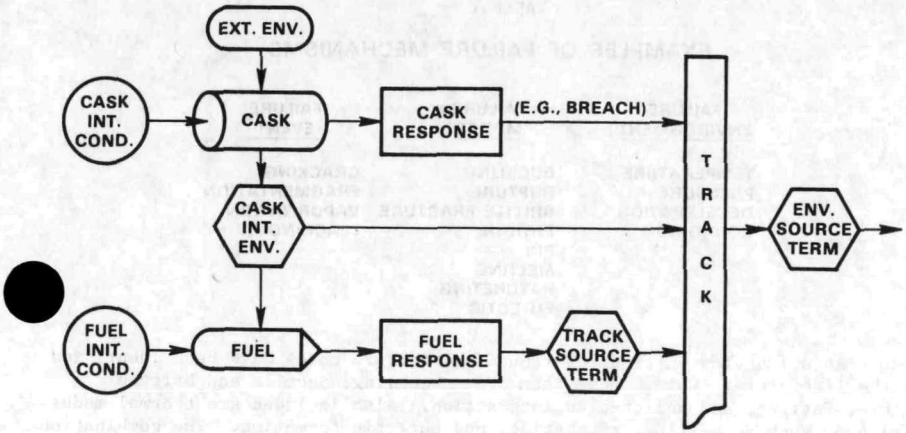


FIGURE 1. FLOW DIAGRAM DEPICTING FUEL/CASK/ENVIRONMENT INTERACTION

identify those conditions within the cask that produce fuel failure and to understand radionuclide transport and release to the environment, assuming a cask breach.

FAILURE MECHANISMS

A literature survey was made of available experimental and analytical data concerning fuel and cladding material behavior. Results of this survey indicate:

(1) Irradiated Material: Most of the tests were performed on unirradiated material. In some respects, highly stressed cladding does react much like irradiated cladding, but in many important areas, they act differently. In addition, comparison between irradiated and unirradiated fuel rod cladding test results are very limited. Most of the data available are intended for material characterization only; hence, failure threshold information is sorely lacking.

(2) Clad/Fuel Coupling: In the majority of cladding tests, the fuel pellets are removed and the rod is tested as a hollow tube, as specimens (tensile, Charpy, etc.) which are irradiated in a reactor prior to testing, or as samples machined from a spent fuel rod. These results do yield the properties of the irradiated cladding, but do not account for clad/fuel coupling that may affect the response of a "filled" fuel rod in a transportation environment.

(3) Strain Rate Effects: Fuel failure environments are expected to involve relatively high strain rates. However, little test data for irradiated cladding exist for strain rates above 0.05/sec. It is anticipated that strain rates on the order of 10^2/sec will be required for fuel failure.

Based on this survey, an investigation of fuel failure mechanisms was initiated. For purposes of our research, and in the context of Figure 1, fuel failure is defined as a release of fission products, actinides, or radioactive structural material (e.g., cladding fragments) to the cask inner cavity. A failure mechanism is defined as the action of a particular environmental parameter or combination of parameters acting through a particular failure mode to produce a particular failure event. Table 1 gives some examples of failure-inducing environments including deceleration (impact) and extremes in temperature and

TABLE 1

EXAMPLES OF FAILURE MECHANISMS

FAILURE ENVIRONMENT	FAILURE MODES	FAILURE EVENT
TEMPERATURE	BUCKLING	CRACKING
PRESSURE	RUPTURE	FRAGMENTATION
DECELERATION	BRITTLE FRACTURE	VAPORIZATION
CORROSION	FATIGUE	LEACHING
	PCI	
	MELTING	
	RATCHETING	
	EUTECTIC	

pressure as a function of time. Various fuel failure modes have been identified from the literature. Chief among them are: buckling, ductile and brittle fracture, fatigue, and pellet-clad interaction. Also included are thermal modes of failure, such as melting, ratcheting, and eutectic formation. The combination of a severe environment and a failure mode produces a failure event, such as circumferential and longitudinal cracking of cladding, fuel fragmentation, vaporization, and leaching.

Table 2 illustrates the relation between failure environment, mechanical failure mode, and failure event. Note that all possible combinations are not reasonable. For instance, pressure would not cause a failure event by melting.

TABLE 2

EXAMPLES OF LWR FUEL FAILURE MECHANISMS: FAILURE EVENTS IN TERMS OF FAILURE MODES AND FAILURE ENVIRONMENTS

	FAILURE MODES							
Failure Environment	Ductile Fracture	Brittle Fracture	Ductile Rupture	Brittle Rupture	Buckling	Melting	Corrosion	Creep
Temperature	C. Cracking* or Vaporization	Fragmentation	L. Cracking* or Thinning	—	C. Cracking or Thinning	Thinning or Vaporization	Thinning	L. Cracking or C. Cracking or Fragmentation
Pressure	Thinning	C. Cracking	L. Cracking	L. Cracking	—	—	—	L. Cracking or Thinning or Fragmentation
Deceleration or Acceleration	C. Cracking	C. Cracking	—	—	C. Cracking or Thinning	—	—	—
Corrosive Vapor	—	—	—	—	—	—	C. Cracking or L. Cracking or Fragmentation	—

*C = Circumferential
L = Longitudinal

However, not all of the failure environments act independently on sealed fuel rods. For example, as the temperature rises, the fuel rod internal pressure will increase and, depending on the rate of strain and physical constraints, failure may occur in a ductile or brittle mode causing a thinning or cracking event, respectively. However, the effect on the cladding and on the fuel will certainly be quite different and a separate matrix of events may be required for each.

As mentioned before, available test data were surveyed to provide a perspective on fuel failure. Of these, several mechanical tests were identified, including tensile, burst, bending, torsion, and compression. In terms of failure mechanisms, these tests and the data generated comprise only a small part of the table. However, these are related to the most common failure modes. Table 3 shows the correspondence between three of these tests and the most common failure events for cladding. It should be noted that, except for burst tests, the fuel is removed prior to testing. The "X's" in the matrix indicate that the particular test involved the corresponding failure event. Our FY-80 report includes further discussion of these test data, many of which were obtained in Battelle's Hot Cells in Columbus, Ohio.

TABLE 3

CLADDING FAILURE EVENTS

TEST	CIRCUMFERENTIAL CRACKING	LONGITUDINAL CRACKING	THINNING
BURST	—	X	X
TENSILE	X	—	X
BEND	X	—	X

For thermal failure mechanisms, Table 4 shows the types of failure events that result from various failure modes as related to the nature of the temperature change.

TABLE 4

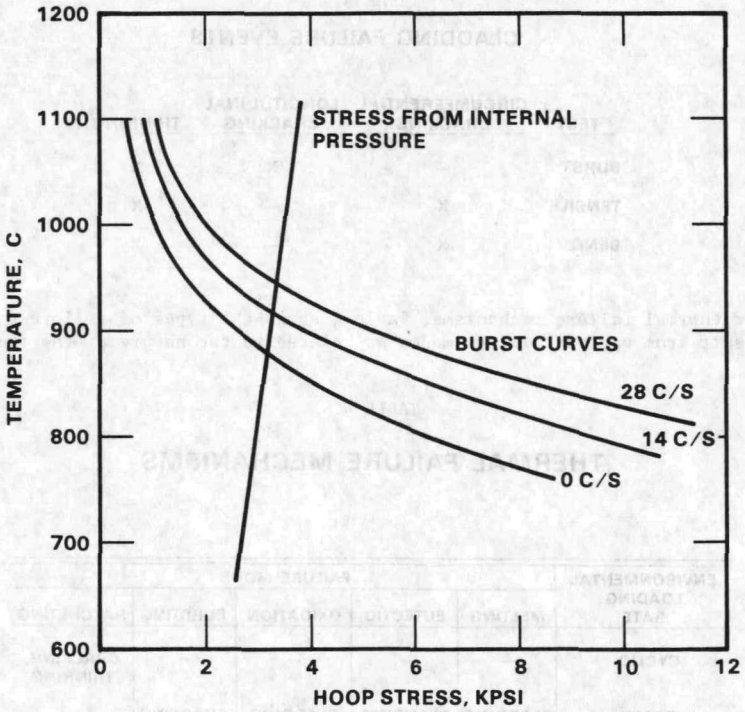
THERMAL FAILURE MECHANISMS

ENVIRONMENTAL LOADING RATE	FAILURE MODE				
	MELTING	EUTECTIC	OXIDATION	BURSTING	RATCHETING
CYCLIC	—	—	—	—	CRACKING, THINNING
RAMP	CLADDING LOSS	CLADDING LOSS, THINNING	CLADDING LOSS, THINNING	CRACKING	—
IMPULSE	—	—	—	CRACKING	—

As an example of the data that are available, Figure 2 illustrates the relation between burst (most probable thermal failure mode) temperature and stress as a function of heating rate (ramp loading). In this figure, bursting failure curves are shown for irradiated cladding. Superimposed on these curves is a curve relating the internal pressure, and hence the stress in the cladding, to the temperature for a typical LWR fuel rod. The intersection of the latter curve with the burst curves determines the cladding burst stress as a function of internal pressure, and therefore, temperatures at which clad failure will occur from internal pressure. From this figure, one can see that the burst temperatures are lower for the lower heating rates and that at the lowest heating rate, a typical rod would not burst below about 880 C. It has been estimated that, for a typical cask, a fire temperature of about 820 C would be necessary to achieve this fuel

FIGURE 2

PWR CLADDING HOOP STRESS FROM INTERNAL PRESSURE COMPARED TO CLADDING BURST CURVES



pin temperature. Although this is only slightly greater than the regulatory (IAEA) condition, it should be noted that for typical cask designs the fire must endure several hours in order to raise the fuel to the temperatures indicated. This indicates that a fire lasting only a half hour would need to be much hotter than the IAEA specification to cause rod burst.

The results of the failure mechanism study indicate that much data needs to be generated to quantify fuel failure thresholds. Meanwhile, available data will be used as input to the source term model in the radionuclide transport code.

SOURCE TERM

The fuel failure mechanisms described earlier were used as bases for source term (TRACK-input) model development. These models have been divided into mechanical and physico-chemical in origin. Mechanical source term generation mechanisms are typified by fragmentation and pulverization processes. Physico-chemical mechanisms investigated this year were limited to leaching. The results of this source term investigation will be incorporated in the cask radionuclide transport code, TRACK.

It is important to characterize fragmentation of spent reactor fuel for two reasons:

(1) The pellets contain trapped fission products which may be released because of the creation of new surfaces.
(2) Respirable particles of UO_2 may be formed.

In order to determine the extent to which these processes can occur, it is necessary to characterize fragmentation. To do so, the concept of an equivalent fragment diameter (D_e), shown in Table 5, has been invoked and applied to both normal and accident conditions. From these diameters, values can be calculated for other system parameters, such as number of fragments formed and total new surface area formed.

TABLE 5

SUMMARY OF TYPICAL PARTICLE DIAMETERS

EVENT	D_e (EQUIVALENT FRAGMENT DIAMETER)
NORMAL OPERATION AND REMOVAL FROM REACTOR	1.15 mm
SEVERE IMPACT	72 μm
THERMAL SHOCK	10 μm

Based on a literature review of the fracture behavior of UO_2 under impact and thermal loadings, conservative estimates of D_e have been made for normal reactor end-of-life and for transportation accident conditions. It must be strongly emphasized that these estimates rely on an extremely limited data base and that some of the inputs have had to be drawn from experiments on brittle materials other than UO_2.

Table 5 lists the values of D_e assumed in our calculations. In addition to estimating the equivalent fragment diameter, the distribution of fragment sizes was also determined. As an initial estimate, a Schuhmann distribution was assumed. (Details of this calculation will appear in our FY-80 report.) The important point

$$f = 0.81\ (D/D_e)^{0.667}$$

where

D = Particle Diameter
f = Fraction smaller than D

regarding this analysis is that the results are based on as yet insufficient data, but do give conservative estimates of what fuel particle sizes can be expected from extreme events.

A number of mechanisms for the release of radionuclides from fuel are possible in cask accident conditions. Examples include: phase changes, aerosol agglomeration, particle and vapor deposition, leaching, and chemical reaction. Of these, phase changes and vapor/particle deposition are incorporated into the TRACK code. Deposition mechanisms are intended to account for radionuclide transfer onto fuel assembly and cask structural components, or onto other particles. Present models assume constant rates or velocities of deposition for radionuclides considered. Current models are being developed to account for effects of temperature, flow rate, particle size, etc. The other mechanisms will be developed as part of the continued code development effort.

During the past year, efforts have concentrated on leaching mechanisms. Available experimental data for radionuclide leaching from spent fuel and waste glass forms have been used to develop a preliminary leaching model. Large uncertainties could be associated with the values calculated using the model.

TRANSPORT CODE

TRACK (originally developed for the U.S. Nuclear Regulatory Commission) performs calculations of radionuclide transport and deposition within a spent fuel shipping cask. The code accepts, as input, the source rates of the various radionuclides being considered, the thermal hydraulic conditions, such as temperature and pressure, within the cask, and geometrical data describing the cask and the locations of the breaches and/or leaks. It calculates the depletion of the radionuclides during their transport through the cask interior and computes a release rate to the environment. This environmental source term may then be used to calculate the health effects of the accidents analyzed. An environmental consequence code, such as CRAC, may be utilized for this purpose.

TRACK models the transport and deposition of radionuclides in both vapor and aerosol (particulate) form. The transport of both vapor and particulate radionuclides is assumed to take place with the ambient fluid flow. This transport is imposed on the code by the time-dependent thermal-hydraulic data read into the code. Deposition of radionuclides within the cask is modeled as a first order rate process. The cask itself is treated as a series of interconnected compartments within each of which the fluid is well mixed. Under these assumptions, the transport and deposition models can be cast into a set of linear first-order differential equations which account for particle or vapor generation, deposition, and flow between control volumes. This set is solved numerically under appropriate initial conditions, to provide the radionuclide concentrations as functions of time.

TRACK output consists of inventories of material released to the environment, as well as that remaining in the cask. The environmental component is in the form of isotopic compositions and masses as a function of time, as well as the physical or chemical form of each species. A similar inventory is produced within the cask, for each control volume, as a function of time.

Although TRACK is an operating code, several features have been and will continue to be in a state of development. The input source term development described above is an example. Models for particle deposition rates as functions of particle parameters have already been incorporated into the code. Other

processes such as particle agglomeration, fragmentation, and chemical reactions are to be included in the code at a later date. This will allow more realistic analyses of cask accidents.

The results of the first year's analytical effort are being used to guide the planning of future experiments. Future research objectives encompass further development, including experimental benchmarking, of the TRACK code, as well as the initiation of an experimental program to provide confirmatory data on failure mechanisms, input source term generation, and radionuclide transport. A series of experiments are being planned to start in FY-81, which will involve tests of single and multi-rod assemblies, using irradiated and surrogate materials. Other research activities related to response of radioactive materials to extreme environments will involve high-level waste forms. Finally, once fuel failure and TRACK input source term have been specified in terms of the cask internal environment, a concentrated effort needs to be made to correlate cask behavior with these factors. This area of study represents the upper portion of Figure 1. By determining the failure thresholds of package contents (i.e., fuel) in response to internal cask environments first, one can then determine what external environment is required to produce the necessary cask response. In this way, the cask response can be coupled to that of the contents. The eventual development of this entire scenario will permit a risk assessment of spent fuel shipments during severe transportation environments.

CONCLUSION

Battelle-Columbus' research activities for FY-80 have been described above. Failure mechanisms were developed, to the limit of the available data, for fragmentation and leaching processes. These mechanisms, along with others to be developed, can be used to benchmark the TRACK code with experiments, as well as provide guidance for experimental planning.

Preliminary input source term models were developed from these mechanisms, for typical spent fuel materials. These models will eventually be inserted into TRACK.

The TRACK transport code was updated to include some of these improved source term models. In addition, other features have been revised. Updating of the code is planned to continue in FY-81.

The response of fuel to extreme thermal environments was assessed in light of available data. The most probable failure mechanism is expected to be bursting, in environments more severe than currently required for safety analysis.

Finally, this research has demonstrated a clear need for further analysis, and especially, for experiments that will fill the gaps in the available data base.

DISCUSSION

Question by S. Williamson, United Kingdom: You referred to a temperature of about 800°C as being a failure temperature of fuel cladding in a transport flask. What was the cladding that you considered? Was it Zircaloy or stainless steel and is there a temperature difference between these two materials for failure to occur?

Answer: The data are for Zircaloy. There is a difference between stainless steel and Zircaloy, but I do not have any stainless steel data available.

Question by N. A. Eisenberg, USA: The model described in your paper appears to treat only gaseous, volatile, and solid fission products from the fuel itself. However, some US and British literature suggests that corrosion products deposited on the spent fuel cladding may be an important part of the source term for spent fuel in severe environments. Could you comment, please?

Answer: We are not currently looking at the crud. However, we expect to carry out experiments on actual spent fuel. If the crud is an important contributer to the radiological source term it will show up in our sampling equipment. We will not treat the crud analytically until our experiments show that it is important.

Question by G. Kaspar, FR Germany: You have stressed the lack of relevant data about the characteristics of spent fuel elements. Have you only taken published data into account or have you also tried to obtain data from manufacturers and nuclear power plant and reprocessing plant operators? It must be possible, for example in reprocessing plants, to make use of experience gained about the mechanical properties of spent fuel elements in the case of heavy mechanical load (mechanical reduction of the fuel element). It must be possible in this way, for example, to obtain information about behaviour under strong impact stresses by examining some fuel elements which were dropped during handling in the reactor pool. I repeat the question. Do you only take published data into account or do you try to systematically make use of the large amount of unpublished data for your study?

Answer: The data are only the data available in the published literature. We have not obtained any data from fuel handlers or any unpublished data.

Transportation of Nuclear Material by sea and inland vessels considered under security aspects

Dipl.-Ing. Gernot Baier

DORNIER SYSTEM GMBH

In the FRG transports of nuclear material by various traffic means have been carried through. The basis for the security of these transports are featured in the security measurement catalogues published by the ministry of the interior.

Corresponding to the frequency of already performed transports measurement catalogues are presently valid, e.g. for the transport of nuclear material on roads. For the up to now seldom resp. not yet performed transports such as sea- and inland-vessel transports, corresponding catalogues are in preparation. The examinations, carried through by Dornier System, will be the basis for its elaboration.

In this lecture it shall be reported on examinations - which were carried through within the frame of an order of the ministry of the interior - on the security of transportation of nuclear materials.

The two essential points of view, to be considered for the secure transportation of nuclear materials, are

- the endangering of the environment by disperging of radioactive material due to accidents or violent attacks,

- the general endangering due to robbery and misuse of highly enriched nuclear material.

It must be assumed that sabotage as well as violent robbery of third persons may lead to this target. Following this procedure, it may be differed in internal as well as external delinquents, whereas the target of the attack may be the same. Internal delinquents are such persons, which are involved in some way or the other with the transportation procedure (e. g. orderer, sender, expedient, receiver etc.)

External delinquents are correspondingly such ones, attacking from outside, this means, attack during the transportation procedure.

Transportation Practices

In order to make a statement on the security-measurements to be carried through for sea- and inland-vessel transports, the presently employed transport practices are considered. Subsequently a future-present-comparison has to be carried through.

The relevant security-measurements for relevant transports are divided in various security categories resp. endangering degrees. For the existing examination only the two most important were considered.

A summary of essential points of view for the transportation of nuclear material resulted in the following ranges:

- existing laws and determinations,
- transported goods (number and kind),
- employed transport containers,
- transport means and corresponding infra structure
- transport ways
- transport practices
- frequency of accidents and reasons
- communication
- navigation.

With the knowledge of these ranges belonging to each transport of nuclear material a complete transport procedure may be elaborated starting from the proposal up to the procedure. Because of control reasons this transport procedure had to be divided in three transport phases. These are

- planning and approval procedure
- transshipment on the port premises
- travel.

This arrangement has been chosen as for sea-going vessels has mostly to be dealt with broken transports and therefore a transshipment on the port premises will be necessary.

The individual transport phases were examined with the help of a future-present analysis. The basis for this analysis was the security level described in the safety measurement catalogue for road transport.

Corresponding to this requirements for the safety against disperging of robbery of nuclear material during a transport measurements resp. proposals were made . Some of them are for short-term others for long-term realization.

Because of the possibility of transporting specially heavy and huge containers, the ship-transport offers itself primarily for the transportation of radiated nuclear material, where huge containers are employed.

With regard to the transports of nuclear material by ship executed in the past years it must be significantly differed in inland- and sea-going vessel transports considering the past employment. Thus the transport statistics of the Physikalisch-Technische Bundesanstalt - approval authority for such transports - shows that up to now with inland-vessels, no transports

corresponding to the categorization mentioned above , will be carried through with inland-vessels.

The possibility exists, to employ this kind of transportation system for the transport of spent nuclear fuel.

As you surely know, a proposal has been put in, to first carry through this kind of transport for the nuclear plant Grafenrheinfeld. In the field of our examination a nuclear transport concept has been elaborated for all nuclear plants in Germany which are situated near a shipable water way. It has been proceeded from the consumption that Gorleben will be the recycling plant. This point must be taken in closer consideration.

Most of the nuclear transports carried through by ship transports from or to overseas will be executed via Bremerhaven. Therefore the practices for this port for transshipment and transport of nuclear material may be considered as representative for the FRG. A large number of this transports will be carried through in containers and by the RoRo-line.

An evaluation of the existing accident statitics of shipment accidents has led zu results, which are also very important for future security measurements.

Subsequently,some examples concerning the security will be explained by means of solution proposals.

Thus it was found out, that in the starting phase of the transport, that means during the planning and approval phase - a great number of persons will be involved with the exact data of transportation goods, the date of transport and the transport procedure. This could possibly give an idea to an internal delinquent, to get exact transportation data (e. g. internal postage, discussions with colleagues, open letters etc.).

This could be avoided by a number of coordinated measurements in order to reach a corresponding confidential degree. This means also, that the circle of persons involved in the individual transportation planning institutes, should be kept as small as possible and that the persons involved must be credible. In addition to this, the transport papers have to be confidential and the exact transportation date is only to be passed by code to the directly involved persons.

A critical point for sea- as well as for inland-vessel transports is the authorized shipment company. As the kind of shipment-company is extremely important for the secure transport it must be ascertained, that only reliable shipment-companies will be authorized. It is expected from the shipment-company, that only ships with highest technical standard and well-educated personnel will be employed. This excludes the employment for transports with so-called "substandard-ships" as they are well known for fuel-transports.

It is well known, that port-premises are difficult to observe because of their bad arrangement. This fact is even hardened by the hardly controllable access from the water side.

It is therefore suggested, that the nuclear material to be transported in the port premises, must undergo a gapless control. This is also valid for the transport procedure, as for possibly essential intermediate stockage on the port premises and also during the lay-days of the loaded ship. The kind of control should be such, as it corresponds to the confidential level of road transports.

For the transport of nuclear material, perhaps being carried through in future with inland-vessels, a number of security measurements should be employed. Here, only some of them are mentioned.

This includes a convoy first of all - regarding the past security measurements for road transports - which possibly may be executed by the water police. Besides a secure communication between ship and transportation central, this means,the transport-arranging company, as well as between ship and convory must be ascertained. The Rheinfunk-Service alone, practiced on the Rhine, would not be suitable because of high overcharge at only 12 channels and the facile interception.

As the waterways on German inland-waters is often interrupted by locks, a significant marking of nuclear material transporting ships should be demanded (e. g. by corresponding flags, lock-pass privilege etc.).

The evaluation of accident statistics on ship accidents has delivered several ideas for improvement-proposals. Thus it is considered efficient to support measurements for a traffic surveillance system as well as to settle the communication ship-ship. Of great significance, not only for the security of nuclear transports, is the development and employment of route surveillance procedures (look container ship "München")and anti-clash device.

The convention ratified by the EG-commission and dismissed in Oct.79 by the IAEO on the protection of nuclear materials means a great step towards the international security of nuclear material.

A basic problem, already discussed during the 5. PATRAM 1978 at Las Vegas, is the transport container. The usual test conditions for the approval of transport containers following the IAEA, do not comprise the possible basic stresses for ship accidents. Corresponding measurements and procedures for the expansion of test criteria for transport containers employed in shipping, should further be supported.

Summed up it can be stated, that the transport of spent nuclear material with inland-vessels could be considered as an unbroken transport following the existing security conditions . It really could be a sensible alternative to the present transport possibilities.

DISCUSSION

Question by R. Schüler, FR Germany: How do you intend to ensure that information remains confidential when, in the case of sea transport, up to 15 authorities must be informed up to 4 weeks before shipment?

Answer: As security checks are carried out on civil servants, it may be assumed that all information remains confidential.

Question by K. Schneider: Security measures have to be carried out and each country has its own philosophy. I fear that if all your proposals were introduced by the BMI (Federal Ministry of the Interior in the Federal Republic of Germany) this kind of transport cannot be established in the future?

Answer: As the BMI wants to assimilate where possible the safety levels of all systems of transport for shipment of nuclear fuel, the proposed measures are only a consequent continuation of what is already successfully practised in the case of road transport.

Comment by J. Jahn, FR Germany: 1. Marking and labelling of cargo ships and police escort of these ships will cause public concern as it is not very common that ships are escorted by police. 2. Cargo handling personnel in the harbour have to be security checked and I think this will be impossible. 3. Information about nuclear shipments is still quite considerable inside fuel fabrication facilities etc. as the fuel has to be handled inside the facility (handling personnel) and fabrication has to be planned.

Answer: 1. Marking of radioactive packages is laid down by law. When it can be assured that the exact shipment date can be fixed at relatively short notice, this would be a possible way of making the possibility of seizure considerably more difficult. A police escort is not unusual and can contribute to preventing seizure. 2. Security checks on the harbour personnel are in fact hardly feasible. This is exactly the reason why administrative measures should ensure a sufficient level of security. 3. For this very reason, personnel should be kept to a minimum. When this is not possible, only information about the exact date of transport would be advantageous for attackers.

Transport of Nuclear Material by Air Carriers

Dr. Volker Behrendt
Dornier System GmbH

1. Introduction

Nuclear Material (NM)-transports by air may principally be carried through by two different kinds of air carriers:

- Rigid-wing airplanes
- Rotating wing airplanes (helicopters).

As for a long time transports with rigid wing airplanes (subsequently called: airplanes) have already been executed; for the helicopter transport of nuclear material in the FRG merely a concept proposal exists.

For the air-transport following reasons are important:

- Great transport speeds
- bridging of medium/large distance without interstop
- nearly gapless, from the transport means almost independent surveillance during a flight
- extraordinarily small seizure possibility of unallowed third persons to the transport means during the flight.

For the helicopter additionally exists the possibility of

- transports without transshipment from one nuclear plant to the other

2. State of the Art of Air transports (1975 - 1978)

In fig. 1 the civil air transports of nuclear material (NM) for the years 1975 - Sept. 1978 from and to the FRG are summarized, divided in European and overseas transports. Fig. 2 shows a rough summary of all kinds of NM and mentiones the quantities.

As far as exclusively uranium was transported, it could be assumed, that type-A-containers have been employed. But in the case of plutonium transports, nearly exclusively type-B-containers have been used.

NM with various frequencies and quantities has been transported on the following routes:

Within Germany:
- Frankfurt - Berlin

Within Europe:
- Frankfurt - London
- Frankfurt - Bornemouth
- Frankfurt - Manchester
- Frankfurt - Brüssel
- Frankfurt - Paris
- Frankfurt - Kopenhagen
- Köln - London

Out of Europe:
- Frankfurt - Bombay (Transit
- Frankfurt - New York
- Frankfurt - Chikago
- Frankfurt - Toronto
- Frankfurt - Boston

- Frankfurt - Washington
- Frankfurt - San Francisco
- Frankfurt - Los Angeles
- Frankfurt - San Diego
- Frankfurt - Seattle.

Since 1978 all NM-air transports in the FRG were carried through exclusively via airport Frankfurt. National NM-air transports have not been carried through except for the mentioned case.

The most different types of airplanes were employed, concerning dimensions (e.g. B 737, B 747) as well as application (passenger aircrafts with additional freight, mere freight carriers, mixed-version-aircrafts) and the technical data (function of air routes).

Except for large radio active sources, which may only be transported in freight carriers, transports of nuclear material are principally permitted for passenger aircrafts, too. A permission regarding various national and international regulations for airplances must be in hand.

Helicopter-transports of NM have not yet been carried through. The employment of an helicopter is interesting for distances of about 100 km - 400 km. Various types of helicopters are known, which may be applicable for this purpose. But it is essential that the type belongs to category A of commercial helicopters (FAR Part 29), that means

- bimotor design
- higher safety level specially regarding flight performance
- IFR-equipment.

The payload and the loading-volumina of the helicopter must be determined dependent on the kind of freight, the frequency of the transports and the transport distances. Helicopter flights in the FRG must be authorized by the air surveillance of the Bundesanstalt für Flugsicherung, like all other flights.

The following regulations for air transports regarding the FRG are to be considered:

- Law on the peaceful use of nuclear energy and the protection from its dangers (atomic law)
- regulation on the protection of damages by ionisizing rays (ray-protection regulation)
- regulation on the covering protection following the atomic law
- law on the transport of dangerous goods
- recommendations for the safe transport of radio active goods (IAEO, 1973)
- air traffic law
- Announcement of the ministry of traffic on the authorization of transporting dangerous goods in airplanes
- IATA-restricted articles regulation
- air traffic authorization -regulation

- nonproliferation treaty, transfer authorization from EURATOM

- and others.

3. Generalized Transport Procedure

The transport procedure may be divided in the following phases:

- authorization
- planning
- transport to the take-off airport and transshipment in the airplane
- flight
- transshipment at the homing airport.

A short, more than trivial description of the transport procedure is not possible because of the many conditions to be regarded, considering the atomic and traffic laws. Relevant features of the individual phases are regarding safety and security items:

A great number of persons of the various institutions is involved in the authorization and planning procedure of the transport. They all know, which material must be transported under which circumstances. But the exact transport date is mostly secret.

The transport from the take-off airport and the loading of the airplane will be executed by specialists with special devices.

The first section of the transport procedure will be executed by attendance of anarmed convoy. The transmission of the radio active material to the air service will be carried through at the air port frontier. Due to activities in the forefield of the airport a great number of persons may reach the radio active material.

During the passenger flight the radio active material is placed unsurveyed in the hold. The crew knows of the kind of freight. The flight will externally be surveyed by radar by the air traffic control. The visibility range of the radar reaches only up to 250 km per device.

The transshipment at the homing airport will be performed similarly to the take-off airport.

The most relevant difference in the transport procedure for helicopter transports besides the small maximal payload and the small flight height and -speed is that the helicopter has not to start from an air-traffic controlled specifically equipped area and not to land there either. Further only few experiences with regular helicopter freight services are in hand.

4. Safety Problems

The part of the whole freight for airplanes lies up to now unter 1 % for NM. This statement qualifies the statements on accident statistics of commercial airplanes. No statistic is known which ranges accidents resp. the significance of accidents and the whole flown routes- and ton-kilometers per type of airplane. Regarding this statement the following survey, which is based on the evaluation of 449 accidents in the USA from 1970 - 1975, may only show a relative picture (fig.3).

Reasons for accidents were in 47 % of all cases human failure, in 22 % the airplane itself, in 4,5 % the airport and navigatory grounds, in 20,5 % the weather and in 6 % other reasons.

For helicopters there exists no comparative statistic material. Here, only the statement can be given that the accident possibility for helicopters is generally higher than for airplanes.

In case of an accident the transport container is the most relevant barrier against a release of radioactive material. Besides, it must be resistable against rough handling during the transshipment procedure. Type-A- as well as type-B-Container meet the latter requirement. In case of an accident it must be thought of failure of type-A-Containers. The type-B-package is therefore much more resistable as it is laid out for higher stresses. In spite of this the air transport of plutonium was prohibited in the USA in 1975 in containers used up to then. The development of a container has been recommended

which resists any stress possibly occuring at crash and following crash-fire of an airplane designed for flying in great height. These stresses are partly far above the presently executed and internationally accepted container tests of the IAEA transport regulations (free fall test out 9 m height; fall test out of 1 m height on to bolt with 15 cm diameter; heating tests in fire of 800°C for 0,5 hrs; dipping test 0,9 m). The expenditure for the development of such resistant transport container must be seen regarding the background of the low transport frequency and the slight accident frequency of the airplanes per flown kilometer. Nevertheless, container tests should correspond to the most important stresses occuring in case of an accident of a specific transport mean.

The external surveillance of the transport procedure with airplanes should be the best due to the safety requirements in air transport and the technical means for surveillance radio radar etc.), compared with the different transport systems. Over the continents it may be considered as gapless (even without cooperation of the pilot), but over the oceans it must be counted on the cooperation of the pilot. The air traffic control directs the air transport in such way, that at less expense a relative minimum of danger exists for the individual plane.

An improvement of the safety of transports of nuclear material by air with the objective of minimizing the possibility of release of nuclear material in an accident may specially be seen in the procurement of containers proved to resist any stresses in accidents. Furthermore, an improvement could be made in the take-off and landing phase, where the guidance of the airplane could further reduce the accident

risks (e.g. take-off and landing out of rush-hours; anticyclic flying times; flying outside the airways with inertia navigation; employment of experiences air controllers and special regulations for the air traffic control).

The mentioned statements are also valid for flights with helicopters, as far as they don't concern exclusively long range flights.

5. Security Problems

The objective of the secured transport of NM by air would be to avoid the loss of control over the load during the transport. This loss may be the consequence of

- robbery in all transport phases
- violent robbery during the loading and deloading-procedure
- sabotage on the transport means
- attack from in- or outside the transport means
- kidnapping of the transport means and the load.

In order to reach this objective, various measurements are necessary. These are partly already executed due to the technique of common air transportation, partly they must still be considered as recommendation.

Planning and authorization procedure:
Due to the large number of persons, which are involved in the planning and authorization procedure of nuclear material transports, the information about the transport concerning load, procedure, date of the transport is widely scattered. This information could be misused.

When increasing the transport of NM is has to be aimed at a very small number of persons to be involved in the transport procedure. The announcement of the transport date should only be made known to a small circle of persons.

The use of passenger aircrafts for nuclear material transports includes the danger of kidnapping the airplane. Therefore it should be thought of only employing freight-carriers for nuclear transports. Opposite to this, the trand in air-transportation removes from the mere freight-carrier to the mixed-version aircraft. Mere freight carriers will be employed only for few profitable routes. In case of using a mixed-version aircraft the examination of the plane and the passengers has to be carried through by suitable measuremtns. Besides that, all transports should steadily be surveyed by well trained and examined escorts (couriers). In case of incidents due to unforeseen traffic-conditions (e.g. intermediate landings, technical defects) these couriers can also arrange further measurements for the security of the transport. For this reason they must be authorized by exact, generally accepted regulations.

Transshipment:
On transshipment a sufficient surveillance of the transshipment procedure and the airplane itself (as well as an eventually necessary intermediate landing) has to be guaranteed. The transshipment-times must be kept as short as possible by exact planning. The personnel should also be confidentially tested. The NM has to be transshipped only in hardly accessible parts of the hold.

Take-off and landing:
For aircrafts with NM as load part special take-off and landing procedures can be arranged if the possibility of a danger for the plane from the ground exists. These procedures prescribe another flight path than usual. During all flight phases it is aimed at an employment of experienced and tested air controllers, who are informed on the special load of the airplane.

Flight:
Measurement can also be taken against a crew who intends to kidnap the airplane with nuclear material load.
This is restricted to the employment of couriers, short-term selection of crew and transport date and similar administrative measurements. Besides that, it must be thought of the possibility of special technicak measurements released by confidential persons (courier) which avoid or complicate the unloading of NM without great technical expenditure.

Similar problems occur for the employment of helicopters. Due to the extreme lift and dropping speed the possibilities of an attack from outside in the take-off and landing phase are reduced. Besides that, a small helicopter can be much easier be checked through before the take-off than a large airplane. An endangering by unknown passengers is not possible, as in case of helicopter transport unknown persons (passengers) will not be taken on board.

In case of tried kidnapping by the helicopter crew some difficulties will occur for the external flight control because of the extreme manoeuvrability and deep flight possibilitеis of a helicopter.

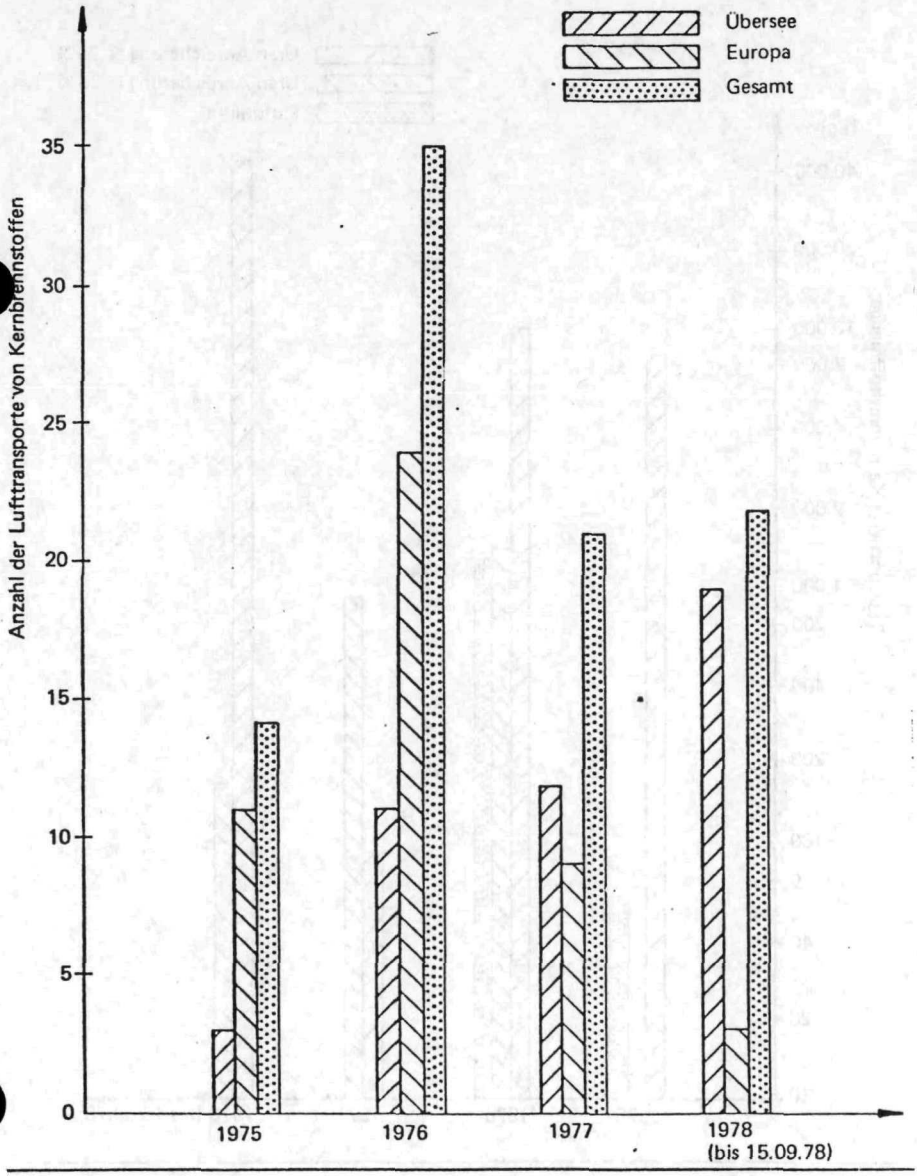


Fig. 1: Air Transports of Nuclear Materials from and into the Federal Republic of Germany

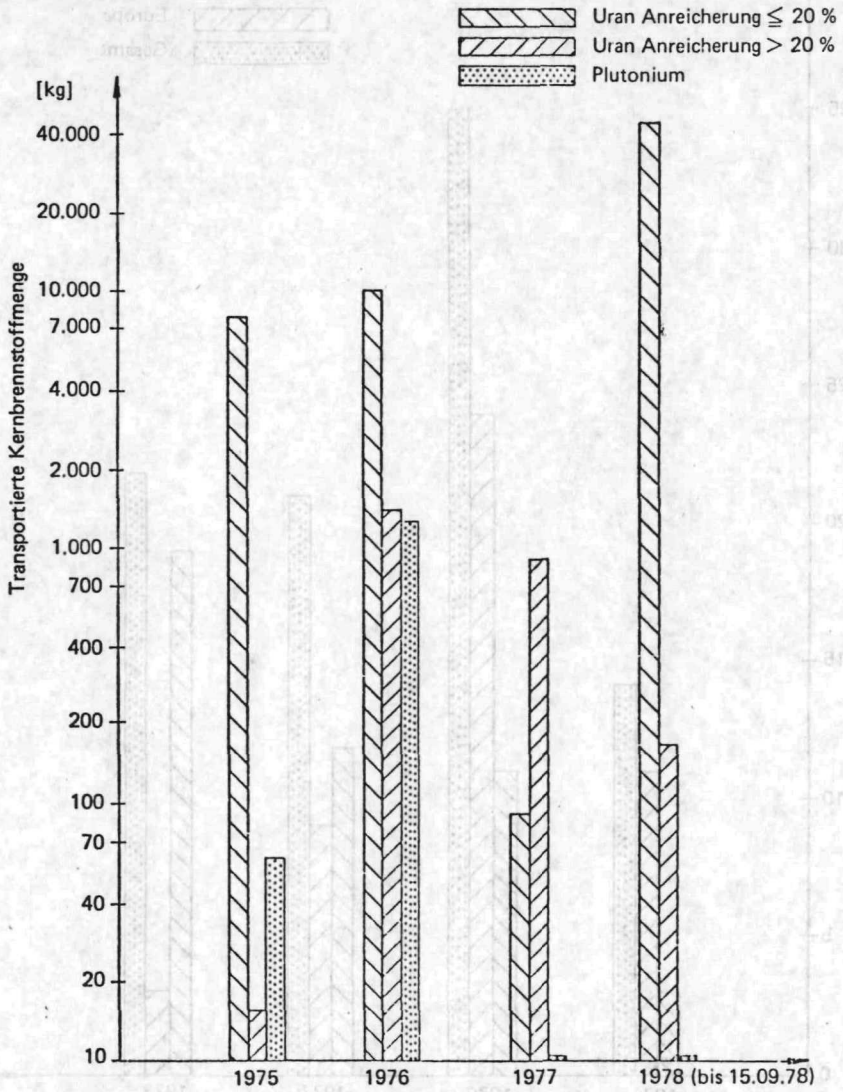


Fig. 2: Survey of Transports of Plutonium, Low Enriched and High Enriched Uranium

Ursachen (Hauptgruppen) \ Betriebsphasen (Hauptgruppen)	Stand	Rollen	Start-phase	Flug-phase	Lande-phase	unbe-kannt	Ges.	%
Verantwortlicher Luftfahrzeugführer	2	18	11	33	59	0	123	(20,4)
Zweiter Luftfahrzeugführer	0	0	1	3	9	0	13	(2,2)
Flugprüfer	0	0	0	0	1	0	1	(0,2)
Sonstiges Personal	19	11	15	85	20	1	151	(25,0)
Flugwerk	3	9	22	19	20	2	75	(12,4)
Triebwerksanlagen	2	4	11	27	7	0	51	(8,4)
Systeme	7	3	4	17	9	0	40	(6,6)
Instrumente, Ausrüstung	1	0	0	3	1	0	5	(0,8)
Flugplatz- und Nav. Bodenanlagen	0	0	0	0	11	0	11	(1,8)
Wetter	0	0	2	81	6	0	89	(14,7)
Sonstige Unfallursachen	7	0	6	21	10	1	45	(7,5)
Gesamt	41	45	72	289	153	4	604	

Fig. 3: Analysis of Accidents in Dependence of the Reasons (Max. Start Weight of the Airplanes > 5700 kp)

DISCUSSION

Comment by R. Schüler, FR Germany: First I would like to point out that no hardships should occur because of the statements. Nevertheless this proves how much theory and practice diverge. Secondly, in my view helicopter transport is not a miracle solution for air transport. There are no flight devices which are approved and the establishment of landing facilities is more expensive as shown.

Answer: Indeed, helicopter transport is no miracle solution. In my opinion it is a proposal to be discussed which really has realistic foundations. There are enough helicopter prototypes which are feasible for transport; an application for permission for such transport has not yet been made.

Comment by. J. Jahn, FR Germany: By using a helicopter for nuclear transport each facility has to be able to withstand aircraft crashes. As far as I remember, not every facility has to be able to withstand such an event at present. The other fact is that the risk of aircraft crashes will increase.

Answer: By introducing special flight procedures and specially situated landing sites in the nuclear facility one could avoid increasing the risk mentioned. The weight and velocity of a helicopter is considerably less than the weight and velocity of a fixed-wing aircraft, against which some nuclear facilities are constructed.

Question by J. A. Andersen, USA: Since air travel routes that have been established and approved by federal aviation authorities are deemed safe for the air transport of the public, why would special flight paths be suggested for the air transport of nuclear material? Non-standard routes could produce aviation problems that may not be as satisfactory as established methods.

Since air traffic controllers are trained safety specialists whose primary duty and goal is the safe passage of aircraft, why would specially-selected controllers be suggested? Both of the above unusual procedures could be counter-productive by drawing special attention to nuclear transport.

Answer: The air traffic density in Europe is the highest in the world, so sometimes dangerous goods or very important people are flown on non-standard routes to further reduce the risk of an accident. Corresponding to that practice this suggestion has been made.

Specially selected controllers could be used to minimise the number of people who know about these special flights.

SESSION XII
OPERATION AND SYSTEMS EXPERIENCE II

EXPERIENCE IN SHIPPING A RADIOACTIVELY CONTAMINATED PWR STEAM GENERATOR BY OCEAN BARGE

J. M. Taylor and V. F. FitzPatrick
Pacific Northwest Laboratory*

INTRODUCTION

This paper describes the shipment of a portion of a contaminated, pressurized water reactor (PWR) steam generator from Surry, Virginia to Richland, Washington.

The shipment of the 200 metric ton steam generator was a first of a kind shipment. Prior to this shipment, no contaminated steam generator had ever been removed from a power station site.

The generator that was shipped was removed from operation because of defects, and was transported by ocean barge some 11,000 kilometers (6000 miles) from the Surry Power Station to Richland, Washington.

The availability of the generator presented a unique opportunity to investigate the behavioral characteristics of defected generator tubing, to gain valuable information related to PWR primary system integrity and degradation with time, and to validate nondestructive testing techniques.

The research work that will be done with the generator is associated with the Steam Generator Tube Integrity Program and is sponsored by the Metallurgy and Materials Branch, Reactor Safety Research Division of the Nuclear Regulatory Commission (NRC).

DESCRIPTION OF GENERATOR

The steam generator shown in Figure 1 is a vertical shell and U-tube evaporator with integral moisture-separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical channel head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Steam is generated on the shell (primary) side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The primary system of a reactor operates at temperatures in excess of 316°C (600°F) and at pressures of about 2.07×10^7 Pa (3000 psi). These extreme conditions contribute to the formation of corrosion products, some of which are circulated through the reactor core and become radioactive. The corrosion products over a period of time plate-out on the inside of the Inconel U-tubes. Essentially, all of the radioactive material that is associated with the generator is contained inside the U-tubes. The radioactive material, which is principally Co-60, is contained in a thin oxide layer, Figure 2, and is tenaciously attached to the Inconel tube wall.

*Prepared for the U.S. Department of Energy under contract DE-AC06-76RLO-1830.

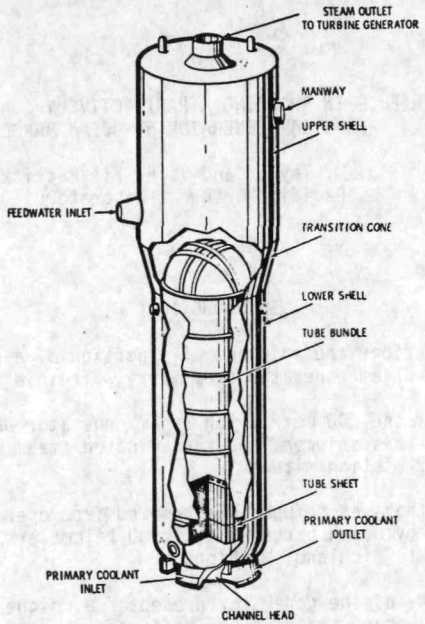


FIGURE 1. SECTION VIEW OF STEAM GENERATOR

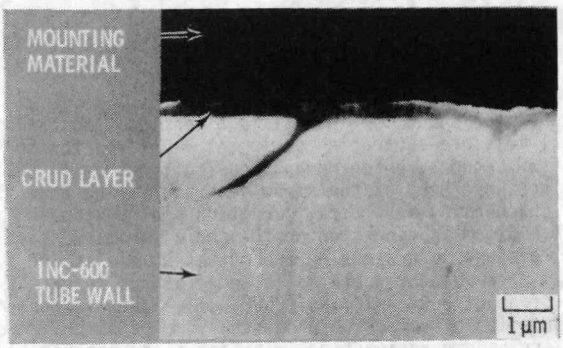


FIGURE 2. CROSS SECTION OF STEAM GENERATOR TUBING

The radioactive material is about 1 w/o of the corrosion product film. The radioactive material is distributed throughout the tubing and is about 3 microcuries per square centimeter. The total activity inside the generator is estimated to be between 120 and 180 curies. The estimate was made from dose rate measurements taken on the generator.

The part of the steam generator assembly that was moved to Richland was the bottom two-thirds of the generator, from the transition cone to the channel head, Figure 1. After the lower shell was separated from the upper shell a two inch steel shielding cap was placed over the open end of the generator.

PROGRAM ELEMENTS FOR TRANSPORT OF STEAM GENERATOR

The complexity of shipping the generator required a multidisciplinary approach. The various elements are shown in Figure 3. The principal elements involved in shipping the generator were: feasibility studies, program management and transport tasks. The block entitled other elements refers to tasks that will be associated with construction of a facility to house the generator for future research studies. This paper will deal only with the principal elements involved in shipping the generator.

Feasibility Study

The feasibility studies included determining whether a failed generator could be obtained, how it could be transported and where it could be located. It was determined that the NRC was very much interested in doing research on a full scale generator that had been removed from service. The studies also indicated that it would be possible to transport the generator to a number of sites where the generator could be examined and tested. Various ways were studied to ship the generator, these included: rail, barge, and rail plus barge. The conclusion of these studies based on cost, site support facilities, technical competence and a company's desire to take on the program was that the generator should be shipped to Richland, Washington by ocean barge.

Program Management

The program management for the project required a concerted effort to organize and administer the various tasks that were necessary to ship the generator. The various tasks involved: negotiation with Virginia Electric Power Company to obtain the retired generator, issuing the carrier contract and the contract to load and unload the generator on the barge, writing an environmental impact assessment, writing a management plan for DOE, issuing marine consultant contracts and obtaining approvals to ship the generator.

To accomplish these various tasks required the services of several support groups such as: the legal office, subcontracts personnel, engineering, health physics and transportation safety.

A significant task was obtaining a exemption to ship the generator. Because of the size (13.4 m long by 3.7 m diameter) and weight (200 ton) of the steam generator, it was not practical to package it in a container. The only reasonable way to ship the generator was to ship it as a Low Specific Activity (LSA) package. The generator shell would have to serve as the container for the contaminated tubing. The generator shell could be classified as a strong-tight container since

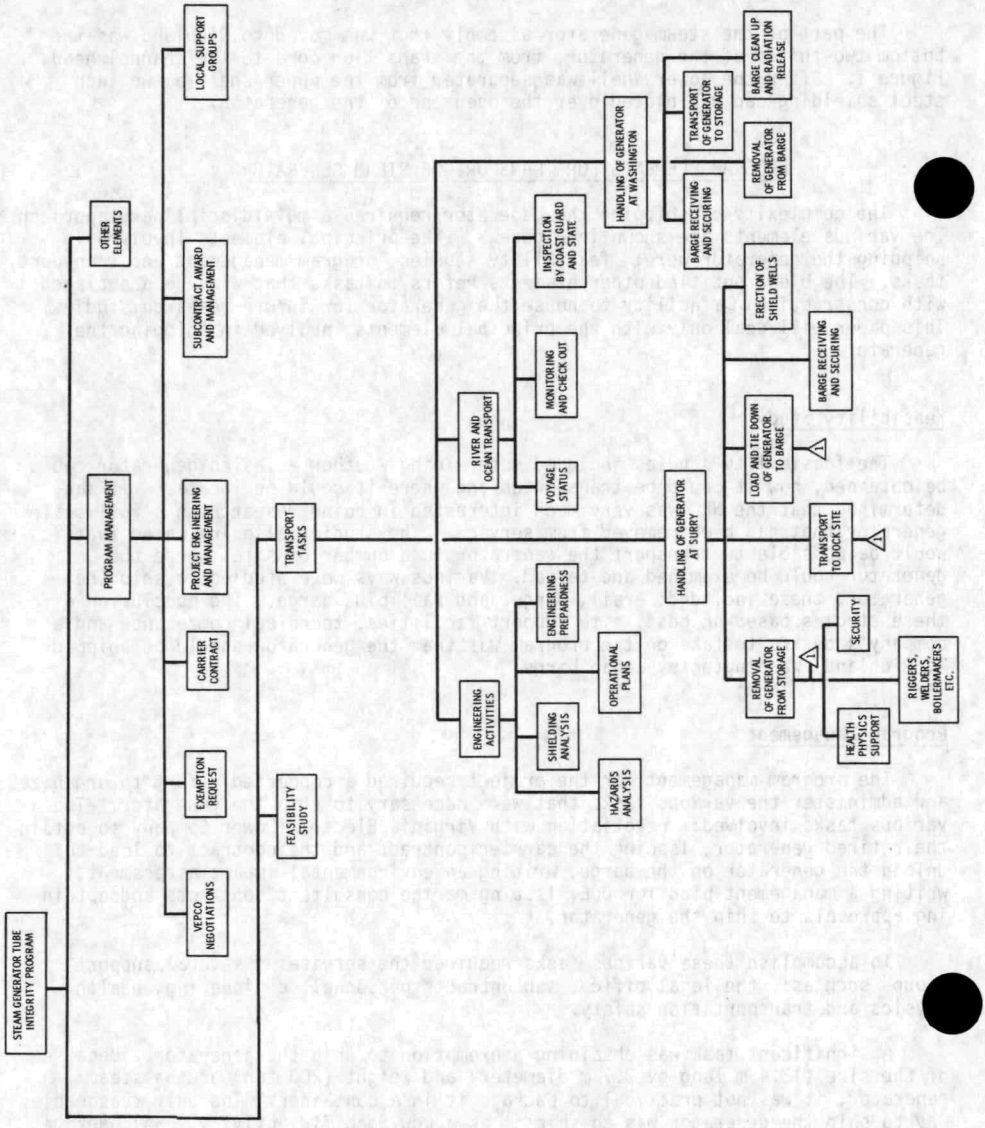


FIGURE 3. PROGRAM ELEMENTS FOR TRANSPORT OF STEAM GENERATOR

it was ~7.62 cm (3 inches) thick and designed to withstand pressures up to 2.07×10^7 Pa (3000 psi) at temperatures of 316°C (600°F).

United States regulations state that in order to qualify as an unpackaged bulk shipment of LSA material, the object that is externally contaminated with radioactive material must not exceed 1 microcurie per square centimeter. This pertains to the kind of radionuclides that were associated with the steam generator. The contaminate concentration inside the heat exchanger tubing exceeded this contamination level by three times. Consequently, it became necessary to obtain an exemption from the Department of Transportation (DOT) in order to ship the generator as a LSA package.

Transport Tasks

The transport tasks included: engineering activities, handling of the generator at Surry, river and ocean transport and handling the generator at Washington.

Engineering Activities

It was necessary to do a hazard analysis that assessed the radiological impact to the environment in the event the generator was involved in accidents during loading, unloading, shore transport and water transport. The accidents assessed were dropping, sinking, fire and sabatoge. The results of the analysis showed that for the accident cases analyzed, the population dose would be significantly less (10^{-4} to 10^{-8}) than the annual dose received from natural occurring sources. The hazard analysis became part of the environmental assessment for the Steam Generator Tube Integrity Program.

Detailed operational plans were written covering the handling, loading, shipping and unloading of the generator. These plans were reviewed by DOT, the U.S. Coast Guard and the NRC before the exemption was granted. The plans included emergency procedures to cover remedial action in the case of fire, loss of shielding and operational problems with the tug or barge.

The radiation dose rate at the surface of the generator (highest reading 750 mR/hr) required that shielding be used to reduce the dose rate to meet the requirements for an exclusive use shipment. Based on the shielding analysis and other considerations, the generator was placed between two concrete walls that were each 25.4 cm (10 inches) thick. This type of shielding proved to be the most cost effective.

In addition to the concrete shield walls, a 2" steel plate was added to the capped end of the generator. This provided a total of 4" of steel shielding on the capped end of the generator. The highest dose rate at the edge of the barge with the generator inside the shield walls was 0.5 mR/hr. The dose rate was about 20 times less than the allowable 10 mR/hr at 1.83 m (6 feet) from the edge of the barge.

Handling the Generator at Surry

The bulk of the work associated with the actual shipping of the generator took place at Surry, Virginia. Complete radiation surveys were made of the generator to assess surface contamination and radiation dose prior to removing it from an onsite storage vault. After the surveys were completed the generator was removed from the

storage vault and loaded onto a multi-wheeled trailer. The generator was unloaded onto the barge. The generator was tied down and secured between the concrete shield walls as shown in Figure 4.

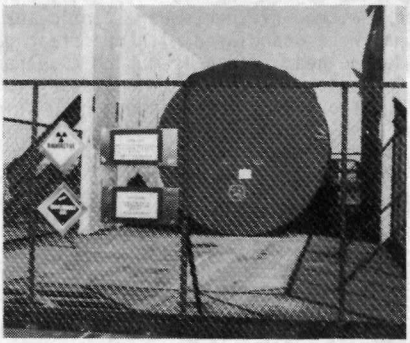


FIGURE 4. STEAM GENERATOR ON BARGE INSIDE CONCRETE SHIELD WALLS

After the generator was secured to the barge, a thorough inspection was made by the Coast Guard and by state inspectors. After the inspection was completed, approval was given to ship the generator. On April 20, the tug and barge proceeded down the James River to the Atlantic Ocean. The start of an 11,000 kilometer voyage, Figure 5, that would go through the windward passage to the Panama Canal and then up the West Coast to the Columbia River.

FIGURE 5. STEAM GENERATOR TRANSPORTATION ROUTE

After 35 days the generator arrived on May 25 at the Port of Benton unloading site near Richland, Washington. The entire voyage was made without incident.

Unloading the Generator at Port of Benton

Unloading the generator from the barge was similar to the loading operation. Two large capacity cranes (200 ton each) were used to load the generator onto an awaiting trailer. The generator was then transported about a mile and a half to a temporary storage facility. Eventually, the generator will be housed in a permanent building where research studies will continue for three to five years.

ACKNOWLEDGEMENT

The authors acknowledge the Reactor Safety Research Division of the Nuclear Regulatory Commission (NRC) for their support and sponsorship that made it possible to ship the steam generator. The authors also extend their thanks to the Virginia Electric Power Company (VEPCO) for enabling Battelle, Pacific Northwest Laboratories, to obtain the generator and for their assistance in preparing the generator for shipment.

DISCUSSION

Question by A. A. Anselmo, USA: Did you consider or make a study of transporting the steam generator on the transporter to eliminate the unloading at barge plus the reloading at destination? Possible time and cost savings? What was the deciding factor to remove and double the handle?

Answer: The possibility of leaving the steam generator on the transporter and tying the generator and transporter to the barge was considered. However, because of the 1.5 g tie down requirement suggested in the proposed ANSI standard N-14-N552 it was not practical to leave the generator attached to the transporter. The extra structural attachments and modifications to the barge and transporter were not cost effective compared to lifting the generator off the transporter and tying the generator to the deck of the barge.

PACKAGING AND TRANSPORT OF BULKY WASTE FROM NUCLEAR POWER PLANTS

H.-G. Knackstedt
R. Christ
K. Schneider

TRANSNUKLEAR GMBH

1. Introduction

Reactor operation and backfitting will result in voluminous and heavy parts which have to be shipped off-site in shielded containers. Depending on weight and dimensions as well as on the strategy of disposal the parts may be transported in one piece or after size reduction.

Typical examples for the preparation and transport of such components will be given below.

2. Transport of a contaminated BWR steam dryer

The steam dryer is a stainless steel structure composed of several sheets and pipings. Dimensions: 5140 mm diameter, 5230 mm hight. Total weight: 35 t. The component had to be put out of service and was temporarily stored inside the plant.

Due to contamination the dose rate at contact was about 800 mr/h. The contamination revealed to be very adherent. An analysis of activity inventory and contamination showed that the component complied with the regulatory definition of LSA material for which a strong industrial package was to be used.

The utility decided to dispose the component by shipping it to a research center with installations for final conditioning. Transnuklear got in charge for preparation and transport.

2.1 Shipment of the complete component

In an early stage of the project the possibility of shipping without prior size reduction was investigated in detail. A shielding container was designed and the use of different shielding materials including concrete was studied (fig. 1). Container
weight was about 110 t resulting in a total package weight of 150 t. For the transport itself, a combination of road and barge shipment was the preferred solution. Limiting factors such as bridge loads for the total vehicle weight of around 250 t had to be taken into account (fig. 2).

Though the investigation showed the project of shipping to be technically feasable, it had to be deleted primarily for financial reasons.

2.2 Shipment after size reduction

It was finally decided to cut the component prior to shipment into several pieces each smaller than 5 t. A lead shielded container with a weight of 25 t was designed and built in order to make several shipments by road under close to "legal weight" conditions (fig. 3).

The size reduction at the reactor site was done by means of the oxygen lance method. A special working area was set up. In that area the component was placed on a rotating table and cut according to a plan which was taking into account the useful dimensions of the container. Prior to loading the parts are placed in plastic bags.

The first shipments have already been performed successfully in compliance with German regulations for the road transport of dangerous goods and a relevant shipping licence for nonfissile radioactive material.

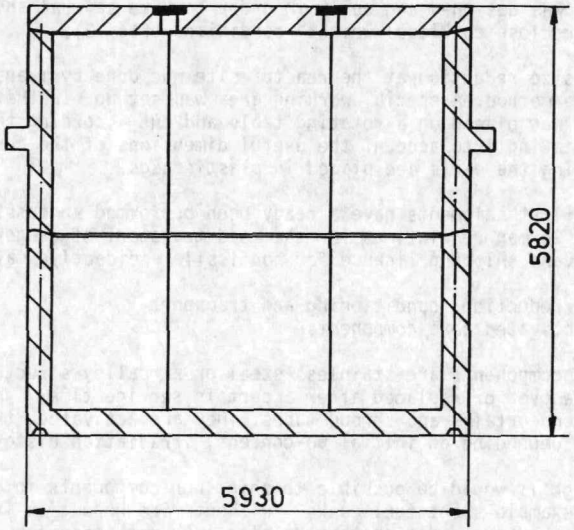
3. Size reduction, conditioning and transport of activated core components

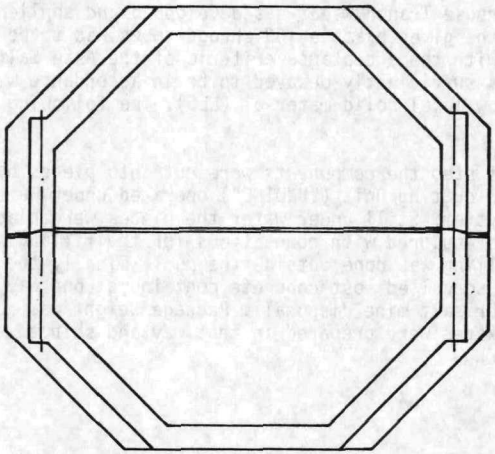
Core components are stainless steel or zircalloy structures which have to be removed or replaced after a certain service time. Typical examples are poison curtains and shroud tubes. They are activated, the specific activity depending on initial Co-content, irradiation history and decay time.

Though it would be possible to ship such components in existing casks - for example spent fuel casks - without size reduction it is sometimes preferred to condition them on site into packages ready for disposal.

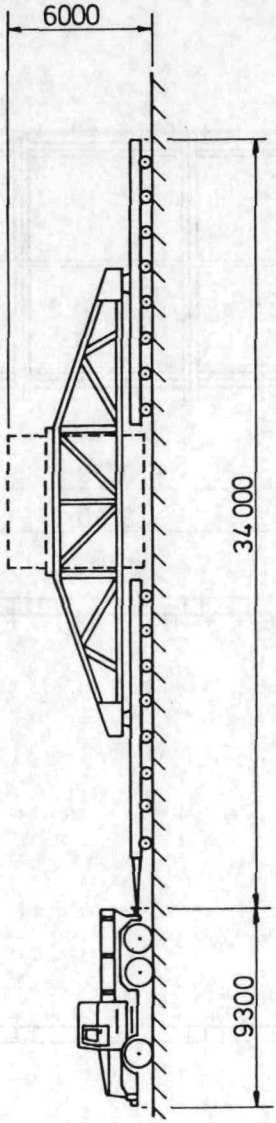
For this purpose Transnuklear has developped and applied an appropriate system. In the given example 169 shroud tubes had to be disposed of in compliance with the acceptance criteria of the Asse Salt Mine. The material was sufficiently decayed to be in accordance with the definition of low level solid material (LLS). The following procedure was applied.

In the first step the components were cut into pieces by a special pressing and cutting unit ("UNDINE") operated under water by means of a hydraulic system. Still under water the pieces were placed into lead shield drums equipped with connections for the filling with concrete. Concrete filling was done outside the pool. Finally the drums were placed into so called lost concrete containers (one way specification container for salt mine disposal). Package weight was about 5 t. More than 50 packages were prepared in that way and shipped to the disposal site by truck.





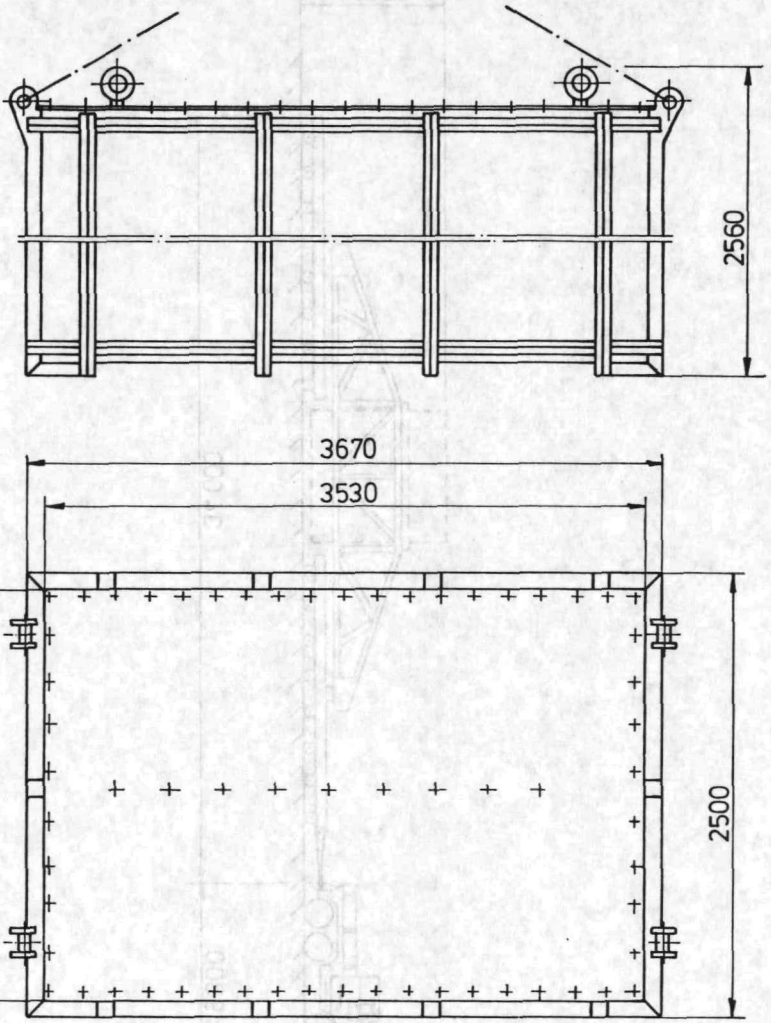
TN	Concrete container	Fig. 1



TN

Road transport vehicle

Fig. 2



TN 30 container

Fig. 3

DISCUSSION

Question by H.-J. Engelmann, FR Germany: Have you optimised the size reduction due to the conditions of road transport or have you made an analysis of costs?

Answer: The size reduction of the parts was adapted to conditions of road transport.

Question by Mr. Schmidt, FR Germany: Concerning costs for packages and transport, what would have been the cost for alternative 2? What was the cost for the chosen alternative 2.

Answer: The analysis and conditioning of the steam dryer was part of a customer order. We are therefore under obligation to keep this information confidential.

ROCKWELL INTERNATIONAL'S PACKAGING AND SHIPPING PROGRAM AT THE ROCKY FLATS PLANT

J. D. McCarthy
D. M. Krieg
Rockwell International
Rocky Flats Plant
P. O. Box 464
Golden, Colorado 80401

INTRODUCTION

Rockwell International operates the Rocky Flats Plant under contract to the Department of Energy. The plant has been in operation since 1952, and has been engaged in handling many forms of radioactive material, including large quantities of fissile material. Operations principally involve plutonium and uranium in the form of metal, solution, oxide, and other compounds. General operations consist of metallurgical processes and chemical processing and recovery operations. Support services are provided in the areas of quality, laboratory facilities, waste management, research and development, and health, safety, and environment protection.

Associated with these operations is the packaging and shipping of radioactive material. Through the years, thousands of shipments have been made from Rocky Flats without incident. Many forms of radioactive materials, including waste material, have been included in these shipments.

In order to assure compliance with Company, DOT, and DOE requirements, an extensive system was established that encompasses all essential aspects of packaging and shipping. This integrated system is defined by a Material Packaging, Shipping, and Transportation Plan that dictates not only the development, initial use, and reinspection of containers for subsequent shipments, but also the required documentation, procedures, training, and management responsibilities. Imperative to this program is the certification of operating personnel to assure extensive training and knowledge in all pertinent aspects relating to packaging and transporting radioactive material. In addition, the evaluation of the package is an essential element to insure the integrity of the container and confinement of the contents and to assure compliance with health, safety, and environment commitments.

A Quality Program Plan has also been established which defines the responsibilities, authorities, and controls in effect at Rocky Flats. This plan supplements the Material Packaging, Shipping, and Transportation Plan.

MATERIAL PACKAGING, SHIPPING, AND TRANSPORTATION PLAN

This plan documents the overall system and the corresponding compliance with regulations governing the packaging, shipping, and transportation of material. The plan is controlled by the General Manager, and responsibility for implementation and compliance is delegated to the Traffic Manager.

The plan applies to all aspects of the packaging, shipping, and transportation of material to and from the Rocky Flats Plant. Radioactive material, hazardous material, classified nonhazardous material, and contaminated waste generated during normal plant operations are included within the scope of the plan.

Requirements apply to inbound and outbound movement of material. These requirements provide the means of control and verification whereby those responsible for the packaging, shipping, and transportation of material can assure that the operations are completed in a safe, environmentally-sound, and economical manner.

The plan also contains information of a general nature relevant to packaging, shipping, and transportation of material, as well as specific details of well-defined activities.

Emphasis is placed on those activities which require interface between plant personnel and regulatory and advisory agencies, which include the Department of Transportation (DOT), the Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), the Federal Aviation Administration (FAA), the International Air Transport Association (IATA), the General Accounting Office, the State of Colorado Department of Health, and the Rocky Flats Monitoring Committee.

In addition to the above agencies, the Traffic Department interfaces with the DOE Albuquerque Operations Office and the Rocky Flats Area Office, NRC licensees, other DOE contractors, the DOE contractor air carrier, airport authorities, and common carriers.

Internal responsibilities as delineated in the Material Packaging, Shipping, and Transportation Plan include the following:

Finance and Administration:

Traffic: Approves all packaging procedures; plans shipments; notifies authorities; interprets regulations and requirements; obtains approvals and exemptions; certifies all off-site shipments; maintains the Transportation Plan.

Product and General Warehouse: Stores and issues shipping containers and components; receives, verifies, and issues material; packages and ships material as required; certifies correct preparation of packages by Warehouse personnel.

Health, Safety, and Environment: Reviews packaging procedures; provides radiation monitoring support; reviews requests for shipment of nonweapons-related radioactive and hazardous

material; maintains source and isotope registry; determines nuclear materials safety limits and calculates transport indices.

Personnel: Trains or qualifies plant personnel; maintains training and qualification records.

Safeguards and Security: Controls security of inbound and outbound shipments; provides accountability control for source and special material; obtains authorization which assures compatibility of incoming materials from other contractors.

Operations:

Material Handling and Packaging: Prepares and approves applicable procedures for support of packaging and related activities for all material covered by the plan except waste; packages material; certifies correct preparation of packages.

Waste Processing: Final packaging and loading; certifies correct preparation of packages; prepares and approves applicable procedures for support of waste packaging and related activities; provides design criteria for Waste Processing shipping containers and components.

Quality Engineering and Control: Implements and maintains Quality Program Plan; audits compliance to Quality Program Plan; verifies container qualification tests; inspects and certifies shipping containers and components; analyzes proposed packaging materials; provides standards and calibrations; reviews all packaging procedures.

Research and Engineering:

Product Definition: Designs, develops, and tests shipping containers and components; approves packaging procedures; provides configuration control for shipping containers and components; specifies packaging, marking, and labeling requirements for off-site shipments.

Component Development: Evaluates and tests proposed packaging materials.

The following is a summary of activities that are controlled to assure compliance with requirements specified in 10 CFR 50, 49 CFR, DOE Manual Chapters 0529 and 5201, and DOE Order Number AL 5610.1.

Container Development

All containers and corresponding packaging configurations are designed to meet DOT and DOE regulations and appropriate standards. Included are Type A, Type B, Special Form, LSA, and limited and large quantities as specified in 10 CFR 71, 49 CFR 173, DOE Manual Chapter 0529, and DOE Order AL 5610.1. During this phase, the

test packages are designed, fabricated, and tested. The testing, where appropriate, considers both normal and abnormal conditions.

Health, safety, and environment protection evaluations, based on the design and the results of the testing, are also performed. This data is included in the Safety Analysis Report for Packaging (SARP), which is submitted to the appropriate government agency for approval. The packages primarily in use for fissile material are DOT specification or have approved DOE Certificates of Compliance.

Initial Use and Reuse

An essential part of this program is the preparation and control of the drawings, specifications, and associated inspection check lists to assure the package meets requirements initially and for reuse. This function is controlled by an integrated system for both incoming and outgoing shipments. Included are activities pertaining to verification of package conditions, marking, labeling, monitoring for radiation, material identification, etc.

Personnel Training

In order to assure quality operations, personnel involved in packaging and transportation activities (material handlers, packagers, security personnel involved in transfer vehicle functions) must fulfill training, qualification, and certification requirements. This program includes indoctrination, on-the-job training, and formal classroom instruction. Proficiency must be demonstrated in all aspects pertaining to a particular position prior to certification. The substance of the training is dependent on the complexity of the operations, the expectations, and the consequences of an error.

Documentation

Imperative to the effective functioning of the overall system is completion of the necessary documents, forms, and reports to monitor and verify fulfillment of the control activities. The system has been established to assure that activities do not proceed until the previous control point activity has been completed and verified. These documents are maintained in a manner whereby they are readily accessible for monitoring, verification, and auditing purposes.

These functions associated with the overall packaging and transportation plan are implemented through individual procedures that define the responsibilities and the performance of the operations in detail. In addition, the plan designates the appropriate signature authority responsible for the various functions. Rockwell departments that participate in this program and that have a vested interest in its success are varied and include Product Definition, Nuclear Materials Control, Traffic, Safeguards and Security, Health Sciences, Radiation Monitoring, Quality, Nuclear and Facilities Safety, Operations, Warehouse, and Material Handling and Packaging.

QUALITY PROGRAM PLAN

An oversight of the program is provided by periodic auditing by Rockwell and DOE personnel. These reports provide management with a means for assessing the effectiveness and implementing modifications where necessary to meet requirements.

An overview is provided by the Quality Program Plan that is controlled by the General Manager and which delineates responsibilities for development, implementation, and auditing activities involved with packaging and transportation of hazardous materials, including radioactive material. Management, guided by various DOE and DOT regulations, establishes the requisite policies and approves the implementation procedures. The Quality Program Plan defines the appropriate control points to assure that the packaging and transportation activities are performed in a manner that does not present a hazard to the public, employees, or the environment. Although the plan provides the necessary guidance and control, the packaging and transportation system at Rocky Flats functions effectively because of the close cooperation of all departments.

SUMMARY

The public continually needs to be made aware that radioactive material can be handled safely, especially in an area as sensitive as shipping. Therefore, Rockwell International monitors the program carefully to minimize risks to the public, employees, and the environment. This program has functioned effectively in meeting present requirements, and provides a framework that can be extended to accommodate future modifications.

A REVIEW OF US ACCIDENT/INCIDENT EXPERIENCE INVOLVING THE TRANSPORTATION OF RADIOACTIVE MATERIAL (RAM) 1971-1980*

J. D. McClure and E. L. Emerson
Sandia National Laboratories, Albuquerque, NM 87185 USA

This paper analyzes the transportation accidents and incidents which have occurred in the United States in the period 1971-1980 based upon the information in the Radioactive Material Transportation Accident/Incident Data Base developed by the Transportation Technology Center (TTC) at Sandia National Laboratories. The accident/incident data base incorporates the files of the Hazardous Material Incident Report (HMIR) system operated by the Material Transportation Bureau of the US Department of Transportation (DOT) with additional information obtained from the files of the US Nuclear Regulatory Commission (NRC). A principal objective of this paper is to summarize US accident/incident experience for the past ten years,[1] providing a concise statement of radioactive material (RAM) package failure description for the transport modes of truck, rail and air.

Operated for the US Department of Energy (DOE) by Sandia National Laboratories, the TTC acquires hazardous material incident reports from the (DOT) on a continuing basis. This information, plus similar information acquired from the NRC, provides a radioactive material accident/incident data base which presently (June 1980) consists of 659 data entries. The present holdings of the HMIR system approximate 86,500 entries for all classes of hazardous materials, including 524 radioactive material entries, or 0.6 percent of the total incidents reported to DOT. This information represents US transportation experience for radioactive materials dating from 1971 to the present. The accident/incident data base is stored on TTC's on-line data base management system for rapid storage, retrieval, editing and analysis.

The DOT regulatory requirements for reporting a hazardous material incident are specified by the Code of Federal Regulations (49 CFR, Para. 171.15). These regulations specify reporting after each incident that occurs during the course of transportation (including loading, unloading and temporary storage) in which as a direct result of moving hazardous material: (a) a person is killed, (b) a person receives injuries requiring hospitalization, (c) estimated carrier or other property damage exceeds $50,000, (d) in the case of radioactive material there is suspected contamination, (e) fire, breakage or spillage occurs involving etiologic agents, (f) continued danger of life exists at the accident scene. From this summary of the requirements for reporting a hazardous material incident, one can conclude that the requirements for reporting an incident vary widely. The 659 data entries in the accident/incident data base are events which meet the regulatory requirements for notification of DOT and NRC of the event occurrence. The main theme of this presentation will be that a detailed examination of the reports in the data base will reveal that some of the events can be classified as transport accidents, some events involve handling accident conditions and others are simply incidents which meet the regulatory requirement for reporting the event and do not involve accident conditions. Such a classification of events is shown in Table I and is necessary to produce, for example, accurate accident rate estimates for

*Work sponsored by the US Department of Energy under Contract DE-AC04-76DP00789.

risk assessment studies. Table I represents an up-to-date representation of US radioactive material transportation experience.

TABLE I

US Radioactive Material Accident/Incident Experience
(1971-1980)

Transportation Accidents	85
Handling Accidents	110
Reported Incidents	464
Total	659

In the interpretive analysis of the basic DOT and NRC information, a distinction is made between an accident and a reported incident. Two types of accidents are categorized, as shown in Table I. For the purposes of this analysis, a reported incident is classified as a transportation accident if there is a vehicular accident involving the vehicle transporting the radioactive material. Other incidents are classified as handling accidents if during the course of handling, loading, etc., an accident occurs, for example, a package is dropped or run over by a vehicle.

The remaining classification, the reported incident, excludes transportation accident conditions. An example of a reported incident typically involves the detection of surface contamination on a radioactive material package or on a transport vehicle. These observations are often noted during loading or unloading operations. Such events generally involve no package failure, no release of radioactive contents and the absence of any accident conditions during transport. For this type of event the terminology "reported incident" has been used to classify the event. For the US experience to date, it can be observed from Table I that the total reported events include 85 transportation accidents and 110 handling accidents.

The 85 transportation accidents listed in Table I were classified by transportation mode as shown in Table II.

TABLE II

Radioactive Material Transportation Accidents by Mode

Transport Mode	Number of Accidents	Number of Packaging Failures
Air	5	1 Type A
Rail	5	1 Type A
Highway	75	39 Type A
	85	41 Packaging Failures

Table III summarizes some of the information that can be retrieved from the accident/incident data base. For example, the 85 transportation accidents involved a total of 711 radioactive material packages. Some accidents had only a single package on board the transport vehicle. Other accidents involved a single transport vehicle with a number of packages on board. Table III traces an analysis of the radioactive material packaging failures in the 85 accidents.

TABLE III

Transportation Accident Analysis Summary

RAM Packages Involved in Accidents	Package Description	Description of Events		Description of Material Released
Packaging Failures* (with release)	38 Type A -->	5 Release --> Events	3 Urban ------>	•Uranium Ore •Sand (LSA) •RAM (not otherwise specified)
			2 Non-Urban -->	•Uranium Oxide
Packaging Failures* (no release)	3 Type A			
Packagings in Accident with No Failures	660 Type A 10 Type B			
TOTAL	711 packages involved in 85 accidents			

* The 41 packaging failures involved 38 Type A paekages which released radioactive material.

Multiple numbers of packages can be carried on a single vehicle and five separate and distinct release events involving these packages have been defined. Three of the release events occurred in an urban area. Two of the release events occurred in a non-urban area. For example, the Springfield, Colorado, accident involved 29 packaging failures which released approximately 5400 kg of uranium oxide; this was defined as a single release event. For the five release events, the material released was described as uranium ore, radioactive sand (low specific activity), uranium oxide and radioactive material (RAM -- not otherwise specified). It is interesting to note that an examination of the accident data reveals that ten Type B accident resistant packagings were included in the set of packagings which were subjected to accident conditions, but produced no packaging failure and, consequently, no release of radioactive contents.

Tables IV, V and VI present a chronological summary of the data base with respect to the number of reported events, transportation accidents, handling accidents,

TABLE IV

Accident/Incident Data Base Chronological Summary

Year	Annual No. Releases	Annual No. Events	HMIR* Events	NRC** Events	Transportation Accidents	Handling Accidents
1971	5	10	10	N/A	5	0
1972	4	19	19	N/A	1	2
1973	7	24	24	N/A	0	11
1974	9	59	59	N/A	5	14
1975	11	44	44	N/A	5	23
1976	14	59	57	2	2	6
1977	16	112	96	16	10	16
1978	18	140	85	55	19	14
1979	25	170	120	50	29	23
1980 (through 5/8/80)	5	22	10	12	9	1
	114	659	524	135	85	110

*Reported events received from HMIR system, DOT.
**Reported events received from NRC.

TABLE V

RAM Transportation Accident Summary

Year	Transportation Accident Total	Transport Mode				Releases
		Rail	Highway	Air	Other	
1971	5	1	4	0	0	1 Release-1971-Hwy
1972	1	0	1	0	0	
1973	0	0	0	0	0	
1974	5	1	4	0	0	
1975	5	0	5	0	0	
1976	2	0	2	0	0	
1977	10	1	9	0	0	1 Release-1977-Hwy
1978	19	0	18	1	0	1 Release-1979-Air
1979	29	2	23	4	0	1 Release-1979-Rail
1980 (through 5/8/80)	9	0	8	0	1 (Courier Service)	1 Release-1979-Hwy
	85	5	74	5	1	5 Release Events

TABLE VI

RAM Handling Accident Summary

Year	Handling Accident Total	Transport Mode				Releases
		Rail	Highway	Air	Other	
1971	0	0	0	0	0	0
1972	2	0	0	2	0	1, Air
1973	11	0	2	9	0	2, Air
1974	14	0	1	13	0	2, Air
1975	23	0	6	14	2, Freight Forwarder 1, Marine	2, Hwy 1, Air 1, Frt Frd 1, Marine
1976	6	0	2	4	0	1, Air 1, Hwy
1977	16	1	4	9	1, Freight Forwarder 1, Warehouse	4, Hwy 1, Warehouse 1, Rail 1, Air
1978	14	0	7	7	0	1, Hwy
1979	23	0	9	13	1, Courier Service	3, Air 3, Hwy
1980	1	0	0	1	0	0
	110	1	31	72	6	26

release of radioactive contents and accident occurrences by transport mode. The following observations apply to Table IV. The accident/incident data base receives most of its data entries from the HMIR system of the DOT. The other principal data source is the NRC. The HMIR data originated in 1971 and the NRC data originated in 1976. Any calculation of rate of occurrence of accidents and incidents should be performed with caution since the overall accident/incident data base consists of elements with differing chronological origination dates.

The most representative view of accident and incident rates is to express their occurrence as fractions (or percent) of total shipping volume. The most recent estimate of total shipping activity in the US was performed in 1975.[2] Estimates of present shipping activity are presently being investigated and are the subject of another paper at PATRAM 80 (see PATRAM paper 185). Until new estimates of shipping activity are available it is not possible to gain an accurate assessment of the percentage of accidents and incidents based upon numbers of radioactive material shipments.

To conclude, it is important to note that the accident/ incident data base described in this paper represents the accumulation and organization of existing data sources available in the United States. This data acquisition is followed by interpretive analysis of the information, and this analysis and data can be shared with several government agencies such as the DOE, NRC, DOT, Environmental Protection Agency (EPA), Federal Emergency Management Agency (FEMA) and the International Atomic Energy Agency (IAEA), as well as other interested state and

local governments and public interest groups. One important fact that has come from the preparation of this report is the importance of performing a detailed study of the incident reports and any attached documentation. It is recognized that after such an examination of radioactive material transportation incident data that all of the reported events are not transport accidents, hence the stratification of the accident/incident data as shown in Table I. Such a classification as shown in Table I is necessary because the varying requirements for reporting hazardous material incidents allow a broad spectrum of data entries which must necessarily be sub-classified (i.e., transport accidents, handling accidents and reported incidents).

The information presented above represents an analysis of a transportation accident/incident data base for radioactive materials. The results of such an analysis can be used to provide information about the environmental impacts associated with the transportation of radioactive materials. In addition, such an analysis can provide information which is useful in the formulation of the regulations governing the safe transportation of radioactive materials.

REFERENCES

1. A. W. Grella, A Review of Five Years Accident Experience in the U.S.A. Involving Nuclear Transportation, IAEA-SR-10/5, International Atomic Energy Agency, August 1976.

2. Transport of Radioactive Material in the U.S., A Detailed Summary of Survey of Radioactive Material Shipments in the United States, BNWL-1972, NUREG-0073, Office of Standards Development, U.S. Nuclear Regulatory Commission, May 1976.

DISCUSSION

Question by D. H. Lohmann, United Kingdom: Could you comment please on the type A packages from which activity was released? Have any steps been taken to improve packaging as a result?

Answer: I do not recall the exact description of each of the packages (type A). However, in the Colorado yellowcake incident the packages were 55 gal. drums. For specific improvements, perhaps closures of such drums could be redesigned. However, it is important to note that from the health hazard point of view, the Colorado yellowcake spill, while it did occur (and released 12000 lbs. of yellowcake), there were no injuries due to the radioactive material. Thus, from an engineering viewpoint the packages performed properly. We must realise, however, that it is our technical and professional responsibility to make needed packaging improvements when that need is observed.

Question by Y. Sousselier, France: Which criteria do you use to distinguish between accident with package failure and accident without failure?

Answer: The HMIR system uses a form called Form 5800. We simply examine the entries on the form and use the information in the failure type entry information to make the judgement that there was a packaging failure. Sometimes, however, the information is not clear so we must exercise engineering judgement to ascertain failure occurrence.

TRANSPORT OF RADIOACTIVE MATERIALS

A REVIEW OF THE DAMAGE TO PACKAGES FROM THE RADIOCHEMICAL CENTRE DURING TRANSPORT

D.H. Lohmann
The Radiochemical Centre, Amersham, England

INTRODUCTION

The Radiochemical Centre, Amersham, is one of the world's largest producers of radioisotopes for use in medicine, research and industry.

Its products are delivered to all parts of the world, with approximately 80 per cent of the products being exported, mostly by air, from the United Kingdom. The quantities of activity packed and transported vary from fractions of a microcurie - eg for in vitro diagnostic kits - to hundreds of thousands of curies - eg for cobalt-60 for industrial sterilization.

This is the first full report of damage that has occurred to packages being transported from The Radiochemical Centre though the subject has been briefly covered in two earlier papers dealing with the packaging and transport of radioactive materials (1,2).

This paper summarizes the results of our experience of the transport of radioactive materials from The Radiochemical Centre. Those occurring in the period 1975 to date are covered in detail.

PACKAGING

The packaging of radioisotopes produced at The Radiochemical Centre is in accord with the IAEA regulations for the safe transport of radioactive material (3). Exempt and Type A packages are designed, tested and approved at The Radiochemical Centre. Type B packages are designed and tested at The Radiochemical Centre and are approved by the U.K. competent authority.

Exempt Packages

The basic pack for most exempt packages consists of a number of vials containing inactive and small quantities of active materials packed into a moulded polystyrene support and this then packed in a rigid cardboard case.

Type A Packages

A typical Type A package consists of a vial containing a solution of a radiopharmaceutical or radiochemical. This is packed into a secondary container that provides containment and radioactive screening as appropriate - either a plastic container providing purely containment or a lead pot with wall thickness

varying from 3 - 36mm for routine shipments. The vial is held in the secondary container by small pieces of absorbent material. The secondary container is then supported - usually by expanded polystyrene mouldings - and sealed into a metal can. The metal can is sealed inside a rigid cardboard case and this labelled with the appropriate warning label.

Increasingly, Exempt and Type A packages are being grouped together in multipacks (or overpacks). This minimises the number of final packages transported and hence reduces the risk of loss or damage.

Type B Packages

Type B packages vary in size and composition from basic Type A assemblies sealed inside simple heat resisting shields to large, bolt-down containers with cooling systems used for transporting hundreds of kilocuries of activity.

DAMAGE OF TRC PACKAGES DURING TRANSPORT

Road

In the period under review, there have been a number of minor accidents to vehicles carrying TRC's products. These have resulted in no more than superficial damage to the packages.

There have been two major accidents to vehicles carrying TRC products. (1) In December 1978, a contractor's vehicle carrying four caesium-137 sources (total 30mCi) in standard Type A containers was involved in an accident that completely destroyed the vehicle and resulted in the driver being killed. The outer packages were severely dented but there was no damage to the inner containers (lead pots with wall thickness of 12mm) or any loss of activity. This incident was monitored under the scheme known as the National Arrangement for Incidents involving Radioactivity (NAIR) and the damaged packages were returned to TRC for detailed examination. They were despatched again after replacing only the outer containers. (2) In January of this year, a TRC van making a regular delivery of materials to the continent skidded off of a motorway on the outskirts of Brussels down on to an embankment with a vertical drop of approximately 5 metres. This resulted in the vehicle being damaged beyond repair. At the time, the vehicle was carrying a load of 116 packages (made up of 17 inactive, 36 exempt, 61 Type A (1 Category I, 15 Category II and 45 Category III) and 2 Type B packages, Three of the Exempt packages were damaged superficially and the outer wooden casing of one of the Type B packages suffered severe damage. In no case was there any damage to the inner container or loss of activity. This incident was monitored by Controlatom (the organisation for physical control of radioactivity in Belgium). Again, the whole consignment was transported on to its final destination after reloading on to other vehicles.

Air

Each year we despatch approximately 70,000 packages by air from London Airport to all parts of the world. During the period under review, there was one major accident to an aircraft carrying radioactive materials for TRC. This was at Athens Airport in December 1979; this is reported in detail in another paper (4).

Airport Handling

The largest proportion of accidents to packages containing radioactive materials from TRC has occurred during handling at London Airport. The damage has usually been the result of careless handling of the package and it's being crushed by a vehicle. The most common type of accident is when a package falls from a pallet of a fork lift truck and is subsequently run over. If accidents of this type occur, the package is isolated, checked by local control, NAIR assistance requested if necessary, the package sealed into a PVC bag or metal drum and then returned to TRC for detailed examination.

The number of packages transported and those reported damaged are summariz in Table 1.

TABLE 1

Survey of Packages Damaged at London Airport during Transport 1975 to date

	Total individual packages from TRC (x1000)	Total final packages (x1000)	Total final packages through London Airport (x1000)	Damaged Packages
1975	200	124	61	13
1976	214	138	73	22
1977	236	154	86	20
1978	255	147	77	16
1979	263	155	67	12
1980 (Jan-Sept)	263	124	51	6
	1,431	842	415	89

These may be further subdivided into categories as follows:-

TABLE 2

Summary of Damage to Packages by Transport Category

	1975	1976	1977	1978	1979	1980 (to date)	Total
Inactive	-	-	1	1	-	-	2
Exempt	2	5	8	5	3	4	27
I	1	6	3	3	2	-	15
II	-	-	-	-	-	1	1
III	10	10	8	7	7	1	43
B	-	1	-	-	-	-	1
Total	13	22	20	16	12	6	89

The damage experienced by a package can vary from a slight surface scuffing to a complete crushing of the package with loss of activity. The severity of damage may be summarized as follows:-

TABLE 3

Summary of Damage to Packages by Severity of Damage

	1975	1976	1977	1978	1979	1980 (to date)	Total
A. Slight damage to outer	-	4	6	3	1	1	15
B. Extensive damage to outer	-	8	6	3	4	3	24
C. Slight damage to can	2	3	3	4	3	-	15
D. Extensive damage to can	10	7	5	4	2	1	29
E. Damage to shielding pot	-	-	-	-	1	-	1
F. Vials broken	1	-	-	2	1	1	5
G. Loss of activity	-	-	-	-	-	-	-
Total	13	22	20	16	12	6	89

On some earlier occasions, packages that had been reported as leaking were subsequently found to be free from external contamination. This was due to incorrect use of monitors. With close liaison and a period of training, London Airport staff now have a good appreciation of the handling of radioactive materials and the use of monitors.

In only one instance (in 1974) has there been any confirmed leakage of activity. This was when a shipment of 50mCi Yttrium (^{90}Y) Silicate colloid - used for internal therapy - in a standard Type A container was crushed, the vial broken and activity spread.

DISCUSSION

The object of this review has been twofold - to summarize data on accidents that have occurred and to try to reduce the numbers of accidents occurring.

The very low incidence of leakage from damaged packages is a reflection of the standards of packaging that are employed for the transport of radioactive materials by TRC. Even the comparatively flimsy packaging used for the exempt packages has withstood most of the incidents. With the Type A packages, the containment system of inner vial, secondary container and metal can has proved to be perfectly satisfactory in most cases. In those instances where lead shielding pots have been used, further protection is provided by the lead (hardened by alloying with 4 per cent antimony). Tests with an 8 ton road roller (5) have revealed no apparent deformation of lead pots of 18mm wall thickness or greater and only slight deformation with thinner pots. Even in those cases where the vial was broken inside the lead pot, liquid was retained by use of a suitable securing tape.

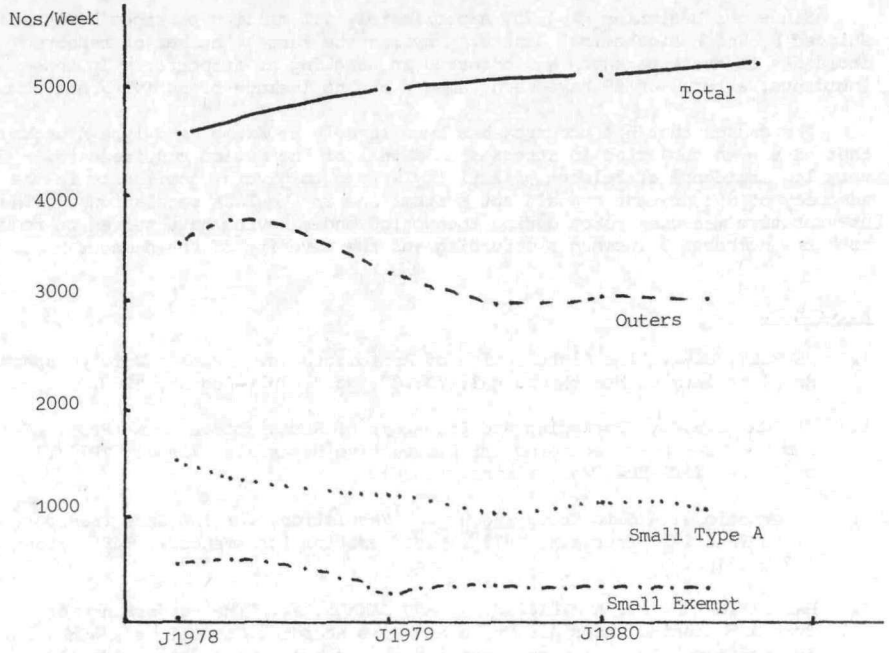
From an early stage, it became apparent that the majority of incidents occurred with the smaller Exempt and Type A packages (up to 500 cu.in.). These packages could easily fall off a pallet and be run over by the wheels of a fork lift truck. Means of reducing the number of smaller packages and of containing others was sought. A twofold approach was followed:-

(1) Reduction of number of small packages

The numbers of individual packages transported has been reduced by a wider use of multipacks (or overpacks). This had had the twofold effect of reducing the number of final packages transported - Table 4 - and at the same time made those that are being transported larger and less vulnerable to damage.

TABLE 4

Reduction in numbers of small packages despatched



(2) Closer liaison with Shippers

In contrast to the situation in the United States, (6), TRC has the advantage of being the only major shipper of radioactive materials out of the U.K. This advantage is strengthened by the fact that we are shipping principally through one major airport - Heathrow. As a consequence, we are able to maintain a close liaison with the major airlines and transport materials with the minimum of problems.

By maintaining close liaison with shippers, we have given them a greater awareness of the packaging and handling of radioactive materials and encouraged them to use, for example, cages rather than open pallets when transporting radioactive materials.

The overall effect of these actions has been a decrease in the number of incidents involving radioactive packages - Table 1. This decrease is at a time when there is a greater awareness of the transport of restricted articles and more stringent acceptance standards by airlines.

There has also been a change in type of damage reported from that involving a complete crushing of the package to that involving largely superficial damage to outer packaging.

CONCLUSIONS

Since the beginning of 1975, approximately 1.5 million packages have been shipped by The Radiochemical Centre. By far the largest number of reported incidents to these packages has occurred in handling at airports. In these incidents, a total of 89 have been damaged but no leakage of activity has occurred.

The damage that has occurred has been largely to Exempt and Type A packages that have been subjected to stresses in excess of the design requirements. The very low incidence of release of activity serves to give us confidence in the adequacy of our present containment systems and in the IAEA regulations. The preventative measures taken during the period under review have served to reduce both the numbers of incidents occurring and the severity of the damage.

References

1. NEWBERY, G.R., "The Distribution of Radioisotopes. Public Safety Aspects", Royal Society of Health Journal, Vol.80, No.4, July-August, 1960.

2. TAYLOR, C.B.G., "Packaging and Transport of Radioisotopes". (Proc. of Sem. on Transport Packaging for Radioactive Materials, Vienna, 1976), pp 71-89, IAEA-SR-10/45, Vienna, (1976).

3. International Atomic Energy Agency. Regulations for the Safe Transport of Radioactive Materials, 1973 revised edition (as amended), IAEA, Vienna, 1979, 171pp.

4. HADJIANTONIOU, A., ARMIRIOTIS, J. AND ZANNOS, A., "The Performance of Type A packaging under Air Crash and Fire Accident Conditions", 6th International Symposium on Packaging and Transportation of Radioactive Materials, Berlin, 1980.

5. COHEN, L.R., "Road Roller Test on 'W' Pots". Harwell Test Report No.140, April 1968.

6. GRELLA, A.W., "A Review of Five Years (1971-1975) Accident Experience in the USA involving Nuclear Transportation". (Proc. of Seminar on Transport Packaging for Radioactive Materials, Vienna, 1976), pp 225-240, IAEA-SR-10/5 Vienna (1976).

DISCUSSION

Question by T. Musialowicz, Poland: 1. How do you apply activity limits for type A packages in the case of overpacks? To a single package in the overpack or to a whole overpack? 2. In your table you show us the number of damaged packages according to the transport category. What is the category of the type B package in the last line?

Answer: Each individual pack is within type A limits. The overpack is limited so that the sum of individual A2 fractions is less than unity. In practice sums are mostly of the order of 10^{-3} x A2. 2. Category III.

Question by K. Kirchhoff, FR Germany: What experience have you had with damage to RAM packages during handling at transit and destination airports?

Answer: We have had no experience of damage at destination or transit airports. The most widely used transit airport is Frankfurt. At Frankfurt, either a direct plane to plane transfer is made or packages are handled in cases of the type described in my presentation.

Question by A. W. Grella, USA: In using overpacks, how do you limit the aggregate activity of the material contained within the overpack?

Answer: The answer on the limit on activity has already been given to Dr. Musialowicz. The transport index given to the overpack is the sum of the individual transport indexes. This is of course always higher than the measured transport index due to shielding of other packages.

THE PERFORMANCE OF TYPE A PACKAGING UNDER AIR CRASH AND FIRE ACCIDENT CONDITIONS

A.Hadjiantoniou, J.Armiriotis and A.Zannos
Health Physics Division
"Democritos" N.R.C.
Greek Atomic Energy Commission

Preface

The type A packaging provides a means for shipping intermediate quantities of radioactive materials, as compared to small quantities in excempt category and large quantities in type B packaging.

In this paper the performance of type A packaging is presented and discussed under an air crash and fire accident.
It is known that type A packaging should meet certain performance standards. However it is recognised that under accident conditions the packages are liable to be damaged. This fact is reflected in the activity limit of the content. The content limits are prescribed so that in the event of a release in a severe accident the consequent hazard is not unacceptable. These activity limits are tabulated for each individual radionuclide (1,5).
When a radionuclide is in "special form" that is in a state which would be virtually non dispersible after the release from its package, the activity limit A_1 applies which is greater than A_2.

The packages carried by the aircraft were either of Type A or in the Exempt category containing radioisotopes for medical, research and industrial applications (Tables 1,2).
All the Type A packages contained a sealed metal fruit can, inside a cardboard box. Inside the fruit can the radioactive material was in one or more small glass bottles, sometimes in a closely-fitting plastic container, sometimes in a small lead pot.
All the packages came from the Radiochemical Centre, Amersham, exept one of exempt category which came from C.I.S. France. The radionuclides, their form and activity , transported by the aircraft are presented in Table 1.

Accident Conditions

On October the 8th,1979, around 10 pm,a DC-8 passenger aircraft during its landing procedure at the Athens airport failed to stop at the end of the runway and crashed on a public road passing parallel to the airport fence and at a level 5 meters lower than the runway. The crash was followed by fire without explosion. The fire was spread quickly inside the aircraft.

As soon as the airport authorities were notified that the aircraft was carrying radioactive materials, they informed the Health Physics Division of the Greek Atomic Energy Commission according to the existing emergency plan. This call came about 5 hours after the accident. The first information concerning the radioisotopes on board (type,

form, activity) was very poor. Complete information was received during next day.

The Health Physics emergency group reached the accident scene in less than an hour from the moment of notification.
Although the fire was not completely put out, the group approached the aircraft and located the area where the radioisotopes were stored taking immediately the necessary measures. Using proper portable radiation detection instruments, the rescue workers, the policemen, the firemen e.t.c. were checked for contamination. No such contamination was detected.

Findings

As soon as the fire was completely extinguished the Health Physics group entered the cargo compartment and started the work of locating and collecting the packages of radioactive materials, the broken vials, the open fruitcan containers and also all other materials which either were part of the destroyed packaging or other contaminated material (fire trash e.t.c.).
A total of about 5 cubic meters of such material was collected in metalic barrels and transfered to Democritos N.R.C.

A detailed radiation survey was carried out inside and outside the aircraft. The exposure rate in that compartment did not exceed 1mr/h and the γ-ray spectroscopic analysis of the contaminated trash showed that this was mainly due to the presence of Na-22.
No detectable contamination was found outside the storage compartment.

From the type of the accident, the overall picture of the crashed aircraft and also from the fact that none of the fruitcan containers was found squeezed, it is concluded that the packages did not experience any external mechanical stress. On the contrary most of fruitcan containers were found in a swelling condition and/or ruptured violently. This was caused by the increased presure of the air inside the cans due to the rise of the ambient temperature (3).
For the same reason most of the vials were found broken or without their rubber seals. Observations on the external appearance of the fruitcans and on other materials in that compartment which were found partly burnt or unaffected and also on packages still containing dry ice, indicate that the fire did not have the same intensity and did not last for the same time everywhere.
None of the lead containers inside the fruitcans were found to be intact. In some cases the lead melted completely and it was traped inside the ruptured fruitcan container (Fig.1) and in other cases the lead container found itself outside the can where the lead melted and so the fruitcans were found empty. Only in one case the lead container was found partially melted (Fig.2).

All these observations and also the presence of melted Aluminum alloy lead to the conclusion that the maximum temperature at the hotest points of that compartment must have exceeded 640°C (melting point of the aluminum alloy) and the average temperature must have exceeded the melting point of lead and it lasted long enough as to cause the melting of the lead containers (3,4).

In the laboratory the main concern was to locate and recover the Pu-238 sources which were in the form of pellets with total activity of 3 Ci (100 sources, 30 mCi each, special form). There were 5 separate fruitcans packed together in a single overpack, each in its own small carton, containing 20 such sources inside a thin lead container. Each pellet consisted of plutonium adsorbed in ceramic material and was sealed by monel metal with a beryllium window from one side and with dimensions 11 mm in diameter and 5 mm thick (2).

The results of this effort were the following :

Three Pu fruitcans were found partly opened at one end and the Pu pellets were trapped inside the melted lead container.

The two last fruitcans were found with one end burst open. Their content was missing and only a few pellets were stuck on the lid.

The total number of pellets collected this way was 74. 18 more pellets were found in the trash either stuck inside the melted lead or as individual ones. The small number of missing pellets (about 8) is believed to be within the rest of the collected trash.

So far no attempt has been made to evaluate the damages to the pellets from the fire. Macroscopic observations show that a large number of the plutonium pellets have undergone extensive damage and also in some of them alpha transferable contamination was detected. Smear test from two pellets gave 420 c/min and 65 c/min which correspond approximately to 630 and 100 pCi of alpha activity.

Discussion

Fire is always an inevitable fact in any aircraft crash. This particular accident did not exhibit any unusual caracteristics, on the contrary because the crash was not very violent and occured to a place where it was easily accessible, almost in the airport, the various emergency teams reached the accident scene very quickly. The fire was soon under control' this prevented the explosion of the fuel tanks and limited the radiological consequences of the accident (possible dispersion of the damaged Pu pellets within a populated area).
From the above we can state that in this particular accident the radioactive packagings have not suffered any severe mechanical shock or squeeze and all the damage is attributed to fire.

In table 3 an attempt is made to classify the damages. From this table it is apparent that most of the packages were extensively or heavily damaged and their content remained within the cargo compartment only because of the fact that explosion did not take place.

Probably, in case of an accidental release, one would assume that a high degree of dispersion is an advantage from the radiation protection point of view. But still there remains the question what the situation would be if, as a result of an explosion, the 3 Ci of Pu contained in 100 damaged special form sources were dispersed within a large populated area with no possibility of recovery.

Afterall the question arises' should the transportation in type A packaging of high radiotoxicity isotopes like plutonium in this case, even in special form, be reconsidered taking into account the conditions encountered in real accidents ?

T A B L E 1

Packages carried by the aircraft

Number of Packages	Category	Label	Index	Radioisotopes and Activities			
3	III	Yellow	2.2	Pu-238	3 Ci	Special form (Disc. Sources)	
				Na-22	2 mCi	Injection	
				Pm-147	10 mCi	Solution	
				I-125	63.5 µCi)	Labeled Comp. and RIA kits	
				TRITIUM	37.76 mCi)		
				C-14	100 µCi)		
26	I	White	0	I-125	10 µCi)	Labeled Comp. and RIA kits	
				TRITIUM	33.4 mCi)		
				C-14	180 µCi)		
11	XMPT			I-125	220 µCi)	Labeled Comp. and RIA kits	
				TRITIUM	322 µCi)		
				C-14	1.9 µCi	Liquid Scint.Std	

T A B L E 2

Radioisotopes and activities carried by the aircraft

Total Activities	
Pu-238	3 Ci
Na-22	2 mCi
Pm-147	10 mCi
TRITIUM	71.5 mCi
I-125	293.5 µCi
C-14	281.9 µCi

T A B L E 3

Fruitcan damage classification

(approximate estimation)

Damage	% of the total number
Fruitcans without any loss of their content	40
Fruitcans with partial loss of their content	14
Fruitcans with total loss of their content	46

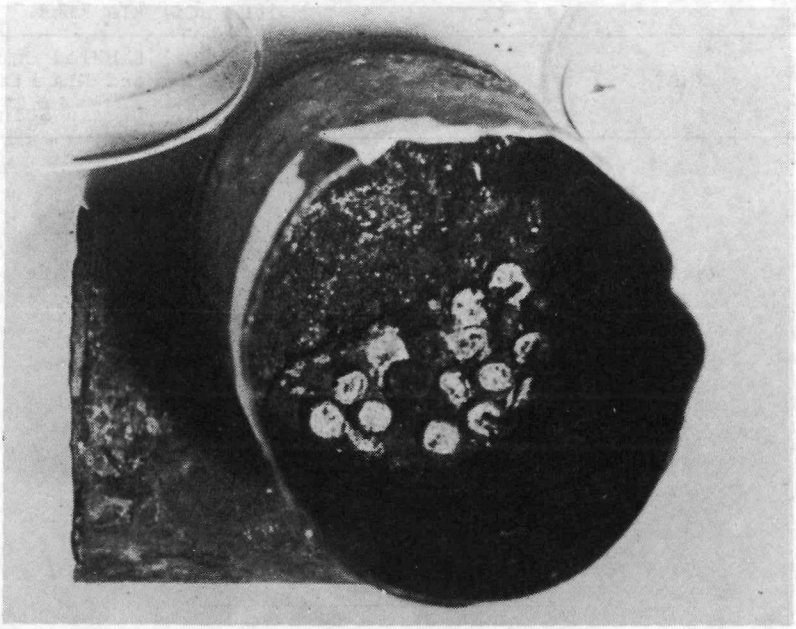


Fig.1. Fruitcan with Pu pellets traped in the melted lead container.

Fig.2. Na-22 inside its partially melted lead container.

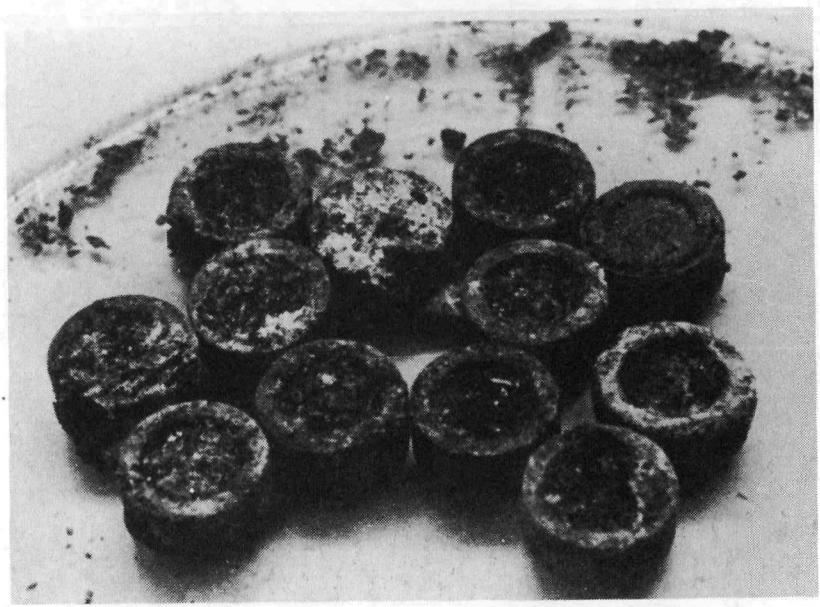


Fig.3. A group of Pu sources affected by the fire.

REFERENCES

(1) Regulation for the Safe Transport of Radioactive Materials IAEA, 1979, S.S. No 6.

(2) TAYLOR,C.B.G. Radiochemical Centre, Amersham, Personal Communication,1980.

(3) MORIYA,I. Studies on fire resistivities of radioisotope containers. Proc.Int. Symp. for Packaging and Transportation Radioactive Materials. Albuquerque, New Mexico, 1965.

(4) Examination of the IAEA Regulations for the Safe Transport of Radioactive Materials. EUR.3485 e,f VIII, 1968.

(5) ASPINALL K.J. & FAIRBAIRN,A. The Classification of Radionuclides for Transport Purposes. AHSB(RP) R23, 1963.-

ON THE DOMESTIC TRANSPORTATION BY "HINOURA MARU" THE SHIP FOR THE EXCLUSIVE USE FOR SPENT NUCLEAR FUEL

KEISUKE KAGE
KANEOMI KUWASHIMA
HIROMI KAJISHIMA
NUCLEAR TRANSPORT SERVICES CO., LTD.

1. INTRODUCTION

There are totally 21 nuclear reactors in operation in Japan. Spent fuel discharged from 8 reactors (total electric generation capacity: 4,200 MW) among them is transported by "Hinoura-Maru", the ship exclusively used for transportation of spent fuel, to the reprocessing plant of the Power Reactor and Nuclear Fuel Development Corporation located at Tokai Village in Ibaraki Prefecture. Since the reprocessing plant does not have it's own port, spent fuel is to be unloaded at the port belonging to the Tokai Power Station of the Japan Atomic Power Company, 4 kilometers apart from the reprocessing plant, and then transported to the plant by trailers. The reprocessing plant is now in trial operation with design capacity of 0.7 tons of uranium per day and 210 tons/year. The aforementioned 8 reactors belong to 6 different power stations, each of which is operated by respective electric power companies. 5 nuclear power stations out of these 6 have their own port for shipping spent fuel within the station site, and the rest uses the general-purpose port 12 kilometers apart from the station.

2. GEOGRAPHICAL EXPLANATION OF JAPAN'S MARINE TRANSPORTATION

1) Locations of Nuclear Facilities

Locations of Japan's nuclear power stations, the Tokai port, (the port of delivery) and the reprocessing plant are located as shown in Fig. 1.

Spent fuel coming from Mihama and Shimane power stations is transported via Tsugaru straits and that from Genkai power station via Ohsumi Peninsula.

2) Distances and Required Hours of Navigation

Distances and Required Hours of Navigation from the each power station to the Tokai port are shown in Table 1. The longest navigation is from Genkai power station and it takes about 86 hours. The shortest is from Fukushima power station and it takes 7 hours.

3) Distribution of Water Depths

The distribution of water depths on navigation route is summerized in Table 2. Accumulatively, 54% of the navigation route has the water depths less than

200 meter, and 75% less than 500 meters. The maximum depth is 2,700 meters.

4) Meteorological Features

On the Japan Sea side generally strongly northwest wind blows in winter, so transportations from the 3 power stations facing on the Japan Sea are seldom executed in this season. In addition, the ship navigation is sometimes interfered by typhoon which attacks the Japan Islands or near places between August and October.

3. OUTLINE OF "HINOURA MARU"

As presented in detail in the last symposium in Las Vegas, "Hinoura Maru" has the double hull structure to ensure the collision-resistance, and horizontal beams are provided in order to absorb the collision energy in event of being crashed on the sides as illustrated in Fig. 2. The ship is also provided with the front bulkhead of the hold as a countermeasure for collision and the double bottom structure for stranding-resistance.

The ship has the hold cooling equipment to keep the cask surface temperature below 82°C and the hold inside temperature below 38°C, as well as the cooling devices to cool down the water cooling type cask. The devices pouring sea water into the holds are also equipped in order to cope with hold fire or another accidents due to abnormal increase of temperature.

In order to keep radiation exposure to the crew below 0.18 mrem/h, the shield water tanks are placed around the holds and the cask covers and hutch covers on the top of the holds.

The main specifications of "Hinoura Maru" are shown in Table 3.

4. TRANSPORTATION CASKS

Casks used for light water reactor fuel in Japan are Excellox-3A, HZ-75T and NH-25. Their main specifications are listed in Table 4. As to the cask HZ-75T made in Japan, the capacities for PWR assemblies and for BWR assemblies are 7 and 17 respectively, and its transportation weight reaches about 90 tons.

5. OUTLINE OF TRANSPORTATION PROCESS

The transportation process of casks charged with spent fuel from a nuclear power station to the reprocessing plant are as follows;

(1) Prior to shipping out from a power station, the casks undergo the air leak tests and the measurements of radiation dose rate, surface contamination density and surface temperature by the inspector of the Government (The Science and Technology Agency) to be certified that they do not exceed the regulatory limits.

(2) The casks are transported by trailers to the pier.

(3) Cask radiation dose rate, surface contamination density and surface temperature are again measured by the inspector of Ministry of Transport, Ship Bureau or its agency at the pier.

(4) After loading onto the ship hold by crane, the stowage inspection such as fixing of casks, radiation dose rate in accommodation areas are conducted by the same inspector as in the 3rd step.

(5) During the navigation to the Tokai port, surveillance of radiation dose rates of the casks and the accommodation areas, and temperatures of the casks and the holds are made by the radiation supervisor on board.
(6) Arriving at the Tokai port, the casks are unloaded and stowed on trailers, after being measured their surface contamination densities on board. In this step, the cask fixing check and the radiation measurement concerning the vehicles are conducted by the inspector of the Road Transport Bureau of Ministry of Transport.
(7) Then they are transported by road to the reprocessing plant.

6. OPERATION MANUAL OF "HINOURA MARU"

More consideration has naturally been given to the safe operation of the ship than in the case of ships for general use. "Manual for Dangerous Goods Handling" has been set up in accordance with "Regulations for Carriage and Storage of Dangerous Goods in Ships", stipulated by the Government.

Some specific features of this Manual are as follows:

(1) In order to ensure, the safe operation of the ship and to prevent dangers and accidents, an "Operation Supervisor" has an exclusive responsibility for handling the operational problems in constant communication with the ship.
(2) During handling and transporting the spent fuel casks, a "Radiation Supervisor" is on board to conduct constant surveillance on radiation, and to give the captain the necessary advices concerning radiation control and accident prevention.
(3) At the entry into and departure from a port, a harbor pilot and tug boat are employed in principle.
(4) Entry into and departure from a port by night is not permitted.
(5) The ship must harbor in case of wind speed more than 23 m/s or wave height more than 4 meters.

7. THE EXPERIENCE OF DOMESTIC SEA TRANSPORTATION

1) Amounts of Spent Fuel Transported

Since her first domestic transportation of spent fuel from Fukushima No. 1 Nuclear Power Station to the Tokai port in January 1978, Hinoura Maru has carried 50 casks containing 121 tons of uranium in total by 23 navigation services from 6 power stations until the middle of October of this year. Transported quantities year by year and cask by cask are summerized in Table 5.

Remaining quantities of fuel scheduled to be transported within this year are 11 tons, and about 100 tons of fuel are expected to be transported annually from the next year.

2) Radiation Data

Radiation Dose Rates of Casks

Table 6 shows the maxima radiation dose rates observed until now for 3 types of casks used for the transportation of BWR and PWR fuel. This indicates that radiation levels were much below the IAEA limitations. The spent fuel conditions such as burn-up and cooling times were not so severe as far as fuel handled until

now concerned.

Surface Contamination Densities of Casks

The casks HZ-75T are ensured not to contact with spent fuel pool water by a protective bag while charging fuel into them, so that their surface contamination densities have scarcely been detectable until now.

Space Radiation Dose Rates

Space radiation dose rates in the accommodation areas of the ship were mostly within the range from 0.001 to 0.003 mrem/h, satisfying the national limitation of 0.18 mrem/h.

8. SAFETY ASSESSMENT OF SEA TRANSPORTATION

According to the report on safety assessment of the sea transportation conducted by the Advisory Committee on Nuclear Fuel Safety of Japan Atomic Energy Commission;

(1) Submersion probability of the ship is no more than 7.6×10^{-7} a year.
(2) With present-day Japanese salvage technique, it is possible to do salvage work at a dept of up to 200 meters.
(3) Although the deformation of the casks used in Japan would become conspicuous at a depth of 2,300 meters, no serious deformation enough to influence the fuel body and lead to leakage of radioactive materials would not produced. Fuel rod could withstand depth of about 5,000 meters.
(4) Local corrosion might be produced between the cask lid and the flange surface of the cask, and the tightness of that part might be lost in about 20 years.
(5) Consequently, cask would not deform so seriously as to affect fuel at the maximum water depth of 2,700 meters on route of the ship.
(6) In order to assess individual exposure dose to the general public, however, two cases of leakage model have been set up.

The case I is that a cask is submerged to the 200 meters water depth and radioactive materials are leached from fuel twenty years later.

The case II is that radioactive materials are leaked from the whole fuel immediately after submerged at the 2,000 meters water depth.

For both cases it is assumed that radioactive inventory inside each cask is 2.1×10^{7} Ci and numbers of casks are four.

The resulting individual whole body dose are estimated to be 12 mrem/y and 6 mrem/y respectively for the case I and the case II.

Therefore it would be concluded from the above results that sea transportation of spent fuel would be fully safe.

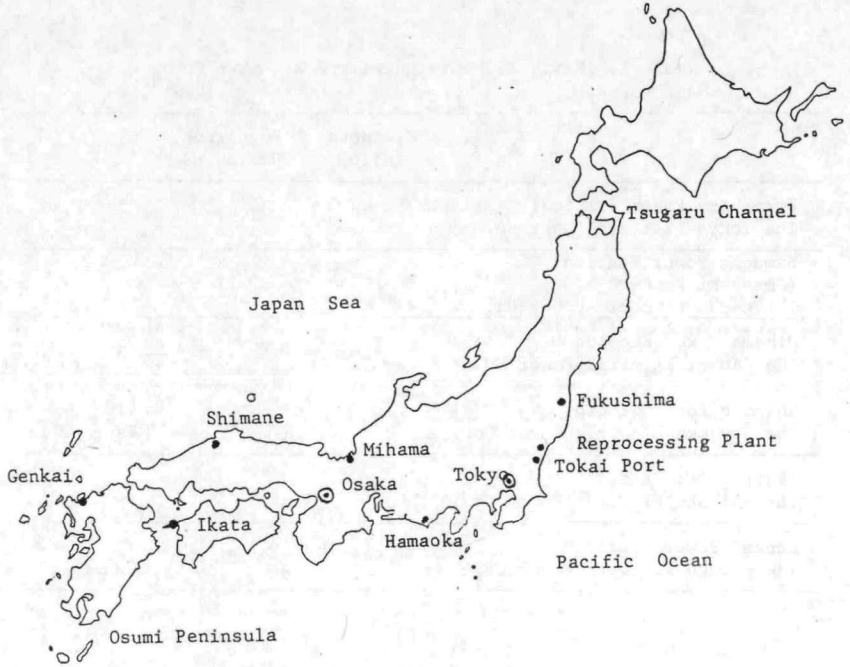


Fig. 1 Locations of Nuclear Facilities

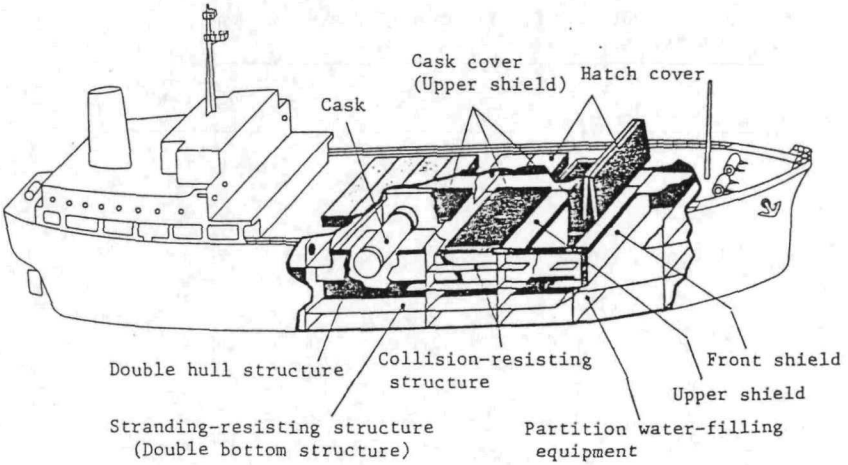


Fig. 2 Structure of "HINOURA MARU"

Table 1. Navigation Distances and Required Hours

	Distance (mile)	Required Hours (H)	Route
Fukushima Power Station The Tokyo Electric Power Co.	70	7	
Hamaoka Power Station (Omaezaki Port) The Chubu Electric Power Co.	230	21	
Mihama Power Station The Kansai Electric Power Co.	820	75	Via Tsugaru
Shimane Power Station The Chugoku Electric Power Co.	930	85	Via Tsugaru
Ikata Power Station The Shikoku Electric Power Co.	590	54	
Genkai Power Station The Kyushu Electric Power Co.	940	86	Via Osumi

Table 2. Distribution of Sea Water Depth

depth (m) rate	%	accumulative %
0 - 100	22.1	22.1
100 - 200	31.9	54.0
200 - 500	21.2	75.0
500 - 1000	14.5	89.7
1000 - 1500	8.8	98.5
1500 - 2000	0.5	99.0
2000 <	1.0	100.0

Table 3. Main Specifications of "Hinoura-Maru"

Gross Tonnage	1,290 Tons
Length Overall	78.25 M
Depth (to upper deck)	5.85 M
Full Load Service Speed	abt 11 Knots
Main Engine	2,500 P.S.
Loaded Casks	4 Casks

Table 4. Outline of Casks Used in Japan

	Excellox-3A	HZ-75T	NH-25*1
Dimension			
Length (m)	5.8	5.9	5.8
Diameter (m)	2.2	2.3	1.5
Weight (t)*2	71	91	33
Loading Capacity			
BWR No. of Assemblies	12	17	2
Weight (MTU)	2.4	3.3	0.4
PWR No. of Assemblies	5	7	1
Weight (MTU)	2.3	3.2	0.4
Cooling	Natural	Natural, Forced Air or Forced Water	Natural

*1 The Cask for the Spent Fuel Testing Laboratory
*2 Including Fuel and Skid

Table 5. Amounts of Fuel Transported, 1978-1980

	HZ-75T		Excellox-3A		NH-25		Total	
	No. of Casks	Fuel (MTU)	No. of Casks	Fuel (MTU)	No. of Casks	Fuel (MTU)	No. of Casks	Fuel (MTU)
1978	—	—	14	30.1	—	—	14	30.1
1979	10	28.0	4	9.2	2	0.8	16	38.0
1980	16	47.4	2	4.6	2	0.9	20	52.9
Total	26	75.4	20	43.9	4	1.7	50	121.0

Table 6. Radiation Dose Rate of Casks

		Case 1	Case 2	Case 3	Case 4	Case 5
Fuel						
Type		BWR	PWR	BWR	PWR	BWR
Weight	(MTU)	2.4	2.0	3.3	2.8	0.4
Burn up	($\frac{MWD}{MTU}$)	9,451	19,300	15,100	27,400	5,972
Cooling Time	(Day)	1,722	1,145	965	576	450
Radioactivity	(Ci)	6.85×10^5	1.5×10^6	1.6×10^6	4.7×10^6	2.5×10^5
Cask Model		Excellox -3A	Excellox -3A	HZ-75T	HZ-75T	NH-25
Radiation Dose Rate on Cask Surface						
on Surface	(mrem/h)	1.4-0.08	4.6-0.2	1.49-0.09	1.00-0.12	2.66-0.48
at 1m from Surface	(mrem/h)	0.12-0.03	1.25-0.08	0.57-0.07	0.75-0.09	0.76-0.11

DISCUSSION

Question by S. F. Johnson, Sweden: 1. In table 1 of your compact you show two different capacities for the casks. Both are for BWR assemblies. How do you explain this? Is this a misprint? 2. What are the total weights of the casks?

Answer: 1. This is a misprint in the compact, page 369. Please correct the second BWR to PWR in table 1. 2. The total weights of the casks are as follows: Excellox 3A - 71 ton, HZ-75T - 91 ton, H-25 - 33 ton.

Question by P. Blum, France: I would like to know if peak or mean G loads experienced by casks during actual sea transport have been recorded?

Answer: We have never measured or recorded G loads received by casks during actual sea transportation.

Question by E. G. Tomachevsky, France: What preparatory measures are taken to recover casks in case of sinking?

Answer: In order to minimise the probability of accident at sea, the captain obtains a confirmation of transportation method from the Ministry of Transport and instructions from the Regional Maritime Safety Headquarters for safety navigation. He also consults with the Operational Supervisor who has exclusive responsibility for the ship at head office. On recovering the casks submerged, a salvage company reported to us that they had a salvage technique which could be carried out at a depth up to 200 m using floating cranes. For the behaviour of casks not to be salvaged, please refer to article 7 on page 368 of the Compacts.

Comment by S. Aoki, Japan, to question by Mr. Tomachevsky: When an accident occurs at sea and if the casks sink to the bottom, salvage is generally possible if the depth is less than 200 m, not possible if the depth is greater.

However, we have already carried out a risk assessment for the case of "not less than 200 m" and recognised that the effect of all F.P. release is negligibly low. The pressure resistance test is required for an external pressure under 500 bar which corresponds to the pressure at the bottom of the sea at a depth of 5000 m. This test result confirms the integrity of the cask at the bottom of the sea.

In addition, the design criterion for sea-going vessels carrying spent fuel has been developed, in order to make the probability of collision and submerging accidents as small as possible.

FABRICATION EXPERIENCE OF HZ-75T SPENT FUEL SHIPPING CASK

Nagao Niomura, Akira Onodera
Akira Nishikawa, Ryoji Asano
Masayoshi Yamashita, Susumu Ozaki
Yoichi Hara
Hitachi Shipbuilding and Engineering Co., Ltd.

SUMMARY

Four units of the HZ-75T Spent Fuel Shipping Cask have been fabricated to the design which was presented at the last Symposium and some of the experiences obtained in the course of fabrication are described: (1) shrinkage of lead bonded stainless steel cylinder was quantitatively obtained by numerical analysis and a mock-up test, (2) gamma probe test was applied to all over the lead shield to detect defects, (3) heat dissipation test was performed with electric heaters in the cask cavity as a part of the performance test and temperature distribution in and around the cask was obtained and evaluated. The HZ-75T Spent Fuel Shipping Casks are in operation for transporting LWR spent fuel in Japan.

SHRINKAGE OF LEAD BONDED STAINLESS STEEL CYLINDERS

The design of the HZ-75T Spent Fuel Shipping Cask specifies that both inner and outer cylinders of lead shield shall be lead bonded to meet the thermal conduction requirements through lead and cylinder walls. It was foreseen that the lead bonding process would cause considerable amount of shrinkage in axial direction, as the bonding was to be applied in axial direction and the part being bonded would be locally heated and plastic compression would occur therein. To obtain the amount of shrinkage quantitatively, a numerical analysis was applied using finite element method, which was to be proved by a mock-up test.

The numerical analysis was carried out on the assumption that lead is bonded to Type 304 Stainless Steel. Bonding is applied in axial direction as shown in Fig. 1 in the order of E-1, E-2 and so forth.

First the part E-1 of the cylinder is heated to approximately 320°C while the cylinder remains at room temperature. E-2 is then heated while E-1 has cooled down to approximately 170°C. E-3 is then heated while E-1 and E-2 are approximately 70°C and 170°C respectively. The temperature transient at each bonding step is shown in Fig. 2. From Fig. 2 and the material properties of Type 304 Stainless Steel, a hysteresis curve is obtained as shown in Fig. 3.

Compression yielding occurs by local heating. However, compression stress decreases as the temperature rises, because the yield stress decreases as the temperature rises. After the temperature has reached 320°C (Point A), E-1 cools down and the stress changes from compression to tension, maximum tensile stress

is observed around 70°C (Point B). Subsequently, E-2 and other elements generate residual tensile stress, then change to compression stress as the cylinder undergoes shrinkage, and when bonding is completed (Point C), compression yielding is about to be reached. The element being reheated, plastic compression (strain) is generated.

For the initial heating, compression starts at zero stress, while for the second heating, compression starts at near-yield compression. Therefore plastic compression strain becomes greater than that from initial heating and accumulative strain increases as shown in Fig. 4.

The calculated shrinkage thus obtained was approximately 0.6 per cent of the total length, that is, 24 mm for the length of 3,939 mm.

A mock-up test was performed using half length cylinder and the shrinkage obtained was approximately 0.5 per cent, which was in good agreement with that obtained by the analysis. Thus the numerical analysis was verified and applied to the fabrication. The cylinders of the HZ-75T Cask were measured of their length after bonding and the rate of shrinkage was approximately 0.6 per cent.

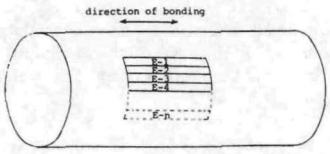


Fig. 1 Lead bonding applied to lead shield cylinders

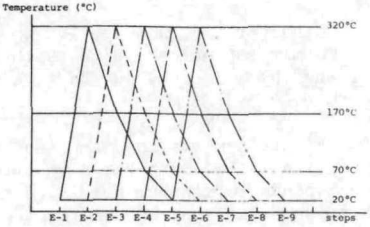


Fig. 2 Temperature transient of lead bonded elements

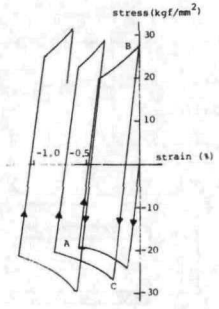


Fig. 3 Stress-strain hysteresis curve of lead bonded stainless steel (E-1)

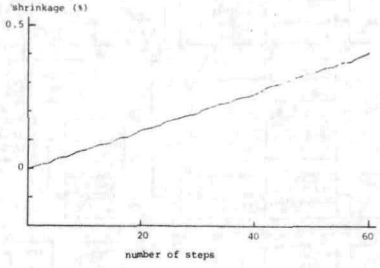


Fig. 4 Process of shrinkage of lead bonded stainless steel

GAMMA PROBE TEST FOR LEAD SHIELDING

The gamma probe test is one of the unique inspection items which was applied to the lead shielding after lead pour. For this test a special device was developed to provide full remote controlled probing and high detectability of defects and/or reduction in shield thickness.

A machined probing yoke and turning rollers were driven by pulse controlled motors to keep geometric position of scanning mechanism at any points. About 10 curies of Cobolt 60 source was used as radiation source. The detector consists of scintillation type NaI (Tl) crystal, photo-multiplier tube and pre-amplifier. For counting and data recording, single channel and multichannel pulse height analyzer, count rate meter and chart recorder were used. To collimate scattered radiation, lead collimaters were provided on both radiation source and detector. The layout of testing equipment and instrumentation are shown in Figure 5.

A standard mock-up with specified minimum lead thickness was used for prior test calibration of maximum acceptable counting rates. The photo-peaks of Cobolt-60, 1.17 and 1.33 MeV were adjusted by discriminating window width and monitoring check of peak drift was periodically carried out by means of multi-channel pulse height analyzer.

Gamma probing was applied to all over the lead shield by moving the set of the Cobolt 60 source and the scintillation counter in axial direction and then turning the cask body so that the scintillation counter can cover the whole surface.

Figure 6 shows an example of the result of gamma probing in an axial direction. The count rates measured were below the preset acceptance level which corresponds to minimum specified thickness of lead for the HZ-75T Spent Fuel Shipping Cask, i.e. 180 mm for the cylindrical part.

With the development of the special devise and testing procedures, the full remote controlled gamma probe test proved practicable with the detectability of 0.5 per cent of reduction in lead thickness, which correspond 1 mm of defects and/or reduction in lead thickness.

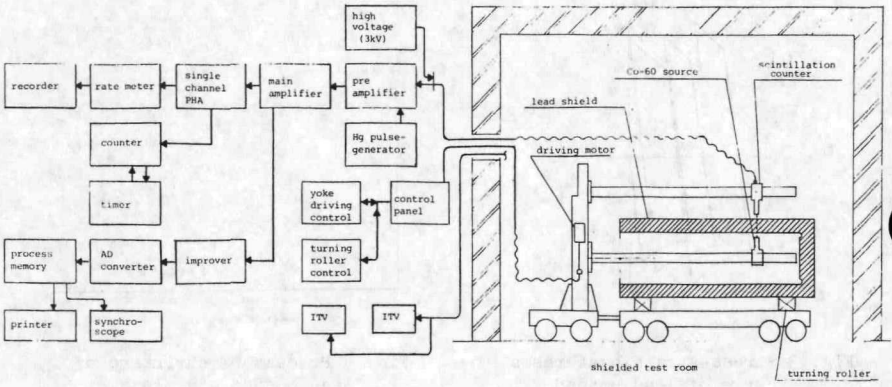


Fig. 5 Layout of gamma probe test and instrumentation

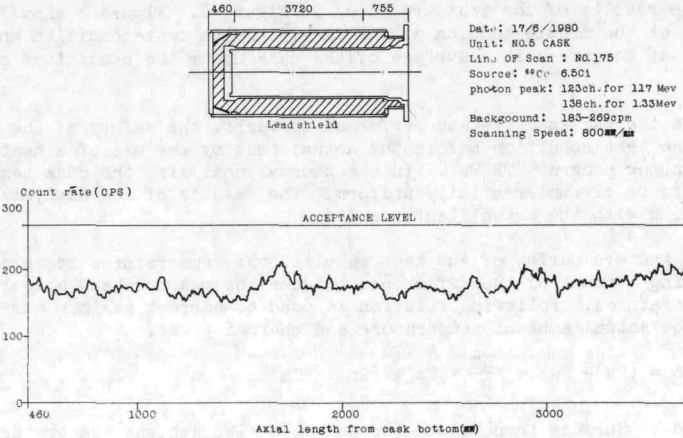


Figure 6 An example of the result of the Gamma Probe Test

HEAT DISSIPATION TEST

Upon completion of fabrication, a series of heat dissipation test was performed for evaluation of the accessible surface temperature and of the heat dissipation performance.

The Cask is designed to be forced cooled over 40 KW by either air or water and it was assumed in the analysis that forced cooling devise becomes inoperative under test conditions for Type A Package at specified maximum heat load, that is, 84 KW.

Therefore, tests were performed under the following conditions and sequence;

Case 1 : 84 KW and natural air cooling
Case 2 : 84 KW and forced air cooling
Case 3 : 40 KW and natural air cooling
Case 4 : 84 KW and forced water cooling

The arrangement of the test equipment is shown in Figure 7. The cask was mounted on the skid and placed on the floor of the factory.

Electric heaters simulating irradiated fuel assemblies were inserted into the cask cavity and temperature measured at the external surface, cavity water, fin tip and ambient air by thermocouples and contact pyrometer.

The thermocouples were fixed at the points shown in Figure 7 and were connected to automatic recorders. Contact pyrometers were used to measure the surface temperature of the cask.

The results of the test are shown in Figure 8. Figure 8 shows the temperatures at the middle section of the cask for each test condition and the temperatures at the accessible surface of the cask under the conditions of Cases 1 through 4.

The thermal analysis was performed to verify the safety of the cask and to predict the test condition before the actual test by the use of a heat transfer computer program "TRUMP". In the thermal analysis, the cask temperature was assumed to be circumferentially uniform. The results of the analysis are shown in Fig. 8 with the actual test results.

In the evaluation of the test results, the temperatures measured were modified taking account of the difference between the measured and stipulated ambient temperatures. Following relation is used to correct maximum surface temperature on actual ambient temperature and thermal power.

$$Tso = (Ts - Ta) \times \frac{QN}{Q} + Ta + (Tao - Ta)$$

where
- Tso : Surface temperature for the specified ambient temperature in the regulation (°C)
- Tao : Specified ambient temperature in the regulation, i.e. 38°C
- Ts : Measured surface temperature (°C)
- Ta : Measured atmospheric temperature (°C)
- QN : Design thermal power (KW)
- Q : Net thermal power (KW)

Then, the temperatures modified were evaluated. It was found that the accessible surface temperature would not exceed 82°C under the ambient temperature of 38°C.

Then, the temperature modified were compared with the analytical results and evaluated.

It was found that although there was a considerable difference in temperatures at different parts of the surface, the accessible surface temperature would not exceed 82°C under the ambient temperature of 38°C.

The temperature gradient at each part of cask was evaluated in comparison with those measured and calculated from the view point of heat dissipation performance.

It was concluded from these evaluations that the heat dissipation ability of the cask between cavity water and outer shell was almost equal to the value expected from the analysis and that of fin on the outer shell exceeded the analytical value. The heat dissipation by the test is about 10 to 25 percent greater than the expected value as a whole.

It was confirmed that the cask had greater heat dissipation ability than design value and that the thermal analysis method gave conservative and adequate results.

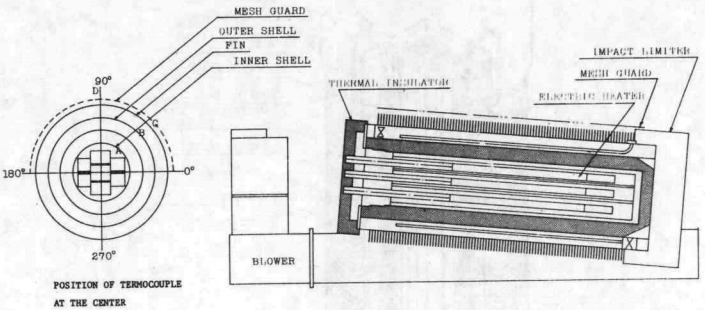


Figure 7 Arrangement of Heat Dissipation Test

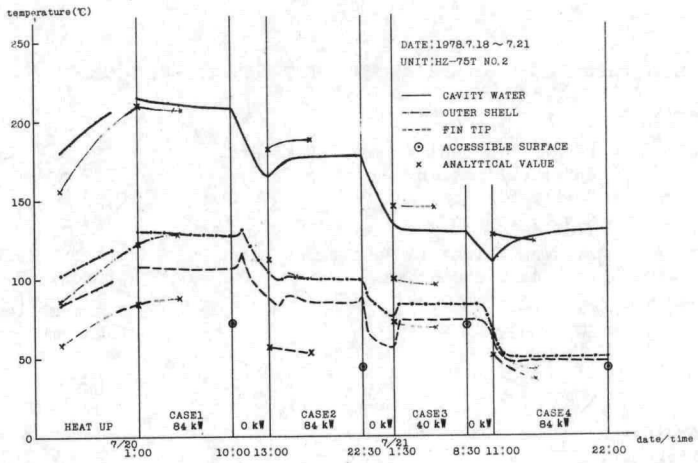


Figure 8 An example of the result of Heat Dissipation Test

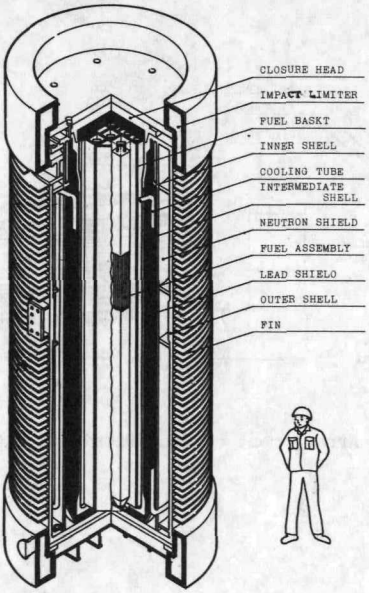


Figure 9 HZ-75T SPENT FUEL SHIPPING CASK

Main Particulars of the HZ-75T SPENT FUEL SHIPPING CASK

1.	Classification		Type B (M)
2.	Dimension	Length (including Impact Limiter)	5.91 m
		Outside Diameter	2.27 m
		Cavity Diameter	0.95 m
		Cavity Length	4.54 m
3.	Weight	Lifting Weight in Reactor Building	approx. 75 ton
		Cask on Transporter	approx. 80 ton
4.	Capacity	PWR Fuel	7 assemblies
		BWR Fuel	17 assemblies

5. Design Parameter of Fuel

	PWR	BWR
Fuel	UO_2	UO_2
Cladding	Zircaloy	Zircaloy
Enrichment, U235	3.4 %	2.8 %
Average Burn-up (max.)	33,000 MWD/MTU	27,500 MWD/MTU
U Weight per Assembly	450 kg	195 kg
Cooling Time (min.)	120 days	120 days
Assembly Length	4,203 mm	4,470 mm

DISCUSSION

Question by T. Musialowicz, Poland: What is the activity of the gamma source Co-60 used for testing?

Answer: About 10 curies.

Question by J. Kerjean, France: For the test conditions described in Fig. 6, is natural cooling consistent with the regulations, i.e. ambient 38°C, external side 82°C?

Answer: For the test conditions described in Fig. 6, the accessible surface temperature of the HZ-75t cask was evaluated to be below 82°C for natural cooling under the ambient temperature of 38°C.

CONTAMINATION STUDIES ON POND-LOADED FLASKS

W.D. CURREN and R.D. BOND
UKAEA, ATOMIC ENERGY ESTABLISHMENT WINFRITH

1. INTRODUCTION

It has been known for some years that flasks which are handled in fuel ponds absorb contamination which is subsequently released, especially if the flask is exposed to humid conditions or to rain. This problem has been encountered at Winfrith with a 50-ton flask which is loaded with spent fuel from the SGHW Reactor and is shipped by rail to Windscale for reprocessing. The IAEA regulations demand that, before shipment, the flask surface should be cleaned until the contamination detectable by smear-testing is less than 1 derived working level, defined as:

1 d.w.l. = 100 pCi/cm^2, averaged over 300 cm^2

It has been noted on many occasions that the level of surface contamination, detected by smear, increases during a journey - sometimes to above 1 d.w.l. on arrival at the destination. To avoid any possible release of contamination during transit, a metal cover has been fitted to the rail wagon. This protects the flask from rain, and prevents access to the flask by unauthorised staff. The flask, with its cover, is shown in fig. 1

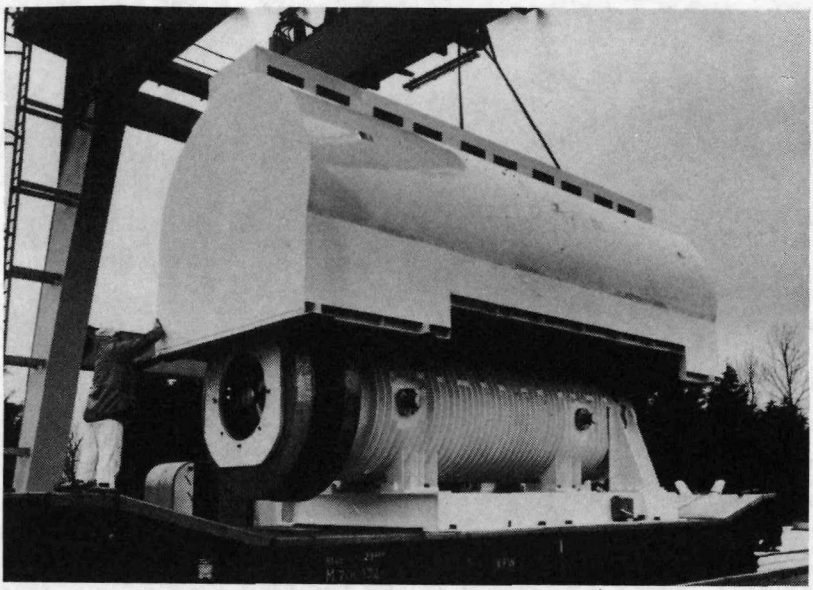


fig. 1

An undertaking was given to the Competent Authority that no relaxation in surface contamination levels would be permitted, and the flask is cleaned to below 1 d.w.l. before each shipment.

The flask cover effectively solves the operational problem by protecting the public from contact with a possibly contaminated flask. Nevertheless, it is not possible to arrange for all flasks to be covered in transit, and it was felt desirable to study the phenomenon of growth of smearable contamination with time, which has become known colloquially as "flask sweating".

The Winfrith flask is painted according to CEGB System 6, which is formulated to provide

minimum absorption of contamination
maximum ease of decontamination
maximum resistance to abrasion, weathering and ultra-violet light

The paint is applied in 4 coats, 2 of epoxy-based resin, followed by 2 coats of polyurethane. Each paint contains 15% by volume of titanium dioxide pigment. The total paint thickness is 250 um (ref 1).

The ponds in which the Winfrith flask is handled contain radiocaesium in solution in the concentration range 0.1 to 1.0 uCi/l. Chemically, the ponds have low conductivity and a neutral pH.

When the study began, the Winfrith flask had been coated with several layers of paint during an operational life of about 5 years. Samples of paint were removed in order to assess the total activity burden of the paint, which was found to be 1.2 mCi of radiocaesium. Dividing this figure by the surface area of the flask gave a result of 2000 pCi/cm^2, which is capable potentially of releasing 20 d.w.l.'s. At the outset, therefore, flask sweating was seen to depend on absorption of radioactive species from the fuel pond into the flask coating, with subsequent release of some of this material to provide smearable contamination on the surface.

2. EXPERIMENTAL PROGRAMME

A programme was set up at Winfrith with the following objectives:

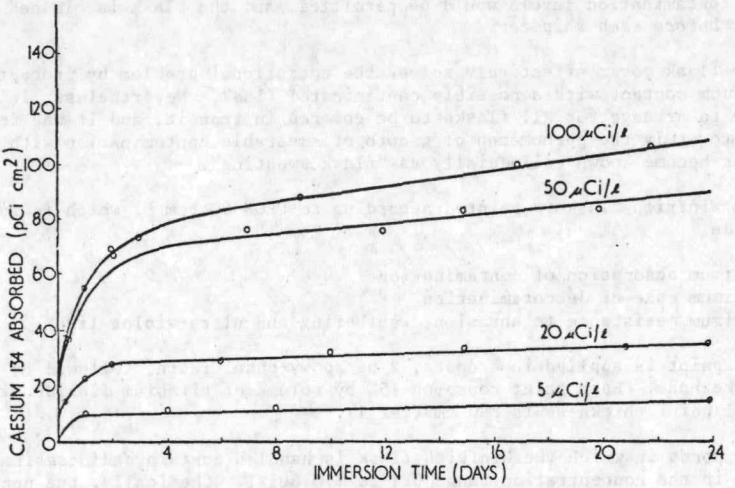
1. To quantify the absorption of radiocaesium by paints, and to investigate methods to limit this absorption.

2. To demonstrate the release of radiocaesium from paints, and to investigate accelerated decontamination.

Metal coupons 50 mm square, coated with the paint system under test, are used for laboratory experiments. Preliminary tests with samples of water taken from the Winfrith pond showed that the activity absorbed was too low for accurate counting. Therefore, samples of simulated pond waters have been prepared, containing ^{134}Cs in the concentration range 5 - 100 uCi/l, for this work.

2.1 Absorption of Radiocaesium

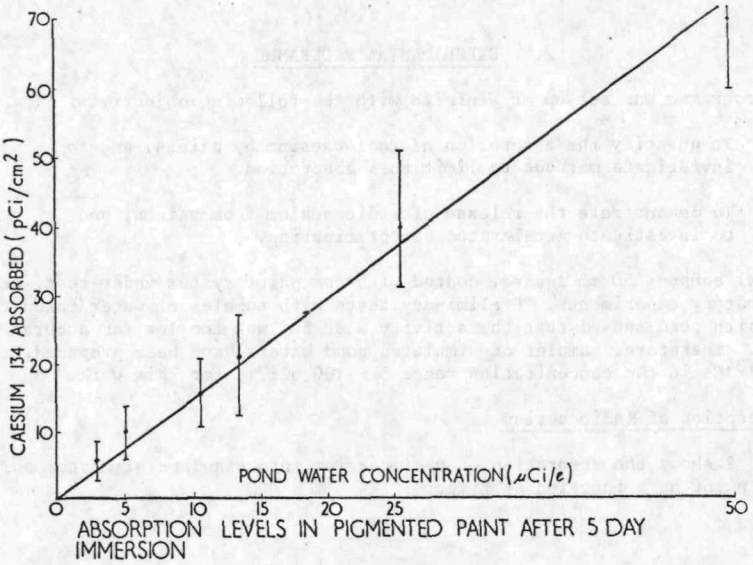
Fig. 2 shows the absorption of radiocaesium into standard pigmented polyurethane paint as a function of time.



ABSORPTION OF RADIOCAESIUM FROM SIMULATED POND WATERS.

fig. 2

The curves show, for each pond water concentration, a rapid initial uptake followed by a slower uptake which increases linearly with time. Once the linear part of the curve has been reached, the total caesium absorbed after a defined number of days of immersion is proportional to the pond concentration. This is illustrated in fig. 3.

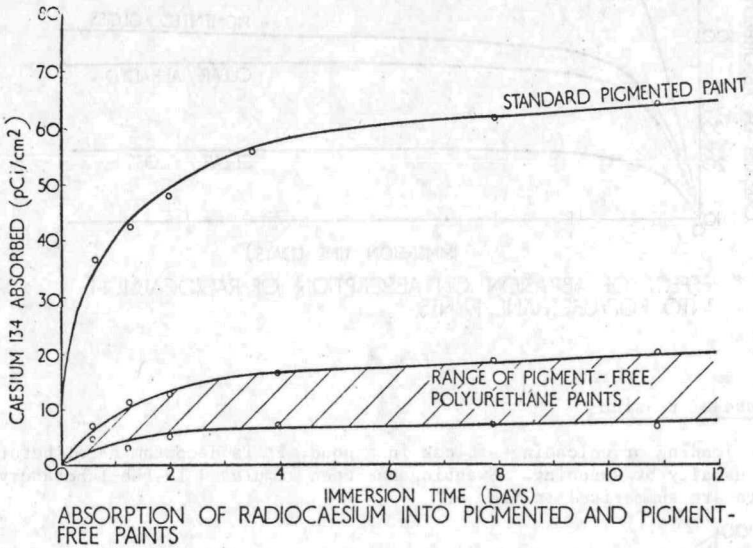


ABSORPTION LEVELS IN PIGMENTED PAINT AFTER 5 DAY IMMERSION

fig. 3

The samples used in this experiment were removed from the pond-water samples for counting after 5 days' immersion.

When paints were first formulated for pond-loaded flasks, it was realised that absorption of radioactive species from ponds would be favoured by presence of pigment in the paint. This has been confirmed by experiment, and is illustrated in fig. 4, which shows that polyurethane containing 15% volume loading of TiO_2 absorbs about 5 times as much radiocaesium as do several pigment-free polyurethanes.

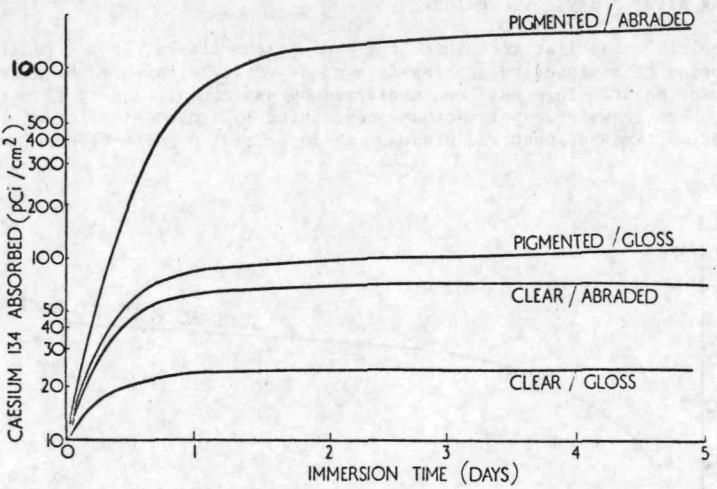


ABSORPTION OF RADIOCAESIUM INTO PIGMENTED AND PIGMENT-FREE PAINTS

fig. 4

Clearly, caesium absorption in the paint must exceed 100 pCi/cm^2 before release of activity can produce smearable contamination above 1 d.w.l. The results shown in fig. 4 are based on a pond water concentration of 100 uCi/l, but absorption of caesium did not reach 100 pCi/cm^2. The Winfrith flask is loaded and unloaded in ponds containing not more than 1 µCi/l of radiocaesium, yet contamination levels detectable by smear frequently exceed 1 d.w.l. Therefore there must be some change in the paint structure, with time, which increases absorption of radioactivity by at least 2 orders of magnitude. An attempt has been made to simulate weathering of paint by abrading it with silicon carbide paper. The results are shown in fig. for absorption from water containing 100 uCi/l.

It should be noted that the vertical axis of the graph has a logarithmic scale. The figure shows that caesium absorption by pigmented paint is increased by more than an order of magnitude by abrading the surface; this must be due to exposure of the pigment, which absorbs radiocaesium strongly. Abrasion of pigment-free paint increases caesium absorption by a factor of only 2, presumably simply by increasing the exposed surface area. Operationally, it has been observed that absorption of caesium increases as the paint on the flask ages. Painted specimens have been exposed outdoors to produce a weathered surface for further experiments.

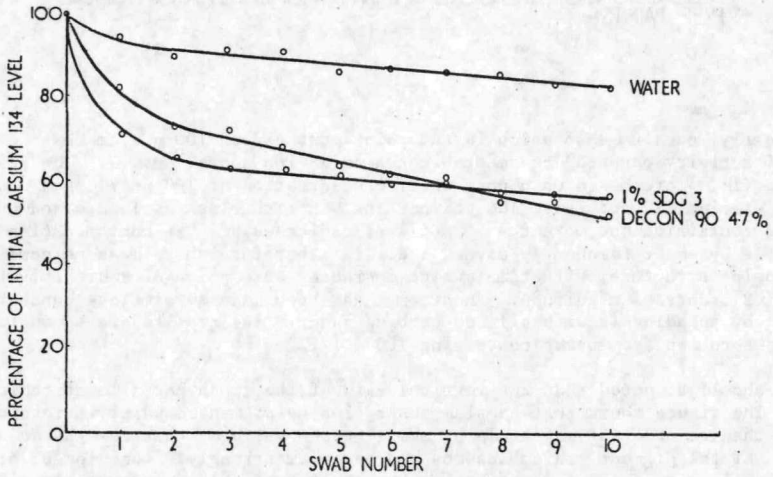


EFFECT OF ABRASION ON ABSORPTION OF RADIOCAESIUM INTO POLYURETHANE PAINTS.

fig. 5

2.2 Release of Caesium

After loading or unloading a flask in a pond, it is decontaminated before shipment, usually by swabbing. Swabbing has been simulated in the laboratory, and the results are summarised in fig. 6.



DECONTAMINATION BY SWABBING

fig. 6

The swabs used are pre-soaked either in water, in detergent, or in a complexing reagent. Whilst swabbing is effective in removing smearable activity from the surface, the total activity burden of the paint is hardly reduced, and the effects are only temporary.

A much better method is to wash the flask in a suitable reagent, either by immersion or by the use of a shower. Laboratory results are summarised in fig. 7.

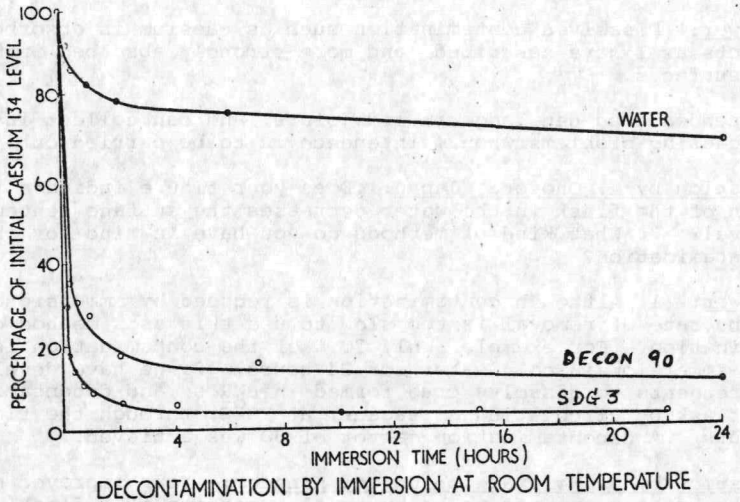


fig. 7

3. DISCUSSION AND PLANS FOR FURTHER WORK

The absorption of radiocaesium from pond waters into paint films has been studied in the laboratory, and this work has provided an explanation for the phenomenon of "flask sweating". Absorption by a newly painted surface is lower than absorption by abraded or weathered paint. Absorption by pigment-free paint is very low, even after abrasion, and it can be argued that pigment-free polyurethane should be used as the top-coat on transport flasks. It is known of course that paint degradation by ultra-violet light is enhanced in the absence of pigment; on the other hand, many flasks avoid exposure to U.V. by the use of covers. The Winfrith 50 Te flask has been repainted recently using standard pigmented polyurethane; half of the flask area has had an additional coating of pigment-free polyurethane.

A decontamination process which can reduce total absorption of caesium to less than 100 pCi/cm^2 would avoid operational problems due to flask sweating. Effective methods have been identified in the laboratory, and will be tested using full-size flasks.

Ref 1: D.A. BAYLISS, P. WALKER: Some aspects of the Surface Protection of Nuclear Plants. Energia Nucleare, Vol. 22, 1975.

DISCUSSION

Question by P.-J. Pöhlitz, FR Germany: During your tests on spent fuel element transport casks did you find any contamination during transport on any other areas besides painted ones, e.g. on joints, gaskets, etc?

Answer: Dissolved contamination such as caesium is absorbed into paints as I have described, and more strongly absorbed into non-painted surfaces.

Suspended crud can lodge in lid joints, and can collect inside flasks, causing problems when maintenance is to be carried out.

Question by A. Onodera, Japan: 1. Does your figure indicate that immersion of the flask in the water decreases the surface contamination level? 2. What kind of methods do you have in mind for internal decontamination?

Answer: 1. Although contamination is reduced by immersion in water, the rate of removal is too slow to use this as a method of decontamination. For example, only 20 % of the contamination is removed by immersion in cold water for 24 hours. 2. We have developed special reagents to dissolve crud formed in BWR's, and decontaminated a 30 Te flask by circulating a reagent, at 90 °C, through the flask for 8 hours, A decontamination factor of 80 was achieved.

Question by S. Williamson, United Kingdom: Your improved method of decontamination appears to be by total immersion of a flask. Is there a relationship between the volume of the liquor to be used and the flask and also the immersion time?

Answer: In the laboratory, using newly-painted coupons, we have established the rate of decontamination. For example, using a complexing reagent, 90 % of absorbed activity can be removed after 4 hours immersion. As far as volume of liquor is concerned, this is not very important in the laboratory, but with a full-size flask we would avoid the cost of immersion in large tanks, and would set up a "shower-bath" system using recirculated liquor.

Question by A. Sihvola, Finland: What is the average time needed for decontamination operations of the outer surface of a cask after wet loading or unloading?

Answer: About 8 to 16 hours.

SESSION XIII
NEUTRONICS AND CRITICALITY

THE USE OF THE MONTE CARLO NEUTRONICS CODE MONK FOR THE CRITICALITY SAFETY ASSESSMENT OF TRANSPORT PACKAGES

by

G Walker
Safety and Reliability Directorate
UKAEA

INTRODUCTION

In the early 1960's the Monte Carlo neutronics Code GEM[1] was developed within the UKAEA for the Criticality safety assessment of arrays of transport packages and was subsequently extended to enable it to deal with more complex systems. About ten years ago GEM was replaced by the MONK[2] code which, whilst fulfilling the same basic role, is more versatile and enables a wider range of problems to be treated with fewer approximations in the geometry. In some types of systems the specification of the problem is also simpler with MONK than it was with GEM. Since its introduction MONK has been continuously developed and is, in principle, capable of dealing with any configuration and virtually any material.

BRIEF DESCRIPTION OF THE CODE

MONK uses point nuclear data derived from the UK Nuclear Data File although "experimental" versions of the code have been produced using group cross sections e.g. from the WIMS library.

The geometrical description of a system such as a transport package is built up of Types each consisting of one or more Regions generally comprising a single material. Types may be spherical, cylindrical or cuboidal. Regions may enclose one another to form a Nest or a number of discrete Regions may be enclosed within an outer boundary to form a Cluster. Types may be arranged as a Cluster, or as an Array like an arrangement of blocks.

A distinctive feature of MONK is the concept of Hole Routines. This technique allows certain types of geometry to be described in a fairly simple way. Some examples of geometrical arrangements which can be described by this method are square or triangular arrays of fuel pins, sets of parallel plates, concentric cylinders, concentric rings of fuel pins, sections of cylinders or spheres and cones. The versatility of this approach is further increased by the ability to use one Hole Routine as a "material" in another. By this means one can, for example, describe lattices of fuel pins in which the pins are hollow and in which there is a gap between the fuel and cladding. Thus in addition to allowing non-standard shapes to be described Hole Routines permit a Region to contain more than one material. These features make geometry specification with MONK very flexible and relatively simple.

Two types of neutron tracking are used in MONK, namely fission-to-fission and boundary-to-boundary and the user chooses the one most relevant to the problem

being considered.

In fission-to-fission tracking neutrons are tracked from birth in fission until they escape, are absorbed or cause further fission. In boundary-to-boundary tracking neutrons are tracked from a specified core/reflector boundary into the core, back across the boundary into the reflector and back to the boundary. Fission tracking calculates k_{eff} directly whilst boundary tracking calculates two parameters, M and R. The surface multiplication, M, is the ratio of the number of neutrons leaving the core to the number entering. Similarly, the reflection, R, is the ratio of the number of neutrons reaching the boundary from the reflector to the number entering the reflector from the boundary. The condition for criticality is that MR = 1. The program also derives a value of k_{eff} from MR but this is somewhat problem dependent and if k_{eff} is required it is preferable to calculate it directly using fission tracking.

An albedo may be specified at any or all of the outer surfaces of a system. If the outer boundary is cuboidal the use of an albedo of unity on appropriate faces will simulate an array of units which is infinite in one, two or three dimensions.

APPLICATION TO PACKAGE ASSESSMENT

The IAEA Regulations for the Safe Transport of Radioactive Materials require that a transport package shall be safe from criticality in the undamage stage (i.e. as presented for transport) and when damaged to the maximum forseeable extent.

An area of major interest at the present time is the transport of irradiated LWR fuel. This type of package typically comprises a heavy metal flask composed of steel and lead containing a number of fuel pin Clusters separated by water gaps and, in many instances, neutron absorber plates.

The geometrical flexibility of MONK allows such a package to be specified in detail and with little or no approximation.

The fuel pin clusters can be represented by a Hole Routine and individual fuel pins can be omitted or replaced by poison pins where appropriate. The basket or frame which keeps the fuel elements in place can also be represented together with any absorber plates and the flask body can be described in as much detail as is deemed necessary.

The "accident case" presents rather more problems, not so much in performing the calculations as in deciding the model on which to make the calculations. Tests on the flask (or a scale model of it) will establish what happens to the packaging under accident conditions but the extent of damage (if any) to the fuel is largely unknown.

It has been the practice in the UK to assume that under accident conditions 10% of the fuel in a flask breaks up into particles sufficiently small to allow them to mix with water and then settle to form a slurry of UO_2 and water at the most reactive concentration - namely around 1.5 gm U/cc. It is appreciated that this assumption, which is quite arbitrary, is very pessimistic and some recent work has indicated that the upper limit of fuel break-up is between 3 and 4%.

Following an accident of sufficient severity to cause fuel to disintegrate one cannot assume that the flask will still be in its normal transport attitude. It may, for example, have rotated about its longitudinal axis or be on its end

at some angle to the vertical or a combination of both conditions. The UO_2/water slurry will obviously settle in the bottom of the flask whichever section this happens to be. Hence in making calculations on the damaged package it is necessary to investigate a number of configurations with the layer of slurry in different positions relative to the fuel clusters. It is found that different configurations do vary in reactivity due to the variations in interaction between the fuel pins and the layer of slurry. To a rather rough approximation, the worst, that is, most reactive, damaged configuration has a value of k_{eff} which can be up to about 0.06 greater than that of the corresponding undamaged package. Some typical values (although with somewhat narrower range of k_{eff} values) are given in Table 1.(3)

TABLE 1

CONFIGURATION	k_{eff}
Undamaged Normal transport position	0.8712
Damaged Normal transport position	0.8611
Damaged Rotated through 45°	0.8852
Damaged Rotated through 90°	0.9069
Damaged Rotated through 135°	0.8819
Damaged Rotated through 180°	0.8931

It should perhaps, be made clear that normal practice within the UK when making assessments of irradiated fuel packages is to assume fresh fuel composition and to ignore the presence of any boron in the flask water. This approach imposes quite a severe penalty in terms of increased reactivity and hence reduced payload. For example, the concentration of boron normally present in the water in a PWR fuel flask will more than counterbalance the increased reactivity from the assumed disintegration of fuel under accident conditions. For example, the results in a typical case were:-

	k_{eff}
unboronated water	0.9722
water containing 1000 ppm boron	0.7810
water containing 1600 ppm boron	0.7112

In making criticality safety assessments one generally uses the Fission Tracking option of MONK which yields a value of k_{eff} for the package which can be compared, if desired, with the results obtained using other codes.

The Boundary Tracking option is useful in some cases. If the Boundary is placed on the outer surface of a package the surface multiplication of the package can be found and this can be used in some interaction calculations such as the Interaction Parameter method. Also, provided that the outer boundary of the system is cuboidal the critical number of packages can be estimated from a calculation on one package by using the PQR method.[1] The requirement of a cuboidal boundary can be met for a cylindrical package by enclosing it in a cuboid of air.

Because MONK is a Monte Carlo code the results of calculations made with it are subject to statistical uncertainties and the output from MONK includes an estimate of the standard deviation associated with k_{eff} (or MR). The criterion for accepting a package as safe which is generally used in the UK is that k_{eff} + 3 S.d is not greater than 0.95 after allowing for any bias. This criterion is applied to the damaged case - the undamaged case in general then has a value of k + 3 S.d of about 0.9 (after allowing for the bias).

It will be evident that the standard deviation referred to above applies only to the numerical value calculated by MONK and takes no account of modelling errors, manufacturing tolerances or possible errors in the code or its nuclear data. Modelling errors are overcome by specifying the geometry as accurately as possible and ensuring that all necessary approximations are pessimistic i.e. err on the side of safety. Similarly, one selects the most reactive combination of tolerances such as the highest enrichment and the thinnest pin cladding. Any errors which may exist in the code (there is no reason to believe that there are any significant errors) or the nuclear data should be revealed during code validation and either corrected or allowed for in the bias.

CODE VALIDATION

MONK has been extensively validated by making calculations on a wide range of critical experiments. The experimental systems of particular relevance to transport packages are:-

(a) lattices of low enriched UO_2 fuel pins in water

(b) lattices of mixed $(U,Pu)\ O_2$ fuel pins

(c) Arrays of MTR type fuel elements in water

Experiments with low enriched UO_2 fuel clusters in simulated flask configurations have been made recently by the Experimental Criticality Group at Battelle's Pacific Northwest Laboratories[4][5] and by a CEA group at Valduc [6]. Some of these experiments included neutron absorbing plates between fuel pin clusters and others included lead or natural uranium reflecting walls. The Battelle group have also made experiments on mixed oxide lattices [7].

Experiments with arrays of MTR type fuel elements in water were made some time ago at Oak Ridge [8].

The results of calculations on some of these experiments are given in Table 2. It will be seen from these results that MONK tends generally to over-estimate k_{eff} of these systems although in three cases the reactivity was under-estimated by 1 or 2%. For the uranium fuelled lattices the agreement with experiment was within 2-3% in k_{eff} but for the mixed oxide lattices the agreement was not so good and k_{eff} was over-estimated by almost 5% in some instances.

TABLE 2

Uranium Enrichment	Fuel Type	Poison Plates	Reflector	Neutron Tracked	k	S.d.	Ref
2.35	UO_2 Clusters	None	Water	10,000	0.9939	0.0120	4
2.35	"	"	"	10,000	1.0265	0.0129	4
2.35	"	"	"	10,000	0.9820	0.0134	4
2.35	"	"	"	10,000	1.0028	0.0123	4
2.35	"	Boral	"	12,000	1.0107	0.0114	4
2.35	"	"	"	12,000	1.0215	0.0107	4
2.35	"	"	"	12,000	1.0141	0.0116	4
4.75	UO_2 pin lattices	None	"	12,000	1.0132	0.0119	6
4.75	"	"	"	12,000	1.0213	0.0117	6
4.75	"	"	"	12,000	0.9931	0.0117	6
4.75	"	"	"	12,000	1.0110	0.0112	6
4.31	UO_2 Clusters	None	lead & water	13,000	1.0249	0.0111	5
4.31	"	"	"	13,000	1.0193	0.0110	5
4.31	"	"	"	13,000	1.0301	0.0107	5
-	Mixed oxide (22.5% Pu O_2) Pin Lattices	"	Water	12,000	1.0436	0.0110	7
-	"	"	"	12,000	1.0341	0.0107	7
-	"	"	"	12,000	1.0358	0.0118	7
93.2	U/Al alloy plates	None	Water	12,000	1.0136	0.0104	8
93.2	"	"	"	12,000	1.0015	0.0109	8
93.2	"	"	"	12,000	1.0213	0.0118	8
93.2	"	"	"	12,000	1.0003	0.0108	8

ACKNOWLEDGEMENTS

The author thanks Mr K C Rushton and Mr P Hague for providing data for inclusion in this paper.

REFERENCES

1. Hemmings P J, The GEM Code AHSB(S) R 105 (1967).

2. Sheriffs V S W, MONK - A General Purpose Monte Carlo Neutronics Program SRD R 86 (1978).

3. Rushton K C and Hague P, Unpublished Work.

4. Bierman S R, Clayton E D and Durst B M, Critical Separation Between Subcritical Clusters of 2.35% ^{235} U Enriched UO_2 Rods in Water with Fixed Neutron Poisons PNL 2438 (1977).

5. Bierman S R, Durst B M and Clayton E D, Critical Separation Between Sub-Critical Clusters of 2.35% and 4.31% ^{235} U Enriched UO_2 Rods in Water with Lead or Uranium Reflecting Walls. Nuclear Technology 47 51-58 (1980).

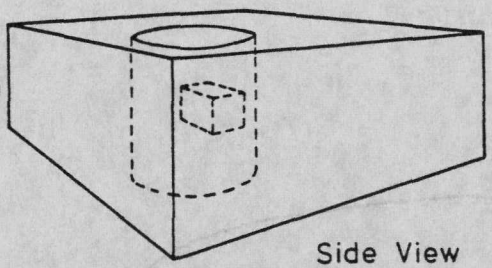
6. Manaranche J C, Mangin D, Maubert L, Columb G and Poullot G. Critical experiments with Lattices of 4.75 wt.% ^{235} U Enriched UO_2 Rods in Water. Nuclear Science and Engineering 71 154-163 (1979).

7. Bierman S R, Durst B M, Clayton E D, Scherpelz R I and Kerr H T. Critical Experiments with Fast Test Reactor Fuel Pins in Water. Nuclear Technology 44 141-151 (1979).

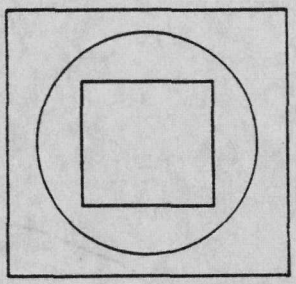
8. Johnson E B and Reedy R K. Critical Experiments with SPERT-D Fuel Elements. ORNL-TM-1207 (1965).

FIG. 1 BASIC TYPES USED IN MONK

(a) NEST

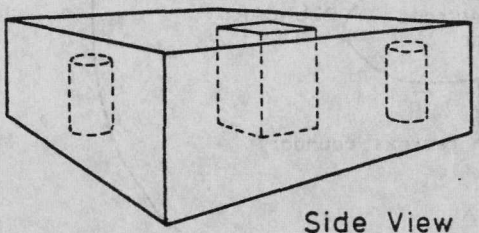


Side View

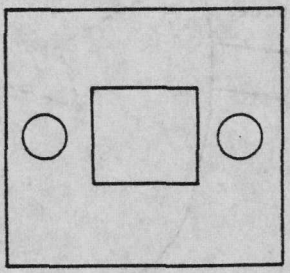


Plan View

(b) CLUSTER



Side View



Plan View

(c) ARRAY

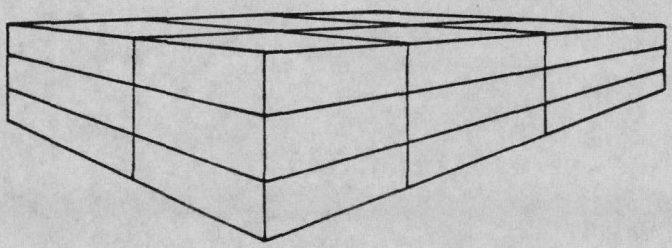
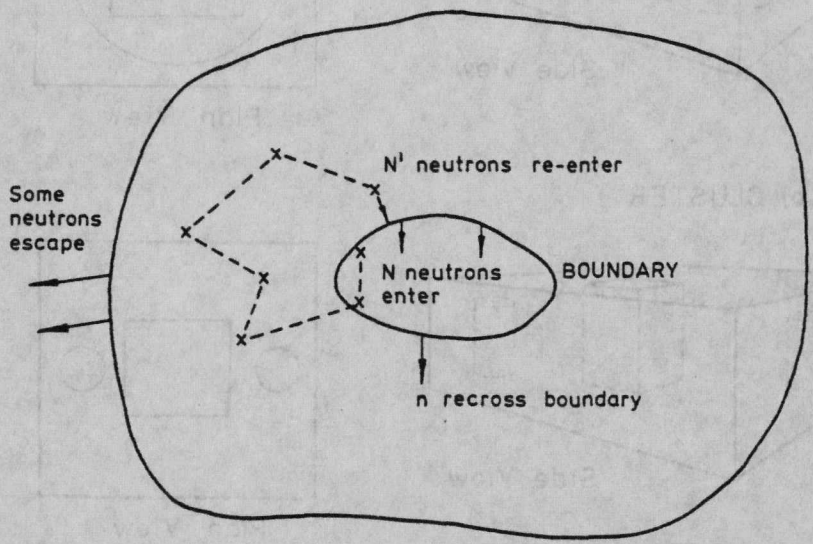


FIG. 2 BOUNDARY TO BOUNDARY TRACKING



MR = N'/N M = n/N R = N'/n

Internally, neutrons will be absorbed, scattered or created from fission, (n,n'), (n,2n), (n,3n) reactions

DISCUSSION

Question by A. Olivieri, Italy: What water densities were considered in your criticality calculations?

Answer: The water inside the flask was assumed to be at full density (i.e. 0.998 g/ml). Previous calculations to assess the effect of reduced water density corresponding to increased temperature showed that such reduction of the water density gave slightly lower values of k_{eff}.

When considering arrays with water outside the packages we assume that the water is at full density if the packages do not have significant spaces between them since in such cases the water acts as a reflector. Where there are appreciable spaces between packages (other than heavily shielded irradiated fuel flasks) the effect of variation of water density on the reactivity of the array is investigated and the assessment is based on the most reactive configuration.

Question by M. S. T. Price: A modern LWR transport flask carrying, say, 12 PWR fuel elements if calculated by the methods described in your paper would have a high k_{eff} value. Should there be any trend to the use of higher initial enrichment in PWR fuel I anticipate that the k_{eff} would be unacceptable using your stated criterion

$$k_{eff} + 3\sigma \leq 0.95$$

Can you comment therefore on the need to develop burn-up or fissile content monitors for use at reactors during loading.

Answer: Yes, you are right. Indeed the conservative assumptions described in the paper have led already to the limit being reached in some cases. The IAEA Regulations recognise the effect of burn-up in reducing reactivity and MONK calculations can readily take the specific isotopic composition of the fuel into account. The problem is the difficulty experienced by a consignor in being certain of the burn-up of the fuel. As we know, it varies with the location of the fuel in the reactor and varies even along the length of a fuel element.

It really is a question for the reactor operator to solve. I believe that the development of a reliable on-line burn-up measuring system would be invaluable if it enabled a consignor to ensure compliance with a criticality clearance based on a specified burn-up. In that way, package designs would be approved on a less conservative and hence more realistic basis.

THE HANDLING OF CRITICALITY PROBLEMS DURING THE APPROVAL OF FISSILE CLASS PACKAGE DESIGN BY THE PHYSIKALISCH-TECHNISCHE BUNDESANSTALT (PTB)

H.-H. Schweer
Physikalisch-Technische Bundesanstalt
Ref. SE 3.1
Postfach 33 45
3300 Braunschweig

In the Federal Republic of Germany the PTB is the competent authority for the approval of packages for radioactive materials. Close contact exists between the PTB and the "Bundesanstalt für Materialprüfung" (BAM), which studies the mechanical and thermal behaviour of the packages. The PTB deals primarily with criticality and shielding problems.

This paper deals with the usual procedure for the approval of fissile class package design. According to the transport regulations, the applicant has to make a safety report in which he shows that the design meets the relevant requirements.

In general the applicant starts the criticality calculations simultaneously with the provision of the package or a model of the package for testing. This procedure implies the risk that some assumptions (leak tightness, deformation of the package after the tests) must be modified when the tests have been finished. In most cases criticality safety is proved by means of computer programmes. This is caused by the fact that great difficulties arise when simulating by experiments the same hypothetical conditions which must be supposed according to the transport regulations. Since experimental data are often not available, it is very important to check the calculation method and the cross section data sets used for the calculations. The applicant must supply all necessary information in this field. Of course, it is helpful if the applicant

can show that the method of his calculation and the basic data are applicable to similar known, critical systems.

As an example, a modern package is taken to show by which methods and which basic data the safety report is controlled regarding criticality aspects. A short description of the package is given (Table 1). The package is intended for the transport of four irradiated fuel elements for the PWR-Reactor "Bilbis". The main features of this type element are shown in table 2. The criticality safety report of the applicant was written by the "Gesellschaft für Reaktorsicherheit". With the well-known code "Gamtec" (1) and the cross-section data which belong to this code, 16 group constants were produced for the different materials. These group constants were given in the Monte Carlo-Code "KENO" (2) to calculate the effective neutron multiplication factor (k_{eff}) of the flooded package. The results of these calculations show that an infinite number of flooded packages is subcritical. This arrangement was simulated by placing a perfect reflector at the outside of the package in the Monte Carlo calculation.

Table 1: Main data of the package "Castor 1A"

Outer container:	dimensions:	length	6530 mm
		width	1730 mm
		height	1730 mm
	material:	cast iron	
	weight:	about 70 t	
Inner container:	length	about 5000 mm	
	width	600 mm	
	height	600 mm	
	number of elements:	2 x 2 (quadratic)	
	neutron absorber:	boron steel (0.8 % Boron)	
	thickness of absorber:	2 x 10 mm between neighbouring elements	
	spacing between absorber plates:	70 mm	

Table 2: Description of fuel element

type:	DWR-Element Reactor "BIBLIS"
fuel material:	UO_2
fuel density:	10.35 g/cm^3
fuel mass (UO_2):	605 kg
enrichment:	3.2 %
number of rods:	236 = 16 x 16 (- 20 control rods)
active length:	3900 mm
diameter of rods:	9.08 mm
outer diameter:	10.75 mm
cladding material:	Zircaloy-4
pitch:	14.3 mm
arrangement of rods:	quadratic

Some further aspects were studied, namely

- the reactivity effect of higher temperature
- the influence of changing the moderation condition by reducing the number of rods in the fuel lattice.

In order to check the criticality calculations, the PTB asked for the complete listing of the computer programmes for the case which resulted in the highest reactivity. All input data for the two codes were checked and could be verified. Calculations with the "GAMTEC" code were repeated by the PTB and by another Monte Carlo code "MORSE-K" (3) - a revised version of the "MORSE" (4) code - the k_{eff}-value of the flooded and fully water-reflected package was determined. The k_{eff} result of the KENO calculation was about 1 % lower than the result of the PTB calculation (0.9351 against 0.9445). Since the statistical standard deviation of the two calculations was 0.6 % and 0.8 % this difference seems to be acceptable.

In order to avoid systematic errors of the cross section data sets, it was found necessary to use other cross section data and calculation methods. We used the modular programme system "RSYST" (5) which we received from the "Institut für Kernenergetik" of the Stuttgart University. This system is well-tested against benchmarks problems and can be used for reactor and shielding

problems. Fast and thermal cross-section libraries are based on ENDF/B-IV data. The following steps must be taken in order to generate problem-dependent few-group constants:

- two zero-dimensional spectrum calculations are made in the fast and the thermal range by the B_1 method. The flux in the resonance domain is calculated by the Nordheim or NR approximation (code GGC-IV)
- by means of a special series of modules the 99 fast energy groups (Gam-II-structure) and the 126 thermal energy groups are collapsed in order to generate microscopic group constants with 60 energy groups
- by one dimensional transport calculations using the S_n method or the collision probability method, problem-dependent macroscopic group constants (60 energy groups) can be produced
- these group constants are collapsed to a lower group number (e.g. 16 or 20) using the corresponding flux in each region of the one-dimensional S_n calculation and are inserted into the Monte Carlo code.

The calculations made for a number of 186 rods for each element yielded the following result: $k_{eff} = 0.9022 \pm 0.007$. This result is about 3.5 % lower than the result of the calculation with GAMTEC cross-sections and the KENO code. Changing the number of rods per element reduces the k_{eff} volue which also confirms the criticality safety report of the GRS. The MORSE-K calculations confirm that the conclusions of this report are acceptable. Regarding the criticality aspects it was therefore decided to give the approval for the package in the fissile class I according to the transport regulations.

The examples given here show the principles which will be observed by the PTB when approving the fissile class package design:

1) The criticality safety report is to be prepared by the applicant who must consider the criteria of the relevant transport regulations
2) The PTB may require the complete listing of computer calculations in order to check the input data and to control the conformity of this data with the structure of the package including the fissile content
3) Parts of the calculations will be repeated in order to ensure that the calculation method (e.g. the Monte Carlo Method) available in the PTB used with the same cross-sections is applicable
4) Independent calculations using newer cross-section data based on ENDF B IV (tape) will be carried out to avoid systematical errors of the basic data.

By means of a one-dimensional S_n Programm, it is possible to take into account the special structure of the package when reducing the number of energy groups.

These principles ensure an adequate control of the criticality calculations given by any applicant. Thus the public safety requirements involved in transport of fissile material are duly fulfilled.

References:

1. L. L. Carter et al.
   GAMTEC II: A CODE FOR GENERATING CONSISTENT MULTIGROUP CONSTANTS UTILIZED IN DIFFUSION AND TRANSPORT THEORY CALCULATIONS
   BNWL-35
   March, 1965

2. G. E. Whitesides, N. F. Cross
   KENO: A MULTIGROUP MONTE CARLO CRITICALITY PROGRAM
   CTC-5
   Sep., 1969

3. F.A.R. Schmidt, W. Bernnat
   MORSE-K: DAS MORSE-K PROGRAMMPAKET VERSION 1
   EIN MONTE-CARLO-PROGRAMMSYSTEM FÜR KLEINE RECHENMASCHINEN
   Bericht Nr. 4-11, Institut für Kernenergetik, Universität Stuttgart (IKE)
   Dez. 1972

4. E.A. Straker et al.
   The Morse Code - a Multigroup Neutron and Gamma Ray Monte Carlo Transport Code
   ORNL - 45 85 (1970)

5. R. Rühle
   RSYST, EIN INTEGRIERTES MODULSYSTEM MIT DATENBASIS ZUR AUTOMATISIERTEN BERECHNUNG VON KERNREAKTOREN
   IKE-Bericht Nr. 4 - 12
   Jan. 1973

DISCUSSION

Question by D. Haon, France: What is the neutronic material employed in CASTOR casks, as this cask is accepted in class I?

Answer: The CASTOR casks have Boralsteel (0.8 % Bor) between the elements. In the outer part of the cask there are additional holes filled with polyethylene.

Question by A. R. Baker, United Kingdom: Is it intended to establish a library of nuclear data which can be used in the Federal Republic for calculations of criticality and also for design calculations for fast and thermal reactor?

Answer: Up to now there are no activities to be seen in this field. Mr. W. Thomas from the "Gesellschaft für Reaktorsicherheit" can probably make some remarks on this problem.

One additional remark is that the programme system "RSYST" mentioned in my paper seems to be one which is used increasingly in the FR Germany. The cross-section data used in this system may serve as the basis of a "norm" in the near future.

Comment by W. Thomas: As Mr. Schweer mentioned, there are no efforts to establish such a cross-section library suitable for all types of criticality problems. It seems better to us to use different cross-section data for specific problems, cross-section data, which have been verified for specific or similar problems. In some cases it seems a good approach to use different calculational procedures and data for comparison. In this procedure one should always be aware of the limitation of the various data sets. So a high degree of reliability can be reached.

VALIDATION OF THE APOLLO-MORET NEUTRONIC CODES ON CRITICAL EXPERIMENTAL CONFIGURATIONS SIMULATING THE SHIPPING CASKS FOR LIGHT WATER FUELS

D. HAON, J.C. MANARANCHE, D. MANGIN, L. MAUBERT, G. POULLOT
Commissariat à l'Energie Atomique

The present development of electronuclear power in France is based on light water reactors and requires to design and to build large capacity shipping casks for the corresponding fuels. In the design of such casks, the criticality safety aspect is extremely important and imposes some limitations. The planned shipping casks have a complex geometry and particularly include neutron absorber shields in order to achieve a maximum compactness. The criticality safety of a shipping cask or of an arrangement of casks can only be based on calculation results obtained by sophisticated neutronic codes which have to be validated by critical experiments simulating as closely as possible the true configurations.

In this purpose, critical approaches have been performed in the criticality facility of Valduc (Commissariat à l'Energie Atomique) with four $U[4.75]O_2$ fuel rods clusters. These experiments consisted in determining the critical water heights of the various assemblies by extrapolating the inverses of the counting rates recorded during a progressive rise of the water around the rods, this water acting at the same time as moderator and reflector.

DESCRIPTION OF EXPERIMENTAL ASSEMBLIES

The characteristics of $U[4.75]O_2$ rods are presented table I and figure 1.

Table I - Characteristics of fuel rods

FUEL		CLADDING	
Material : UO_2		Material : Aluminium alloy	
Pellet diameter (mm) : 7.9		I. Diam. x O. Diam. (mm) 8.2 x 9.4	
Height of fuel (mm) : 900		Composition (wt %) :	
Exact enrichment : 4.742 % ^{235}U		Al	98.85
Oxide density (g/cm^3) : 10.38		Mg	0.47
Sum of impurities in Boron		Si	0.43
equivalent (g/cm^3) : $5.19\ 10^{-6}$		Fe	0.22
		Zn	0.03

The studied assemblies include four rods clusters with square disposition ; each cluster is composed of 18 x 18 rods (320 fuel rods and 4 stainless steel rods) disposed according to a 16 mm square pitch (print 1). Each cluster is lateraly surrounded by Boral shields (250 mg Boron/cm^2) (prints 2 and 3). The bottom is reflected by water and the sides are reflected either by water only or by 10 cm of lead plus water or by 15 cm of steel plus water (print 4 shows

the lead reflected assembly, outside the experimental pool). Two variables have been studied : space (Δ) between two clusters and distance (S) from the outside of the shields to the lead or steel reflecting wall. The dimensioned diagram of a quarter of the studied assemblies is presented figure 1.

EXPERIMENTAL RESULTS

The experimental results are given in table II and plotted figure 2.

Table II - Experimental data and results

Reflecting walls Material	E (mm)	Δ (mm)	S (mm)	Critical Height HC (mm)	keff ± 3σ
Water	0	3	-	507.4	1.011 ± 0.014
		5	-	530.1	-
		10	-	574.3	-
		15	-	661.5	1.001 ± 0.014
		20	-	729.6	1.016 ± 0.014
		25	-	841.4	1.004 ± 0.014
		30	-	> 900	-
Lead + Water	100	20	2.3	539.8	1.019 ± 0.014
		20	25	638.5	1.008 ± 0.014
		20	50	694.0	1.005 ± 0.014
Steel + Water	150	20	2.3	515.5	1.008 ± 0.014
		20	25	618.4	1.006 ± 0.014

In order to use these critical experimental data as benchmarks, the composition (in 10^{24} atoms/cm^3) of all the mixtures is specified in table III.

Analysis of experimental results leads to state the following informations :

- Water contained between clusters increases the neutronic absorption of the Boral plates and the general reactivity is lowered.
- A fairly thin (25 mm) gap of water in front of lead or steel walls noticeably decreases their reflecting effect.
- The reflection due to a 10 cm thick lead wall + water is nearly the same than the reflection due to a 15 cm thick steel wall + water.

DATA ANALYSIS BY CALCULATION CODES

Calculations are carried out in two phases using the APOLLO [1] and MORET [2] codes.

APOLLO code solves the multigroup form of the neutron transport equation in one dimensional plane, cylindrical or spherical geometry and uses the collision probability with one of the following approximations : "transport

correction", B1 approximation. Here, APOLLO is used with the "transport correction" option, in order to calculate the macroscopic cross-sections of the homogeneous mixture which is equivalent to UO_2 rods infinite lattice in ambiant medium (water or air). Cross sections are collapsed in 16 energy groups (group structure of Hansen and Roach). The cell is assumed to be cylindrical.

Critical experimental configurations are exactly described by MORET code. For the lattice it uses APOLLO calculated cross-sections, for the other materials it uses Hansen and Roach 16 groups cross-sections.

CALCULATION RESULTS

In table II, are given the average calculated keff ($\overline{keff}$) with a statistical dispersion equal to three times the standard deviation : 3σ (99.73 confidence interval, assuming a normal law).

The calculation methods, which are used, bring very satisfying keff values, close to one ($\overline{keff}$ is between 1.001 and 1.019) but slightly higher, what is leading to more security.

SUBCRITICALITY CRITERIA

Choosing the lowest value among the $\overline{keff} - 3\sigma$ derived from table II, that is 0.987 (1.001 - 0.0014), it can be said that there is a probability higher than 0.9987 for the subcriticality to be certified according to a calculated keff lower than 0.987. Concerning the calculation of assemblies very close to those experimentaly studied, the following subcriticality criteria could be retained :

$$\overline{keff} + 3\sigma < 0.987$$

(The probability corresponding to this criteria is $\# (0.9987)^2 = 0.9974$).

But, for the calculation of assemblies near those experimentaly studied (this is the case of shipping casks), it is necessary to take an additional safety margin that we arbitrarily appoint about 0.02 ; then, we may keep the following subcriticality criteria :

$$\overline{keff} + 3\sigma < 0.97$$

($\overline{keff}$ calculated with the preceding method)

It is to be noted that this criteria is only valuable for calculation results ; nevertheless in the event that a conservative set of data would be introduced in the calculations, this last criteria may be used.

CONCLUSIONS

The presented experimental data provide benchmarks.

The practical lessons drawn from the experiments and the good agreement of calculation results (APOLLO + MORET) with experimental determinations give us confidence in achieving nuclear criticality safety studies of shipping casks for light water reactors fuel elements.

REFERENCES

[1] Alain HOFFMAN, Françoise JEANPIERRE, Alain KAVENOKY, Michel LIVOLANT, Henri LORAIN
APOLLO - Code multigroupe de résolution de l'équation du transport pour les neutrons thermiques et rapides
Note CEA - N 1610 (1973)

[2] R. CAIZERGUES - J.P. MARIOT - J. MORET-BAILLY - G. POULLOT - J.R. TEILLET
MORET : un programme Monte Carlo pour le calcul rapide de coefficients de multiplication effectifs de milieux fissiles dans des géométries complexes
Rapport CEA - R - 4752 (1976)

TABLE III COMPOSITION

(atomic concentrations in $10^{24}/cm^3$)

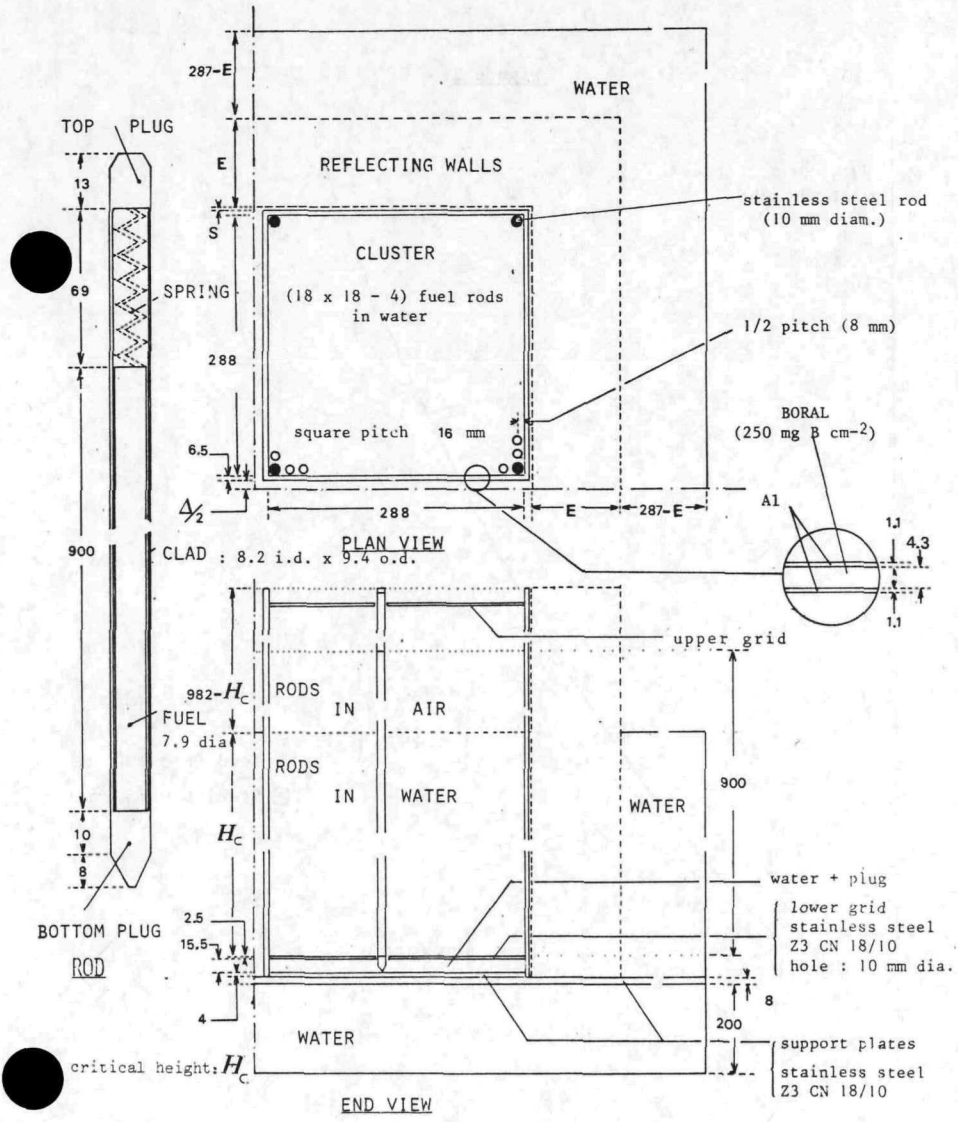
	air	stainless steel Z3CN18/10	bottom plug + water (b)	lower grid water + plug + stainless steel (b)	water	boral plate clad	boral plate boral	reflecting walls lead	reflecting walls steel	fuel rods fuel (c)	fuel rods clad (a)
density	0.001293	7.90	1.2989	6.2394	0.99820	2.651	2.6189	11.34	7.8	10.38	2.70
element											
Al			0.01068	0.01626		0.059177	0.041858				0.059535
B							0.032375			0.00000029	
C							0.008089		0.000548		
Cr		0.016467		0.01141							
Fe		0.061341		0.042520					0.083831		0.000064
H			0.054872	0.002380	0.066742						
Mg											0.000334
Ni		0.008107		0.00562							
N	0.00004325										
O	0.0000108		0.027436	0.001190	0.033371					0.046406	
Pb								0.032965			
Si									0.0003348		0.00024
^{235}U										0.0011118	
^{238}U										0.022051	

(a) wt % : Al 98.85 - Mg 0.5 - Si 0.43 - Fe 0.022

(b) average atomic concentrations

(c) exact enrichment : 4.742 %

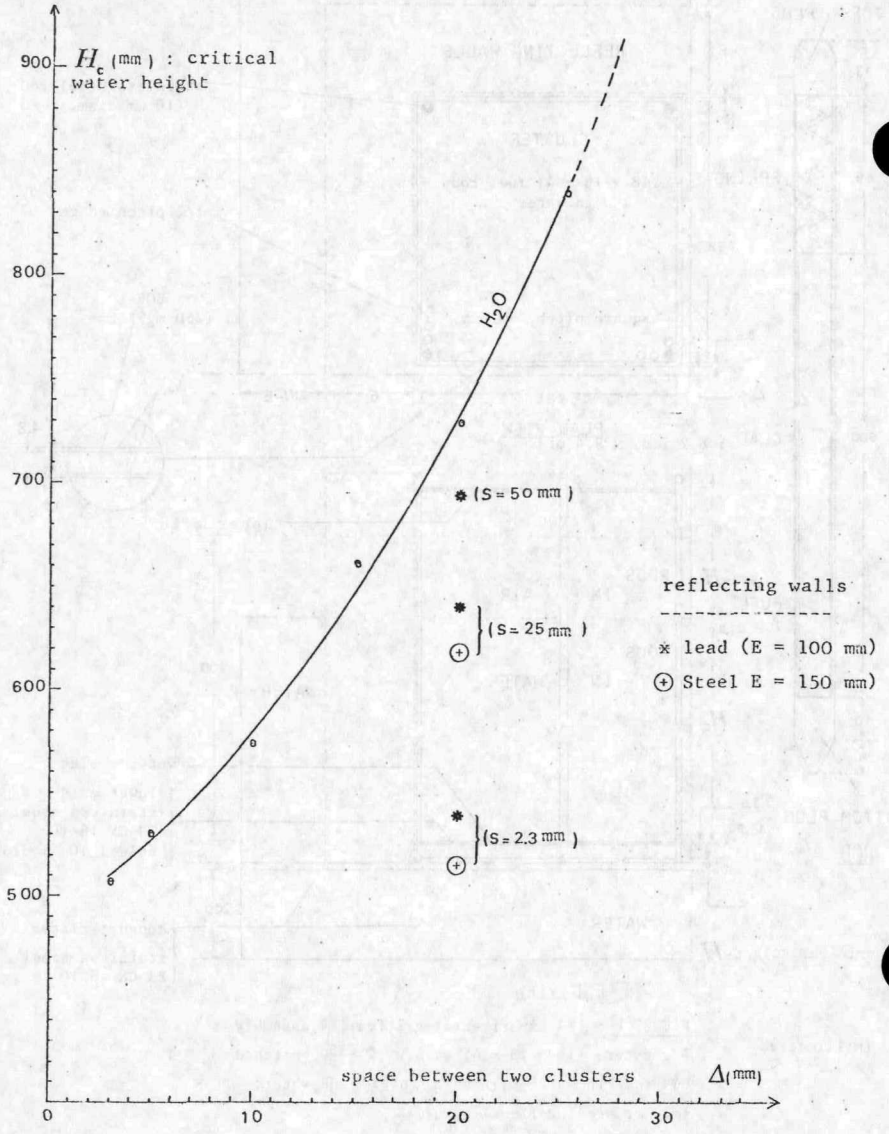
Remark : upper plugs, upper grid and spring may be neglected

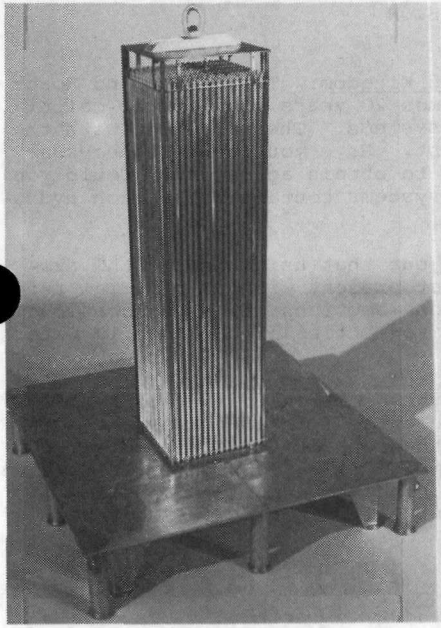


(millimetres)

FIGURE 1 - 1/4 experimental criticality assembly 4 clusters (18 x 18 - 4) 4.75 wt % ^{235}U enriched UO_2 rods in water with boral shield and water, lead or steel reflecting walls.

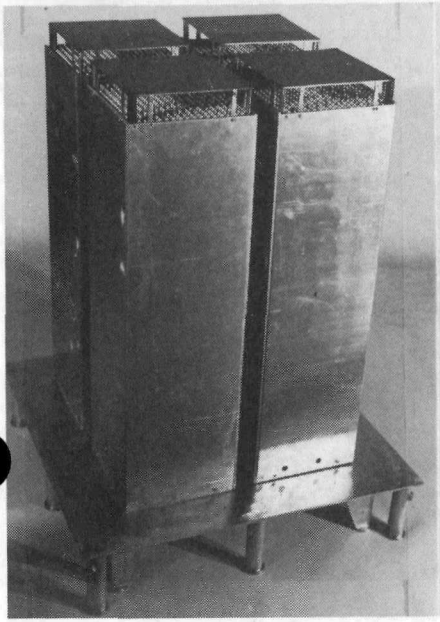
Figure 2


900
800
700
600
500
H_c (mm) : critical water height
H_2O
(S = 50 mm)
(S = 25 mm)
(S = 2.3 mm)
reflecting walls
* lead (E = 100 mm)
⊕ Steel E = 150 mm)
space between two clusters
Δ(mm)
0
10
20
30

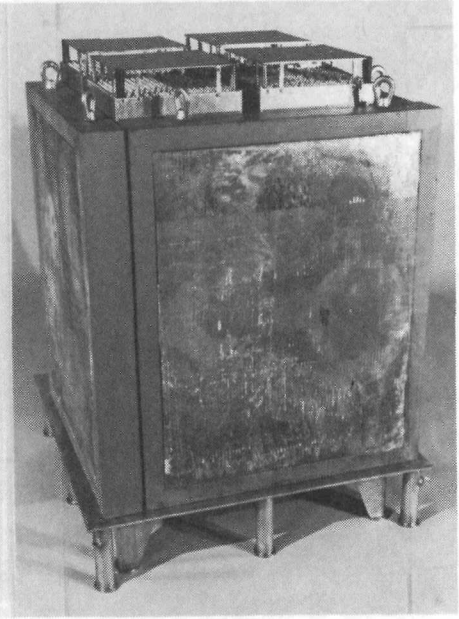


Print 1 : One fuel cluster

Print 2 : Four fuel clusters



Print 3:.. surrounded with boral shield



Print 4: ..and with lead reflecting walls

DISCUSSION

Question by A. R. Baker, United Kingdom: The Hansen and Roach data set was established at Los Alamos 20 years ago for the calculation primarily of highly enriched systems. The agreement with the experiments is nevertheless excellent. Have you found it necessary to improve the Hansen and Roach set to obtain agreement? Could you also comment on its usefulness for systems containing so much hydrogen moderator?

Answer: I would like to point out that using the APOLLO programme (which has its own library, at present without official reference), the average macroscopic cross-sections of the homogeneous environment equivalent to the infinite lattice of UO_2 rods in water or air have been determined.

For all other materials, the cross-sections based on the 16 energy groups of Hansen and Roach were used; in all cases the total microscopic cross-section of the 16th group of the table relative to hydrogen was modified, taken from 7.2 to 29.29 Barn. (This value is justified by the fact that it enables the experimental value of 0.160 cm of the diffusion coefficient D of water to be recovered.

NEUTRON ISOLATION OF PACKAGINGS WITH A BORATED COMPOUND USUALLY USED IN FRANCE

D. HAON* - L. MAUBERT* - C. BOCHARD**
* Commissariat à l'Energie Atomique
** Société ROBATEL-SLPI

For the shipment of new or spent fuels, the packagings manufactured in France contain a layer of borated compound. A packaging so designed satisfies the conditions concerning the neutron radiation shielding (Art. 229 of I.A.E.A. Regulations).

In 1968, a patent was taken out for this material by CEA and SLPI Co (now Robatel-SLPI Co). It consists of polyethylen, plaster, water and boron. At the present time, three compounds called N8, N9 and N10 are utilized. The compositions (wt %) are specified in Table I. The densities are of about 1.25, what is advantageous for packagings transport.

TABLE I - Compounds composition

Compound	Polyethylene	Polypropylene	Plaster	Colemanite (1)	Water
N 8	41	0	28	6	25
N 9	41	0	31	8	20
N 10	0	40	32	8	20

(1) Colemanite is a mineral which composition (wt %) is :

B_2O_3 = 43.3 - CaO = 24.5 - H_2O = 20 - SiO_2 = 3.7 - others = 8.5

The compounds preparation process includes the following steps : dry incorporation of colemanite in plaster, mixing with water, mixture homogeneization, polyethylene or polypropylene granules incorporation, pouring into the packaging.

PROPERTIES OF THESE COMPOUNDS

1. The fire behaviour, owing to the presence of plaster, is satisfying. For instance, a thermal test has been performed with a packaging FS 33 agreed with n° F/134/B(U)-F. This one destined to the transport of new fuels of the fast neutrons reactor PHENIX contains a layer of borated compound (thickness : 84 mm), between two walls of steel. Thermocouples are situated in different points of the packaging :

- in connection with the external wall
- at the interface external wall-compound
- at 15 mm and 20 mm within the compound

The temperatures registered at the end of the test (30 minutes) are respectively 1030° C - 650° C - 105° C et 40° C. 40 minutes after the extinction of the fire, the temperatures are 147° C - 107° C - 104° C and 74° C. The two last temperatures are much lower than those leading to the dessication of the compound. An examination performed after the test shows that only the first millimeters of the superficial layer were calcinated and that the interior layer was preserved. The withstanding to fire is ameliorated if the compound is surrounded with a thermal protection, for example N2 which composition (wt %) is :

Al_2O_3 = 48 - Cement = 18 - Plaster = 8 - Water = 26

2. The mecanical properties are very near these of building wet plaster.

3. An integrated flux of 1.8 . 10^{17} neutrons . cm^{-2} has not degraded this material (which only looks a greyish colour).

4. After dessication, the contents in Hydrogen and Boron are respectively 0.078 g/cm^3 and 0.010 g/cm^3. This is a necessary condition for an effective neutron shielding. The fission neutrons being fast, they must be slowed down by light atomic mass nuclides (H and C) before their absorption by neutron thermal poisons (B and H).

CALCULATION RESULTS

1. The first study has been performed to evaluate the reflection and isolation effects of compound N8, with the transport code DTF IV using the Hansen and Roach 16 group cross sections. Keff are calculated for two configurations :

- an infinite slab reflected by a variable thickness of compound
- an infinite array of slabs separated by a variable thickness of compound.

This geometric shape was chosen because it leads to the maximal effect of reflection and neutron interaction.

The fissile material is :

- metallic uranium highly enriched in ^{235}U, the thickness of the slab is 1.5 cm ;
- plutonium nitrate aqueous solution (300 g/l Pu).

Both considered fissile systems were chosen on account of their different neutron spectra, fast in the case of metallic uranium and thermal in the case of plutonium nitrate.

The compositions of materials introduced in the calculations are given in the table II.

TABLE II

Material	Density	Atomic concentrations (in 10^{24} atoms.cm^{-3})					
Initial Compound N 8	1.244	Carbon	0.0220078	Calcium	0.0016503		
		Hydrogen	0.0673644	Boron	0.0005617		
		Oxygen	0.0186126	Silicon	0.0016884 (S + Si)		
Compound N8 after dessication	0.932	Carbon	0.0220078	Calcium	0.0016503		
		Hydrogen	0.0464691	Boron	0.0005617		
		Oxygen	0.0081826	Silicon	0.0016884 (S + Si)		
Metallic uranium	18.8	^{235}U	0.048184				
Plutonium nitrate aqueous solution	1.438	^{239}Pu	0.0007560	Oxygen	0.03671		
		Nitrogen	0.003024	Hydrogen	0.05527		

The calculation results are plotted on fig. 1, curves 1 and 1' for metallic uranium, curves 2 and 2' for plutonium nitrate aqueous solution. On fig. 1, we see :

- for both fissile materials considered, the thickness of compound N8 saturating the reflection effect is about 15 cm and the thickness isolating is about 30 cm (double thickness) ;
- for a thickness less than 30 cm, the upper limit of interaction is given by keff(1) - keff(1'). For instance this value is about 0,06 for a 20 cm thickness.
- The coumpound N8 is a poor reflector since the keff of metallic uranium slab, reflected by compound, is 0.715 while reflected by water, keff = 1. The critical thickness of a slab containing plutonium nitrate aqueous solution (300 g/l Pu) reflected by compound is 10 cm while reflected by water, this thickness is 6.1 cm.

2. The second study has been performed with cylindrical configurations more like these of packagings. The calculations have been derived from the Moret code that is a Monte Carlo code using the same Hansen and Roach 16 group cross sections. The fissile materials are the same as in the first study. The cylinder diameters are respectively 7.6 cm for mettallic uranium and 15.4 cm for plutonium nitrate aqueous solution, these values would be critical with a water reflector. The results are given in table III and plotted on figure 2.

TABLE III

CASE		Compound thickness (cm)	keff ± 3σ (a) Metallic U	keff ± 3σ (a) Plutonium nitrate solution
A single cylinder (infinite height) reflected by a variable thickness of compound	Initial compound	2	0.815 ± 0.012	0.750 ± 0.019
		5	0.881 ± 0.014	0.822 ± 0.019
		7.5	0.904 ± 0.017	0.822 ± 0.019
		10	0.905 ± 0.017	0.822 ± 0.020
	Compound after dessication	2	0.776 ± 0.011	0.703 ± 0.019
		5	0.845 ± 0.013	0.780 ± 0.017
		7.5	0.873 ± 0.015	0.785 ± 0.019
		10	0.873 ± 0.016	0.798 ± 0.020
Infinite number of cylinders disposed in triangular pattern array surrounded by a variable thickness of compound	Initial compound	2	1.631 ± 0.023	1.258 ± 0.020
		5	1.074 ± 0.030	0.935 ± 0.022
		7.5	0.998 ± 0.028	0.874 ± 0.021
		10	0.941 ± 0.022	0.847 ± 0.022
	Compound after dessication	2	1.799 ± 0.020	1.365 ± 0.016
		5	1.213 ± 0.027	1.002 ± 0.020
		7.5	1.034 ± 0.025	0.901 ± 0.021
		10	0.952 ± 0.024	0.852 ± 0.023

(a) σ : standard deviation

From this results it is possible to say that :

- in the safety study of individual package considered in isolation, it is necessary to consider the initial compound because it is a better reflector.

- in the study of an arrangment of "damaged" packages, it is necessary to consider the compound after dessication because the interaction is higher.

- the thickness of compound making subcritical an infinite arrangment of packagings is inferior or equal to 10 cm.

EXAMPLES

Packaging FS47 : It is cylindrical and its dimensions are : diameter 710 mm - height 2055 mm - a layer of compound 157.5 mm thick (necessary value for the neutronic shielding) surrounds the internal cavity which diameter is 143 mm. This package contents 17 kg Pu in the form of oxide powder at any moderation (H/Pu), contained in a cylinder which diameter is 130 mm. The number of packages in transport is not limited. This is very interesting. It is scheduled to take the place of the package FS52 which dimensions are comparable, 600 x 600 x 1705 mm, actually agreed in fissile class II, if its content is 12 kg of Pu in the form of oxide powder at limited moderation (H/Pu $\leq$ 2). The number of packages in transport is 38.

Packaging FS55 : It is cylindrical and its dimensions are : diameter 360 mm - height 1140 mm, positionned in an undercarriage 600 x 600 mm section. A layer of compound, 78.3 mm thick surrounds the internal cavity which diameter is 138 mm. This package contents 3 kg of ^{235}U in the form of U-H_2O mixture at any moderation. The keff of an infinite arrangment is only 0.8 and the weight of uranium could be increased. It is very interesting . It is scheduled to take the place of the package FS51 which dimensions are 600 x 600 x 1255 mm, agreed in fissile class II with the same content but with a number of packages in transport lower than 34.

CONCLUSIONS

This compound has two properties which make its use attractive in transport or in storage : poor neutron reflection and very good neutron isolation.

A package with a layer of compound, 100 mm thick, can be agreed in fissile class I or fissile class II with a very large allowable number.

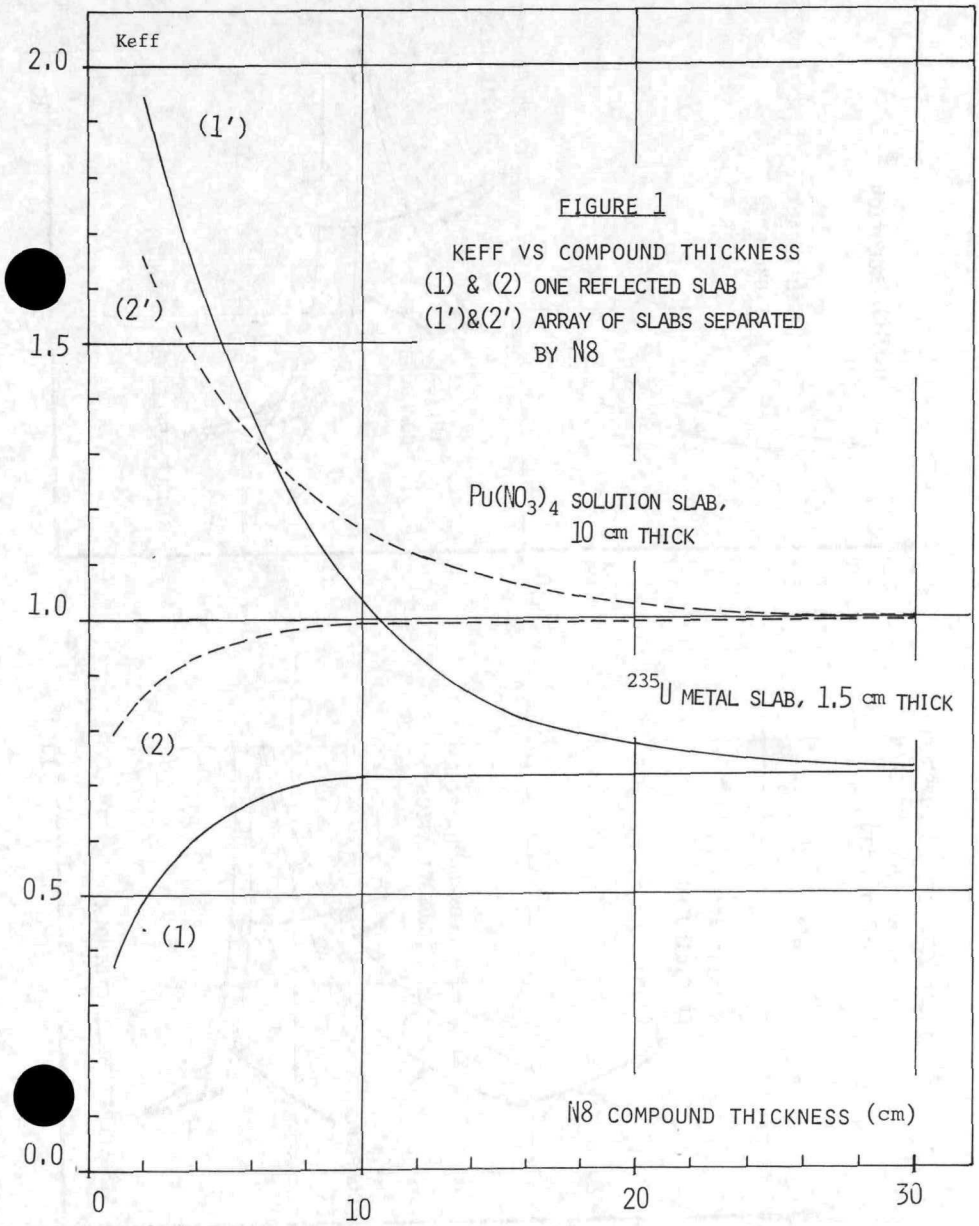
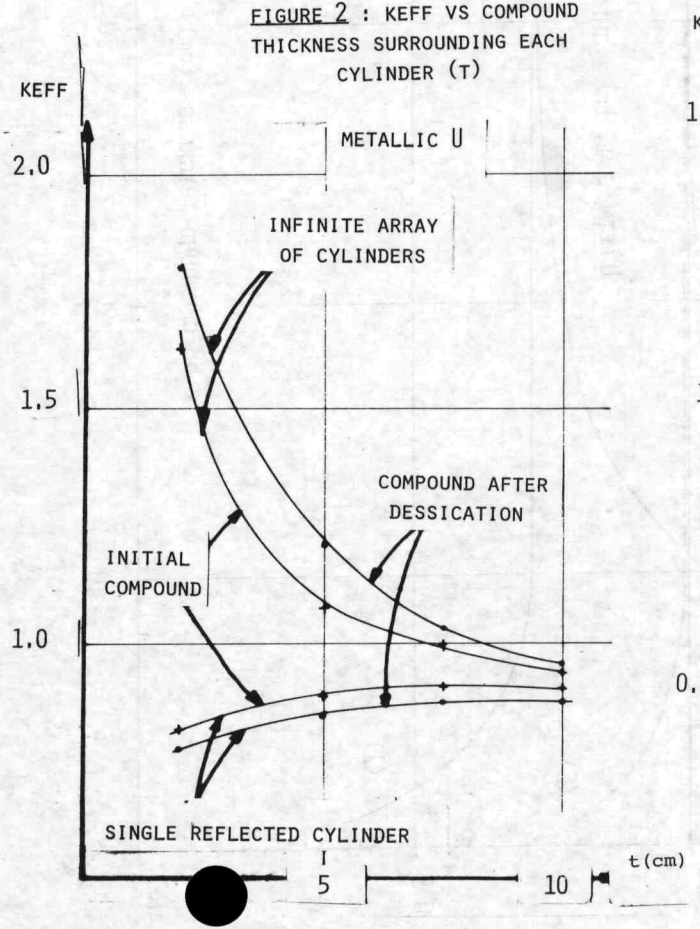
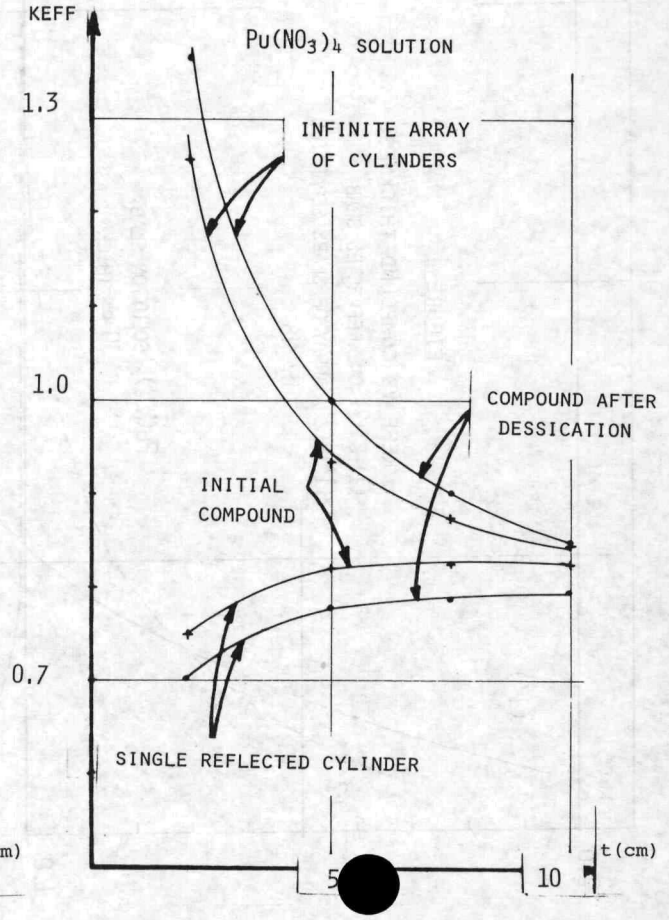

Keff
2.0
1.5
1.0
0.5
0.0
(1')
(2')
(2)
(1)
FIGURE 1
KEFF VS COMPOUND THICKNESS
(1) & (2) ONE REFLECTED SLAB
(1')&(2') ARRAY OF SLABS SEPARATED BY N8
$Pu(NO_3)_4$ SOLUTION SLAB, 10 cm THICK
^{235}U METAL SLAB, 1.5 cm THICK
N8 COMPOUND THICKNESS (cm)
0
10
20
30

FIGURE 2 : KEFF VS COMPOUND THICKNESS SURROUNDING EACH CYLINDER (T)





THE DOSE RATE SURVEY OF SPENT FUEL PACKAGES

P.T. BLUM and P. MORIN
TRANSNUCLEAIRE

S. IAZAREVITCH
NUCLEAR TRANSPORT LIMITED

1. INTRODUCTION

This paper concerns methods proposed for the routine or periodical surveys of packagings used for the transport of spent fuel assemblies, methods which are based upon radiological measurements and which can be implemented with minimum perturbation of normal operations.

These surveys concern mainly the efficiency of the flasks gamma and neutron shieldings but, as indicated hereafter, they may also be extended to their nuclear safety as they permit follow up of the condition of the built-in neutron poisons, provided to control their criticality.

Furthermore, we should recall that one single radiological measurement allows instantaneous checking that the power of the contents does not exceed the allowed limit.

2. CONTINUOUS GRAPHIC RECORDING OF DOSE RATES AROUND CASKS

The survey of dose rates in the vicinity of casks loaded with spent fuel is generaly performed by means of hand counters (ionisation chamber, Rem-meter), as for any packaging containing radioactive materials.

For large casks, this punctual survey may be replaced by a continuous recording of the counter outputs during handling, a method already experimented for the periodical control of shield efficiency.

Method Principle

This consists of moving the cask at even speed in front of fixed counters connected to a recorder.

Indeed, equivalent records may be obtained by moving the counters (at even speed) along or around a stationary cask in a vertical or horizontal position.

Fig. 1 represents a typical set-up experimented at different sites. The gamma monitor is a scintillation counter and the neutron one, a He^3 counter surrounded by a parafine moderator. The counter outputs are recorded on the paper which gives a continuous record of dose rates as a function of the position of counters with respect to the cask.

Above instrumentation may be adapted according to requirements:

- for a routine dose rate survey, it is sufficient to use standard counters (ionisation chamber and Rem-meter) which have just to be calibrated and installed at regulatory distance (2 meters) from cask surface.

- for a periodical control of shield integrity it seems preferable to use a gamma probe with adjustable energy threshold (scintillation counter) and to equip it with a collimator, both precautions to reduce the effect of scattered gamma radiations.

- this type of instrumentation will also be preferred for measurements used as bench marks for qualification of shielding calculation codes, or for the determination of the radioactive sources in a cask.

Exemples of recording and interpretation

Fig. 2 concerns the results of measurements performed on 25 april 1979, with a collimated gamma monitor, along a TN 8 cask loaded with 3 PWR fuel assemblies having following characteristics:

Active length	: 2985 mm	Burn-up	: 35000, 32000, 32000 MWd/tU
Uranium mass	: 338 kg		
U5 enrichment	: 3.18 %	Irradiation time	: 990 d
Av. specific power	: 38.5 MW/tU	Cooling time	: 390 d

As shown on Fig. 1, this cask has 3 fuel compartments forming a T, a lead gamma shield of variable thickness (150 to 185 mm), a steel outer shell (20 mm thick) and a neutron shield consisting of a 150 mm thick resin layer, crossed by copper cooling fins.

The gamma monitor (type SPP3 of SAPHYMO-STEL) was a scintillation counter (NaI-Tl) surrounded by a 10 cm thick collimator with a rectangular slot (3 x 50 mm) perpendicular to the direction of the movement.

This counter was connected through an integrator/energy discriminator to a potentiometric recorder.

The cask was moved at a speed equal to 0.5 m/mn and the maximum gamma dose rate at counter level was 22 mRem/h.

One can check that records are well centered with respect to the active part of the fuel assemblies and that irradiation decreases rapidely beyond the limits of this active part.

The effect of Cobalt impurities in the stainless steel parts (end pieces, plenum springs) is visible on records corresponding to energy thresholds lower than 1500 keV.

On Fig. 1, one can also see (after its transformation into polar co-ordinates) the record of gamma dose rates obtained on 7 june 1979 while rotating the TN 8 cask at 1 rpm speed. The monitor was bare and located in front of fuel assembly at mid height. The fuel burn-up was about 15 000 MWd/tU.

The irregularities of this graph ($\frac{\max}{\text{mini}} = 1.45$) seem due to the fact that calculated Build-up factors were merely approximative for such a complex geometry (shielding and radioactive source).

3. CHECKING OF EFFICIENCY OF CASK NEUTRON POISONS BY NEUTRON DOSE MEASUREMENTS

The neutron dose rates at proximity of a cask loaded with spent LWR fuel may be used to check its multiplication coefficient and therefore the integrity of the built-in neutron poisons that control its criticality.

The method consists of comparing the neutron dose rates at a given point at the external surface of the loaded cask, with or without water in its cavity.

Therefore it is especially interesting for the periodical control of casks of the dry type, as the measurements may be easily performed before and after their filling with water (required before their unloading in a pool).

For a periodic control, it is generaly sufficient to verify that the cask multiplication coefficient does not change significantly from one transport campaign to the next one from the same reactor. But, it is also possible to derive from neutron measurements the actual multiplication coefficient and to use this method to establish a Bench Mark for the control of the codes used for criticality calculations when the package characteristics are well defined.

Method principle <1>

The U238 contained in the low enrichment oxyde used as the fuel of LWR reactors, generates by irradiation neutron emitting isotopes such as Cm 242 and Cm 244.

The amount of these isotopes is sufficient, even after a single irradiation cycle (15000 MWd/tU) to require a transport cask provided with a neutron shield.

The neutron dose rate (M) at a given point P at cask surface depends upon the intensity (So) of this source. It also depends upon:

- the cask multiplication coefficient (k)
- the neutron transmission coefficient from the fuel to the point P

$$M = \frac{So}{1-k} \times T \qquad (1)$$

The value of the multiplication coefficient for the package filled with water is derived from 2 neutron dose rate measurements at a same point P:

- a first measurement (M_1) when the package cavity is dry (transport conditions)
- a second one (M_2) after its filling with pure, cold water (cooling before pool unloading).

The ratio between these measurements gives :

$$\frac{M_1}{M_2} = \frac{1-k_2}{1-k_1} \times \frac{T_1}{T_2} \qquad (2)$$

where subscripts 1 and 2 refer respectively to the package without and with water.

For a given package, the ratio T_1/T_2 is a constant which can be calculated accurately by neutron transport codes. Therefore Eq. (2) enables us to derive k_2 from k_1, using the formula:

$$k_2 = 1 - \frac{M_1}{M_2} \times \frac{T_2}{T_1} \times (1 - k_1) \qquad (3)$$

As for a package containing LWR fuel, k_1 is close to 0.20 (comprised between 0.15 and 0.30 according to the number (assemblies) and not significantly influenced by the conditions by built-in neutron poisons, one may assume, in first approximation that T_2/T_1 $(1-k_1)$ is a constant (A) for a given package type.

$$k_2 = 1-A \times \frac{M_1}{M_2}$$

Eq. (4) may be used to asses the multiplication coefficient of the package and therefore to check the integrity of its built-in neutron poisons.

Calculation of A

Eq. (3) indicates that k_2 is not significantly sensitive to variations of k_1 around its mean value.

Therefore, for an approximative calculation, one may assume that k_1 equal to 0.20 as this corresponds to a relative error $\frac{\Delta(1-k_1)}{1 - k_1}$ of 6.25 %

For more accuracy, it is always possible to calculate k_1 by code calculations of the MONTE-CARLO or Transport type, as indicated later.

The ratio of neutron transmission coefficients (T_1/T_2) essentially depends on:

- attenuation of fast neutrons by water in case 2
- differences between spectra and spectral distributions of neutron sources due to different levels of induced fission neutrons in the 2 cases.

On the other hand, this ratio is not affected by possible uncertainties in material compositions, neutron cross sections and detector reponse function.

This ratio may be accurately calculated, using for instance the DOT-III discrete-ordinates transport code <2>.

A sensitivity analysis indicated that the assumptions concerning spectra and spatial distribution of neutrons have no significant influence on calculation results. In particular, the assumption of a homogeneous neutron distribution and U 235 spectrum, does not cause more than 2 % error on the calculated value of A.

Using above simplified assumptions, one obtains for the TN 8 cask:

$\frac{T_2}{T_1}$ = 0.347 (0.354 for actual spectrum and neutron distribution)

and A = 0.278 (8.25 % precision)

Comparison between experimental and code calculation results

Neutron measurement at surface of TN 8 cask loaded with PWR fuel irradiated at 30 400 MWd/tU gave :

$$\frac{M_1}{M_2} = 1.26$$

Introducing this value M_1/M_2 and that of A previously calculated in Eq. 3, one obtains:

$$k_2 = 1 - 1.26 \times 0.278 = 0.65 \pm 0.03$$

The above result should be compared to that obtained using the fuel and cask characteristics to calculate the multiplication coefficient by means of usual codes.

The table 1 indicates the results using a MONTE-CARLO code (MORET <3>) and a 2 dimensional transport Code (DOT-III).

These code calculations used the HANSEN and ROACH cross sections for materials and assumed infinite length and homogeneous sources, the homogeneisation being performed by the transport code APOLLO <4>.

The conservation of this model (infinite length, absence of grids, perfect pin array, etc.) could explain the differences between experimental and code results.

The table 1 also indicates experimental multiplication coefficients (k_2) obtained:

1. Using the results of the simplified calculation of A, previously exposed (arbitrary value of k_1, transmission coefficients calculated with simplified assumptions).

2. Using more accurate values for k_1 and T_2 and T_1 obtained by code DOT-III ($k_1 = 0.159$, $T_2/T_1 = 0.354$).

Medium in cask cavity	Code MORET	Code DOT-III	Neutron Measurements	
			1	2
water (k_2)	0.674 ± 0,07	0.6465	0.65 ± 0,03	0.62
air (k_1)	0.159 ± 0,05	0.159	-	-

Table 1 - TN 8 Package - Multiplication coefficient when loaded with 3 PWR fuel assemblies irradiated at 30 400 MWd/tU

4. CHECKING OF POWER GENERATED BY CASK CONTENTS

The transport regulation for radioactive materials requires a checking of any package before its expedition.

For that reason casks are often provided with means to measure their temperature and their operation procedure provides for long immobilisation before departure.

In fact, to verify that a spent fuel cask will remain at acceptable temperature during transport, one just has to assure that its power does not exceed the allowed limit.

Now, the fission product activities and fuel power are both monoton functions that decrease with longer cooling time and/or shorter irradiation time.

In practice, to assure that package temperature will remain acceptable during transport, one has just to check that gamma dose rate at a given point at cask surface (preferably at mid length) does not exceed a given limit which may be calculated or preferably derived from the measured gamma dose rate at that point corresponding to fuel assemblies of known characteristics.

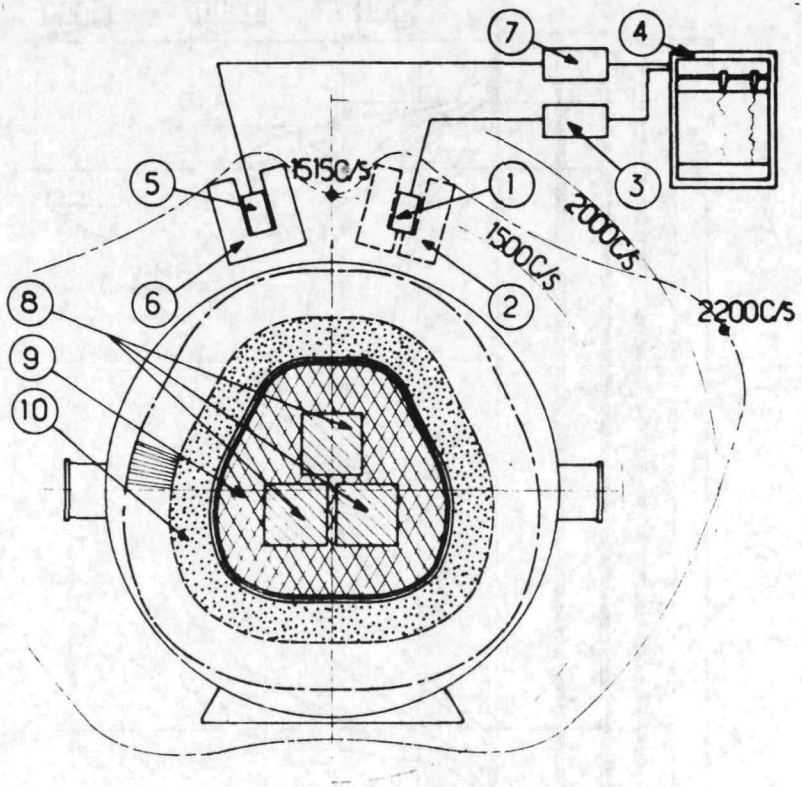
5. REFERENCES

<1> C. DEVILLERS and P. BLUM: Neutron Multiplication and Shielding Problems in Pressurized Water Reactor Spent Fuel Shipping Casks Nuclear Technology. Vol. 35 Mid. August 1977.

<2> F.R. MYNATT, W.W. ENGLE, M.L. GRITZNER, W.A. RHOADES, and R.J. RODGERS, "The DOT III Two-Dimensional Discrete Ordinates Transport Code, ORNL-TM-4280, Oak Ridge National Laboratory (1973).

<3> J. MORET-BAILLY, G. POULLOT and J.R. TEILLET : A Monte Carlo Program for Rapis Calculation of Effective Multiplication Coefficients of Fissiles Mediums, Note CEA R-4752 (1976) (MORET Computer Program).

<4> A. HOFMANN and F. JEANPIERRE : Multigroup Code for Resolution of the Transport Equation for Fast and Thermal Neutrons, Note CEA N-1610 (APOLLO Computer Program).



1. Gamma scintillator
2. Gamma collimator (optional)
3. Discriminator
4. Recorder
5. He3 Counter
6. Neutron moderator
7. Amplifier
8. Fuel lodgements
9. Gamma shield
10. Neutron shield

Fig. 1 - Set up for dose rate record and results obtained around the TN8 cask

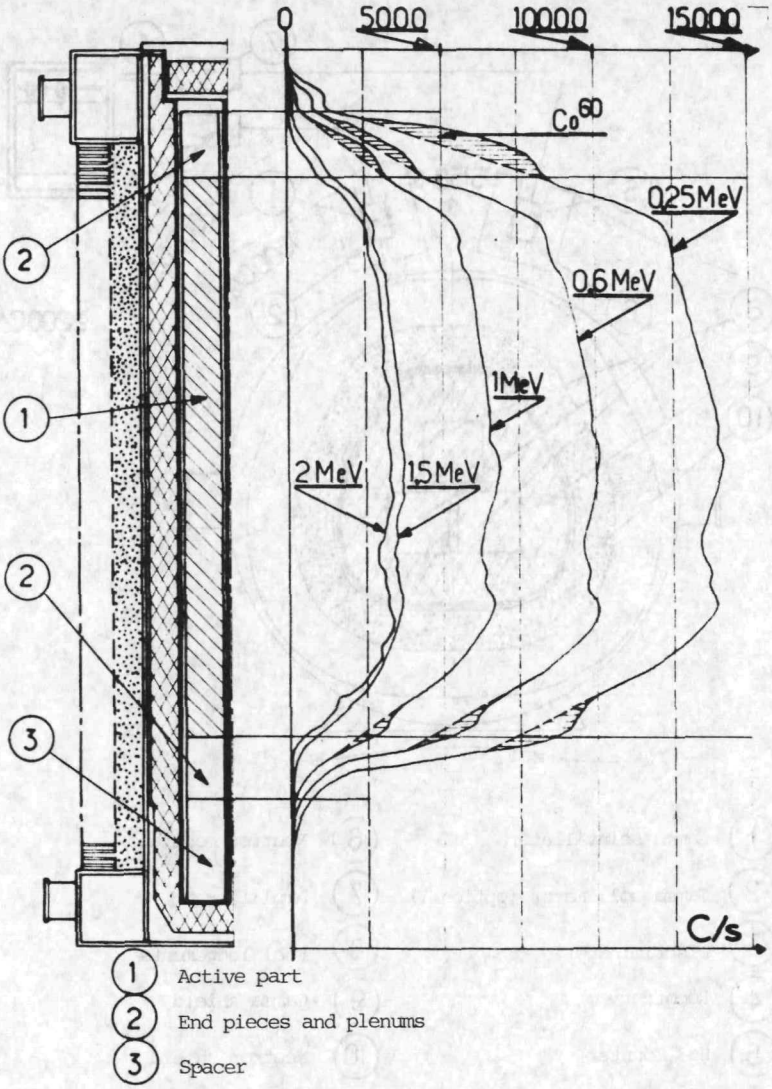


Fig. 2 - Regards of gamma irradiation along the TN8 cask

DISCUSSION

Question by Mr. Ottati, Italy: I would like to know if you have performed γ and neutron dose rate measurements along the axis of the container, and what the results are compared to the calculations?

Answer: The dose rates that we measured in axial direction at extremities of containers generally differ significantly from that predicted by code calculations. This may be due to the simplifications of source model used in these calculations.

Question by H.-H. Schweer, FR Germany: Do you think that your method is exact enough to test criticality calculations?

Answer: This method may be used to test criticality calculations as well as more classical critical experiments which also generally require corrections. A potential problem concerns the difficulty of introducing, in calculations, all actual conditions which influence test results (actual geometry, crud deposit on fuel rods ...).

SESSION XIV
PLUTONIUM SYSTEMS

AIR TRANSPORT PACKAGE FOR PLUTONIUM SAFEGUARDS SAMPLES
SAND80-1735

John A. Andersen
Sandia National Laboratories, Albuquerque, NM, USA

BACKGROUND AND CURRENT STATUS

Radioactive materials safeguards samples, particularly plutonium and enriched uranium, are obtained in numerous countries that participate in the Non-Proliferation Treaty (NPT) and are then transported to an IAEA (International Atomic Energy Agency) or other safeguards analytical laboratory. Air transport is necessary to meet the objective of timeliness of detection (of a safeguards problem), especially for over-water routes and long-distance routes, and also to enhance physical protection and security in transit.

Legislative and regulatory events in the United States, commencing late in 1975, resulted in very stringent crashworthiness requirements being placed on packages for the air transport of plutonium. The PAT-1 package, for up to 2 kg of uranium oxide or plutonium oxide in any isotopic composition, limited to 25 watts thermal activity, was developed to meet the criteria of NUREG-0360 and was licensed by the US NRC in 1978.

The U.S. rulings have also affected the operational policies in other countries, resulting in the consideration of the PAT-1 package for the international movement of safeguards samples.

The certificate of compliance (license) for the PAT-1 included a stipulation that conflicted, at the time of issuance, with another U.S. regulation concerning aircrew access to hazardous materials; and, the PAT-1 license included specific limitations on contents, which affected the accommodation of a grouping of individually-packaged safeguards samples. The regulatory problems have recently been resolved, with a revision to 49 CFR (aircrew access to radioactive material is deleted) and by a revision to the PAT-1 certificate of compliance (safeguards samples, including traditional packaging methods, are accommodated). However, the PAT-1 is licensed only for the oxides of uranium and/or plutonium; the PAT-1 SAR (Safety Analysis Report, NUREG-0361) stipulates the use of a welded can inside the containment vessel for radioactive material exceeding 20 curies; and, the PAT-1 license restricts the on-board location of the package on an aircraft to an aft position. These regulations, in conjunction with the relatively large size (62 cm diameter, 108 cm length, 227 kg mass) of the PAT-1 package for safeguards samples (which are typically a few grams

This work sponsored by the DOE Office of Safeguards and Security, International Support Branch.

each), caused an interest in the development of a smaller, lightweight safeguards sample shipping package that would satisfy the same stringent requirements that the PAT-1 package survives, and that would have fewer regulatory limitations on usage.

The lightweight air-transportable accident resistant container (LAARC), to be designated the PAT-2 package (plutonium air transportable model 2), is now in development to better meet specific U.S. and IAEA needs for the air shipment of plutonium samples under very stringent accident-modelling conditions.

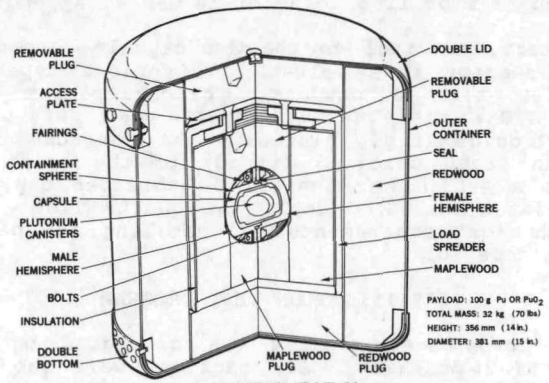
PACKAGE CONTENTS AND DESIGN

The LAARC package has been designed to meet the stringent requirements of NUREG-0360, to accommodate approximately 40 grams of mixed oxide (PuO_2/UO_2) powders, or approximately 100 grams of fuel pellets containing plutonium and/or uranium, or small quantities of anhydrous salts of plutonium. A typical safeguards payload would be less than 8 grams of plutonium, in various chemical and physical forms and in several individual internal packagings. LAARC is sufficiently minimized in size and weight to be accepted by small-package air freight services, as well as to be one-man portable when necessary.

The essence of the LAARC design problem was to provide a virtually unassailable sealed pressure vessel in the NUREG-0360 sequential test environment: a high-speed impact (129 m/s) perpendicular to an unyielding target (an environment comparable to a supersonic collision with hard earth); followed by a 32-tonne crush test; followed by a puncture test (3 m drop of a 227 kg steel probe); followed by a double slash test consisting of two 46 m guided drops of a 45 kg steel angle beam onto two locations on the package (the package being propped up at a 45° angle to enhance the damage potential, and rotated between drops of the steel beam, to make two holes in the package); followed by a one-hour (minimum) JP-4 aviation jet fuel pool fire that extends from 1 to 3 m beyond the package in every direction (the fire attains a minimum temperature of 1010°C); followed by underwater submersion. Each test is directed at the weak point resulting from the previous test(s), all on the same package. Also, the requirements of 10 CFR 71 (less severe tests, but, adds a requirement for double containment and defines post-accident shielding and criticality requirements) and 49 CFR 173 (defines normal transport shielding requirements) must be satisfied. These requirements were then synthesized with the design challenge to keep the resulting package reasonably small and reasonably simple to use while still retaining a payload space that is useful for various chemical and physical forms of plutonium, all of which must meet various State (national) requirements of health physics protection in glove-box bagouts and initial packaging.

"LAARC" is the development project that resulted in the PAT-2 (Plutonium Air Transportable Model 2) Package (see cutaway illustration, Fig. 1). The total package mass is 31 kg (70 lbs); the

height is 356 mm (14 inches); and the diameter is 381 mm (15 inches, not including the protruding handles). The PAT-2 package comprises the AQ-2 (Air Qualified Model 2) overpack, the TB-2 spherical containment vessel, the C-1 capsule, and various brass (or other metal) canisters.



L A A R C (PAT-2)
Lightweight Air-Transportable Accident Resistant Container
(Plutonium Air Transportable Model 2)

FIGURE 1

The AQ-2 overpack consists of a double-walled outer drum of 0.9 mm (0.035 inch) 304 stainless steel sheet metal, an outer region of 91 mm(3.6 inch) thick grain-oriented redwood within the drum, a 6.35 mm (0.25 inch) thick titanium load spreader (190 mm in diameter) with ceramic felt backface insulation, and an inner region of grain-oriented maplewood, 40 mm(1.6 inch) thick. Redwood was chosen for high specific energy absorption on a volumetric and mass basis and for good char performance (fire protection). Maplewood was chosen for high ultimate crush strength within the smaller region of the load spreader. Titanium was chosen for its high strength-weight ratio and for its high liquidus temperature.

The TB-2 spherical containment vessel, Figure 2, is made of A-286 ironbase superalloy and is bolted shut with twenty bolts, also made of A-286. The bolts are installed to a specified torque and the vessel is sealed by a copper gasket formed in place between knife-edge sealing beads which are in each hemisphere. An integral shear shoulder is included in the joint. This is a very

tough sealed high-temperature and high-pressure containment. The TB-2 remains leaktight in all normal environments and has proven to be leaktight after the severe accident-modelling tests of NUREG-0360.

The C-1 capsule is fabricated from Nitronic 60 stainless steel and utilizes a torqued screw-thread joint with a sealant. The C-1 capsule satisfies the double containment requirement (for RAM > 20 Ci) of 10 CFR 71, remaining sealed in all normal environments and after the accident-modelling tests of 10 CFR 71 Appendix B.

The canisters, internal to the C-1 capsule, accommodate various sizes of reactor fuel pellets and various types of uranium and/or plutonium salts and powders. The canisters may be equipped with quartz (SiO_2) vials or liners so that safeguards samples extracted as liquids (e.g., plutonium nitrate) can be evaporated to dryness, and then calcined (fired) to the oxide state. The canisters have a PVC gasket inside the threaded cap; this gasket seals the canister and also seals the quartz liner, if used, to meet health physics requirements for handling and packaging into the sealed C-1 capsule.

CERTIFICATION TEST PROGRAM

A certification test program was conducted on a group of final-design PAT-2 packages. Six packages were subjected to the sequential qualification criteria for air transport of plutonium in the U.S., NUREG-0306. These packages, listed in Table 1, were assembled in a helium atmosphere, with various surrogate uranium payloads representing the entire spectrum of plutonium safeguards samples that would be utilized by the IAEA. The attitude (top, bottom, side, etc.) of each package was varied, and the high (104°C; 220°F) and low (-40°C; -40°F) temperature conditions were tested, in the first sequential test, which is the high-speed rocket-pulldown impact perpendicular to an unyielding target. These conditions are indicated in Table 1. Following the impact test, the sequential tests of crush, puncture, double slash, fire, and underwater submersion were performed in a similar manner on all of the packages. Figure 3 indicates the condition of five of the packages following impact, crush, puncture, and double slash. The burn or fire test was conducted using JP-4 aviation jet fuel for one hour at 1010°C (1850°F) or more, in a newly-developed burn stand (figure 4). Following the burn test, the packages were permitted to char and reach ambient temperature, and were then submerged underwater (>1 m) for 8 hours or longer. The packages were then disassembled (figure 5) and the TB-2 spherical containment vessel was recovered, examined, cleaned, and leak-rate-tested in a helium-type mass spectrometer. Five of the containment vessels exhibited no detectable leak, with a test sensitivity of 1×10^{-10} cm^3/s; one TB-2 exhibited a leak in the 10^{-6} cm^3/s range. Then, the containment vessels were opened (unbolted and disassembled) in the presence of a helium detector; strong indications of helium were detected.

One package was subjected to the normal and accident-modelling conditions of 10 CFR 71 Appendix A and B (which is also similar to the package test requirements of IAEA Safety Series No. 6). Following these tests the TB-2 containment vessel was leaktight ($< 1 \times 10^{-10}$ cm^3/s) and the C-1 capsule, which is required to provide secondary containment, in the U.S., for contents > 20 curies, was found to be sealed.

Individual (non-sequential) test or analysis requirements of minal velocity and hydrostatic pressure (deep underwater), uired by NUREG-0360, were also passed.

CONFIRMING ANALYSES

Both test and analysis are applied to confirm that the PAT-2 package meets the various statutory and regulatory requirements. Structural analyses address all of the stated requirements, and additionally treat the TB-2 containment vessel by finite element analysis, and the AQ-2 overpack by an energy balance analysis.

Thermal analyses treat maximum normal operating temperature, with thermally-active radioisotopic contents, and maximum credible accident temperature of the containment.

Plutonium containment analysis treats the problem of plutonium transport through very small apertures or cracks, for various conditions including those measured on the TB-2 containment vessel and the C-1 capsule. Plutonium containment is confirmed to be orders of magnitude in excess of IAEA requirements (essentially, there is no plutonium release).

Nuclear criticality analysis will confirm through the use of benchmarked codes, that the PAT-2 package qualifies as a Fissile Class I package, meaning that infinite arrays of damaged packages, with the most fissionable contents permitted, and with water shielding, remain subcritical.

Nuclear shielding analyses confirm the required protection for biota from the effects of ionizing radiation, in both the normal and post-accident conditions, conservatively satisfying all regulatory conditions. All of these analyses will be presented in the PAT-2 Safety Analysis Report (SAR).

CONCLUSION

It is concluded, by the developer, that the LAARC/PAT-2 will et the requirements of the various applicable regulations and that a Safety Analysis Report (SAR) will be issued. This SAR will be the basis for an application to the US Nuclear Regulatory Commission for a certificate of compliance, or license. It is then expected that this US license, with the supportive evidence of the Safety Analysis Report, will be the basis for the issuance of licenses needed from the IAEA Competent Authority in using States (nations). Then, timely air shipment of plutonium safeguards samples will be feasible on an international basis.

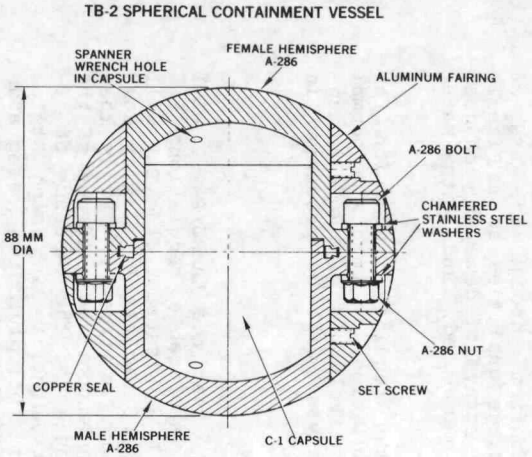


FIGURE 2

FIGURE 3

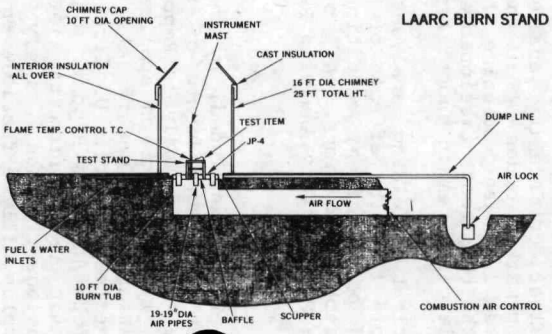


FIGURE 4

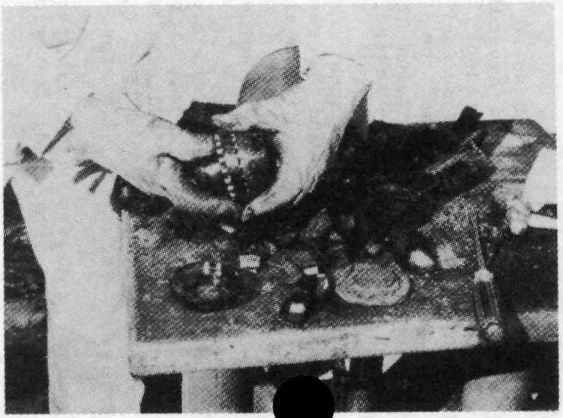
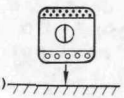


FIGURE 5

LAARC/PAT-2 ϕ4 TEST PROGRAM

X07 — AQ-2: TOP — TB-2: EQUATOR

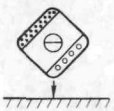
C-1 3 ea. BC-4 CANISTERS
3 gm. Pu $(NO_3)_4 \cdot 2H_2O$ (SURROGATE)



X08 — AQ-2: TOP CORNER — TB-2: POLE

C-1 2 ea. BC-1 CANISTERS
50 gm. PuO_2 (SURROGATE)

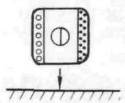
[APPROX. 50 g UO_2, 10 g PVC, 38.6 g BRASS
16 g SiO_2, 0.5 g H_2O]



X09 — AQ-2: SIDE-COLD — TB-2: EQUATOR

C-1 6 ea. BC-5 CANISTERS
120 gm. PuO_2 PELLETS (SURROGATE)

[123.6 g D-38, 10 g PVC, 31.4 g BRASS]

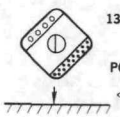


50°F
126 m/s (413 fps)
POST-TEST $\dot{q}$: $< 10^{-10}$ cm^3/s

X10 — AQ-2: BOTTOM CORNER — TB-2: EQUATOR

C-1 2 ea BC-2 & 1 ea. BC-3 CANISTERS
3 gm. Pu $(SO_4)_2 \cdot 4H_2O$ (SURROGATE)

[3 g UO_2, 10 g PVC, 47.4 g BRASS, 19 g SiO_2,
0.66 mℓ H_2SO_4, 0.66 mℓ H_2O]

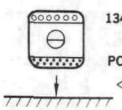


133 m/s (435 fps)
POST-TEST $\dot{q}$: $< 10^{-10}$ cm^3/s

X11 — AQ-2: BOTTOM — TB-2: POLE

C-1 NO CANISTERS
3 gm. Pu $(SO_4)_2 \cdot 4H_2O$ (SURROGATE)

[3 g UO_2, 10 g PVC, 50 g BRASS, 18 g SiO_2,
0.66 mℓ H_2SO_4, 0.66 mℓ H_2O]

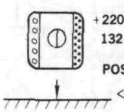


134 m/s (438 fps)
POST-TEST $\dot{q}$: $< 10^{-10}$ cm^3/s

X12 — AQ-2: SIDE-HOT (200/220° F)

C-1 6 ea. BC-5 CANISTERS
120 gm. PuO_2 PELLETS (SURROGATE)

[123.6 g D-38, 10 g PVC, 31.4 g BRASS]



+220°F
132 m/s (433 fps)
POST-TEST $\dot{q}$: $< 10^{-10}$ cm^3/s

X15 — 10 CFR 71 APP A & APP B

C-1 2 ea. BC-1 CANISTERS
3 gm. $Pu(SO_4)_2 \cdot 4H_2O$ (SURROGATE)

[3 g UO_2, 10 g PVC, 39 g BRASS, 18 g SiO_2, 0.66 mℓ 96% H_2SO_4, 0.66 mℓ H_2O]

TABLE I

DISCUSSION

Question by R. Brissier, France: Why are all the PAT-2 approval tests so rigorous? Do these tests correspond to actual conditions? What are they based on? In other words, don't you think you are being too severe in the approval tests for transport casks for shipment of plutonium by air?

Answer: PAT-2 package tests are defined by regulation in the United States. This regulation is NUREG-0360, qualification criteria to certify a package for air transport of plutonium. NUREG-0360 is worst case criteria, combining very severe levels for each individual element of an air transport accident, all in sequence on the same package, even if such a combination is very unlikely, if not practically impossible, in a real accident. The document includes the rationale used by the US Nuclear Regulatory Commission in defining the criteria.

Question by W. Heudorfer, FR Germany: Can the cask concept presented be used for larger containers? In this case, would there be scaling problems?

Answer: In regard to scaling the cask concept of LAARC/PAT-1 to larger containers, we have had success in scaling the energy balance design for impact protection alone, and we have had success in applying the redwood char and temperature performance data to larger casks, for fire protection alone. We have had problems in scaling when all tests are done in sequence including the very arbitrary puncture and slash tests of NUREG-0360.

However, we believe that these effects are mitigated, not worsened, on larger casks, our current applicable experience and problems have come from scaling down from the PARC-PAT-1 design, to the smaller LAARC/PAT-2 design. Our older work with larger accident-resistant containers has been to different criteria which were more amenable to scaling than the NUREG-0360 criteria.

Question by G. Cohendy, France: I would like to ask you if you think that this marvellous package corresponds in practice to the interdiction of the transport by air of significant industrial quantities of plutonium oxide.

Answer: The NUREG-0360 criteria for the air Transport of plutonium only apply within the USA, and then only apply to air transport. IAEA Safety Series No. 6 requirements still apply for other international air shipments, unless the originating nation or state desires to emulate the US regulation. Thoseof us associated with this new work are sympathetic to the possible affect on the air shipment of commercial or bulky quantities of plutonium; but, we have no choice but to conform to the US government's authority in these matters - the US NRC.

ANALYSIS OF THE ACCIDENT ENVIRONMENT AND CONTAINER RESPONSE FOR THE AIR TRANSPORT OF SMALL TYPE B PACKAGES

M L BROWN
UNITED KINGDOM ATOMIC ENERGY AUTHORITY
Safety and Reliability Directorate
Wigshaw Lane
Culcheth
Warrington WA3 4NE
United Kingdom

SUMMARY

Potential accidents in the transport by airplane of small ($\lesssim$500kg) type B packages are considered, with plutonium carrying containers especially in mind. Impact and fire are the major threats. Probabilities are derived for encountering an impact of above any given speed, and a fire of above any given duration. The crash of typical jet cargo aircraft against a rigid normal surface is considered, and cargo hold decelerations are derived from a 1 dimensional model. The response of a cargo load of typical containers to such decelerations is calculated and related to the velocity of impact onto a hard target necessary to produce similar damage in single containers. Free fall of containers and the effect of surface struck are briefly discussed. Cumulative probabilities of exceeding an equivalent hard target impact velocity are derived. The consequences of plutonium releases are estimated and the implications for container testing and safety assessment discussed.

1. INTRODUCTION

The transport of radioactive materials has an extremely good safety record. However, it has been suggested that for some substances, notably plutonium, containers built to satisfy existing standards may not give adequate protection in the very severe environment of an aeroplane crash. US law in particular has required the development of a more rigorously qualified container for the air transport of plutonium (1).

Within Europe recycling thermal reactor fuel could give a significant economic advantage. It has been estimated (2) that for this reason up to 15te of plutonium per year might be transported by 1990; air transport is a possible mode. The work described here was performed against this background, under contract to the Commission of the European Communities, as part of their programme to assess the environmental impact of light water reactors.

Past work is not reviewed here because of lack of space. We note however that an American risk study has been performed (3), that qualification criteria have been discussed (4) and that many basic data used there and elsewhere come from an extensive and thorough study of the accident environment for US conditions (5).

Here we concentrate on defining the accident environment and indicating typical container responses. We estimate accident consequences in order to be able to make some judgements on test criteria and safety. The work is described in more detail in reference (6). Impact and fire are the primary threats to package integrity.

2. AIRCRAFT IMPACT STATISTICS

We have surveyed (6) recent civilian aircraft accidents involving modern jet cargo aeroplanes. Most information came from the World Airline Accident Summary of the UK Civil Aviation Authority, and the International Civil Aviation Organisation's computerised accident data base. Information on impact angle, attitude, and fire duration (if any) is sparse. Impact speed information is available in some cases mostly for near airport accidents. It is known however, in what phase of operation an accident occurred. We have used this to construct more comprehensive speed distributions. For approach accidents we have inferred a normalised velocity distribution characteristic of this phase of flight, using data directly. For landing and take off we have assumed reasonable distributions consistent with the more limited data. For climb-to-cruise and in-flight phases direct data are almost entirely lacking. We have relied on an analysis of military accident data published by Clarke et al (4). Finally we arrive at distributions of cumulative probability (given an accident) vs impact speed as shown in figure 1 denormalised for two aircraft, a BAC-111 and a B-707.

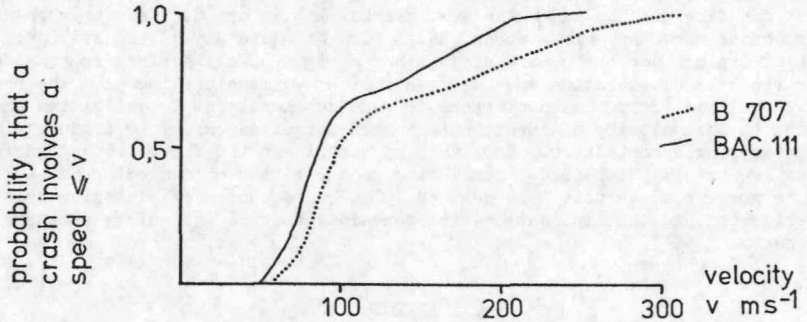


Figure 1. Cumulative Probability Curves for Impact Speed

The definition of an accident associated with these distributions is one destroying the airframe and killing at least 1 person. The associated rate for Western Europe is about 1.3×10^{-6} per flight (The in-flight fraction of accidents is small, ≈ 17%, so it is more reasonable to express the probabilities "per flight" than "per km").

3. AIRFRAME COLLAPSE

Airframe collapse is extremely complex. We present here the results of a 1 dimensional analysis using simplified computer code modelling (6) similar to that of Drittler and Gruner (7) (The method has been validated against experimental test data on simple missiles). Impact on a rigid surface normal to the aircraft is assumed. This is artificial but of some interest since it is the attitude in which cargo interaction effects are likely to be greatest. Figure 2 illustrates typical results; the velocity of the forward cargo bulkhead is plotted as a

function of time for the two typical aircraft chosen in section 1.

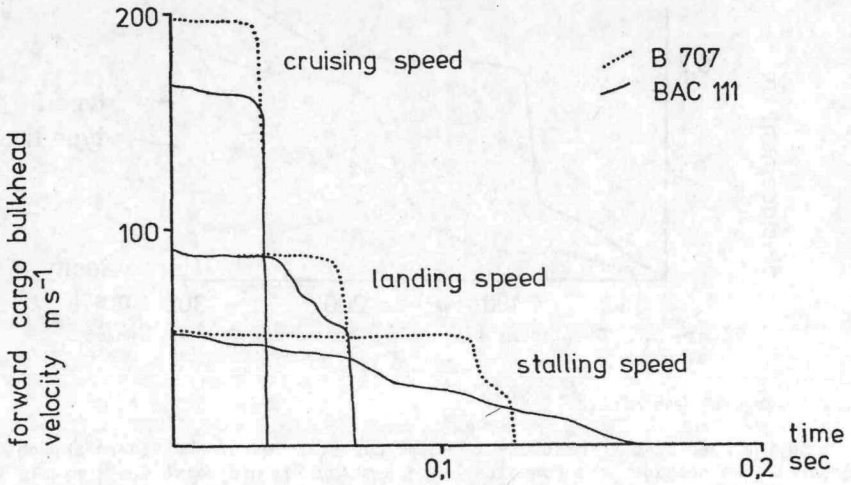


Figure 2. Cargo Bulkhead Velocity Versus Time

The additional weight of the larger aircraft is important at lower velocities - sudden decelerations occur for the B-707 even in a stall speed crash. At cruising speed in both cases the airframe has only a slight influence on the velocity of impact of the cargo bulkhead.

4. THE CARGO IN AN AIRCRAFT ACCIDENT

For container testing it is convenient to use a hard rigid target and to test a single container. But to relate this to safety in a real accident the container damage under test conditions must be related to damage under real conditions. As an aid in this, we have defined two simplified, but structurally typical containers: a type I which resembles an SL-10; and a type II which resembles a PAT-1. (Both these are type B fissile material containers). We consider the behaviour of a single container on a rigid target, and a full cargo load in the aircraft of section 3.

4.1 Single Container

Both typical containers consist of an outer vessel, insulant/shock absorber, and an inner vessel. By a combination of finite element analysis and knowledge of insulant crush strengths we may construct a force/displacement relation for container crushing (6). Integration gives (from initial energy) a relation between displacement (damage) and velocity of impact on an unyielding target. Typical curves are shown in figure 3 for side on impact.

Although the model is simple there is good agreement between predicted and measured (1) deformations for a PAT-1 container. For side impact at 132 ms^{-1}: predicted - 0.196m; measured 0.21m. For end impact at 135 and 142ms^{-1}: predicted 0.178 and 0.203m; measured 0.188 and 0.208m.

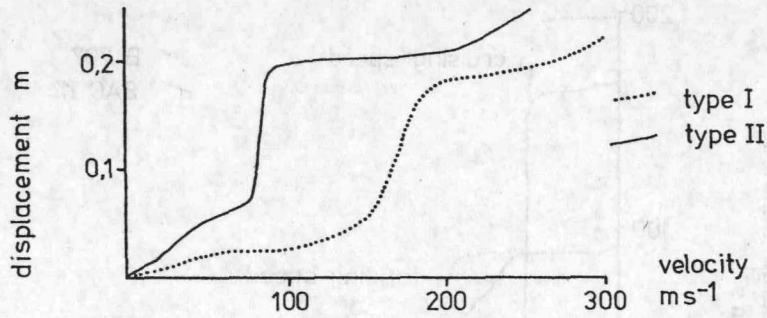


Figure 3. Displacement-velocity Curves for Side on Impact

4.2 A Cargo of Containers

Having specified a force displacement curve, a row of containers may be modelled. We consider one container as a mass/spring and solve the system numerically. The cargo hold decelerations of section 3 may be used as a boundary condition. Typical results of such a calculation are shown in Table 1. This is for a type I container in side on orientation, stowed to the full capacity of each aircraft - 14 in line for a BAC111, 32 in line for a B707.

Table 1. Cargo Deformations

Aeroplane	Velocity ms^{-1}	Approx. Deformation	Number so Damaged	Equivalent Hard Target Velocity ms^{-1}
BAC-111	50	.015	3	20
"	90	.05	1	138
"	120	.07	1	154
B-707	54	.192	8	200
"	90	extreme	several	310

The last column compares the velocity necessary to produce the same damage in impact on a hard target for the most damaged container. The differences are wide. For the lighter aircraft, cushioning by the airframe outweighs inertial crush at stall speed and the hard target velocity is lower. At higher speeds inertial crush predominates. For the larger aircraft cushioning is less and the payload greater so inertial crush always aggravates the damage.

5. FREE FALL OF CONTAINERS

Indications are that cargo carried on the main cargo deck of an aircraft only rarely leaves the airframe before impact in an accident (6). Nevertheless we have evaluated impact velocity for a number of containers in use, released at different heights. This is shown in figure 4.

Variations in air density and local Mach number with altitude are allowed for. All the containers are considered in the fastest falling (end on) attitude,

except for the 0675 where a tumbling attitude is also shown. The theoretically stable attitude for some of these containers is side on but there is some evidence that tumbling occurs in practice (8).

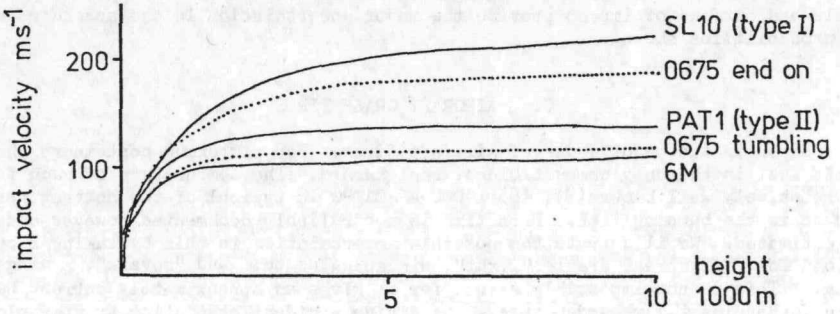


Figure 4. Free Fall Impact Velocity for Various Containers

6. EQUIVALENT DAMAGE PROBABILITIES

Figure 1 gives impact speed probabilities only; to assess equivalent damage allowance must be made for angle of impact and surface struck. There are inadequate data on angle of impact. From accident reports however it is fairly closely (but not uniquely) related to the degree of control exercised just prior to impact. Take-off and landing accidents generally involve shallower impact angles; and these also occur at lower velocities. We relate impact angle to velocity in an arbitrary way that we believe reasonable (6), associating high impact angles with a greater proportion of high speed impacts. For high impact angles we associate cargo and airframe crush effects with the accident. To allow for hardness of surface we review (6) limited data on enhanced severity container testing and deduce that a factor of between 2.5 and 7 in velocity is needed to relate states of equal damage in going from hard to soft surfaces. We choose 3 here, and estimate the probability of striking a hard surface from geographical considerations (6). Finally we adjust the distributions of figure 1 as outlined above for angle and surface of impact, airframe and cargo crush, and arrive at figure 5 for the probability of exceeding an "equivalent hard target" velocity, given an accident.

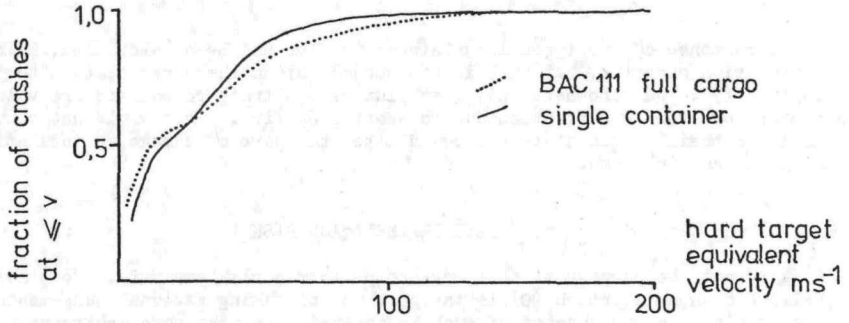


Figure 5. Cumulative Probabilities for Equivalent Damage

The 2 curves shown are for a BAC-111 with full cargo (type II) and for a single container with allowance for angle and surface of impact but no allowance for airframe or cargo crush. The airframe has an effect at low velocities and cargo inertia at high velocities but the difference between the curves is not great. This is important when considering operational restrictions. Assumptions on angle and surface of impact provide the major uncertainties in the absolute values of probabilities shown.

7. AIRCRAFT CRASH FIRES

Crash is the major cause of aircraft fires. For plutonium containers the aircraft fuel is the only combustible of real hazard. The temperature of such fires is relatively well known (5), (6) (1000 - 1010°C is typical of the hottest part) and so is the burning rate. Burn time is not reliably documented however and must be estimated. We illustrate the possible uncertainties in this by taking 2 assumptions for fuel spread area: "light", wingspan×length; and "severe", $\frac{1}{2}$ wing area. (This is not impossibly severe for it gives an approximately correct burn time for at least 1 recorded case). We divide accident statistics by operation and associate appropriate fuel loads. We derive the dashed curves in figure 6. These are for a weighted average of jets in service. To illustrate variation by aircraft type we show spot values for the most severe fire in a BAC-111 and a B-747. As a reference curve we propose the solid line in figure 6. This is arbitrary but it has the features of medium severity with some allowance for extreme fires at low probability.

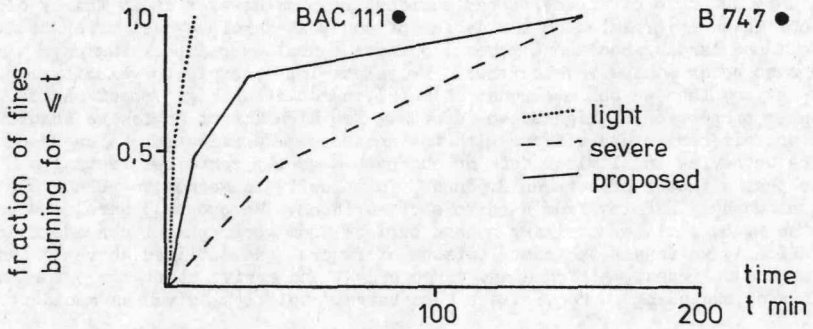


Figure 6. Cumulative Probability for Fire Duration, Given an Accident Involving Fire

The response of our typical containers to fire has been calculated, allowing for insulation charring, but that is the subject of another presentation to this conference (9). We note here only that plutonium nitrate containers are vulnerable to rupture by overpressurisation due to heating by fire. This would not occur normally in their virgin state but could after the sort of damage to insulation possible in an air crash.

8. TEST CRITERIA AND RISK

It is not the purpose of this work to perform a risk analysis. Nonetheless, we believe such an approach (10) is the only way of making rational judgements on test criteria. In the absence of such an analysis, we make some arbitrary but reasonable assumptions to construct approximate risk spectra. This is done to

illustrate our general conclusions only and no exact use of our figures should be made.

We assume 15te per year of plutonium shipped as dioxide powder. Taking a crude 2 group population density division, a 2 group weather conditions division, and "threshold" assumptions for container failure with a fractional release as respirable dust, we arrive at the risk spectra of figure 7.

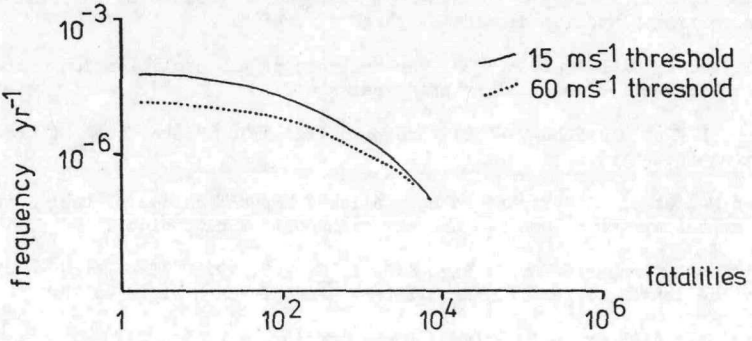


Figure 7. Approximate Risk Spectra

The two curves shown are for two failure thresholds in hard target equivalent velocities. They do not differ greatly; particularly in the low probability/high consequence region where containers qualified to either threshold are assumed effectively failed. This is not realistic since in general containers will fail progressively and only gradually release more material as accident severities increase. This will yield risk spectra substantially below those of figure 7. We conclude therefore that the failure threshold is of less importance for the overall risk than the container's ability to restrict releases in the failed condition. Current designs are likely to have this ability but it will not be tested by current regulatory procedures. We feel any change to international regulations would be premature; more work is needed on release and consequences in extreme accidents before rational decisions on changes could be made.

However, in the meanwhile it is obviously useful for designers to have a performance target in mind. For this purpose we think for fire a 30 minute hydrocarbon fire is appropriate; but that slashing and compaction of insulation must be considered. For impact we suggest 100 ms^{-1} impact onto a hard target. But we emphasise that a satisfactory performance under these conditions would be a limited release of say 1% of payload. This is of course a current opinion and could be modified by further work.

8. ACKNOWLEDGEMENTS

The author wishes to thank the many SRD staff who performed this work and the Commission of the European Communities who paid for it.

10. REFERENCES

1. Andersen J A et al NUREG/CR-0030; SAND 76-0587, 1978. "PARC (Plutonium Accident Resistant Container) Program research design and development".

2. Haijtink B EUR 5936 EN, 1977. "Prospects for plutonium recycling in light water reactors in the European Communities.

3. McSweeney T I, Johnson J F BNWL-2030 1977 "An Assessment of the risk of transporting plutonium dioxide by cargo aircraft".

4. McClure J D SAND-76-0708 1977 "An analysis of the qualification criteria for small radioactive material shipping packages".

5. Clarke R K et al SAND-74-0001 (3rd printing) 1978 "Severities of transportation accidents".

6. Brown M L et al CEC report to be published "Specification of test criteria for containers to be used in the air transport of plutonium".

7. Drittler K, Gruner P Nucl. Engg & Des. 37, 245, 1976 "The force resulting from the impact of fast flying military aircraft on a rigid wall".

8. Cohen L R AERE Harwell Private communication.

9. Young R L D this conference "Transport fire hazards and the calculation of package response".

10. Kinchin G H Proc. Inst. Civ. Engrs 64 431 1978 "An assessment of hazards in engineering work".

DISCUSSION

Question by W. K. Müller, FR Germany: When it comes to finding a solution to pressure failure in the inner container during transport of plutonium nitrate, how much breathable aerosol do you suppose is released?

Answer: Our study addressed the accident conditions and container response rather than the releases and consequences. I showed an approximate risk spectrum only to illustrate my point that testing to a "threshold" (with no leakage) only gives you limited information - you can envelope the "real" risk spectrum but you cannot get close to it.

In generating that approximate risk spectrum we considered only plutonium as dioxide powder, not nitrate. Our release assumption was 1 % of payload as airborne powder when the container "threshold" impact was exceeded. I imagine nitrate releases could be high if driven by a fire. This could overpressurise the inner container (left vulnerable by the impact) and spray out aerosol.

Question by R. Brissier, France: What is the corrective factor between the speed on a rigid target and a natural target?

Answer: It is difficult to generalise. Container damage is a function of both container and target; a flimsy container will see most surfaces as hard. Our studies indicate that a corrective factor of between 3 and 7 (on velocity) is appropriate in going from a normal, untilled, soil surface to a rigid target.

Question by R. S. Butterfield, United Kingdom: Could you please give your views on the intensity and duration of a severe fire on a container after a severe impact. I would expect a severe impact to cause great dispersion of the aircraft wreckage and fuel, and therefore the fire to be dispersed?

Answer: I would expect it too. Unfortunately we know from accident reports that a severe fire has occurred after a severe impact - so the two cannot be completely divorced. But if the probability of a severe impact is low and the probability of a severe fire is low then the probability of seeing both in one accident is very low.

Comment by C. B. G. Taylor, United Kingdom: You spoke of the need for more severe test conditions, together with less severe "pass" criteria, in type B packages which may get into severe accidents. The same applies also to type A packages, which can be very badly crushed by mishandling at an airport, for example, but still retain nearly all their contents. Present packaging is fortunately better than the Regulations require but it would be wrong to stiffen the Regulations without relaxing the pass criteria.

Answer: I entirely agree. Existing containers qualified to present standards may well have the capacity to restrict releases in the failed conditions; this could mean that the risk posed by their use is quite acceptable. I personally think that is so. However, the fact remains that we do not know it is so unless we do some testing beyond regulatory requirements.

Your central point is worth emphasising, however. It is not sensible to associate severe test conditions with a requirement for no leakage at all. The event corresponding to these conditions is so very unlikely.

SUMMARY OF THE RISK ASSESSMENT MADE OF THE TRANSPORT OF PLUTONIUM NITRATE

J C Chicken

1. INTRODUCTION

This paper presents an overall view of the assessment of the significance of the risks involved in transporting plutonium nitrate from Dounreay to Windscale. The assessment was made at the special request of the Secretary of State for Energy and was in addition to the evidence submitted to the competent authority to demonstrate that all the relevant regulations had been satisfied.

The original assessment was made by a team consisting of: J T Daniels, J R Beattie, M L Brown, S F Hall, J Jowett, C Wheatley, R L D Young, L E Lockett, P L Holden, R J Williams, P J Cooper, G D Kaiser, A R Taig, J J Clifton, A N Kinkead and J C Chicken. The interpretation that this paper gives of their original and detailed contributions in no way detracts from the importance of their work.

The discussion in this paper is divided into seven distinct parts which are movement, criteria, probability of rupture, consequences of release, monitoring, decontamination and conclusions.

2. MOVEMENT

The movement consists of moving plutonium nitrate solution in specially designed packages from Dounreay to Windscale. The packages will be taken by road from Dounreay to the nearby port of Scrabster. From Scrabster the packages will be conveyed to Workington by sea on a roll-on roll-off ship, this will allow the packages to remain on the articulated vehicles they travel from Dounreay to Scrabster on. The ship used will be one that was specially designed for carrying heavy generating plant for the Central Electricity Generating Board. Among the special features of the ship are shallow draft (12 ft), twin engines, twin screws, twin rudders, bow thruster, and comprehensive navigation and communications equipment. For the journey from Scrabster to Workington the plutonium nitrate packages and the vehicles they are mounted on will be the only cargo the ship carries. From Workington docks the packages will go by road to British Nuclear Fuels Limited's Windscale works.

3. CRITERIA

In the practical sense provided the operation satisfies IAEA regulations there is universal agreement, among those involved in the safe transport of radioactive material, that the operation will be acceptably safe. But for the purpose of this exercise it was considered desirable to make a different approach and to overtly quantify the risks involved so that they could be compared easily and rationally with other acceptable risks.

Advice on the acceptable level of risk has been given in many publications including ICRP Publication 26, the 6th Report of the Royal Commission on Environmental Pollution, the Rasmussen "Reactor Safety Study" (WASH-1400), and the First Report of the Advisory Committee on Major Hazards. The following extracts, from the reports just mentioned, show there is a considerable measure of agreement about what constitutes an acceptable level of risk.

ICRP Publication 26, paragraph 118, published January 1977 states:-

> "The acceptable level of risk ..... for members of the general public may be inferred from consideration of risks that an individual can modify to only a small degree and which ..... may be regulated by national ordinance ..... From a review of available information related to risks regularly accepted in everyday life, it can be calculated that the level of acceptability for fatal risk to the general public is an order of magnitude lower than for occupational risks. On this basis, a risk in the range 10^{-6} to 10^{-5} per year would be likely to be acceptable to any individual member of the public."

The 6th Report of the Royal Commission on Environmental Pollution - Sir Brian Flowers (1976), shows in Figure 13 for a programme of 100 nuclear reactors on semi-urban sites in the UK that an accident causing 10 fatalities can be expected once in 2000 years, and one causing 100 fatalities once in 10,000 years. In the report attention is drawn to the fact that such risks are considerably lower than many of the widely accepted man-made risks such as dam failures, chlorine releases and aircraft crashes.

The Rasmussen "Reactor Safety Study" (WASH 1400) in Table 5-7 of the Main Report, lists the consequences to accidents from 100 reactors in the USA, the following:-

> "At a probability of once in a million years 900 early fatalities 14000 illnesses and other damages."

The First Report of the Advisory Committee on Major Hazards (published by HSE) considering non-nuclear industry postulates in paragraph 19 that "a serious accident ... more than once in 10,000 ... might perhaps be regarded as just on the borderline of acceptability;" and goes on in paragraph 44 to identify a major accident as one causing casualties in the range from less than 10 to over 1,000.

More recently Lord Ashby, the first Chairman of the Royal Commission on Environmental Pollution, brought the whole issue into perspective with his suggestion that there is a scale of relative values that can be used as a rule of thumb for political decisions(1). Four points identified on this scale are: risks of one in a million are of no concern to the average person, risks of one in 100,000 elicit warnings, when risks rise to one in 10,000 people are willing to pay to have the risks reduced, and risks of one in 1,000 are unacceptable to the public and there would be strong pressure to have them reduced.

From these views the criterion for the acceptable level of risk of an individual becoming a casualty of 10^{-5}-10^{-6}/year appears to be defensible.

4. PROBABILITY OF RUPTURE

Assessments were made of the probability of the package being ruptured as a result of being involved in either a road or ship accident, also the probability of the package being ruptured by a fire during either road or sea phases of the operation was assumed.

For the particular road routes it is proposed to use, and allowing for the number of journeys to be made, it was predicted that the probability of a package failing during the road phase of the journey was 6.2×10^{-7} per annum.

Similarly it was predicted that during the sea part of the operation the probability of the ship being involved in a collision severe enough to damage the package would be about 2.2×10^{-8} per year. The chance of the ship simply being

lost is naturally higher and is predicted to be of the order of 2×10^{-6} per year. In some ways the loss of the package at sea can be regarded as a fail safe type of accident, as will be shown later, it is unlikely to impose any significant hazard on the population.

In the assessment of the possibility of the package being ruptured as a result of being engulfed in a fire, consideration was given to the plutonium nitrate solution being raised to such a temperature that the resulting vapour pressure would be sufficient to cause failure of the containment vessel. An additional risk consisdered was the neutron shielding built into the package being burned away to give rise to a radiation risk to the public and recovery workers. It was estimated that the pressure vessel holding the plutonium nitrate would fail as its temperature approached 300°C. Calculations suggest that the temperature the contents of the package would not rise significantly for the first two hours any fire then the temperature would rise fairly steadily and reach the critic value of 300°C in 5 hours, if the fire lasts so long.

For the road part of the journey it was estimated that the probability of a fire occurring was 3.4×10^{-6}/year. The probability of a fire lasting more than 90 minutes, is naturally lower and was conservatively estimated to be 1×10^{-7}/ year. The only real source of sufficient flammable material to support such a long fire is the oil stored in a tank farm near Scrabster harbour. The total probability of a fire at that point in the journey was estimated to be 8.7×10^{-9}/year.

While the package is on the ship there is a possibility that it could be involved in fire, ship fires are quite common (probability 0.1/year). It was estimated that the probability of a fire that would threaten the package was of the order of 10^{-6}/year. It was concluded that it was unlikely that there could be a ship fire lasting long enough to threaten the package, as a fire of long duration was likely to lead to the loss of the ship. Sinking of the ship leaves the package relatively safe, and salvage is a practical proposition.

If the package is involved in a fire it was estimated that it would take 1 hour 52 minutes to char all the neutron shielding. Complete charring of the shield to charcoal would reduce its shielding properties to about 9% of its pre-fire value. Such a reduction in shielding could result in the need to establish an exlusion area around the package that might hamper recovery. The general conclusion was reached that the mechanical failure, due to a transport accident, was likely to be the dominant cause of rupture, and that the probability of such failure was conservatively about 6×10^{-7} per annum.

5. CONSEQUENCE OF RELEASE

If the container holding the plutonium nitrate fails by some mechanism it is assumed that only 10^{-4} of the leakage would be in the form of a respirable aerosol. On this basis an assessment using the computer program TIRION was made of the consequences in terms of individual and societal risks.

In the assessment of the risk to individuals it was assumed that the in vidual most at risk would be no closer than 500m from the source, and that anyo closer would have been evacuated upwind. The assessment showed that along the shole route the risk to the individual (10^{-8}–10^{-11} yr^{-1}) always lies several orders of magnitude below the generally accepted level of 10^{-5}–10^{-6} yr^{-1}.

The assessment of the societal risk identified the chance of an accident causing the death due to cancer of one or more people being of the order of one in about five million years, and the chance of an accident causing the deaths of ten

or more people might occur once in ten million years. Serious accidents causing the deaths of more than a hundred people might occur once in 10^9 years. The assessed risk is clearly well below 10-1000 deaths per 10^4 years mentioned by the Advisory Committee on Major Hazards as being the borderline of acceptability.

An assessment was also made of the consequences of losing a package at sea. From this assessment it was concluded that neither consumption of sea-food contaminated by any plutonium released, or exposure resulting from plutonium being washed ashore and contaminating the shore line constituted an unacceptable level of risk.

6. MONITORING

To ensure that the position of the ship is known throughout the operation and that communication with the ship is possible at all times additional navigation and communication systems will be installed. These will include Satellite navigation and communication systems. At all times during the operation the position of the ship will be monitored automatically so that any deviation can be checked and corrective action initiated. To help locate the packages in the unlikely event of the ship sinking the package will be fittd with transponders and pingers.

An assessment has also been made of the problems of salvaging the package, and no reason has been found to doubt the ability of salvors to recover the cargo without damage within the safe life of the package on the sea bottom.

7. DECONTAMINATION

The ground contamination that could follow from an accident in which the package leaks would at worst contaminate a few Km^2. In reality the contaminated area is likely to be much less than $1Km^2$. Once the contaminated area has been identified there should be no problem in decontaminating down to an acceptable level using well tried and proven techniques.

8. CONCLUSIONS

The findings of the assessment are sumarised in Fig 1 and the risks identified are compared with acceptance criteria in Fig 2.

Two main conclusions are drawn from the work. First the proposed operation presents risks substantially lower than those common in industrial operations, transport and everyday life[2]. Secondly when a package is designed, tested and approved to the standards by the IAEA Regulations a standard of protection of the public is achieved that is wholly acceptable when measured against other risks. This confirms the overall adequacy of the IAEA regulations and further indicates that there is a large measure of conservatism incorporated in the current standards.

REFERENCES

E ASHBY, Reconciling Man with the Environment, Published Oxford University Press, 1978.

2. The transport of plutonium in the form of nitrate solution between Dounreay and Windscale. A safety assessment by the Nuclear Installations Inspectorate - Published by the Nuclear Installations Inspectorate, 1979.

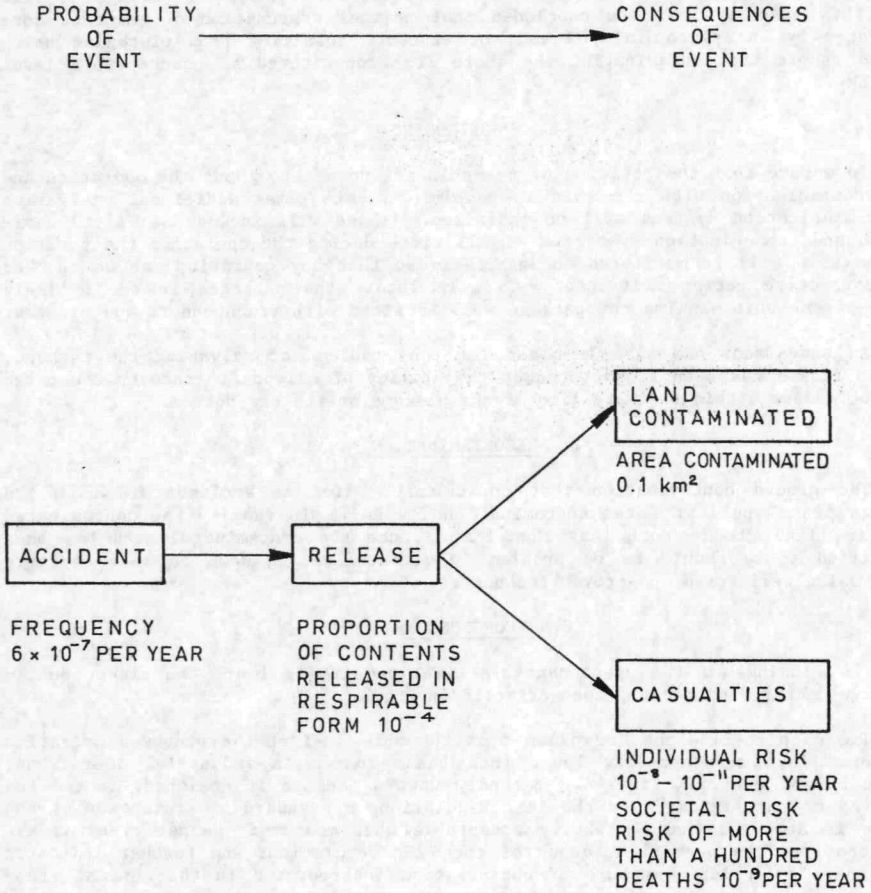


FIG.1 SUMMARY OF ASSESSMENT

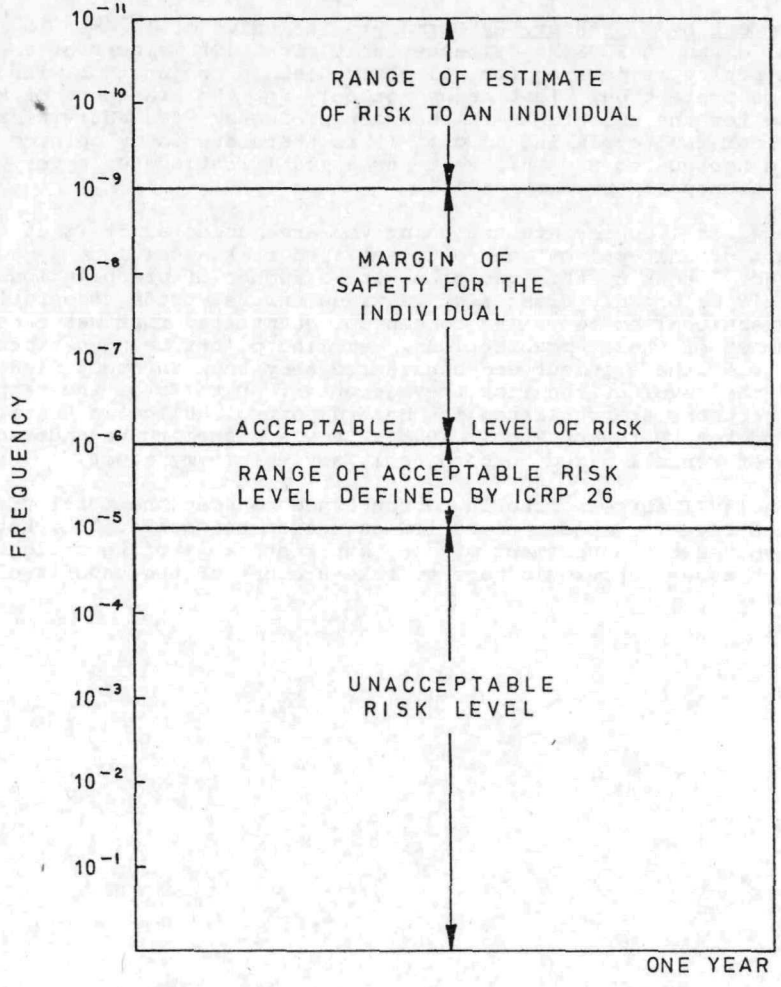


FIG. 2 RELATIONSHIP OF RISK TO ACCEPTABLE LEVEL

DISCUSSION

Comment by G. Wehner, FR Germany: The value of 10^{-5} to 10^{-6} per year given in ICRP 26 as "acceptable risk" for members of the general public is derived for all activities in conjunction with radiation protection. That means not only for the transport of RAM but also for the use of RAM in consumer products, for activities in the nuclear fuel cycle and so on. It is therefore in my opinion a little dangerous to use this value as a justification for a very special transport arrangement.

Answer: I appreciate the point you are making as it is at the heart of all applications of quantified risk assessments. Some years ago, I took up this question with a number of organisations that could be broadly classified as trade unions, trade associations and professional bodies. The concept of quantified risk was accepted by a number of these organisations; airline pilots being particularly au fait with the argument and considered they knew in quantified terms exactly the level of the risk they accepted. Details of the response to my questions are published in "Hazard Control Ruling in Britain", J. C. Chicken, Pergamon Press, 1975. My views seem to be endorsed by the reference I cited to Lord Ashly's views in my paper.

Finally I agree we should be concerned to keep the total risk the human race is exposed to as low as reasonable achievable, but this involves the assessment of many manifestations of technolgy. Any such assessment would have to take account of the benefit of technology.

EuPuC II

A TRANSPORT PACKAGE FOR LARGE QUANTITIES OF PLUTONIUM OXIDE RAW MATERIALS

Authors : I. Lafontaine
TRANSNUBEL S.A. Belgium
J. Draulans
G. Evrard
J.P. Fabry
L.U.B. Bredael
BELGONUCLEAIRE S.A. Belgium

Presented by : J. Draulans

A transport package for large quantities of plutonium oxide raw materials has been studied in the frame of C.E.C. contract, ref. 041-78-12 RPu B, between the Commission of the European Communities on the one hand and BELGONUCLEAIRE-TRANSNUBEL on the other hand.

The C.E.C. prescriptions for the new package require, beside others :

- a concept corresponding to the test specifications, provided in the I.A.E.A. regulations for type B (U) packages,
- a composition in 3 main parts : an outer container, an inner container and a primary one,
- the possibility of remote mechanical opening and closing of both the outer and the inner container and of the new primary container vessel,
- the ability to transport several tens of kg of a reference plutonium having a theoretical composition of : 2.1 w/o Pu 238, 51 w/o Pu 239, 26.7 w/o Pu 240, 13 w/o Pu 241, 7.2 w/o Pu 242; 0.7 w/o Am 241; 8 mCi/kg fission products and a maximum of 6% moisture. The plutonium will be transported either as pure PuO_2 or as a mixed oxide with a content of 40% PuO_2, both can be in the form of powder, Sol-Gel kernels, granules, etc.

The main problems, encountered during the conception, resulted from the requirements related to the remote filling and emptying operations of the product holder, the criticality aspects, dose rates, plutonium heat generation, impact and fire damage limitations.

The concept has been based upon :

- a limitation of all damages to the shock-absorbers and the outer shell, resulting from a drop followed by a fire,
- a complete separation of the energy absorbing function and the leak-tight closing function,
- a doubling of the leak-tight barriers,
- a limitation of the gamma and neutron dose rates at the outside of the inner and of the outer container,

- a solution for remote filling and emptying of the product holder without the need for expensive and in most cases insufficient decontamination work.

Specifications have been drafted in order to obtain basic data for the concept. These specifications cover not only the product to be transported but also the package concept and the package components.

The general specifications cover the nature of the product and some of its physical properties : the product will be in the form of powder, granules or kernels of PuO_2 or mixed oxide (MO_2), with a humidity content below 6 w/o, a Pu content in the mixed oxide below 40 w/o, a density between 1.5 and 6.0 g/cm^3, a heat generation limited to 20 W/kg Pu, and a thermal conductivity between $0.20.10^{-3}$ to $0.26.10^{-3}$ cal/sec cm°C. The quantity of Pu, transported per package is limited 45 kg and the quantity of Pu per product holder is limited to 15 kg.

The basic specifications for the package concept include amongst others :

- the conformity to the I.A.E.A. regulations,
- an obligation to store at least one product holder containing 15 kg of Pu in any of the provided forms and as pure or as mixed oxide,
- a limitation of the maximum allowable weight per length or height to 0.50 kgPu/cm,
- a limitation of the maximum allowed heat generation per storage cavity length or height to 10 W/cm,
- a limitation of the maximum allowed temperature in the centre of the powder to 550° C for normal transport conditions.

Furthermore prescriptions are given concerning the package concept that will be composed in 3 main parts :

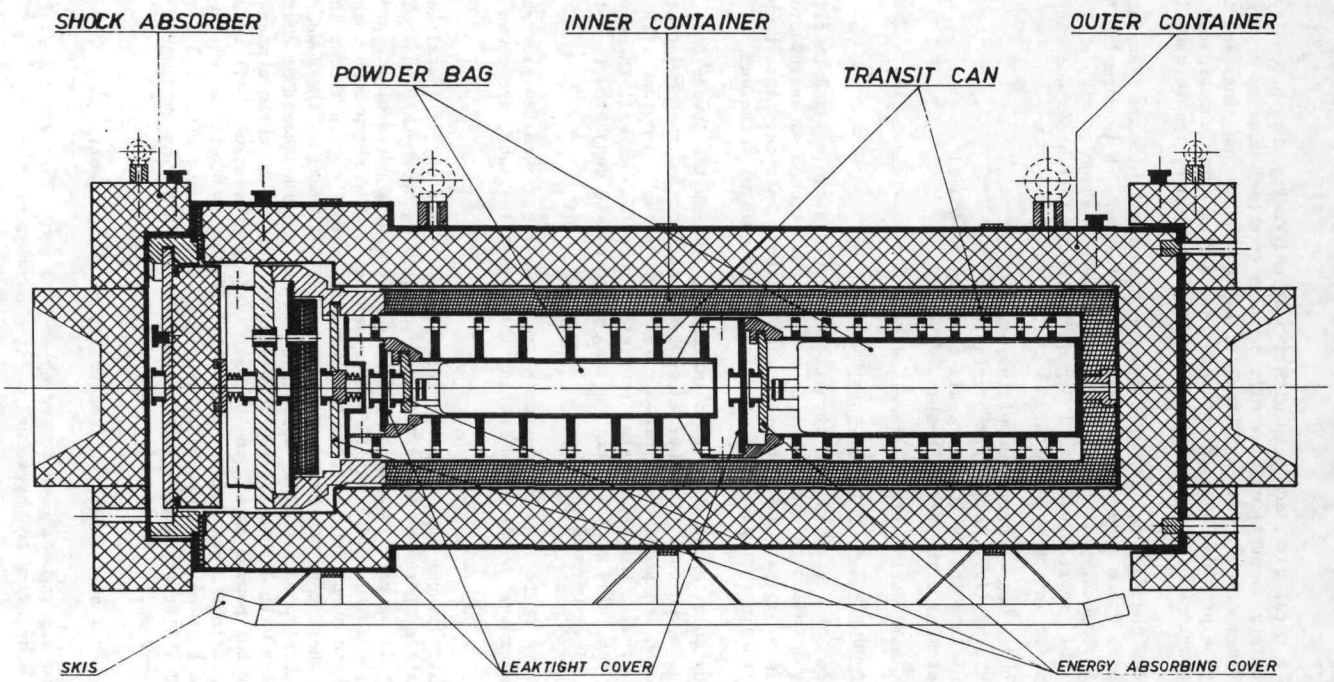
- an outer shell, capable of absorbing any damage resulting from different and successive I.A.E.A. tests,
- an inner container, suitable to be used as a storage container, and build in such a way that manipulation of the full loaded inner container is possible, even in case of a complete destruction of the outer shell,
- a transit can as product holder.

Two leak-tight barriers have to be provided namely one barrier on the transit can and one barrier on the inner container. The outer dimensions will be limited to a length of 3.0 m and a height of 1.75 m, the package weight will not exceed 3000 kg.

The detailed specifications cover the package components. The transit can will be constructed in stainless steel, leak-tightness will be obtained by means of a hollow metal gasket, the construction will resist to an inner pressure of 40 kg/cm^2 and an outer pressure of 20 kg/cm^2. The opening and closing operations and the introduction in the inner container have to be possible by remote operations.

The inner container will be constructed as a solid metal cask, equipped with the required heat resisting and neutron thermalizing and absorbing layer. Leak-tight closing will be obtained by means of a hollow metal gasket. Opening and closing operations of the inner container and introducing and removing of the inner container in or out the outer shell must be feasible by remote operations. Finally a sampling device has to be provided for sampling the inside of the inner container.

The inner container will be protected by an outer shell. The latter will be



EuPuC II
fig. 1

GREATEST LENGTH: 2870mm
GREATEST DIAMETER: 960mm
APPROXIM. WEIGHT: 2300 kg.

constructed as a metal cask, equipped with a heat and energy absorption layer. This heat and energy absorption layer will be made from high quality dry redwood parts, each of them covered with a fiberglass layer. Degassing plugs and lifting devices have to be provided on each separate part of the outer container. Opening and closing operations of the outer shell will be possible by remote handling.

The proposed concept, called EuPuC II from European Plutonium Container is a package into which 2 new type transit cans can be loaded. It is normally transported in the horizontal position. Such a container is shown on fig. n° 1. Each transit can is able to receive up to 15 kg of Pu in the form of oxide. The concept can be divided into 4 main parts :

- the sealed powder bag as primary containment,
- the transit can as first leak-tight barrier,
- the inner container into which 2 transit cans can be loaded,
- the outer container with its shock-absorbers.

Each of these 4 main parts will be described in more detail hereafter.

1) The sealed powder bag as primary containment

The first idea of connecting the transit can onto a glove-box and to fill the powder directly into the can has not been maintained. Indeed no satisfactory process has been found for a quick, cheap and safe decontamination. An attempt has been made to maintain the concept of the plastic bag welding technique.

The following solution is proposed : instead of filling the powder directly into the transit can, which is a rather expensive construction, the powder will be introduced into an inexpensive stainless steel "powder bag", provided with a filling piece made from a thinwalled stainless steel tube (0.5 mm wall thickness). The thinwalled tube will be fixed onto a ring which can be connected in a leak-tight way on a glove-box wall.

The successive operations of connecting the powder bag onto the glove-box, filling the powder bag, welding and cutting the thinwalled tube, are shown on fig. 2.

The whole unit "powder bag - filling piece - connecting ring" is installed below the filling station of the powder filling glove-box. The push-through lock in the bottom of the filling glove-box is closed by the remaining parts of the previous filling operation. The powder bag is now pushed into the push-through lock, during this operation the remaining parts of the previous operations are forced into the glove-box from where they can be evacuated as waste. As soon as the powder bag is in position, the powder filling operation can start. The amount of dust, setting on the inner wall of the filling tube, can be reduced by using concentric tubing, one for the powder supply and one for the air evacuation. After the powder filling operation, a part of the thinwalled filling tube will be mechanically compressed to a flat piece on which a resistance weld will be made on 3 places. The middle weld will then be cut, as it normally is done during bagging-in or bagging-out operations.

Preliminary tests have shown the feasibility of the project.

The closed powder bag has been tested up to 5 kg/cm^2. This value certainly can be increased by using the optimized welding parameters.

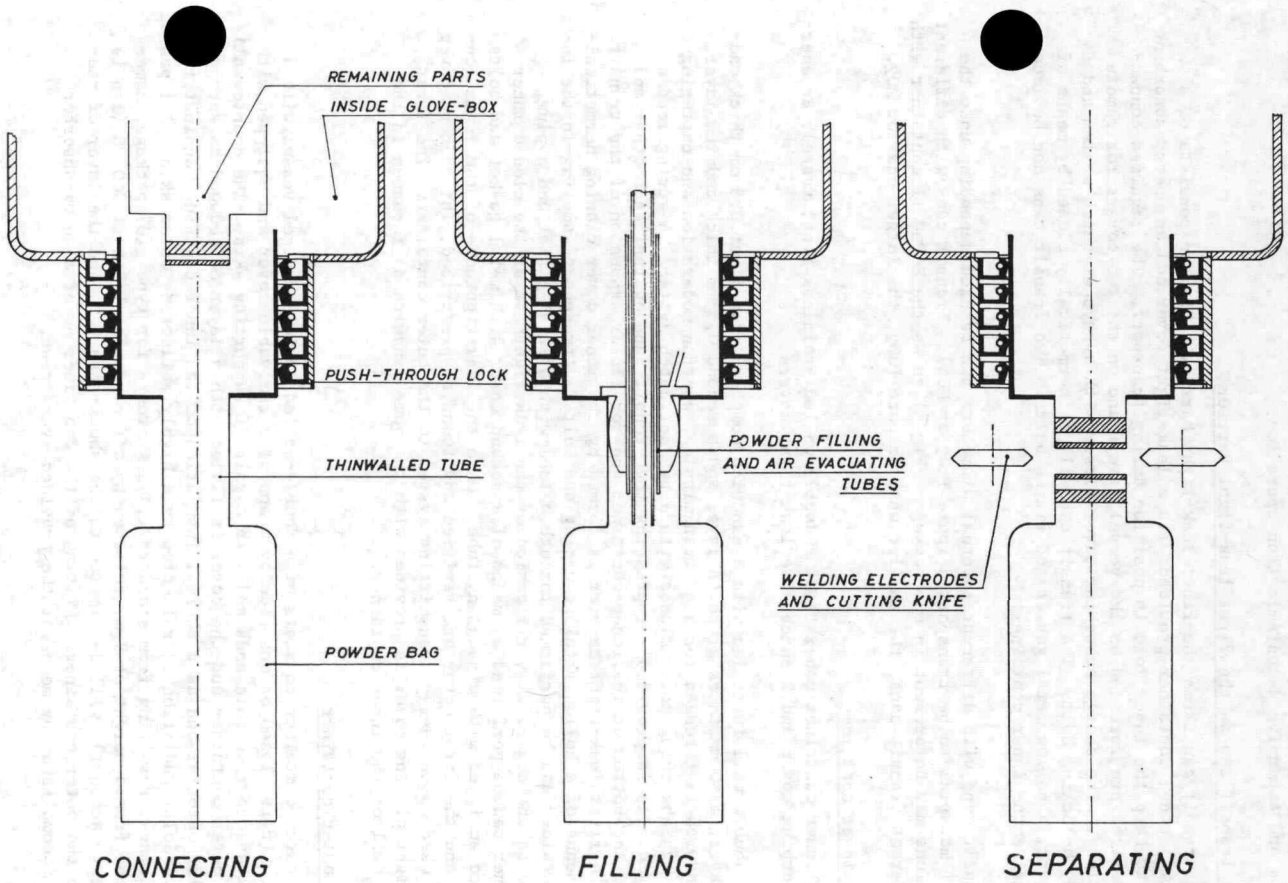

REMAINING PARTS
INSIDE GLOVE-BOX
PUSH-THROUGH LOCK
THINWALLED TUBE
POWDER BAG
POWDER FILLING
AND AIR EVACUATING
TUBES
WELDING ELECTRODES
AND CUTTING KNIFE
CONNECTING
FILLING
SEPARATING

fig 2

The closed powder bag can then be introduced in the transit can without a complicate and expensive decontamination operation.

2) The transit can as the first leak-tight barrier

The transit can forms the first leak-tight barrier. The can consists of a stainless steel container provided with a leak-tight cover and an energy absorbing cover to keep the inner load in position during transport. The required connecting devices are foreseen on the vessel itself and on the two covers for remote operations. The energy absorbing cover closes with a bayonet catch system and is held by a locking bolt. The transit can will be hermetically closed by means of the cover, a hollow metal gasket and fixing bolts. Two transit cans can be located inside the inner container.

Transit cans with different internal diameters can be used depending upon the type of material to be transported (pure PuO_2 or MO_2). Transit cans with different dimensions may be present in one package. Fig. 1 shows the EuPuC II container with 2 different transit cans, the smaller one for pure PuO_2, the larger one for MO_2.

3) The inner container

The inner container consists of a double-walled stainless steel vessel, an energy absorbing cover and 2 successive leak-tight covers.

The double wall is filled with a sandwich composed of a first 0.5 mm thick cadmium layer, a 50 mm thick graphite layer and a second 0.5 mm thick cadmium layer. The sandwich is foreseen for heat transmission, neutron moderation and capturing. The energy absorbing cover closes with a bayonet catch system. A spring loaded pushing device is provided for pressing the transit cans against each other and against the bottom of the inner container and for linear expansion of the transit cans. A first leak-tight barrier is formed by the inner cover closing hermetically by means of a hollow metallic O-ring and bolts. Atmosphere sampling in the inner container can be performed through a tube with a metal gasket and a plug, mounted in the cover. A verification of the leak-tightness of the closed inner container can be performed by means of a second cover with metal gasket and bolts and also equipped with a sampling tube. Once the leak-tightness of the inner container and the first cover are verified and found as prescribed, this second cover can be used as an energy transmitting system to the outer container. The first leak-tight closing cover is provided with the same sandwich as foreseen in the double wall of the inner container.

4) The outer container

The outer container consists of a double-walled stainless steel vessel with a bottom. It is closed on one side by means of a removable cover and equipped with a shock-absorber on both ends and with a pair of supporting skis. The double wall of the outer container and the cover is filled with a layer of redwood as energy absorbing material and as a thermal insulation of the inner container against fire damage. Solar radiation is reflected by a highly reflecting outer skin. All possible damage, resulting from standard I.A.E.A. tests for type B(U) package, namely a drop from a height of 9 m, followed by a fire during 1/2 h at 800° C (1 h is taken into account), will be limited to the shock-absorbers and the layer of redwood in the outer container. Lifting of the container or of a shock-absorber can be accomplished by means of the provided eye-screws.

Calculations concerning impact, mechanical test, heat transfer, criticality safety and dose rates have been performed for the proposed design and have shown that such a package meets the I.A.E.A. regulations.

Three different impact possibilities namely : side-on, corner and end-on impact have been studied. The redwood in the shock-absorbers will be deformed over a thickness of 40 mm in case of the side-on impact. The stress in the outer metallic cask has been verified by means of the STARDYNE computing model; for a weight of 21238 N of the package the highest stress obtained in the outer metal cask is 266 N/mm^2 for a bow of 3 mm. The corner-impact can be divided into 2 impacts, first a corner impact followed by a side-on impact with a rotation speed equal to the initial speed at the moment of the first shock. In the first phase the redwood of the shock-absorber is deformed over a thickness of 80 mm, in the second phase a redwood deformation of 41 mm is obtained. For the end-on impact special parts in the form of a crown are provided at both shock-absorbers, for a given deceleration of 100 g one obtains a deformed redwood height of 143 mm.

In case of the mechanical test calculations give a redwood penetration of 32 mm when the presence of the metal cask is omitted, one may estimate the real deformation comparable to the metal cask thickness.

Heat transfer calculations have been performed to define the temperature in the different layers of the package for a case as defined by the I.A.E.A. regulations for a type B package and for a heat output of 20 W/kg Pu. For an outside temperature of the package of 50° C the following temperatures have been calculated by a simplified evaluation and by the finite element calculations.

Temperature distribution in the container				
Package components	Simplified evaluation °C	°C Finite element calculations		
		Top	Side	Bottom
Outer surface	50	49	50	49
Inner redwood wall	86	84	86	98
Outer graphite wall	122	95	96	98
Inner graphite wall	122	95	96	98
Outer transit can wall	209	127 (1)	127 (1)	127 (1)
Axis of Pu powder	341	259	259	259

(1) evaluated, using a linear energy release, properly increased to take into account the reduced inner exchange surface of the inner container.

Criticality safety calculations have been performed with regard to the I.A.E.A. regulations, giving the specific provisions to fissile class II packages for an individual package, considered as isolated, and for one or more packages. The computing codes NUSYS and DIXY, set up by Kernforschungszentrum Karlsruhe, have been used for these calculations.

A load of 15 kg Pu in the form of PuO_2 is considered per powder bag. The Pu has a fissile isotopic composition of 90 w/o, contains 6 w/o of water and has a density of 4.5 g/cm^3

Assumptions have been made for one or more isolated bags.

In the case of an isolated bag, surrounded by a reflecting water screen of

50 cm, one obtains a k_{eff} = 0.703. If a group of 3 Pu powder bags is considered standing upright in water in such a way that an equilateral triangle is formed, a maximum k_{eff} of 0.944 is obtained. If the bags are placed one against each other in air the k_{eff} decreases to 0.24.

Assumptions have been made for an infinite number of undamaged packages.

If an infinite number of undamaged packages, each loaded with 2 x 15 kg Pu as PuO_2, are stacked together and a radial water reflection of 6 cm and an axial one of 30 cm is considered, a k_{eff} = 0.33 is obtained.

Furthermore several assumptions have been made for damaged packages.

If a waterleak into the transit can is assumed a k_{eff} = 0.67 is obtained. If furthermore an axial volume reduction, equal to 7% of the inner container, is assumed a k_{eff} = 0.71 is obtained (corresponding to a k_{eff} increase of 0.04). If finally a variation of the water density is assumed from 1 to 0.15 g/cm^3 a k_{eff} = 0.49 is obtained.

Increasing the thickness of the waterlayer between packages from 6 to 24 cm does not affect the k_{eff} value.

Remark

The multiplication factor has also been computed for packages in which each primary powder containment bag has been filled with 22.4 kg instead of 15 kg Pu in UO_2 + 40 w/o PuO_2 kernels at a density of 9.0 g/cm^3. A k_{eff} = 0.66 has been obtained.

Conclusion

These calculations show that an infinite number of packages remains largely sub-critical in normal and accidental conditions.

Dose rate calculations have been performed on the base of a BN calculation model. Gamma and neutron emissions out of the EuPuC II package have been defined. These calculations demonstrate that for a load of Pu from LWR origin, the dose rate at the surface of the package is lower than 35 mrem/h or 350 µSv/h. At 1 meter of the package, the dose rate is reduced to 5 mrem/h or 50 µSv/h.

The dose rate emitted by the naked inner container is lower than 80 mrem/h or 800 µSv/h at the surface and 12 mrem/h or 120 µSv/h at 1 m of the container.

A general conclusion may be drawn from the results of these calculations : the EuPuC II package fulfills the standard I.A.E.A. regulations.

DISCUSSION

Question by J. A. Andersen, USA: The design work is most interesting and appears to be very appropriate. What now are the plans for fabricating and testing the EuPuC II package, what might be its cost, and will there be a design drawing published in the proceedings?

Answer: The CEC contract no. 041-78-12-RPu-B between the Commission of the European Communities and BELGONUCLEAIRE-TRANSNUBEL concerns only the elaboration of specifications and preliminary design for a transport package for large Pu quantities. The contract does not include any further development in the field of fabrication and tests. However, we hope that a follow-up of this primary design will be possible in the future. The cost of such a package, fabricated in a series of 5 pieces is estimated at ≸ 80,00.00. The paper includes a design drawing.

SESSION XV
INTERMEDIATE STORAGE IN CASKS

Storage of Spent Fuel in Transport/Storage casks

H. Keese)
W. Anspach) Transnuklear GmbH
R. Christ)
E. Schlich- Nukem GmbH

1. Introduction

Of the various ad-reactor (AR) and away-from-reactor (AFR) storage techniques which have been studied the intermediate storage in type B casks seems to be an interesting and viable solution to the problem.

The casks in such intermediate storage system have equally transport and storage functions. Therefore, they obviously must fulfil the IAEA-regulations as type B transport casks. In addition, however, they have to ensure safe storage of spent fuel and to maintain their integrity under long-term conditions. This requires inherently safe casks without fragile auxillary systems, but requires also their adaptation to specific storage conditions.

For more than 10 years Transnuklear and its foreign partner companies have strictly followed the line of inherently safe "dry" transport casks filled with gas or air and operating with the passive cooling system, i.e. heat dissipation by natural air convection. These casks have been successfully operated for the shipment of spent fuel from European nuclear power plants. The new generation of forged steel casks type TN 12 and TN 17, 39 of which are actually ready for operation, in production or on order, will be used for shipments of European and Japanese fuel elements to the COGEMA reprocessing plant at La Hague.

2. Transport/Storage Cask TN 1300

Based upon the experience gained by development and operation of its casks, Transnuklear has designed a series of casks especially dedicated to medium and long-term storage of various types of spent fuel. The dimensions of these casks were chosen in such a way as to ensure maximum compatibility with the TN 12 and TN 17 transport casks and exchangeability of single components, if possible.

The largest of this group of casks, the TN 1300, is designed for transport and storage of spent fuel elements from the 1300 MW class of light water reactors. It has a maximum capacity of 12 PWR and 33 BWR fuel elements, respectively. The maximum heat dissipation (by natural convection) amounts to about 50 kW corresponding to a cooling period of 1,5 to 2,4 years for the fuel elements. This range is a compromise of different parameters.

Principle dimensions of the packaging are given in figure 1. The weight of the laden cask in transport and storage conditions is about 115 t.

The cask is built as follows:

- The cask body is made of ductile cast iron (German standard GGG 40.3) with cast fins.

- The cylindrical cavity is completely covered by a stainless steel liner, 8 mm thick.

- The cask comprises a double lid system with a shielding lid (inside) and a sealing lid (outside). Both lids are equipped with metal double O-ring sealings with extremely low leak rates of 10^{-8} to 10^{-9} mbar x l/sec. Those metal sealings remain fully functional even under impact conditions.

  This high quality sealing system together with an under-pressure of about 0.5 bar in the cavity insures safe and tight containment of the radioactive content over an extended period of time. There will be no activity release. It goes without saying that the lids can be equipped with Viton O-rings under transport conditions.

- Much emphasis is put on containment control and repetative testing. Therefore all sealings are, as mentioned before, of the metal double-O-ring type and can be tested independently from the cavity and in addition to interspace testing between barriers. These tests can be performed by means of different methods.

- The orifice connections through which the container is filled or emptied, pass through the shielding lid and are protected and sealed independently. One of those tubes leads through the basket to the lowest point of the cask. This design avoids any passage through the cask body and thus contributes to safety and low cost production.

- The shielding of cast nodular iron/hydrogeneous material test models was measured. Californium 252 was used as a neutron source as its spectrum is comparable to the neutron source of the irradiated fuel. Based upon those test results positions

and dimensions of the incapsulated neutron shielding were designed between the fins, at the cask base and on top of the shielding lid. Thus the cask body kept its full integrity (fig. 2).
The container is so designed that the total dose rate (gamma and neutrons) at the surface does not exceed 20 mrem/h, with the neutron dose rate less than 10 mrem/h.

- The TN 12 basket will be used for short cooled fuel. This basket consists of 4 x 3 segments of an aluminium stainless steel structure. CuB_4C sintered plates serve as neutron poison.

  A simplified basket of boronated stainless steel is being developed for longer cooled fuel.

  The choice of different baskets always allow storage of fuel at a maximum fuel rod temperature of less than 400°C, a value which is generally considered tolerable. Recently new test programs were initiated in the US and in the Federal Republic of Germany in order to gain additional data about the maximum allowable temperature and the long term behaviour of spent fuel or rods.

3. Manufacturing

The first TN 1300 was cast May 31st, 1980 at Thyssen-Gießerei AG in Mülheim/Ruhr. The cask body was removed from the mould after 3 weeks and was ready for quality control after another 2 weeks. The quality control tests which were performed in close cooperation with TÜV and BAM gave very good results for mechanical properties and integrity of the casting of body and fins. This is, among others, due to the fact that the cask body was produced by means of a novel casting technique which results in above-standard properties normally only reached with thin walled castings.

4. Licensing

Two 1:3 cask models and several normal size test pieces were produced for the type B tests. These cask models had to undergo a series of type B qualification tests under BAM surveillance. In addition to normal drop tests the casks were drop-tested at minus 40°C and with an artificially damaged wall, namely with a cut of 1/3 of the wall thickness. These qualification tests and all quality control investigations were successful and were confirmed by a BAM-letter of November 6, 1980.

The firetest will be performed soon. We know, however, from the development of the TN 12/TN 17 that a steel cask heats very slowly due to its high heat capacity. The transient computer calculations which have been performed for the TN 1300 therefore

show satisfactory results in case of Viton sealings and very good results in case of metal sealings with which the TN 1300 will be equipped. Thus we are confident that the TN 1300 will be Type B(U) licensed.

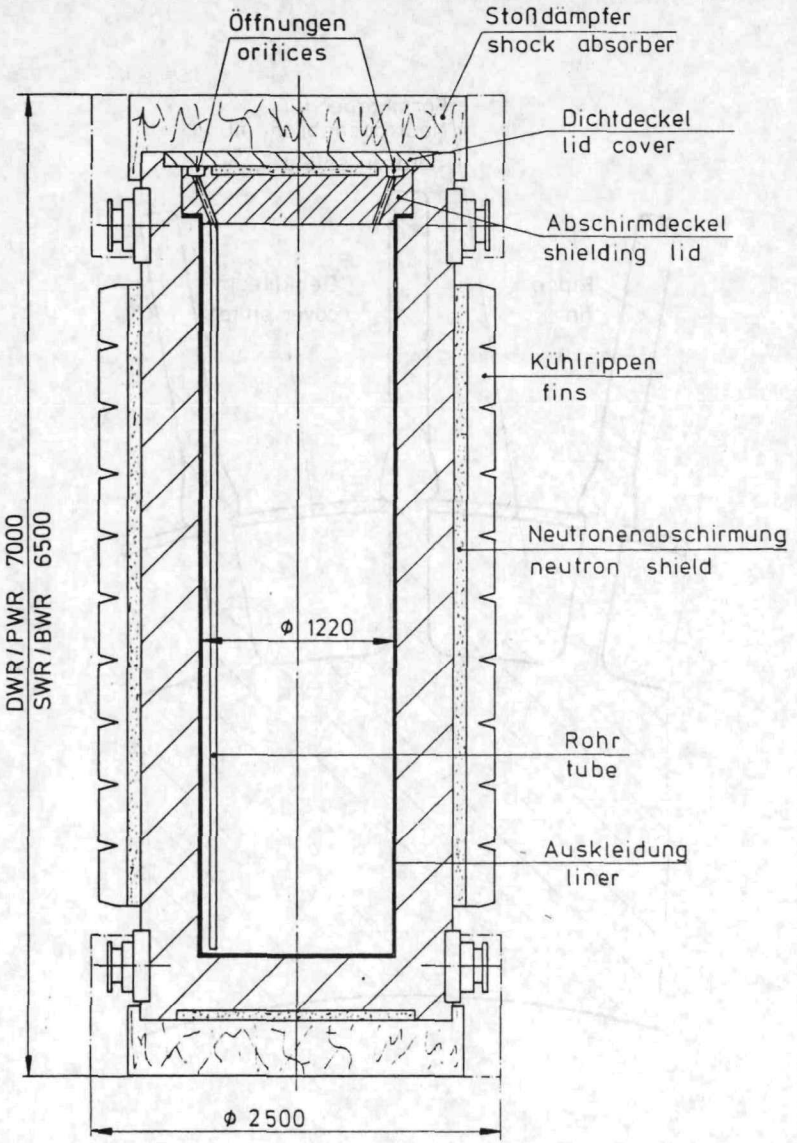
Additional calculations and tests for demonstration of compliance with storage requirements are presently under way.

5. Range of application

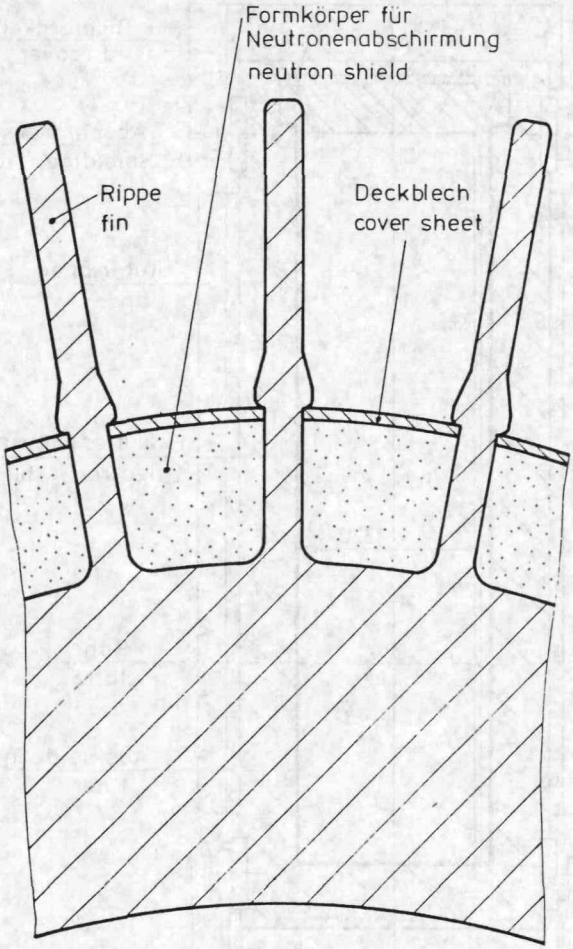
The TN 1300 is the first and largest type of a group of casks which have been designed to meet specific requirements given by the type of fuel and/or reactor handling limitations. The TN 900 for reactors of the 600 - 900 Megawatt class and the TN 300 for nuclear power plants up to 300 Megawatt are being developed. The basic design of the TN 1300 is scaled down to these smaller versions without principle design modifications and with a minimum of additional licensing effort (fig. 3).

The application of the incapsulated demountable and replaceable neutron shielding between the fins also allows an optimal adaptation of the cask to the storage of uranium/plutonium mixed oxide fuel.

The basic cask concept which allows short production time and efficient low cost machining leads to a delivery time of only 6 to 8 months per cask.



TN	Transport-/Lagerbehälter Transport-/Storage Cask	TN 1300	FIG. 1



Neutronenabschirmung TN 1300
Neutron shield TN 1300

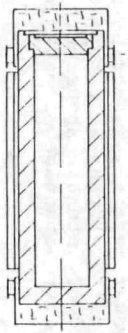
FIG. 2

GGG-Behälterfamilie
Set of nodular cast iron casks

Fig. 3

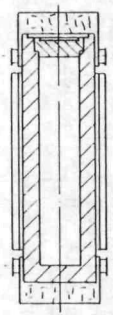
Behälterfamilie GGG - Transport - / Lagerbehälter
Set of nodular cast iron Transport / Storage Casks

TN 1300



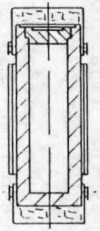
	DWR/PWR	SWR/BWR
Gewicht weight	= 116,5 t	108,0 t
Höhe height	= 7000 mm	6500 mm
Durchmesser diameter	= 2500 mm	2500 mm

TN 900



	DWR/PWR	SWR/BWR
Gewicht weight	= 75,0 t	80,0 t
Höhe height	= 5700 mm	6500 mm
Durchmesser diameter	= 2100 mm	2100 mm

TN 300



= 40,0 t

= 4050 mm

= 1750 mm

DISCUSSION

Question by B. Vogel, FR Germany: Are there such things as flawless castings? Is this rejection of the term"flawless" not also valid for competitive materials?

Answer: I am of the opinion that the term "flawless" may be used but I would like to add that this is "flawlessness" only in the sense of the given specifications. When, for example, we have defined a flaw in a component as a value which is greater than the permissible value, then the component can be considered "flawless", when the established material value is less than the permissible maximum value.

Question by B. Balmer, Switzerland: In the case of type TN-13 casks, is the neutron shielding fixed to the outside? If a serious fire occurs, there could be loss of shielding. In such an event, how will the surface dose rate be influenced?

Answer: In order to establish how much loss of shielding would occur in the event of fire, we will carry out relevant tests. Previous experience with TN casks leads us to expect that part of the shielding will remain intact. Ifwe, however, assume that in the case of unfavourable conditions, there is total loss of neutron shielding, then the total dose rate which is designed at 20 mrem/h will be increased by a factor of 5. This is well below the factor of 100 allowed by the IAEA in the case of incidents.

Question by M. Komurka, Austria: What would be the cost of storage of spent fuel elements in such casks as TN12 or TN17 in comparison to wet storage?

Answer: Based upon publications of a IAEA meeting in Madrid in 1978 and data from the INFCE subgroup 6, we determined that cask storage which we expect to cost about 300 DM/kgU would be cheaper than wet storage and most probably also cheaper than dry storage. Since there is, however, a remarkable depression in specific investment cost for wet and dry storage facilities of higher capacities, the cost advantages of cask storage may be limited to storage capacities of up to 1000 tU.

In the US, however, where less stringent safety requirements (no air crash for instance) than in Europe and particularly in Germany are applied, wet and dry storage shows much lower specific investment costs, and today a cost advantage of cask storage does not exist.

LONG-TERM STORAGE BEHAVIOUR OF SPENT LWR-FUEL

M.Peehs, G.Kaspar, W.Jung, F.Schlemmer
KRAFTWERK UNION AKTIENGESELLSCHAFT
P.O. Box 3220, D-8520 Erlangen/FRG

STRACT

Theoretical estimation indicated that no systematic failure should occur under wet storage of spent fuel. The theoretical results are confirmed by the results of the experimental investigation of the storage behaviour of spent fuel at the KWO-pool. The fission product release from defective spent fuel rods is mostly related to Cs isotopes Cs-134 and Cs-137. The release approaches zero when the Cs-concentration approaches certain equilibrium values. The effective $G(H_2)$ value for the β/γ-radiolysis of storage pool water by a spent fuel assembly is lower than 10^{-3} H_2 molecules/100 eV. Thus the H_2 production in a spent fuel pool is very low.

Theoretical investigations of spent fuel behaviour under dry storage conditions show that hoop strain due to fuel rod internal pressure is the insertion temperatures limiting process. Experiments are performed with non-fueled cladding samples to look for other possible degradation mechanismes. Also storage experiments with complete fuel bundles and with single rods are initiated to assess the results from the theoretical study.

1. INDRODUCTION

The management of the backend of the fuel cycle starts with the interim storage in the nuclear power plant. Thereafter the spent fuel assemblies are transported in special fuel shipping casks for subsequent processing and/or storage. Before further processing in the fuel reprocessing plant the fuel assemblies are stored in the receiving store which serves also as a buffer store between delivery and further processing. If suitable size, the receiving store may also serve as an interim store for longer-term fuel storage. The same results can be obtained when there is an adequate interim storage capacity outside the fuel reprocessing plant. Three concepts of away from reactor storage (AFR) are pursued worldwide: wet storage in large spent fuel pools, dry storage in caskes and dry storage in large storages facilities.

The interim storage of fuel assemblies can extend over periods of 20 years and more. An effort must be made to prove the integrity of the fuel assemblies and their proper handling capability throughout the storage period and during the discharge of the fuel assemblies from the storage position and the transfer to the fuel repro-

cessing plant as well. The following contribution is to provide a summary of the most essential results of this R+D-work.

2. THE LONG-TIME STORAGE BEHAVIOUR OF SPENT FUEL ASSEMBLIES UNDER WET STORAGE CONDITIONS

2.1 Estimate of the Long-Time Storage Behaviour of Spent Fuel Assemblies on the Basis of Available Material Data

The most severe mechanical stress imposed on cladding tubes during long-time storage results from the internal pressure of the fuel rods. The very conservative estimate of the internal pressure in case of a pressurized water reactor as related to an average bundle burnup of 37 MWd/kgU has a result of 100 bar. 50 bar are resulting in case of a boiling water reactor with equivalent conditions. A fuel rod for pressurized water reactors has been taken as a basis for the following reflection, with an internal pressure of the fuel rod of 120 bar at operating temperature. This gives an internal pressure under storage conditions of approximately 63 bar, corresponding to a tangential stress in the cladding tube of 43 N/mm^2. The $\sigma_{0,2}$ strength amounts to 440 N/mm^2, thus yielding a tenfold safety margin of the internal pressure stress as compared to the design limit. Creep processes at temperatures of 40 °C and the stresses stated before can practically be ruled out.

The average hydrogen concentration at the end of the operating period as conservatively calculated by the fuel rod design at 250 ppm [1]. Empirical values which lie at 30 to 150 ppm are distinctly below this value. If the fuel rod is cooled down to approximately 40 °C after the end of operating, most of the hydrogen which was adsorbed by the cladding tube is precipitated in the form of finely dispersed hydride precipitates. Integrity-controlling properties of the cladding such as yield stress, rupture stress, uniform elongation and elongation at rupture however, depend on the hydrogen content of the Zircaloy cladding tube. Yield stress and rupture stress are increased, the ductility decreases due to hydrogen absorption. The long-time storage temperature of 40 °C lies below the ductile/brittle transition temperature for hydrided Zirconium, i.e. in this case the cladding tubes can withstand elastic deformations up to the raised yields stress, but only negligible plastic deformations. However, plastic deformation does not occur during handling of fuel assemblies since bowing is not possible due to guidance of the fuel rods by the fuel assembly skeleton. Therefore the hydrogen absorption in the cladding tube should not exert any negative influence on the handling and storage capacity of the fuel assemblies. As mentioned before, the hydrogen is contained in the cladding tube as a finely dispersed and axially uniform precipitate. The formation of so-called hydride pits by an Osswaldt ripening process cannot be expected at 40 °C. Due to the small heating-up range of the coolant water in the storage pool a redistribution of the hydrogen by thermal diffusion can be ruled out. The root of the average displacement square after a period of 30 years amount to 2.5 cm.

The corrosion behaviour of Zirconium alloys in pressurized water and in water vapour has been sufficiently known for temperatures

above 200 °C. Comparative investigations at other temperatures show that the temperature dependence of the corrosion rate complies with an Arrhenius law in the linear post-transition corrosion area. If the corrosion under storage conditions is compared with the corrosion under operating conditions [2], the result shows that the corrosion rate in the pool water is by 10^{-10} inferior to the value at operating temperature in the reactor, i.e. the corrosion phenomena of a fullpower day correspond to those which may occur over a period of 27 million storage years. This allows the conclusion that Zircaloy is practically immune against oxidative attack by water under long-time storage conditions. The structural parts of the fuel assemblies consist of Inconel and austenitic steels. It must be pointed out in this context that in the Federal Republic of Germany only niobium-stabilized austenic steels are used. Stress corrosion phenomena due to different carbon activity [3] can therefore be ruled out. Electrochemical corrosion attacks are prevented by suitable control of the storage pool water quality. The correctness of this reflection can be inferred from the inspection of high-grade steel component parts which were exposed to the operating conditions mentioned above in various reactor spent fuel pools for a period of 10 years and longer. Any degradation of the component parts could be observed in no case.

2.2 Long-Time Storage Tests for Experimental Confirmation of the Positive Long-Time Storage Properties to be Expected

Relatively little systematically recorded long-time storage experience is available at present with regard to spent fuel assemblies for light water reactors with typical target irradiation. The performance of a fuel-rod storage test in the fuel pool was initiated in the Obrigheim nuclear power plant in 1977 in order to gain a broader data basis. For storage 18 intact and 10 defective fuel rods were selected from four fuel assemblies of the Obrigheim plant with different levels of burn-up. The condition of these fuel rods is being documented over a three year period. Fig.1 shows an overview of the status of this storage test. Eddy current examinations, dimensional measurement and measurements of oxide layer thickness are carried out and evaluated to characterize the original conditions and any change during storage. A detailed visual inspection by television camera and documentation of the takes by video tapes are performed in addition. New defect signals on the fuel rods investigated could not be detected by means of eddy current examinations. This applies to intact as well as to defective rods in the test program. The average fuel rod diameter did not change within the scope of measuring accuracy. An exact comparison within the program did not show an essential change in the defects over the storage period under observation. Therefore the investigations performed emphasize in an impressive way the long-term storage stability of fuel assemblies for light water reactors. The investigations will be continued.

2.3 Investigation of Activity Release from Defective Fuel Rods

In the case of intermediate storage it must be assumed that even fuel assemblies with operational cladding defects are stored. Fission products are released by defective fuel assemblies during

the storage period. Therefore it becomes necessary to investigate the fission product release of individual defective fuel rods. It is especially desirable in this context to clarify the dependence of the release rate on time, the environmental activity, temperature, age, and the type of damage of the defective fuel rod.

Fig. 2 and 3 show the test results which were obtained until now. Primarily the investigations confirm the experience gained during storage in the reactor storage pool i.e. that the activity of the pool water depends nearly complete on the fission isotopes Cesium 134 and 137. The activity contribution of all other fission products can practically be neglected. The measurements on two different defective rods show common characteristic features. A relatively high activity release rate arises at the beginning of the test, which approaches zero after some time. This means that the total activity in the capsule water first increases steeply and then approaches an asymtotic limit value. The activity limit concentrations in the capsule water resulting in this context are the same for both test fuel rods. The time pattern of the activity release is different in both tests. This result underlines the influence of the type of defect and possibly also of the location of defects. Fig. 3 illustrates the results of tests during which inactive esium was added to the capsule water at the beginning of the test. The results show distinctly that the equilibrium concentration of the Cesium activity is determined by the total content of Cesium in the capsule water. This allows the conclusion that exchange mechanisms play a certain part in the activity release from defective fuel rods beside other mechansims. It can be recognized as well that the activity release of individual and defective fuel rods which stand in a spent fuel pool cannot be regarded independently of each other. It is also interesting in this context that the activity release rate depends on the activity concentration in the sourrounding water. The investigations show that an analysis of the release mechanisms from data measured at the reactor fuel pool where several defective fuel rods are possibly present, is extremely difficult and may even lead to misinterpretations.

2.4 Investigation of the Pool Water Radiolysis with Wet Storage of Irradiated Fuel Assemblies

The question of hydrogen formation due to the radiolysis of the water plays an important part for the design of a wet storage facility as a whole as well as for spent fuel pools and shipping casks. The value to be taken as a basis for hydrogen formation according to the guidelines of the RSK for pressurized water reactors is considered as excessive. Therefore orienting tests with an irradiated fuel assembly in the spent fuel pool were carried out in the Obrigheim nuclear power plant. A fuel assembly with an average cluster irradiation of 30 MWd/kg and with an burnup increase in the last cycle of 10.3 MWd/kgU was selected for this purpose. The investigations resulted in an upper estimate of the $G(H_2)$ value. This value is $G(H_2) \leq 10^{-3}$ H_2 molecules per 100 eV. This refers to the hydrogen quantity released from the water and contains the real test parameters used in the experiment, such as gas/water ratio, test geometry, pH value of the water, content of typical additives of the water, water temperature, system pressure, and radiation spec-

trum. Therefore the measured value only applies to n t H_2-production from a spent fuel bundle in wet storage.

3. STUDIES CONCERNING THE BEHAVIOUR OF SPENT FUEL ASSEMBLIES UNDER DRY STORAGE CONDITIONS

3.1 Estimate of the Long-Time Storage Behaviour of Spent Fuel Assemblies on the Basis of Present Material Data

Comparable to the objective of the studies of fuel assembly behaviour under wet storage conditions, the objective of the studies conducted was to identify and to evaluate mechanisms endangering the integrity. Starting from the condition on the light-water reactor fuel assemblies after discharge from the reactor, the following mechanisms are potentially endangering the integrity under dry storage conditions: corrosion of the cladding tubes inside and outside, hydrogen pickup, crack propagation and creep deformation due to internal pressure. Here, too, considerations are based on a PWR prepressurized fuel rod, because in this case the higher stresses under dry storage conditions are to be expected in comparison to BWR-fuel rods. All studies are based parametrically on the two temperature/time-patterns during dry storage, because the assessment of the idividual mechanisms under isothermal conditions would be too conservative.

With the anticipated dry-storage concept the fuel assemblies area sourrounded by air or by an inert gas. The determination of the maximum possible outside corrosion has been based on the oxidation kinetics of Zircaloy in water steam. Expecting the so-called "post-transition"-oxidation rate of Zircaloy in water vapour based on the temperature/time pattern shown in Figure 4, the wall thickness reductions shown in Fig. 5 as a function of the insertion temperature can be expected. Consequently the oxidative outer corrosion should by no means be a factor which limits the temperature in case of dry storage.

To assess corrosion from the inside, we can assume that all fission products having exhalated the fuel during operation have reached their chemical equilibrium state. Further reaction is not possible during storage. The low average temperature of the fuel under storage conditions compared with the temperature under reactor conditions precludes a further fission exhalation. Consequently considerable damage to the cladding from inside may not be expected even under dry storage conditions. This is true especially for iodine, which initiates stress corrosion. To corroborate this statement a long-term experimental program has been started (table 1), dealing with the different influence parameters such as material state, surface finish, oxidation condition and degree of hydriding. Within the framework of this experimental work no damage to the specimens inserted could be proved for the periods of more than one years available at present.

The hydrogen displacement caused by thermodiffusion is unproblematic even under dry storage conditions, although the average temperatures are higher and the expected temperature gradients are greater than at wet storage. Figure 6 shows the estimate of the maximum local concentration of hydrogen by thermodiffusion in a Zircaloy cladding tube after 20 years at constant storage temperature. The allowable concentration of 500 ppm hydrogen is based in KWU's experience that integrity and handling capability of fuel assemblies is ensured. The results show that thermodiffusion causes no problems in dry storage of spent LWR-fuel.

Fig. 7 shows the result of an estimate of the maximum crack length after 20 years of dry storage of Zircaloy cladding assuming that the cracks at the beginning of the storage period have a depth of 50 µm and the critical stress intensity value is very low. This assumption, however, is extremely conservative, because the evaluation of the performance of fuel assemblies has shown that such cracks inititated by power ramp will lead to fuel assembly defects within a relatively short time. Thus, fuel rods, which were still intact at the end of operation should have no cracks at all or much smaller cracks than anticipated. The described results shown that the crack problem is only of importance for initial storage temperatures above 350 °C if at all, if however K_I-values of Cubicciotti or Smith are taken into consideration, the crack propagation mechanism under dry storage conditions should not be any problem.

The stress $\sigma(\dot{T}, \varepsilon)$ resulting from the internal fuel rod pressure depends on the instantaneous temperature and slightly on the diameter increase that has already taken place. All known creep laws having the form $\varepsilon(T, \sigma, t)$ resulted from experiments under temporally constant stress. For their application in this context the creep curve must be divided into time periods of approximately constant stress; the creep over such a time period is determined by the following equation:

$$\Delta\varepsilon = \varepsilon(T, \sigma, t+\Delta t) - \varepsilon(T, \sigma, t)$$

This procedure is permissible, as repeated primary creep is not to be expected with the stress falling over time.

The results of this calculation on the basis of two different creep laws are shown in Figs. 8 and 9. Consequently the circumferential creep expansion of the cladding tubes at an internal pressure at the beginning of storage of 100 bar (reference value related to 350 °C) remains below 1 %, if the initial storage temperature is limited to 400 °C; this corresponds to a decay time of about one year. The total time in dry storage is of less importance for the integral creep deformation, because the creep rate gets negligibl after one year.

Evaluating the results on creep deformation one must consider that the calculations are based on a experimental data base gained from out-of-pile creep investigation. The material state of the Zry-cladding is changed during reactor operation. A better analysis could be done if creep data on post-pile creep would be available.

Because of the lack of those data, the maximum allowable fuel rod temperature has to be reduced by an adequate safety margin. Thus temperatures in the range of 350° - 370 °C should be reliable.

3.2 Experimental Work to Confirm the Positiv Long Term Dry Storage

A fully instrumented KWO fuel bundle will be stored for 1 to 1.5 years under dry conditions to demonstrate the overall storage performance and to confirm the predicted temperature limits for the dry storage insertion temperature. This experiments includes also careful pre- and post-experimental characterization of the test bundle. In order to complete and to assure the results from the integral storage experiment additional experiments will be performed with single rods:

- experimental study of the post-pile creep behaviour of irradiated Zry-cladding
- experimental study of the steady state fission product release from operational defected fuel rods under dry storage conditions
- experimental study of the transient fission product release during defection in the dry storage.

The evaluation of this experimental work will assess the result of the theoretical work already performed. In parallel to this more or less basic research demonstration test storages will be performed using CASTOR [4] shipping and storage caskes. The first demonstration test with BWR fuel assemblies from the Würgassen reactor will already start in December 1980.

4. CONCLUSIONS

In wet storage the fuel assemblies are stored under water at a temperature of 40 °C. The internal pressure load of the cladding tubes is only 10 % of the yield point. Cladding tube weakening due to corrosion may be practically precluded as well as the corrosion of the structural components of the fuel assembly which are made from high-grade steel and Inconel. The cladding tube loads arising during the handling of spent fuel assemblies also present no problems for Zircaloy aged during operation. The micro- and macro-distribution of the hydrogen in the fuel rods is adequately stable. The long-term storage experience available to date underlines the favourable storage behaviour of LWR fuel assemblies which may be extrapolated from the material behaviour.

A storage experiment has started in the Obrigheim nuclear power plant reactor to systematically determine the fuel assembly storage behaviour under wet storage conditions. 18 intact and 10 defective fuel rods from Obrigheim fuel assemblies with different levels of burnup were selected for storage. The first measurements within the scope of the storage experiments were performed

in December 1977. It was partially possible to compare them with earlier results gained with other aspects in mind. The eddy current test produced no new defect indications in the intact and defective fuel rods. The average fuel rod diameter remained unchanged within the scope of measuring accuracy. The visible cladding tube damage showed no essential changes as far as it was possible to see from TV-recording. A further examination of the stored fuel rods will be done in 1980. The examinations will be supplemented by measuring the oxide coating thickness.

Experiments were performed to determine the activity release form operational defective LWR-fuel rods during storage. The activity released originated almost completely from Cs-134 and Cs-137. The release rate depends first and foremost on the degree of cladding tube damage. The release rate of activity approaches to zero after a certain time. Total leaching my be affected by changing the ambient concentration of the fission products in the pool water. Further experiments are necessary for a better understanding of the mechanismes of fission products release.

Experiments were performed in the spent fuel pool of the Obrigheim nuclear power plant to measure the formation of hydrogen due to β/γ -radiolysis by a spent fuel assembly during wet storage. The test device was not the best possible due to the restrictions at the test site. Thus it is only possible to give an upper limit value for hydrogen formation, i.e. $G(H_2) \leq 10^{-3}$ H_2 molecules/100 eV, as a first test result. Nevertheless, this serves to emphasis that the official $G(H_2)$ value is too high for its use to claculate H_2-generation during wet storage of spent fuel.

The following mechanisms potentially endangering the integrity of fuel bundles during dry storage were considered in the estimation for periods up to 20 years long: corrosion of the Zircaloy cladding tubes from the outside and from the inside, H_2-absorption and thermal diffusion, stress corrosion cracking, crack propagation and creep deformation under internal pressure. For dry storage the insertion temperature limiting mechanisms is first and foremost the creep deformation of the cladding tubes as a result of their internal pressure. If one calculates assuming realistically decreasing storage temperatures, the insertion temperature should be in the range of 350 - 370 °C.

Long-term tests are being performed on unfueled cladding tube conditions at 100 °C and 300 °C and internal pressure loading. The specimens have different iodine concentration. Some of the specimens were additionally hydrogenated. Test periods up to 1500 days are envisaged. The specimens are helium leak-tested to check for failure after various test periods. After 400 days no defect occured. Storage experiments with complete spent fuel bundles and with single rods will assess the results of the theoretical studies performed.

REFERENCES

[1] H.P. Fuchs
Brennstabauslegung, Brennelementbetriebserfahrung
VGB-Kernkraftwerksseminar 1976

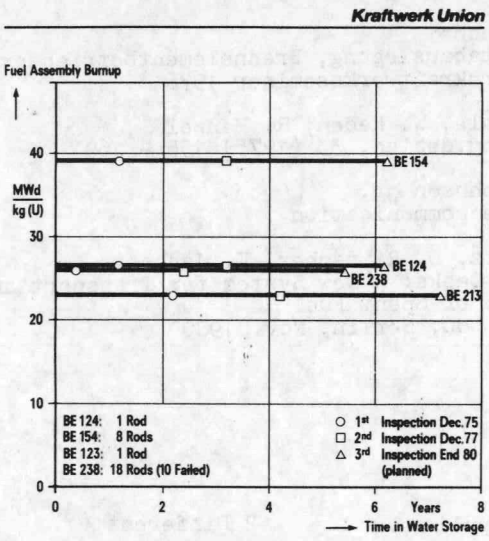
[2] H. Stehle, W. Kaden, R. Manzel
nucl.eng.design, 33 (1975) 155 - 169

[3] A.B. Johnson jr.
private communication

[4] H. Baatz, D. Rittscher, K. Janberg
CASTOR-Casks: A New System for Transport and
Storage of Spent Fuel
PATRAM '80, Berlin, Nov. 1980

Cladding Supplier	:	2 Different
Annealing	:	Standard
		Recrystallization
Additional Treatment	:	No
		Hydrating
		Autoclaving
Hoop Stress	:	62 N/mm^2
		95 N/mm^2
Temperature	:	100 ^oC
		300 ^oC
Iodine Invetory	:	None
		$1 \cdot 10^{-6}$ g/cm^2
		$3 \cdot 10^{-6}$ g/cm^2
		$1 \cdot 10^{-5}$ g/cm^2

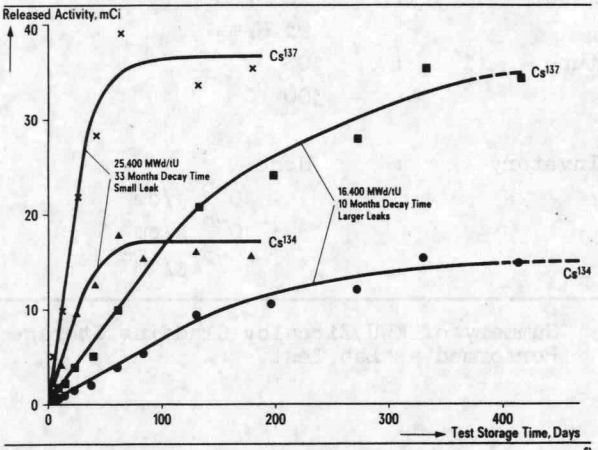
Tab. 1 Summary of KWU Zircaloy Cladding Storage Program, Performed as Lab Test



Summary of KWU Spent Fuel Surveillance Program in KWO Reactor Pool

E 801263 e

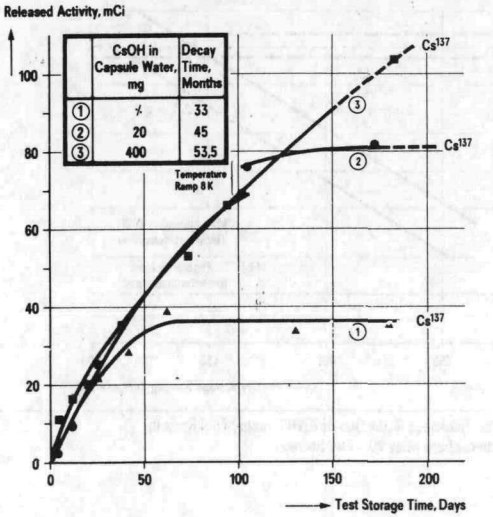
Fig. 1



Fission Product Release from a Single Failed Fuel Rod during Wet Storage in a Capsule

E 801265 e

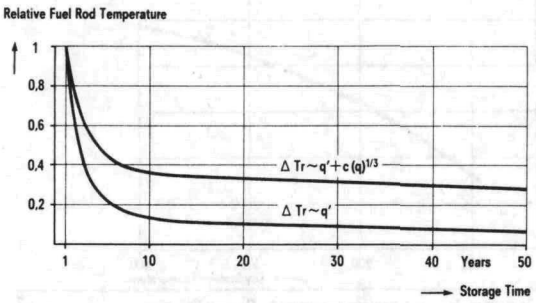
Fig. 2



Influence of Surrounding Cesium Concentration on Cs – Release from a Failed Fuel Rod during Wet Storage

E 80 1266 e

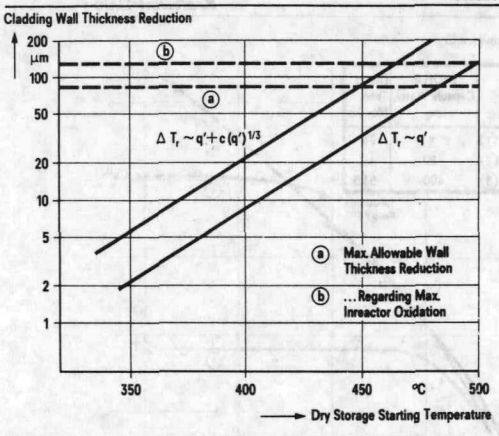
Fig. 3



Temporal Fuel Rod Temperature Decrease during Dry Storage According to Two Different Models

E 80 1262 e

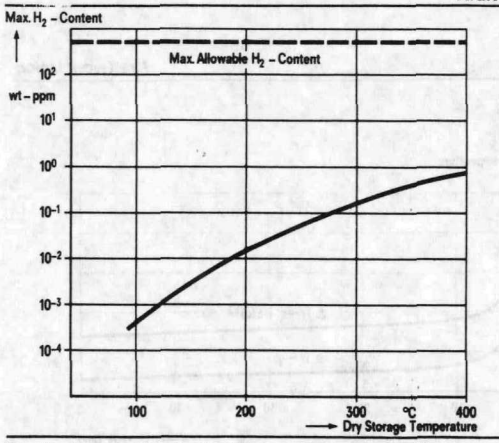
Fig. 4



Cladding Wall Thickness Reduction of PWR Zircaloy Fuel Rods in Oxidizing Atmosphere after 20 y Dry Storage

E 80 1268 e

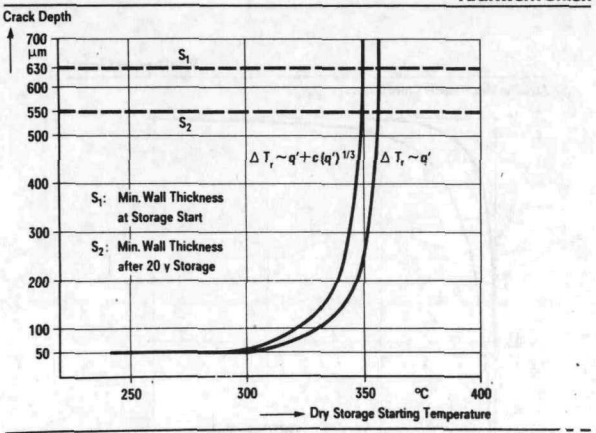
Fig. 5



Max. H_2 – Content of Zircaloy Cladding According to Thermal Diffusion in an Axial Temperature Gradient of 10 K/m at Isothermal (I) Storage

E 80 1264 e

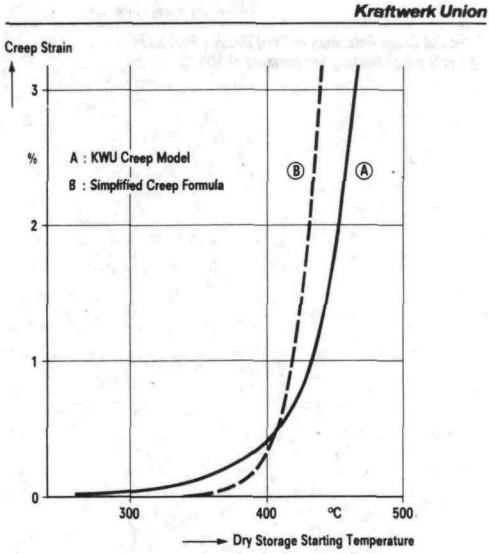
Fig. 6



Max. Crack Depth after 20 y Dry Storage of PWR Zircaloy Fuel Rods with Initial Crack Depth of 50 µm and Inside Pressure of 80 bar (at 350°C)

E 80 1269 e

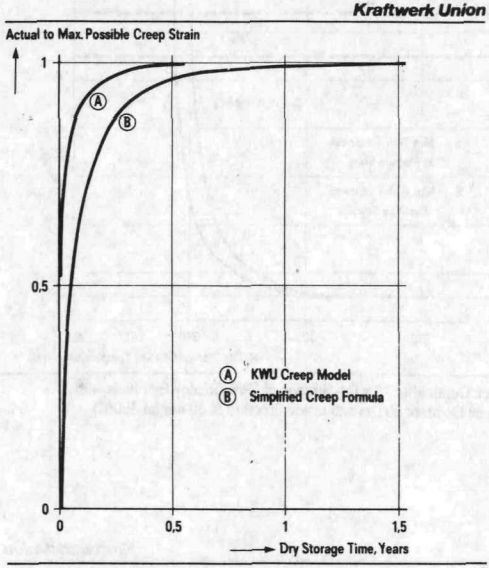
Fig. 7



Diametrical Creep Strain of PWR Zircaloy Fuel Rods with 100 bar (at 350°C) Inside Pressure after Infinite Dry Storage Time

E 80 1267 e

Fig. 8



Temporal Creep Behaviour of PWR Zircaloy Fuel Rods
at Dry Storage Starting Temperature of 400 °C

Fig. 9

DISCUSSION

Question by W. Heudorfer, FR Germany: Due to the use of fuel elements in the nuclear reactor, the pellets show a certain grain size distribution. Is this grain size distribution changed by storage? Are measurements carried out?

Answer: In the dry storage conditions expected the fuel does not affect the intact fuel rods. To answer your specific question, I would like to say that the fuel temperature is so low that there [redacted] no change in grain size. An increase in grain size is only ex-[redacted]ted in the case of temperatures above 1000°C.

Comment by J. Fleisch, FR Germany: Commissioned by the US DOE, high temperature fuel rod tests were carried out at the Battelle Columbus Laboratory at 500°C, to finally verify model calculations. It was found that even in the case of longer dry storage under isothermic storage conditions (500°C) there were no fuel rod defects. The integral creep rate is 1.5-2 % and amounts to only a small part of the estimated 8-10 %.

Question by J. Fleisch, FR Germany: What conservative and pessimistic assumptions are taken as a basis for the KWU model calculations and how can it be explained that the experimental results are decidedly more favourable - and therefore higher storage temperatures will be permitted - than the existing calculations predict.

Answer: The creep of Zircaloy rods depends on many factors. For example, a very decisive factor is the heat treatment of the Zircaloy rods at the end of the fabrication process. It is possible that in the experiment you mentioned, rods were used which had not been finally annealed at low temperatures, which then hardly recrystallised and are then very creep tight. Furthermore, evaluation requires, for example, that the inner pressure of the fuel rods and their wall thickness is known.

Our assumptions are explicitly given. The calculations are carried out for fuel rods with KWU-DWR standard dimensions (in each case the more unfavourable tolerance value) with a cold inner pressure of 47bar (up to now a maximum of 35 bar measured for standard fuel rods) and using both creep formulae mentioned. These are gauged to experimental data obtained, in particular, the KWU creep model to all published creep data of Zircaloy of different manufacture.

In so far it is obvious that, depending on material type, some of the irradiated fuel rods will survive a higher temperature than 400°c [redacted]uring dry storage. Our aim is to find a temperature limit where [redacted]actically all LWR fuel rods - irrelevant of manufacturing differenes [redacted]remain intact during dry storage.

We do not think that it should be attempted to start the dry storage procedure with a maximum storage temperature. A possible large amount of damage would seriously affect the dry storage. We suggest that only when more experience about dry storage is gained should the temperature limit be tested. This seems to the way which is most favourable to the project of dry intermediate and long-term storage.

Question by J. D. Hart, United Kingdom: We heard from Dr. Klingensmith of the oxidation of UO_2 resulting from the transport of damaged fuel in dry air. From your studies can you put a temperature limit on the fuel for transport when it is damaged and exposing UO_2?

Answer: One aim of our study was to find a temperature limit at which, for all types of LWR fuel elements with Zircaloy rods, the fuel rods would remain leaktight during dry transport or dry storage. Our results show that this value is about 400°C as mentioned in the paper.

However, no temperature threshold is given whereby exceeding this threshold would lead to defects. Only in the case of particularly unfavourable parameter combinations, e.g. DWR fuel rod with almost total fusion gas release, particularly high pre-corrosion, wall thicknesses at the lower tolerance limit, etc., could defects occur by exceeding the temperature threshold.

As I said, it was our aim to find a temperature limit below which under all circumstances intact fuel rods remain intact during transport. If already damaged fuel rods are being transported, oxidation of the free UO_2 occurs sometimes. Dr. Klingensmith has reported on oxidation due to ambient air - not dry air. By filling the transport cask with inert gas, for example, this can be excluded.

1500 t-U SPENT FUEL ELEMENT STORAGE PLANT AHAUS

H. Malmström, G. Schroeder / H. Klöpper, A. Lührmann
STEAG Kernenergie GmbH / Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen

1. INTRODUCTION

On September 28, 1979 the Federal and States Governments in Germany have passed a resolution on spent fuel management. As a result the: "Grundsätze zur Entsorgungsvorsorge für Kernkraftwerke" (Principles of spent fuel management for nuclear power plants) of May 6, 1977, have been revised, effective since February 29, 1980. The revision takes into account the developments of the german integrated spent fuel management center concept. In this context the intermediate storage of irradiated fuel elements is regarded as an independent step of spent fuel management.

Since 1977 "Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen" (DWK) and STEAG Kernenergie GmbH (STEAG) jointly prepared the construction of a 1500 t-U spent fuel element storage plant in Ahaus, Northrhine-Westfalia. In January 1978 both companies submitted an application to the competent authority, the Physikalisch-Technische Bundesanstalt (PTB), Brunswick, according to the regulations of § 6, Atomic Law. At that time the concept was based on the wet technology, i.e. storage in water filled pools, protected by a thick wall concrete building.

2. CONCEPT DEVELOPMENT

Since 1977 DWK and STEAG jointly in cooperation with their affiliate "Gesellschaft für Nuklear-Service" (GNS) developed a new type of spent fuel shipping cask using nodular cast iron as a new cask material. This development, in the meantime completed to a large extent, has lead to various cask types called CASTOR. This concept allows for a reduction of the manufacturing time for large shipping casks by more than one half compared to those casks based on the technology in use so far.

Furthermore it is possible to manufacture these casks in series with a high quality standard, thus reducing considerably the manufacturing costs. As a result of these specifics the development of these casks has been extended towards meeting the requirements for spent fuel element storage. In 1979 the site-independent planning work for a spent fuel storage plant based on storage in shipping casks was completed.

The german Reactor Safety Commission has examined this concept and has summarized its position during a session on September 19, 1979 by the statement, "that the safety requirements for transport and dry storage of spent fuel elements in shipping casks can be fulfilled, and therefore dry storage is representing one possible

form of interim storage of spent fuel elements. By this, two alternatives realizeable to meet the safety standards (dry storage and conventional wet storage) are available for interim storage of spent fuel elements."

As a result of the successfully accomplished development, DWK and STEAG have decided to use this technology for their joint fuel storage project in Ahaus. Therefore the license application with PTB for the Ahaus project originally based only on the wet storage concept has been supplemented in October 1979 by dry storage. DWK and STEAG, however, have declared that only one of both concepts will be realized, and have asked the licensing authorities to give first priority to dry storage. This priority is a result of the following specific properties of this concept:

- Storage unit is the cask; due to its short manufacturing time the storage capacity can be adjusted to actual requirements.
- The fuel elements are safely confined in a type-B-package. No radioactive material will be released.
- Short construction time of the plant (about one year; considerably less than with wet storage).
- Easy plant decommissioning as no contaminated systems or parts will have to be removed.

3. SAFETY CONCEPT

The main characteristic of cask storage is that all essential safety functions are assured by the cask itself. It warrants that during normal operation and accident situations

- safe retention of the radioactive materials will be maintained,
- criticality will not occur,
- the decay heat can be transferred to the environment, and
- radiation shielding will be assured.

The suitability of the casks to meet these requirements has already been thoroughly examined during the licensing process for transportation. As other papers deal with the cask technology, this paper will only discuss storage-specific aspects.

An interim fuel element storage facility is a stationary nuclear facility and subject to the relevant licensing specifications and guidelines. The cask as a storage module of the nuclear facility is subject to the following additional safety requirements exceeding the IAEA regulations:

a. A two-barrier containment with monitoring of the interspace has to be foreseen, and

b. the resistance against those external impacts as foreseen for nuclear reactors has to be assured.

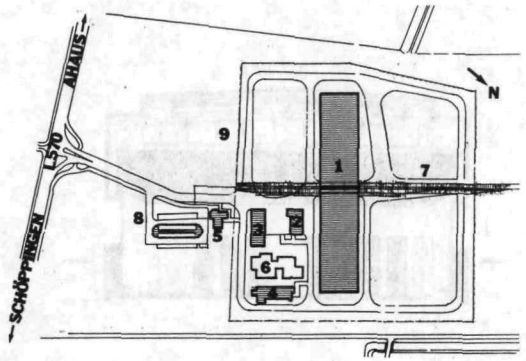
Ad a: The two-barrier containment is warranted by a multiple plug system with continuous leak tightness monitoring. The massive cask body itself represents a safe barrier due to its wall thickness (over 400 mm), material properties and quality assurance. This has been confirmed by the Reactor Safety Commission.

Ad b: Aircraft crash is the only external impact not covered by the tests for the transport license. The casks are specially designed to withstand aircraft crashs. This has been confirmed by tests in which a 1000 kg missile has been impacted onto full-size cask models at about sound velocity.

Furthermore long term aspects for the cask and its components - e.g. the seal rings - have to be considered. The fuel element behaviour does not mean any limiting factor to the storage period, as the casks remain leak tight under all circumstances. Nevertheless it is of interest with regard to unloading conditions. Taking into account the decay time of a minimum of one year the fuel cladding temperature will be limited to data which do not lead to additional damage.

4. STORAGE FACILITY

The size of the site at Ahaus is about 180,000 m^2. The casks normally will be transported to the site by rail. The capacity of the storage facility will be 1500 t-U. The storage building will have positions for 420 casks. 60 % of the fuel will be from PWRs, 40 % from BWRs. As shown in the following layout the facility comprises various buildings.



Plant Layout

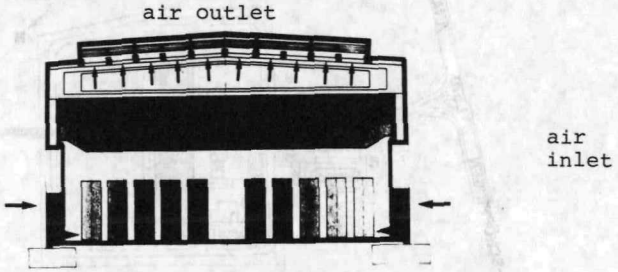
1. Storage building
2. Auxiliary building
3. Workshop building
4. Administration building
5. Guard house
8. Parking area

The storage building will be about 38 m wide, 200 m long, 20 m high. It comprises the receiving and the storage areas. In the maintenance room of the receiving area the casks will be prepared for storage by testing the seal system and installation of the leak tightness monitoring transducer. Furthermore in the maintenance room in case of a leak-signal the necessary measures will be taken on the casks in order to reassure the two-barrier containment. Also in such a case no release of radioactivity occurs.

Each of the 420 storage positions in the storage areas will be linked to the leak tightness monitoring system. A 140 t crane will cover the total building area. The building does not have a safety function, however, it contributes to further reduce the external radiation dose.

5. HEAT REMOVAL

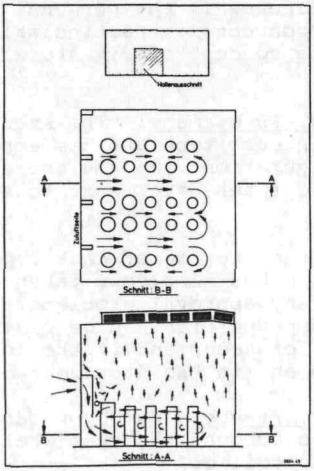
Under conservative design assumptions, and after complete loading of the storage facility within three years, a maximum decay heat of about 8 MW will have to be removed. Heat removal from the cask surfaces and from the storage building will take place by natural convection without technical equipment. Air heated up by the cask surfaces will leave the building through openings in the roof, cool air will flow into the building through openings in the side-walls. The cross section figure below shows the principle of the heat removal from the building.



Storage building (cross section)

The flow pattern and the temperature distribution within the building will be different from those around a free-standing cask. As the flow and temperature profile cannot be determined by calculation, experimental tests have been undertaken at the STEAG affiliate Kessler & Luch at Gießen.

The experiments were done on a 1 : 5 scale part of the building with electrically heated cask models. The experiments showed that the flow reached all casks almost uniformly leading to surface temperature differences between equally heated casks of a maximum of 6 K. This is mainly due to the fact, that the air flow returns from the central gangway, thereby cooling the inner casks. This can be seen on the figure below.



Air Flow Pattern

The cask surface temperature inside the fully loaded building in the worst case is 10 K higher than on a free-standing cask. The air will be heated up by about 16 K.

A disturbance of the heat removal can occur if, in case of an external event, the storage building will be destroyed and natural convection will be prevented by debris covering the casks. The effects of such an event have been determined by simulation tests, in which the debris-coverage of a cask-section has been simulated by thermal insulation. Due to the large heat capacity of the system the temperature increased very slowly and reached a new stable state after a few days. In the worst case the temperature inside the cask increased by 50 K, which will not lead to damage of the cask or its contents.

DISCUSSION

Question by B. G. Pettersson, Sweden: Based on assessments of occupational exposure for the two spent fuel storage options - wet and dry - which option would you prefer? I am also interested in the reasons for preferring one particular option.

Answer: Both storage concepts have to meet the requirements of the radiation protection regulations. In both cases the major part of the radiation load during storage and discharging of the casks results in no major difference in the personal dose rate. Because of this, neither of the concepts is particularly preferred. The reasons why we prefer the concept of dry storage in casks are given in the paper.

Question by H. Pirk, FR Germany: The Ahaus dry storage facility is designed for a certain cask type and has applied for the approval procedure for this configuration. Do you see any problems concerning approval should you, at a later date, want to store larger storage containers at Ahaus?

Answer: Basically the storage of cask types which are not part of this approval procedure has also been taken into account. We therefore do not expect any approval problems, if for such containers at least the same safety properties can be proved as in the case of the types which are now foreseen, and if the containers are compatible with the technical equipment in the storage facility.

Question by D. Haon, France: Have you carried out a study of this storage to take into account the case where an earth tremor would cause two stored casks to collide?

Answer: The effects of an earthquake are examined as part of incident investigations. The mechanical loads due to earthquakes are considerably smaller than those due to drop tests for type B approval. Moreover, the casks are stable with regard to the established earthquake accelerations for this site.

With regard to the criticality it can be ascertained that even in the case of an unlimited number of casks stored very closely together, a chain reaction is prevented. Heat dissipation could for a certain time be reduced by an earthquake which leads to destruction of the storage facility and burial under rubble. This type of incident has been specially investigated and is mentioned in the paper.

Question by E. Schlich, FR Germany: Do you have, and if yes what kind of, supervision of radiation and contamination of spent air from the storage facility?

Answer: Because of the cask construction, release of radioactive aerosols or gases is not possible. Therefore continual supervision of the spent air with regard to radiation and contamination is not foreseen. For routine inspections, mobile measuring equipment is available.

TYPE B CASKS FOR THE INTERMEDIATE STORAGE OF SPENT FUEL ELEMENTS IN THE FEDERAL REPUBLIC OF GERMANY

J.-P. Masslowski, R. Trapp
Bundesanstalt für Materialprüfung (BAM)
Unter den Eichen 87, D-1000 Berlin 45

Owing to the growing number of spent fuel elements and the delayed decision to build a national decommissioning centre in the Federal Republic of Germany, planning experts are being forced to look for a medium and long term solution to the problem of decommissioning spent nuclear fuel.

The various possibilities for decommissioning can be seen in the following table.

SOLUTIONS TO THE PROBLEM OF DECOMMISSIONING

1) Short term decommissioning from nuclear power plants

a.	b.	c.
Reprocessing contracts	Wet storage pools with compact storage facility in nuclear power plants	Small dry storage facility on the nuclear power plant site

2) Medium term decommissioning up to 30 years

Construction of intermediate storage facilities

a.	b.
Wet storage pools	dry storage facilities (cask storage)

3) Long term storage of spent fuel elements up to 50 years

Large long term intermediate storage facility (dry storage)

4) Final processing of spent fuel elements

a.	b.	c.
Reprocessing in a decommissioning centre	Reprocessing in a demonstration plant	Conditioning of the fuel elements without reprocessing

5) Final storage

a.	b.	c.
Low and medium level waste	High level waste	Direct final storage of spent fuel elements

Point 1 shows the possibilities for short term decommissioning of spent fuel from nuclear power plants. These are:

a) Reprocessing of spent fuel elements in appropriate facilities

b) Wet storage in compact storage facilities in nuclear power plants

c) Dry storage in casks on nuclear power plant sites

Point 2 shows the possibilities for medium term decommissioning. Intermediate storage of the spent fuel elements

a) in conventional wet storage pools

b) dry storage in casks.

If reprocessing is not possible by the end of the medium storage term, then a long term storage facility may be used in order to vacate the medium term facility (point 3).

Final processing of the spent fuel elements should be carried out at the latest at the end of the long term storage period, as can be seen under points 4 and 5.

This paper concerns itself with point 2, intermediate storage, and, following a short description of methods, transport and/or storage casks will be dealt with in more detail.

Intermediate storage in wet storage pools means that the spent fuel elements are positioned in water-filled pools. The water acts as a shield against neutron and gamma radiation and as a coolant whereby the decay heat is fed through a secondary circuit cooling tower. To protect against the effects of earthquakes, explosions, fire and aircraft crashes, the safety-relevant parts, for example the building and the pool must be appropriately proof and the reliability of all safety installations redundantly ensured.

Intermediate dry storage provides for the loading of the spent fuel elements into transportation casks, after the required decay period of about one year in the nuclear power plant, and shipment to storage facilities. While in storage the fuel elements remain in the casks which then assume the function of storage casks. The storage building itself acts as a protection only against the weather. Each individual cask for operational loads must then be designed to protect against both natural incidents and those of man-made origin.

Both concepts, intermediate wet storage and dry storage, have been studied by the "Reaktor-Sicherheitskommission" (Reactor Safety Commission) and, with regard to their safety-technical aspects, have both been found to be possible.

CASK DESCRIPTION

During the last few years a new generation of casks for radioactive materials has been developed, which are intended on the one hand for transportation purposes and on the other for use in dry storage.

These casks have wall structures which, contrary to usual construction principles such as steel-shrouding, lead shielding and special heat insulation layers, are made of ferritic nodular cast iron. Wall thicknesses of more than 400 mm ensure integrity, even in the event of extreme mechanical loads. A considerable decrease in radioactive radiation is likewise achieved, especially as the cask is equipped with neutron-absorbing material in the wall, base and lid regions.

The cask cavity and also the seal surfaces in the lid region are protected against corrosion by rust-proof metallic coatings.

Metal gaskets seal the multiple lid system according to the "two barrier principle". During the storage period continual monitoring of the excess pressure between the primary and secondary lid takes place with the help of a pressure box, so that failure of one or both of the barriers can be quickly detected.

If leaktightness of the secondary lid fails, the lid is restored to its former condition by changing the seal. In the event, however, of the primary or primary and secondary lid not being leaktight, leaktightness is doubly ensured by inserting an additional so-called welded lid.

The decay heat is released into the environment by natural convection via numerous integrated cooling fins.

CASK CONCEPT REQUIREMENT

Transportation and storage both make various types of demands on the cask design. The cask must therefore pass numerous tests in order to demonstrate fulfilment of cast design requirements.

As far as transportation is concerned, it is necessary to demonstrate that the cask is capable of withstanding accidents during shipment.

As far as storage is concerned, it must be shown that fixed threshold values are adhered to during normal operation and that in the case of incidents, there is no release of the radioactive contents of the cask so that further operation of the storage facility is possible.

In other words this means that in each case

- integrity of the cask must be guaranteed
- the radioactive radiation must remain sufficiently shielded
- the decay heat must be dissipated from the cask cavity
- subcriticality must be guaranteed.

TESTS

In the Federal Republic of Germany, the "Bundesanstalt für Materialprüfung (BAM)" is the competent authority responsible for testing the type B characteristics of packagings. In the BAM, testing facilities are available which enable full-scale casks of up to 100 t mass to be tested.

In more detail, testing a cask to demonstrate its suitability for shipment purposes includes the well-known 9 m drop tests on to an unyielding target laid down in the regulations, the 1 m drop test on to a mild steel cylinder and an accident fire test of half-hour duration at a temperature of 800°C. The cask must withstand the different loads involved. At least one barrier must remain leaktight. The shielding must not be below certain defined threshold values.

The testing of casks for storage purposes is not explicitly laid down in regulations to the same extent as for shipment as yet. As a starting point, possible damaging incidents which could take place in a storage facility form a basis for test requirements.

The following can be regarded as incidents: earthquakes, fires, aircraft crashes, burial of the cask under rubble, crashes during transportation within the storage facility, effects of harmful substances, chemical reactions, radiation embrittlement and corrosion.

Incidents such as earthquakes, fire, as well as crashes during transportation within the storage facility are assumed to be covered by the type B tests.

Resistance to corrosion, radiation embrittlement and chemical reactions, together with the effects of harmful substances, must be included in quality assurance and/or QA-planning which will be mentioned again later. It therefore remains to be established whether the protecting function of the cask is maintained in the event of an aircraft crash or burial under rubble.

An aircraft crash on to a cask can be sufficiently simulated by a bombardment test using a projectile. Dimensions, mass and speed of the projectile must simulate the actual conditions during the impact of an aircraft. The most endangered cask region must be taken as the impact point.

Burial of the cask under rubble, for example in the event of the storage building collapsing, can be simulated by a thermal test, i.e. preventing heat dissipation using insulating material, whereby the source heat must be determined by the maximum possible heat load. The temperatures must be in equilibrium by the end of the test.

The results of the accident simulations described should not affect the protective function of the cask to such an extent that the criteria already mentioned such as integrity, shielding, sub-criticality or decay heat dissipation are not fulfilled.

A further, important aspect of the casks described, both for shipment and storage, is quality assurance.

Quality assurance must accompany the cask concept during design, during production of the serial model, when the cask is put into use and further during maintenance and recurrent testing.

Mechanical and thermal resistancy, leaktightness and long term behaviour of materials used must fulfill the safety criteria.

An inventory and record of the production and test sequence procedure should then be established which forms a basis for all material cask component requirements, their production and assembly.

In an operation and maintenance guide, the handling procedure defined in the regulations, such as loading and unloading, sealing and renewal of the seals, must be recorded.

Recurrent testing of transport casks and monitoring measures in the storage facility should ensure at all times the necessary safety-technical conditions (type B(U) characteristics), necessary for the casks to survive incidents and accidents.

In conclusion, it should be pointed out once again that the operation of nuclear power plants in the future can only be assured if the problem of decommissioning is satisfactorily solved. In this respect the intermediate storage of spent fuel elements in shipment/storage casks can make a decisive contribution.

DISCUSSION

Question by A. Olivieri, Italy: I would like to know the maximum pressure and the minimum temperature expected to occur during the storage period?

Answer: The inner pressure of the cask is normally 0.8 bar. The area between the primary and secondary lid has a pressure of 6 bar. According to regulations the temperature of the accessible cask surfaces should not exceed 82°C during transport. During storage temperatures of up to 110°C are expected, under unfavourable parameters. This only applies to newly stored casks containing fuel elements with maximum permissible decay heat.

Question by E. Schlich, FR Germany: Is there a difference in the definition of integrity of a cask for the type B (U) approved cask and for a cask approved for storage?

Answer: There could be some differing definitions of integrity. On the one hand, release from a type B(U) cask is, according to traffic law, limited to $A_2 \times 10^{-6}$/h (per radionuclide) and to threshold values for surface and 1 m dose rate. On the other hand, for a storage cask, the threshold values of the whole facility are given in the appropriate paragraphs of the German radiation protection ordinance.

SESSION XVI
OPERATIONS AND SYSTEMS PLANNING

The U.S. Department of Energy's Nuclear Materials Transportation Technology Development Program

Richard B. Chitwood
U.S. Department of Energy, Washington, D.C., U.S.A.

INTRODUCTION

Shortly before the previous PATRAM Conference in early 1978, the Department of Energy (DOE) established a new transportation program to provide for the coordination and direction of nuclear materials transport-related activities in the U.S. Existing nuclear transport activities within DOE were merged into this program to provide a nuclear materials transport focal point within the Department. Since its inception this program has been functioning well. In fact, one manifestation of its success is the extent of participation by U.S. authors at this Symposium. There are 37 papers being presented here which are a direct outgrowth of the DOE program, and there are 10 more which have derived at least partial technical support from the program. Thus approximately 30% of the total presentations at PATRAM 80 derived technical support from the current DOE transportation program.

As significant as the program output has been, much work remains in forging that vital link in the nuclear energy option--transportation of nuclear materials. It is the view of the Department of Energy that under present conditions of uncertainty in regulatory constraints, and of uncertain economic incentive in the private sector, the government must provide the leadership and the support necessary to assure that transportation capability and safety needs will be satisfied in the United States. The responsibility for overall planning and direction of transport functions has been assigned to the Transportation Division in the Office of Transportation and Fuel Storage, under the Assistant Secretary for Nuclear Energy.

In order to meet the need to develop a national level capability to transport all forms of nuclear materials in a safe, reliable and economic manner and cope with the growing complexities of regulation in all its forms, the Sandia National Laboratories were designated as the lead contractor for detailed implementation of the Department's nuclear materials transport program.

This organizational structure, providing both a Headquarters Transportation Division and a national laboratory as the lead transportation center, now constitutes a single focal point for DOE nuclear materials transportation. Manifest in this structure is the acknowledgment that transport of nuclear materials is an indispensable part of the nuclear power option and other nuclear materials applications, both under government aegis and in the private sector. Further, despite the expanding need for transporting nuclear materials over the past several decades, we now approach a period of even more rapid expansion in the 1980s.

In this paper I shall discuss nuclear materials transportation activities in the U.S., the issues confronting us, and how the DOE program is designed to meet the challenge of today and the future.

Radioactive Materials Transported

Nuclear transport within the U.S. can be broadly divided into three categories. By far the largest of these, by number of packages involved, is the shipment of medical isotopes, industrial sources, and radioactive research

materials, and the waste associated with these activities. The second largest involves the movement of radioactive materials in the commercial nuclear power cycle. At present these shipments involve such materials as uranium ore, yellow cake, UF_6, contaminated clothing, demineralizer resins, evaporator bottoms, and new and spent reactor fuel.

The third broad category of radioactive materials to be transported are those wastes associated with the U.S. government nuclear programs. These currently represent a very small fraction of the total and differ from those generated by the first two categories in that they contain transuranic nuclides. They range from contact-handled transuranics to solidified high level materials. Because some government produced wastes are currently being stored in locations considered unsuitable for long-term confinement, the need to transport these materials from their existing locations to interim or final storage exists and is independent of any commitment to use specific waste repository locations. In addition to these wastes, the decision to defer reprocessing has developed a need to move spent fuel from reactor stations to either intermediate (away-from-reactor) storage facilities or to other storage pools at other reactor sites. Since--as many of you know--President Carter has offered to accept limited amounts of spent fuel generated outside of the U.S., our transportation requirements will intensify accordingly.

Program Organization

As mentioned earlier, the U.S. Department of Energy (DOE) established in the fall of 1978 the Transportation Technology Center (TTC) at the Sandia National Laboratories (SNL) in Albuquerque, New Mexico. Figure 1 shows the Organizational relationships within the Department of Energy as well as the TTC. The charter of this technology center was "to establish a DOE organization for nuclear materials transportation technology and systems development, operations, and planning, that will develop a national level DOE capability to cope with the growing complexities of transporting all forms of nuclear material in a safe, reliable, and economic manner consistent with national policy and needs."

This very broad charter embodies a number of judgments made during the early stages of program planning. The most important of these was the judgment that industry within the U.S. and elsewhere is able to solve the technological problems involved in the movement of nuclear materials. A second judgment was that private industry's capability is not now being fully utilized within the U.S. because of uncertainties in the marketplace and in regulation. The DOE Program is therefore aimed at fostering technological and regulatory stability in the transport industry. The third judgment was that there needs to be a dialogue established between the DOE and the nuclear and transport industries to allow for the exchange of ideas and complementary solutions of problems, with a cooperative effort on the part of both parties.

We intend that any program developed should provide the necessary incentive to the industry without the need for subsidies while at the same time avoiding infringement upon its prerogatives. The DOE itself has a significant interest in certain program activities, such as the National Waste Terminal Storage Program and spent fuel storage, which involve transport. So the TTC should provide some contingency capability for the Department to have transport service for radioactive materials available should commercial suppliers be unwilling or unable to meet DOE needs.

In order to meet the goals contained in the charter and to build a program around the principles just covered, the program at the TTC is divided into four principle areas as shown in Figure 2.

Systems Development Area

The program is most easily understood when one views the Systems Development area as the focal point. The Systems Development area covers activities organized in a manner corresponding closely to the nuclear waste management program activities within the U.S. DOE. Funding for the transport work comes from each of the programs being served.

The Defense Waste activity encompasses the development of systems for transporting contact-handled and remote-handled transuranics, military spent fuel, and the high level wastes from reprocessing spent fuel.

The Domestic Fuel Cycle specifically addresses those transport systems for [illegible]ing radioactive wastes which are identified with the current domestic fuel [illegible]cle within the U.S. These would include systems for transporting spent fuel to away-from-reactor (AFR) storage sites and eventually include systems for transporting low-level wastes to shallow disposal sites. There is close cooperation between Domestic Fuel Cycle Transportation Systems activities, Systems Development, and the programs of the Office of Nuclear Waste Isolation coordinated by Battelle Memorial Institute.

In the advanced reactor area there are Systems Development activities under way to provide for spent fuel from the FFTF and the proposed commercial breeder. Although not an active program at this time, it is this area which would address transport related to alternate fuel cycles, such as the proliferation resistant cycles investigated by INFCE.

The International Fuel Cycle activity includes many different interests. The most obvious of these interests is President Carter's announced program to receive spent fuel of American origin from reactors in other countries in support of nonproliferation objectives. The International Fuel Cycle area also addresses activities such as the transport systems for possible seabed disposal.

Supporting these four specific systems areas is a Logistic Assessments activity designed to optimize transport system design for maximum operational economy and availability and thereby serve as a long-range planning tool on a National scale.

In addition to the Systems Development work areas described above and shown in Figure 2, two other systems development activities were recently undertaken by the TTC to support programs assigned to DOE by Congressional initiative. One pertains to safety research associated with the Three Mile Island recovery activities and the other to the high level waste disposition activities at West Valley, New York.

Regarding the Three Mile Island (TMI) recovery research, DOE has recently initiated a program to examine the management of high-activity dewatered resins arising from nuclear plant operations. This work will have an immediate applic[illegible]bility to the handling of high-activity dewatered resins being produced by [illegible] EPICOR II system at TMI. The TTC has been assisting in the program in [illegible]nnection with the transportation of the resins off TMI to a storage/disposal site. Key issues in this TTC work are whether spent resins in their liners should undergo any treatment, processing, or modification (such as spent resin solidification) prior to being transported, and whether the package or overpack to be used in transport will also be used in storage or disposal. The possible options for spent resins management are under investigation and the TTC is supporting this work regarding transport aspects.

Relative to the West Valley Waste Disposal activities, DOE has instituted a program to examine the prospect of solidifying the high level liquid waste (HLLW) at West Valley. In connection with this effort the Department has assigned to SNL-TTC the responsibility of providing for the nuclear materials transport needs of this program. The initial task of this transportation work is the evaluation of the Fort St. Vrain cask for transporting samples of waste from West Valley to Hanford for characterization. After this initial work is completed, the TTC will proceed, through a subcontract with General Atomics in San Diego, with the design, approval and fabrication of casks to transport West Valley solidified HLLW. The GA work is also being coordinated by the TTC for applicability to other DOE high level waste and spent fuel transportation needs.

Technology Areas

The remaining three areas shown on Figure 2 are intended to either support the Systems Development areas or to communicate the results produced in that activity. By far the easiest association among these activities to understand is the connection between Technology and Systems Development. The TTC's technology activity involves programs directed toward the understanding and unification of methodologies used for the development of specific systems. This includes the benchmarking of thermal analysis and structural analysis codes, the investigation of packaging interfaces and the effect that such interfaces have on system operations, and the investigation of the uses of codes and standards. A second technology activity involves the development of information such as materials property data that might be useful in forming the basis for package approvals. A third activity results from past work and involves the utilization of structural and thermal codes to analyze the behavior of transport systems and the subsequent testing of these systems at either full scale or subscale in order to validate the accuracy of such code work. It is this full scale test program that has attracted the most public attention to the TTC program. A fourth Technology activity involves the investigation of the effects of intentional attack on packages while in transit. It is in this area that hard technical information is needed most urgently since most analyses of the results of intentional attacks upon nuclear shipments have been based on conservative assumptions which have been made to compensate for uncertainties. This leads to over estimates of the environmental impacts of such attacks.

Institutional Issues

We believe that the technology base for the design and development of packagings for radioactive materials transport can be developed to a point adequate to satisfy the needs of Systems Development activities. The Systems Development activities can produce competent packagings for transporting radioactive materials. However, the ability to actually move these materials is still dependent upon other factors as well. These other factors have been identified in the structure of the TTC as Institutional Issues and have been broken down into four categories.

The initial goal of the Institutional Issue activity is to identify those instances where public policy, regulation, or the actions of public groups stand in the way of successfully transporting radioactive materials. A second factor in the Institutional Issues area is the accommodation of divergent social and organizational viewpoints. These dissenting views are not always capable of being resolved but do need to be considered and therefore identified. A third activity under the Institutional Issues areas which is receiving additional emphasis is that of risk assessment. Efforts are under way to develop comprehensive, and yet understandable, methodologies for calculating risk to man and

communicating the results to the general public and policymakers. Under U.S. law, it is required that each major activity of the society be evaluated for its possible environmental impact. This activity hypothesizes the release of radioactive materials during transport, usually independently of the probabilities involved, and then investigates the possible impact upon the public. Until recently there was little consistency in these environmental impact assessments involving transportation. The effort has now been consolidated at the TTC.

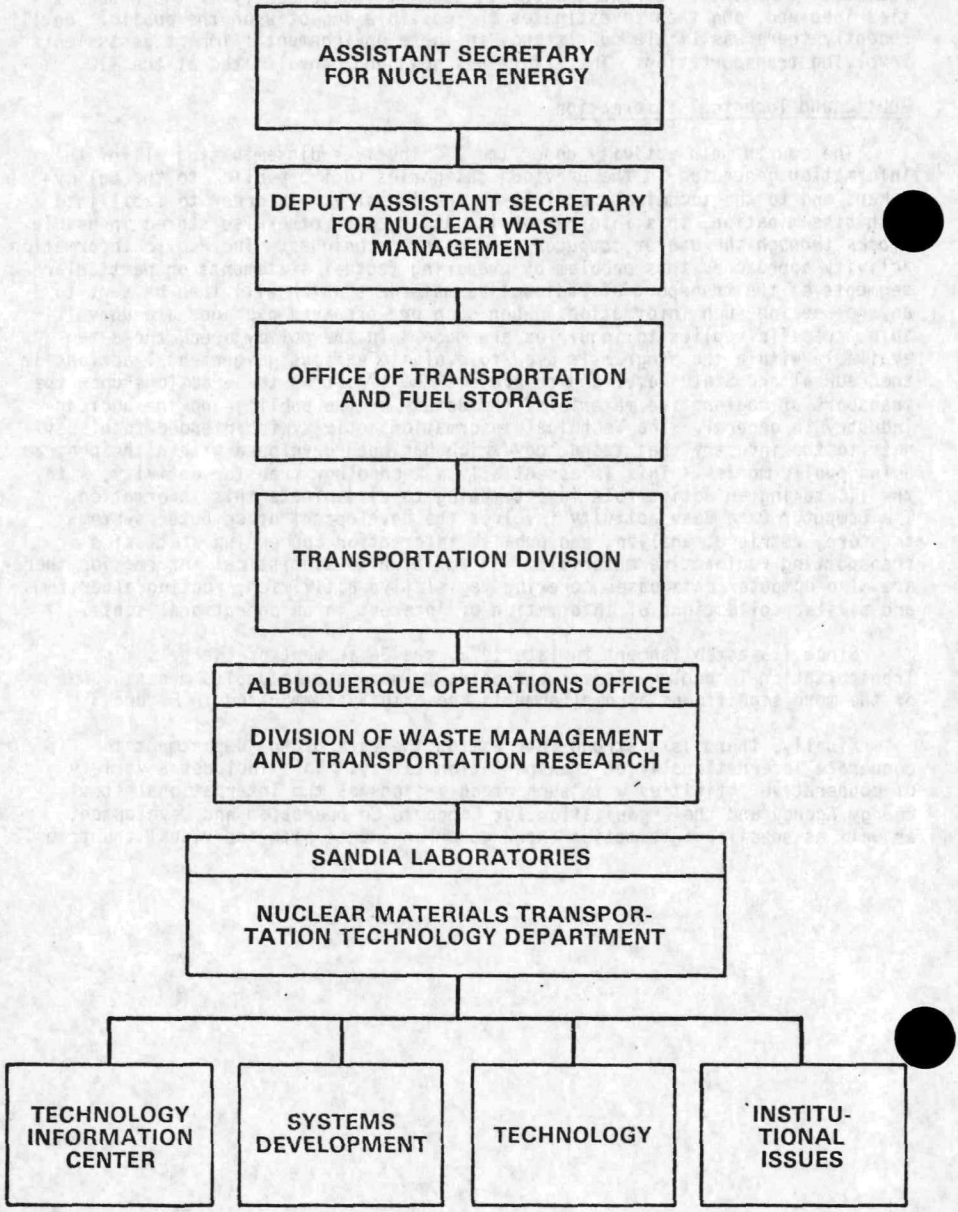
Public and Technical Information

The fourth main activity under the TTC involves disseminating all of the information generated in the previous categories to the public, to the policymakers and to the technical portions of the industry. In order to facilitate such dissemination, this information is cataloged or otherwise stored in usable stocks through the use of computers and other techniques. The Public Information activity approaches this problem by preparing factual statements on particular segments of the transport of radioactive materials which will then be sent to anyone seeking such information. When such pre-prepared packages are unavailable, specific replies to inquiries are made. In the policy area, the data available within the program is used to evaluate various governmental actions at the Federal and State level and to project the impact of these actions upon the transport of radioactive materials, the safety of the public, and the nuclear industry in general. The Technical information activity is intended to disseminate to the industry that technology which has been developed within the program using public monies. This is essentially a technology transfer activity, with the TTC taking an active role in attempting to disseminate this information. The Computer Data Base activity involves the development of computer systems to store, retrieve, analyze, and publish information concerning statistics of transporting radioactive materials. In addition to statistical information there are also computer data bases covering legislative activities, routing algorithms and similar collections of information of interest in an operational sense.

Since its establishment in late 1978, the Department of Energy's Transportation Technology Center can point to numerous accomplishments. Some of the more significant accomplishments are briefly summarized in Figure 3.

Finally, there is a strong interest on the part of the Department to cooperate internationally on transportation safety. This includes a variety of cooperative activities with such organizations as the International Atomic Energy Agency and the Organization for Economic Co-operation and Development as well as specific information exchange arrangements with individual countries.

Figure 1


ASSISTANT SECRETARY FOR NUCLEAR ENERGY
DEPUTY ASSISTANT SECRETARY FOR NUCLEAR WASTE MANAGEMENT
OFFICE OF TRANSPORTATION AND FUEL STORAGE
TRANSPORTATION DIVISION
ALBUQUERQUE OPERATIONS OFFICE
DIVISION OF WASTE MANAGEMENT AND TRANSPORTATION RESEARCH
SANDIA LABORATORIES
NUCLEAR MATERIALS TRANSPORTATION TECHNOLOGY DEPARTMENT
TECHNOLOGY INFORMATION CENTER
SYSTEMS DEVELOPMENT
TECHNOLOGY
INSTITUTIONAL ISSUES

SANDIA LABORATORIES

NUCLEAR MATERIALS
TRANSPORTATION TECHNOLOGY DEPARTMENT

Figure 2

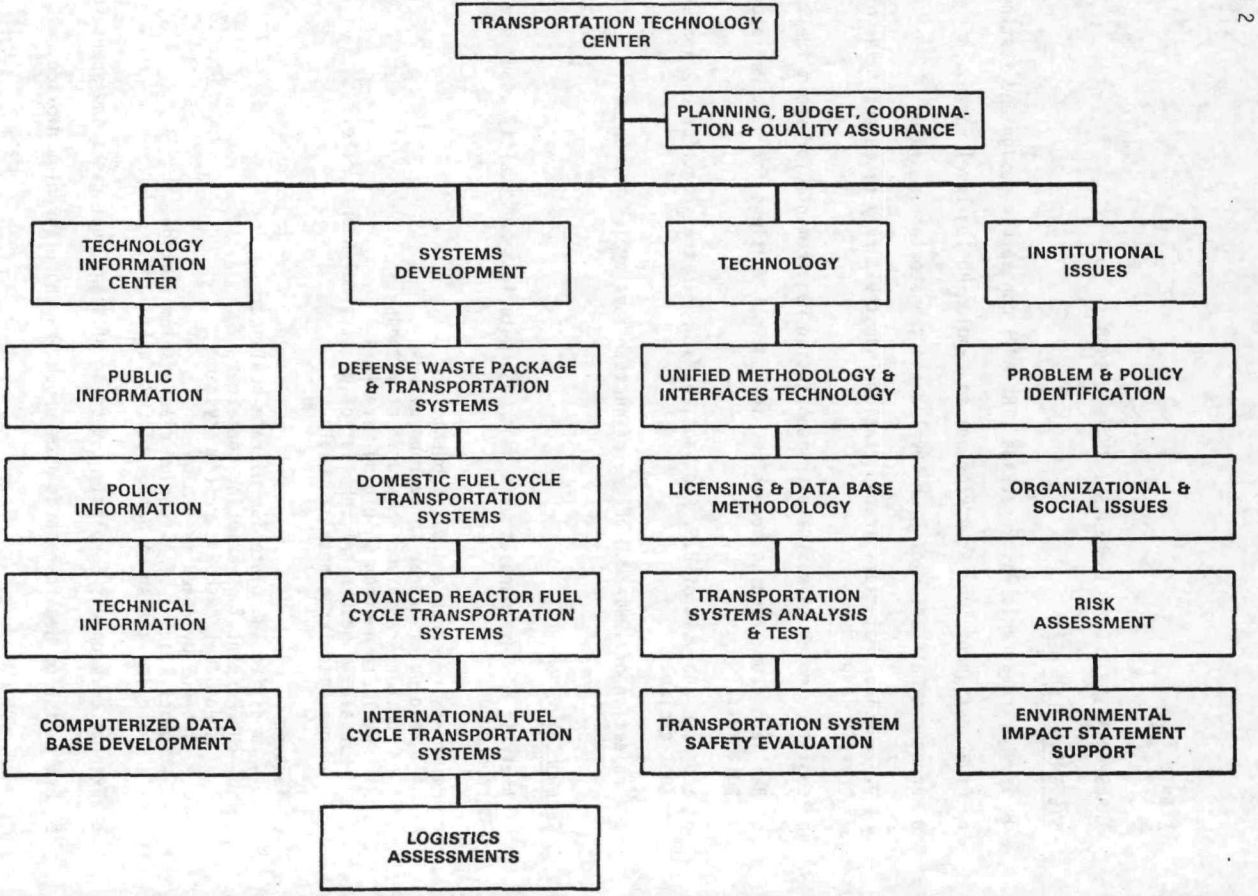


Figure 3

TRANSPORTATION ACCOMPLISHMENTS 1978 THRU 1980

SYSTEMS DEVELOPMENT

- o Type B transuranic waste package (TRUPACT) conceptual design and testing.
- o Final design of Type A modular box to handle TRU low level waste.
- o Preconceptual design of high level waste transportation system.
- o U.S. Nuclear Materials Transportation Industry Survey of capabilities of private industry.
- o Logistics model to evaluate transportation requirements for spent fuel.
- o Rail and truck routing models provide optimal routings for U.S. Radioactive shipments.
- o Economic Analysis of U.S. Radioactive Materials transport for LWR and LMFBR fuel cycles.
- o Projection of 1980-2000 volume of nuclear fuel cycle shipments.

TECHNOLOGY

- o Analysis of cask manufacturing methods, materials and quality assurance procedures.
- o Materials research and development:
  - welding of copper to ferrous alloys
  - puncture resistance-measurement methods
  - evaluate neutron shielding materials
  - evaluate mechanical behavior of uranium-molybdenum alloys
  - evaluate large metallic seals
- o Testing
  - modified 18B plutonium nitrate container
  - HNPF spent fuel shipping container
  - spent fuel cask and railcar system
  - TRUPACT overpack system
  - shock loading of zirconium clad uranium dioxide
  - handling/transport damage of LWR fuel
- o Extreme environment (explosive) testing of spent fuel casks and spent fuel.
- o Evaluation of health hazards from shock loading of uranium dioxide.

Figure 3 (continued)

INSTITUTIONAL ISSUES

- Continuing study and characterization of institutional issues affecting nuclear materials transportation.
- Analysis of effects of U.S. federal and state legislative actions on transportation risks.
- Development of emergency response guidance for nuclear materials transportation accidents.
- Review of U.S. radioactive accidents/incidents 1971-1980.
- Analysis of risks and consequences of transporting hazardous material.
- Application of ALARA principles to shipment of spent fuel evaluated.
- Interface with IAEA, OECD and numerous countries.
- Developed generic accident scenarios for environmental impact statements and analyzed unit-risk accidents.

PUBLIC AND TECHNICAL INFORMATION

- Films "Accident Safe?" and "Five Full Scale Cask Tests." Over 1000 copies distributed worldwide.
- Nuclear Materials Legislative Data Base on regulations and Legislation Affecting Transportation.
- Worldwide Spent Fuel Cask Inventory data base on cask types, ownership, capacity and approval status.
- Transportation Technology Environmental Information Center Data Base on package environments, both normal and accident.
- Transportation Bibliographic Reference File on radioactive materials transportation information.
- Truck and water carrier workshops.

DISCUSSION

Question by R. Kortright, Canada: Why is risk assessment classified as an institutional issue?

Answer: This is primarily a feature of the organisation and staff capability at the TTC and how the work load has been distributed. An additional factor is the utility of information derived from risk assessments to support the public information activities.

Question by G. L. Dooley, Canada: What is the best source of information available to consignors/consignees to keep updated on the magnitude of regulations imposed by state or local municipalities? Is it mandatory for them to have their regulations published in the federal register?

Answer: Publication of state and local - proposed or enacted - legislation in the federal register is not done. However, the TTC has sponsored the continued development and maintenance of a registry of proposed legislation at state, local and federal levels. Work is continuing to have this registry up to date. The registry is being developed and maintained at the Oakridge National Laboratory.

Question by J. F. Gustafsson, Sweden: According to some US industry people, it is at present considered that the DOE might be the future transporter of spent fuel. Could you comment on this?

Answer: We have heard these concerns and speculations since the consolidated DOE Nuclear Material Transportation Technology Development Program began. From the very start we have emphasised the interest in involving the industry and we would take appropriate policy actions to provide incentives to encourage industry involvement. In this connection our first priority is to assure that spent fuel will be transported by the industry.

ASSURING PEOPLE SAFETY IN HANDLING RADIOACTIVE PACKAGES

Charles B. Killian and Robert Devlin
New England Nuclear Corp., U.S.A.

People who are involved with the transport of radioactive materials must have information on the packages they are requested to handle. Experience has shown NEN that time is well spent going to as many groups as possible and at least introducing them to the safety record, the available assistance, the experience gained and the valuable use to society these materials have. The training of people in all walks of life who are, or may be, involved with these packages has resulted in less concern when problems have arisen. Honesty and rapid response to needs has lead to appreciation and mutual respect. It is very important that we constantly respond to the need to know that the package is people safe.

The majority of NEN's products are small in volume and radioactive content (1 to 10ml of liquid or milligrams of solids). These materials are loaded into a variety of initial containers dependent on the chemical and physical form. The multi-dose septum-cap vial is used to package most of these products. This container was designed to eliminate leakage of solutions during transport, particularly by air. Some products are transferred into vials by means of remote tools; this is due to the isotope characteristics or to pharmaceutical requirements. In this latter case, there is always a possibility of surface contamination problem.

Those vials which contain soft energy material are placed into a plastic zip-seal bag, which is in turn put into a snap-capped cylinder. If liquids are involved, absorbent materials are also added.

Other products require lead shielding. In this case, the vial is placed into a lead pig, which contains absorbent for liquid and the pig is placed into the plastic snap-cap container.

The snap-cap container is mounted into a fixed position in a molded styrofoam holder which fits into a certified DOT 7A box.

Some items, such as gaseous Xenon-133, are loaded into small vials which are placed into a lead pipe. The lead pipe is then mounted into a cardboard box. Other products are fixed into sources or on to devices which are less vulnerable than the vial containing liquid material.

The inner container is designed to keep the product in a convenient, safe, usable form for the customer. These products are placed into a transport package, which must hold its integrity through normal transport. The government has written laws which require that specific tests be conducted and passed on prototypes of any package to be used in the transport of hazardous materials before it can be placed into use.

Even though a package has passed all of the U. S. Department of Transportation certification tests many potentially problem situations arise after a number have passed through the transportation system. Once these variables surface, the shipper must review his package to see if it threatens safety which would require redesign.

A review of our history gives a few examples of changes brought about through experience. The multi-dose crimp seal vial designed to prevent leaks failed due to a few weaknesses. On one occasion the leakage occurred when the mold used by the glass manufacturer wore and did not have a satisfactory pressure ridge for the septum to compress against. A second failure was due to chipping the ridge when the vials went through a pre-washing and storage operation. And thirdly, the crimp was either not strong enough or the crimper die wore. Each of these took some time to surface and then to determine cause. In these cases the corrections were made by more frequent changes in dies and more careful handling.

In another case involving the shipment of Xenon-133 two problems surfaced. In the first case, the lead pipe was mounted in a fixed position in the shipping container by using plastic chips. This type packaging can cause problems during accident situations. For example, a box carrying Xenon-133 fell from an airport transport cart and one end was run over by a back wheel. The box, which had plastic chips for a filling, lost a portion of these chips. The small lead pipe containing the Xenon-133 came out of the box and rolled away from the immediate area causing much concern. This packaging was corrected by mounting the lead pipe on a cardboard clip inside the box. However, this resolve was good only for a large five sample pipe. Smaller pipes were wrapped in packing materials and were vulnerable to additional discard at the customers receiving room. We recently designed a styrofoam package which eliminates these problems.

The caps at each end of the lead pipe containing Xenon vials presented a second problem. If these caps were not placed in position properly or if the lead was not added to the cap, a package could pass through the system and have a beam of radiation. The error potential was met by conducting 100% visual and radiation surveys, in addition to, redesigning the pipe so only one end could cause the problem.

No matter how much the manufacturer does to design a safe package there is always something that can go wrong when different circumstances arise. It is most difficult to protect against human error. We have found that close, interested, non-distracted supervision is the most effective way of reducing human error problems.

The finished package is measured, labeled, and transferred over to the driver for transport to the customer.

Our drivers handle thousands of packages each year. He is in close proximity to them as he travels hundreds of miles each day. Deliveries are made directly to the customer, to numerous airline flights, and to other distribution points. Smaller quantities are turned over to selected carriers for regional delivery. The driver travels out of New England as far south as Florida and as far west as Chicago. If one compared these distances traveled from Berlin, our drivers would cover an area which is equal to most of Europe.

The driver loads and unloads his own truck. He spends most of his working hours in vehicles transporting radioactive materials. He must be trained in a number of areas in order to assure safety. In addition to safety lectures given to all employees, he must become aware of equipment, government rules, established routes, emergency action and proper handling as it relates to the product. He is equipped with personal dosimetry equipment to record radiation exposure to his whole body and wrist. His vehicle is equipped with lead shielding when necessary, radiation detection instruments, radio communication systems, and emergency gear. He is in constant communication with the company, with a 40 kilometer radio range. Other equipment and procedures are available to him when at a greater distance.

The exposure record for this group is very important. It indicates, both to the driver and to outside inspection agencies, that we are packaging properly and have control over personnel exposure.

Figure 1, a five year review of our record, shows that we have dramatically increased the number of packages handled by the NEN driver, while at the same time we have maintained a low average dose rate.

Information presented this far in the paper expresses the importance we have placed on following the law in assuring compliance to package criteria. It also attempts to point out the importance of having the NEN employee truly feel that the package he is assembling and transporting is safe. However, no matter what is [redacted]e in assuring our own employees that a package is safe, a real or suspected [redacted]ident in the transit system can cause havoc in the public eye. It is, therefore, very important to get this same information and trust out to the general public.

In addition to the NEN employee, there are many other different groups involved in the direct transportation of our products. These include the baggage handlers who load and unload planes and the pilot who allows the materials on board during his flight. A second group directly involved are the drivers for the selected carriers who handle our products. These people can directly relate to our drivers because they do similar work, although with fewer packages. Those who are involved in this immediate transportation must be advised on the package, its safety and its make-up. It is also good to explain the use of the product and the importance of immediate delivery.

Still another group of people become involved with the transportation of radioactive packages on a less routine basis; these people respond to a real or suspected problem. They include police, firemen, hospitals and local government employees. Emergency people must examine the things that could go wrong in the transportation of this material. They must receive guidance on the best way to identify and handle the problem, so they can intelligently react to emergencies without fear.

NEN has designed training sessions to meet each persons' specialty. Our program is based on slides and simulated experiences. In general, it is divided into three sections, with the greatest emphasis placed on that section which is most important to the group receiving instruction.

The basic program is quite flexible and is constantly being updated as new experiences unfold.

Part One is a basic area used for all groups. It covers items about radiation common to everyday life. Subjects include the electromagnetic spectrum as it applies to radio, the sun, and other causes. We also discuss background radiation as it is found in various materials in nature. The radiation from a radioactive package is related to the routine chest X-ray. It is pointed out that [redacted]iation is not new, it is something that we all have and will live with through[redacted] our lives.

Part Two is directed to those people who may handle, or come in contact with, the package during normal transport. In this section, the person is shown how the package is assembled and he is also given an understanding of the physical quantities involved. This area is designed to take the mystery out of the contents of the box. He gets a clearer idea of the things that could happen. Finally, we interest him in the positive uses of the product and the need to get it to the user fast.

In this section, we introduce radiation and basic safety steps of time, distance and shielding. We also review the proper use of instrumentation, and the rules of thumb to follow to assure good judgment. Special sections are added to meet fire or highway accidents. We review the results of real accidents from our files and show how durable most of these packages are. A special effort is made to supply each geographical audience with the correct name and telephone number of the official government response team in the area.

In Part Three we attempt to put ourselves in the position of the person who must react in an emergency or a threatening situation. This section is designed for firemen, police, emergency people and those public officials who are first to respond.

When time allows, the training groups will simulate road accidents with sample packages. The packages contain vials with colored dyes rather than radioactive materials. This session calls for participation in the crushing and recovery of the materials. If circumstances don't allow for this time, sample packages are distributed and the class is requested to open them by any means they want. In either case, people soon realize that the package is people safe.

The training session ends with an open discussion on any issue. This is followed by a review of handout materials. These materials include specific booklets on radiation protection; "Team", an NEN inhouse emergency system; Pac Bac, an advisory on returning packages; and a slide rule on Transport Index at different distances. Finally, we also provide a booklet which would include basic terms, copies of some key slides that were used in the lecture, and other relative material.

The process of assuring package safety is never complete. This paper has attempted to point out that simply meeting the requirements of the law is not enough to guarantee safety. Every shipper must constantly go beyond this. He must test his package, train the non-technical handler, and maintain communications with other professionals, with whom he can share his experiences.

TRANSPORT OF RADIOACTIVE MATERIALS BY NENC TRUCKS AND DRIVERS

Year	Number of Drivers	Number of Trucks	Hours per Driver per year	Packages per Year	Distance per year (Kilometers)	Drivers Annual Dose (mRem)
1975	10	12	2500	83,485	622,178	1071
1976	11	13	2600	77,485	712,142	1645
1977	16	17	2600	112,772	839,272	2016
1978	18	20	3100	186,113	1,061,670	1341
1979	18	24	2800	234,637	1,244,468	1048

981

Figure 1.

CENTRALIZED MANAGEMENT OF RADIOACTIVE MATERIALS PACKAGINGS

J. FRADIN
Compagnie Générale des Matières Nucléaires - FRANCE

For the needs of all the nuclear research centres and nuclear plants included in the CEA Group (Commissariat à l'Energie Atomique and sub-companies), there has been for 15 years in France, a nuclear packagings and containers park whose management is centralized.

I - SETTLEMENT OF PACKAGINGS PARK.

- Design : in accordance with the needs of users, the packagings are designed according to regulations in force and as far as possible to national and international standards, depending on the existing equipments in laboratories and nuclear factories, as the available means of handling.
- Tests : the regular prescribed tests are operated in a special test-station which uses important means of measures, especially accelerometric lines and thermic measures computer. These tests can be carried out on prototypes of packagings or on models at reduced scale, applying the rules of similarity. Other types of tests can be regularly made in this test-station on particular packagings and containers.
- Safety file and approval : after design, calculations and tests of the packaging, the safety file is transmitted for approval to the Competent Authority.
- Manufacture : after the establishment of technical specifications and cost-book, the packaging is manufactured in a factory. Controls are executed during the manufacture as at the final acceptance. The packaging is after then, identified, indexed and placed at disposal of the customer with a notice of functionning and using.
- Maintenance and repairing : a notice of maintenance which is given to the customer, permit him to execute an elementary maintenance (gaskets, O-rings, screws, ...). The systematic maintenance and the reparations are made by technician specialists, as the regular yearly and pluriyearly retestings. A central store make sure the storage of the component parts and the set of tools. The stocks are renovated according to the needs, and the corresponding card-index ensure the maintenance of the spares.
- Inventory : the inventory of the packagings placed at the disposal of the customers of CEA Group is materialized by the edition of an index established and brought up to date by informatic listings as by catalogues printed according to the categories of packagings and containers, and supply the principal informations for use and transport of these packagings and specially :
  - approval certificate number of the Competent Authority,
  - diagrams and plans,
  - useful sizes and weight,
  - kind of biological shielding,

- list of component parts,
- notice of use.

Nota : All that which concerns the design, manufacture and maintenance of the packagings, is applied in the same way to the transport equipment :

- transport frames,
- devices of stowing and handling, swingle-bars,
- shocks absorbers,
- tank vehicles,
- special vehicles.

- DISTRIBUTION OF PACKAGINGS PARK.

The different radioactive materials (Cf. Appendix I) are transported in adapted packagings distributed in the 8 following categories :

FS	fissile materials
DV	tritium, UF4, UF6, natural U, gas
IL	irradiated fuels (examinations)
IU	irradiated fuels (reprocessing)
LR	liquid materials (tanks and tank-vehicles)
RD	solid wastes
SV	radiation sources
CC	protective shells and overpacks (criticality, fire, impacts).

All the packagings are divided in the following way :

Categories	Number of types	Number of packagings	Value french francs 1980
FS	24	2903	
DV	12	1625	
IL	28	79	
IU	5	12	
LR	13	181	
RD	12	143	
SV	21	346	
CC	14	135	
TOTAL	129	5423	82.000.000

NOTES :

1 - In addition of 5423 packagings, we have about 6.500 specific inner containers, boxes, cases and devices to pack the radioactive materials in the packagings.

2 - Some containers and packagings which have a specific using (one customer only) are managed directly and are not book kept in the central park. Its the case specially for the most part of UF6 cylinders and irradiated fuel casks to reprocessing plants.

III - MANAGEMENT OF THE PACKAGINGS PARK.

The management of the packagings is made by the way of an informatic card-index, including fixed data and variable data which are registred by direct approach of the computer by the way of a terminal. (Cf. Appendix II).

So it is easy to follow the position of the packagings after each transport, and to place them geographically in fifteen research centres and nuclear plants distributed on the national area.

Relevant documents to make up the data.

The customer fills up a form of hiring of packaging by establishing :

- whether a telex of packaging placement at his disposal (beginning of the hiring),
- or a packaging transport form,

The hiring is closed when the customer sends a telex of placing at our's disposal (end of the hiring).

System of management.

1 - The geographical management is made sure by the way of following relevant documents :

- packaging transport form (forecasted data),
- telex with the date of packaging expedition (carrying out),
- consignor's certificate and details of consignment note (correction of the forecasted data).

2 - The commercial management is made sure by additional following relevant documents :

- telex of packaging placement at customer's disposal,
- telex of placing at our's disposal (end of hiring),
- hiring price-list including the packaging paying off, maintenance charges and management charges.

The packagings hiring price-list includes two scales :

. one for the daily hiring,

. the other, twice lower price, for 6 months hiring.

Drawing up of the card-index.

The informatic card-index is formed with :

1 - Fixed data established of a year which identify each packaging :

- category and identification number of the packaging,
- hiring price,
- way of hiring (daily or annually).

2 - Variable data introduced at each movement of packaging (Cf.Appendix I

- signer of transportation or hiring form,
- customer's name and department,
- command number,
- transport and hiring form number,
- beginning date of the hiring,

- place of departure,
- place of arrival,
- unloading date,
- end date of the hiring.

Procedure to work the card-index.

The card-index is continually kept up to date by :

1 - The entry, in the computer terminal, of the changes of variable data (transactionnal form Cf. Appendix III).

2 - The entry, in the computer terminal, of the changes of fixed data (interactive form Cf. Appendix IV).
In addition, this second document shows the five last movements of the considered packaging.

Use of the card-index.

The card-index is used for many purposes :

1 - Geographical localizations of the packagings and containers which are scattered,
- in the different nuclear research centres and plants,
- in the storage centre of PARIS area (SACLAY),
- in the storage centre of SOUTH area (CADARACHE).

The physical management of packagings of fissile materials is operated, and these packagings are placed at the customer's disposal, from the storage centre of SOUTH area.

This storage centre is able to consult the central card-index to know the geographical places of the packagings and their availability, but in anyone case, it is possible for it to change or to entry new data in this central card-index.

2 - Commercial management of the packagings :
- edition of documents allowing the invoicing to the customers, either by month, either immediatly,
- possibility of grouping by customer, all the packagings hired during the month (Cf. Appendix V).

3 - Technical management of the packagings and containers by edition of different data :
- rate of use,
- rate of movements of full packagings,
- rate of movements of empty packagings,
- opportunity to renew, complete or scrap,
- periodical maintenance and repairs,
- frequency rate of regular retesting,
- movements rate by geographical places.

4 - Drawing up of statistic data.

IV - CONCLUSION.

The centralized management of the packagings and containers park, for the needsof CEA Group, is operated by the "Section des Transports Spéciaux" and shows several advantages :

- on the safety level at first. The specialized engineers are in charge of design, engineering, tests, manufacturing and standadization of packagings. These packagings are indexed, identified, maintained, repaired, decontaminated, retested regularly by the way to be always conform to the rules and regulations,
- on the economic level afterwards. A limited number of packagings permit to satisfy all of the needs, the different departments of nuclear research centres and plants having not to buy their own packagings which would be then in a great number, and used with a very small average,
- on the level of customer's aid. The technic departments of research and production will have no trouble in design, quality, good conditions of maintenance, management, nor of the regular prescribed situation of the packaging which is put at their disposal,
- finally, the available statistic data concerning the packagings, containers and the transports can be used by the safety and security Authorities.

APPENDIX I

RADIOACTIVE MATERIALS TO BE TRANSPORTED

(national area, export, import)
by road, ferry, air and sea

1 - Uranium ores.

2 - Uranium concentrated, uranates.

3 - Uranium nitrate (naturel, depleted, enriched).

4 - Uranium tetrafluoride (natural).

5 - Uranium hexafluoride (natural, depleted, enriched).

6 - Uranium oxide (natural, depleted, enriched).

7 - Uranium ingots (metal and alloys).

8 - Fuels (U nat - Ue - Pu - Mixed).

9 - Irradiated fuels and irradiated materials to examination laboratories.

10 - Irradiated fuels to reprocessing plants.

11 - Nuclear materials (Pu - PuO_2 - Ue $>$ 20 %).

12 - Liquid effluents (hight and small activities) and contaminated solvents.

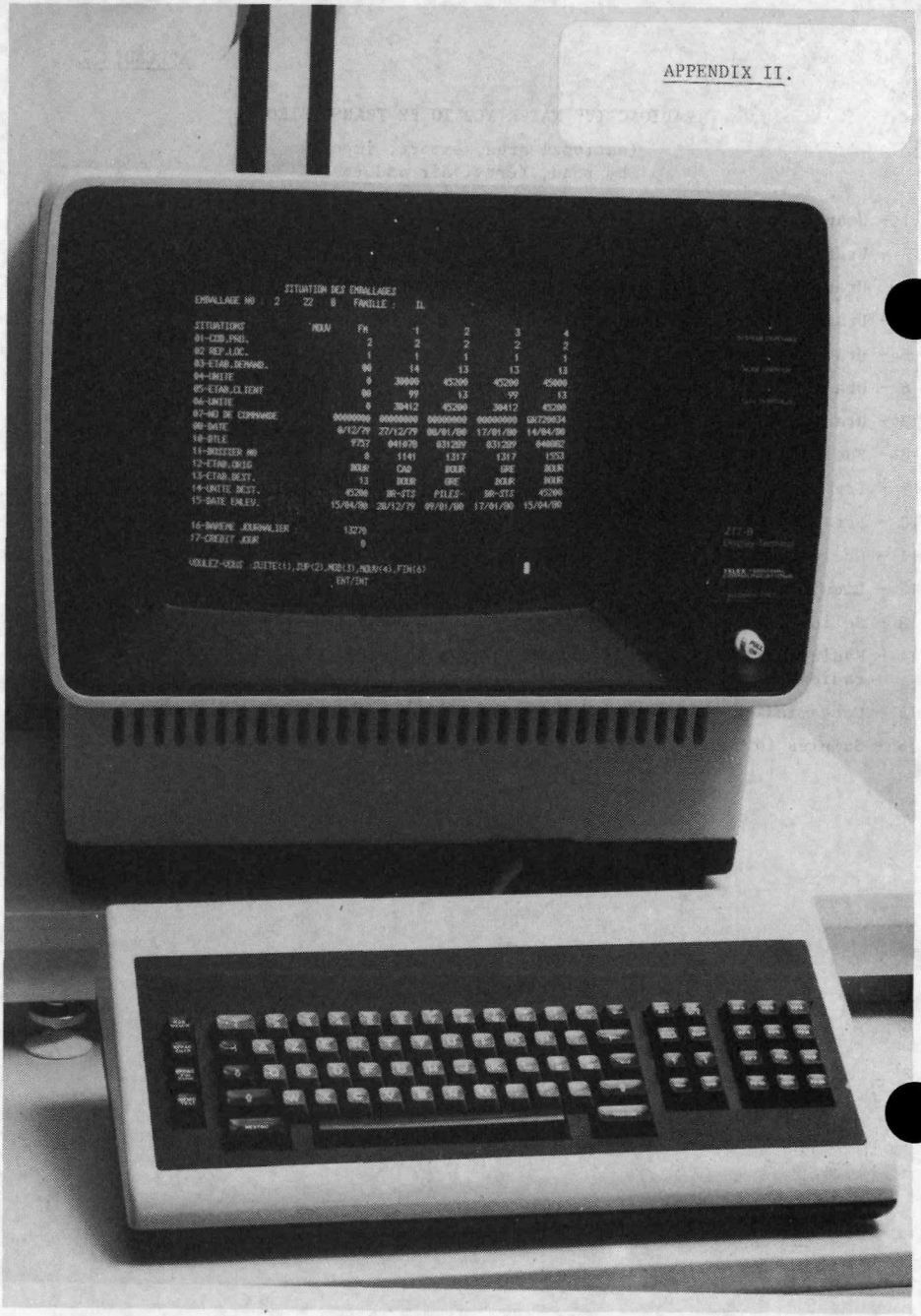
13 - Solid wastes.

14 - Radiation Sources (sealed or unsealed), radioactive solutions and different radioactive materials.

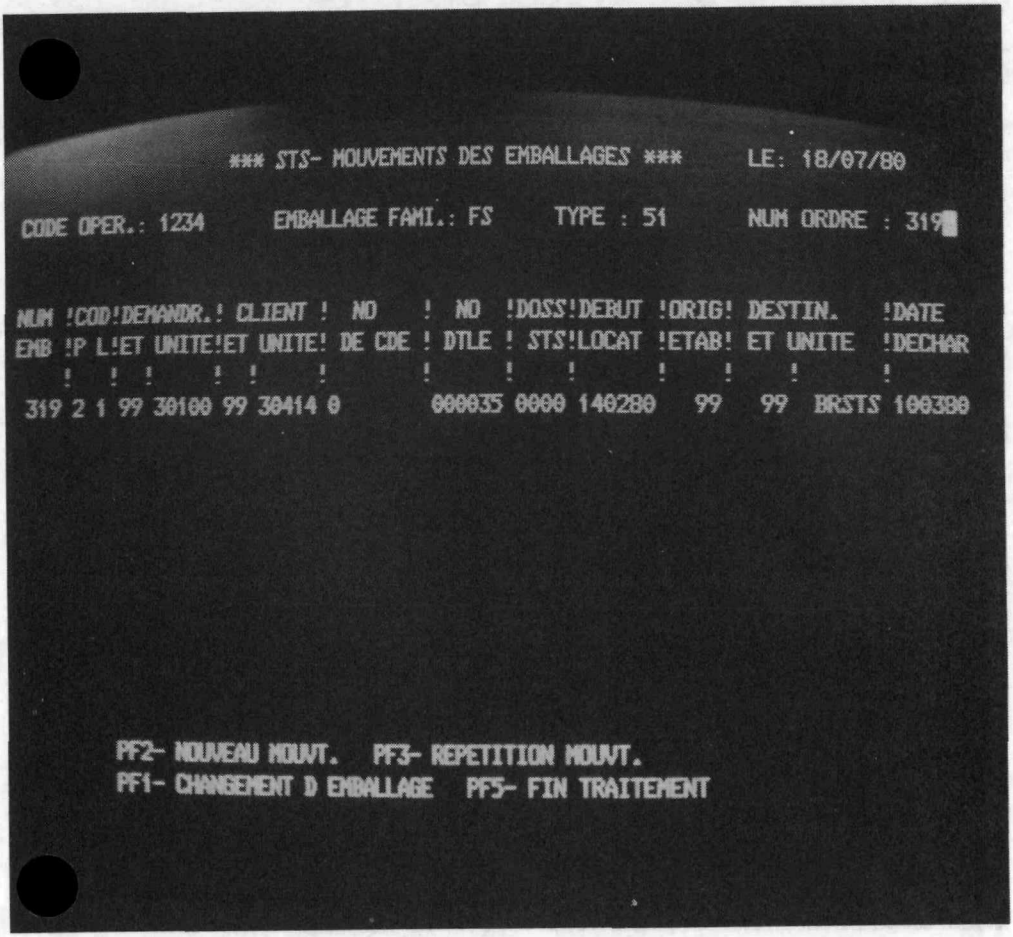
15 - Contaminated materials (gloves-box, metallic structures etc ...).

16 - Samples (by post).

APPENDIX II.


SITUATION DES EMBALLAGES
EMBALLAGE NO : 2 22 0 FAMILLE : D.
SITUATIONS MOUV FR 1 2 3 4
01-CDE.PREC. 2 2 2 2 2
02 REP.LOC. 1 1 1 1 1
03-ETAB.DEMAND. 00 14 13 13 13
04-UNITE 0 30000 45200 45200 45000
05-ETAB.CLIENT 00 99 13 99 13
06-UNITE 0 30412 45200 30412 45200
07-NO DE COMMANDE 00000000 00000000 00000000 00000000 68729034
08-DATE 0/12/79 27/12/79 00/01/00 17/01/80 14/04/80
10-STLE F757 041470 031209 031209 040982
11-DOSSIER NO 0 1141 1317 1317 1553
12-ETAB.ORIG BOUR CAB BOUR GRE BOUR
13-ETAB.DEST. 13 BOUR GRE BOUR BOUR
14-UNITE DEST. 45200 RR-STS PILES- RR-STS 45200
15-DATE ENLEV. 15/04/80 28/12/79 09/01/80 17/01/80 15/04/80
16-BAREME JOURNALIER 13270
17-CREDIT JOUR 0
VOULEZ-VOUS :SUITE(1),DUP(2),MOD(3),NOUV(4),FIN(6)
,ENT/INI

APPENDIX III.



```
                SITUATION DES EMBALLAGES
EMBALLAGE N0 :  3    51  319    FAMILLE :    FS

SITUATIONS              MOUV      FM          1          2          3          4
01-COD.PRO.                          2          2          2          2          2
02 REP.LOC.                          1          1          1          1          1
03-ETAB.DEMAND.                     99         35         35         35         99
04-UNITE                         30414      30524      30524      30515      30100
05-ETAB.CLIENT                      99         99         99         99         99
06-UNITE                         30414      30100      30100      30100      30414
07-NO DE COMMANDE             00030979   93731406   93731406   93731406   0
08-DATE                        0/12/79    07/01/80   22/01/80   12/02/80   14/02/80
10-DTLE                           1006     042752     042752     042755     000035
11-DOSSIER N0                        0       1582       1530       1617       0000
12-ETAB.ORIG                        99       BOUR       BOUR       BOUR         99
13-ETAB.DEST.                     BOUR       BOUR       BOUR         99         99
14-UNITE DEST.                   BRSTS     BR-DIR     BR-DIR      30100      BRSTS
15-DATE ENLEV.                 10/03/80   08/01/80   25/01/80   14/02/80   10/03/80

16-BAREME JOURNALIER :             1155
17-CREDIT JOUR                        0

VOULEZ-VOUS :SUITE(1),SUP(2),MOD(3),MOUV(4),FIN(6)
                          ENT/INT
```

[redacted]STURATION DE LA LOCATION DES EMBALLAGES S.T.S. MARS 1980

CLIENT 99 304[redacted]OUR STS-CAD IMPUTATION STS 0 0

LOCATIONS OCCASIONNELLES

NUMERO DE CDE	NUMERO DE D.T.L.E.	NUMERO DOSS.STS	CODE DEMANDEUR	EMBALLAGE FAM	TYPE	NUM	CODE LOC.	PERIODE LOC. DEB	FIN	MOIS	NB.JOURS LOCATION	BAREME JOURNALIER	COUT
0	35	2212	99 30414	IU	5	2	2 1	1	31	3	31	603.81	18.718,11
0	35	2212	99 30414	IU	5	3	2 1	1	31	3	31	603.81	18.718,11
0	35	0	99 30414	IL	19	5	2 1	1	31	3	31	83.33	2.583,23
0	33724	471	99 30414	IL	28	1	2 1	1	31	3	31	568.50	17.625,98
0	41442	637	99 30414	IL	28	2	2 1	1	31	3	31	568.58	17.625,98
0	94	0	99 30414	IL	30	3	2 1	1	31	3	31	153.91	4.771,21
0	73	0	99 30414	IL	30	4	2 1	1	17	3	17	153.91	2.616,47
0	27023	2622	99 30414	IL	30	5	2 1	1	31	3	31	153.91	4.771,21
0	168	0	99 30414	IL	30	7	2 1	1	31	3	31	153.91	4.771,21
0	130	0	99 30414	IL	30	9	2 1	1	31	3	31	153.91	4.771,21
0	30584	875	99 30414	IL	30	10	2 1	1	31	3	31	153.91	4.771,21
0	4149	0	99 30414	IL	31	1	2 1	1	31	3	31	351.10	10.884,10
0	338	0	99 30414	IL	31	2	2 1	1	31	3	31	351.10	10.884,10
0	308	0	99 30414	IL	31	3	2 1	1	31	3	31	351.10	10.884,10
0	29822	6511	99 30414	IL	31	4	2 1	1	31	3	31	351.10	10.884,10
0	236	0	99 30414	IL	32	1	2 1	1	31	3	31	431.06	13.362,86
0	177	0	99 30414	IL	32	2	2 1	1	31	3	31	431.06	13.362,86
0	67	0	99 30414	IL	32	3	2 1	1	31	3	31	431.06	13.362,86
0	22	0	99 304_4	IL	33	1	2 1	1	31	3	31	609.36	18.890,16
0	17	0	99 30414	IL	33	2	2 1	1	31	3	31	609.36	18.890,16
0	1386	0	99 30414	IL	33	3	2 1	1	31	3	31	609.36	18.890,16
0	0	0	99 30414	IL	34	1	2 1	1	18	3	18	84.85	1.527,30
0	0	0	99 30414	IL	34	2	2 1	1	31	3	31	84.85	2.630,35
0	0	0	99 30414	IL	39	1	2 1	1	31	3	31	97.98	3.037,38
0	0	0	99 30414	IL	46	2	2 1	1	31	3	31	1142.04	35.403,24
0	35	0	99 30414	FS	1	17	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	71	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	75	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	102	2 1	1	18	3	18	11.55	207,90
0	35	0	99 30414	FS	1	146	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	227	2 1	1	31	3	31	11.55	358,05
0	22852	791	99 30414	FS	1	270	2 1	1	31	3	31	11.55	358,05
0	0	0	99 30414	FS	1	288	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	292	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	306	2 1	1	31	3	31	11.55	358,05
0	26720	20	99 30414	FS	1	313	2 1	1	31	3	31	11.55	358,05
0	33111	374	99 30414	FS	1	327	2 1	1	31	3	31	11.55	358,05
0	32409	2831	99 30414	FS	1	338	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	390	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	496	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	582	2 1	1	31	3	31	11.55	358,05
0	35	0	99 30414	FS	1	2017	2 1	1	31	3	31	11.55	358,05

.../...

DISCUSSION

Question by K. Janberg, FR Germany: Do you transport HAW in liquid form?

Answer: Yes, in France we use CENDRILLON packagings to transport highly active liquids. However, we seldom transport such liquids in France.

Question by H. Eick, FR Germany: Is it possible to rent COGEMA packagings?

Answer: COGEMA packagings are available to all companies in the CEA (Commissariat à l'Energie Atomique) group. However, it is possible to rent them out to private French and foreign companies.

PROGRAM FOR THE TRANSPORTATION OF IRRADIATED CANDU FUEL

R. Kortright, S.J. Naqvi, D.W. Souther,
J.F. Tanaka, R.W. Barnes, M.R. Matthias, P.K.M. Rao
Ontario Hydro Design and Development Division

This paper outlines the current program in Ontario Hydro for development of an efficient system for transportation of irradiated CANDU fuel. The place of transportation in future strategies for irradiated fuel management in Ontario is reviewed with reference to a recent major study. This study evaluated the differences between storing fuel at a centralized site and extending storage at reactor sites.

1. PROGRAM HISTORY

In February 1974, Ontario Hydro initiated comprehensive studies on various aspects of irradiated fuel management including transportation.[1] The transportation study covered restrictions on flask weight and the length of the shipping season, costs, and the design of a container to support and protect fuel bundles during storage and transportation. This study formed the basis of the existing program which has developed within the Canada/Ontario Nuclear Fuel Waste Management Program (established in 1978). In this program Ontario Hydro is responsible for developing a system for safe and economic transportation of irradiated fuel.

2. NEED FOR IRRADIATED CANDU FUEL TRANSPORTATION

A recent decision by Ontario Hydro in favour of continued on-site storage of irradiated fuel has made it clear that a full scale irradiated fuel transportation system will not be required until the fuel is sent for disposal or reprocessing.

The transportation systems currently available for irradiated CANDU fuel are not adequate to meet intermediate-scale fuel transportation needs in the late 1980s. At that time irradiated fuel is required for disposal demonstration projects to be carried out by Atomic Energy of Canada Limited. Therefore, the program for an off-site irradiated fuel transportation system will proceed as rapidly as possible to develop an acceptably safe and efficient system for implementation in the late 1980's.

It is expected that a fully licensed transportation system could potentially be in place by 1988 at the present schedule pace.

3. CANDU FUEL TRANSPORTATION EXPERIENCE

Ontario Hydro has experience with three irradiated fuel transportation systems: a small (15 Mg) flask for research shipments, the Pegase flask from Transnucleaire for larger off-site shipments, and the Pickering on-site flask for transfers between storage bays at the Pickering station. None of these systems is suited for large-scale off-site shipment. The Pegase flask has been used for 18 shipments from Ontario Hydro. It has a reasonably large capacity (72 CANDU fuel bundles - 1.4 Mg uranium), but is not efficient to use because bundles must be loaded one at a time. The Pickering on-site flask (Figure 1) is more efficient to use than the Pegase flask because of its larger capacity (256 bundles) and because it is designed to carry fuel storage containers, each holding 32 fuel bundles. However, its use is limited to transportation on the Pickering site. Experience with approximately 150 shipments during 2-1/2 years of operation has been good. Future off-site rail flasks will probably be similar in size, shape and handling concept to the Pickering on-site flask. However, they will carry impact limiters as well as modifications to maximize fire resistance.

4. STATUS OF DEVELOPMENT PROGRAM

The principal elements of the system under development are: a) the module (fuel container for storage and transportation); b) the flask (rail, barge or truck); c) the flask tie down and vehicle; and d) the loading system (wet or dry).

4.1 Module Design

The module (Figure 2) is a stainless steel container for storage and transportation of 48 pairs of fuel bundles. It provides high storage density (2.25 Mg uranium/m^3 of usable storage pool volume), support and protection for the fuel bundles, and efficient handling in a single container from reactor discharge to ultimate disposal. Design and development effort in the program to date has been focused on the module to enable modules to be introduced into existing storage bays at Pickering Generating Station in the near future. The modules will replace an existing type of storage container which provides lower storage density.

Development programs for the module have included thermal analysis and testing, stress analysis, and the shock and vibration program. Initial thermal analysis of modules in a shipping flask indicated that, with fuel sheath temperature limited to 200° C, an air-filled flask could transport 3 year cooled fuel. The shock and vibration program to assess the ability of CANDU fuel to withstand the normal transportation environment is the subject of another paper (by T. Loewen[2]) at this conference.

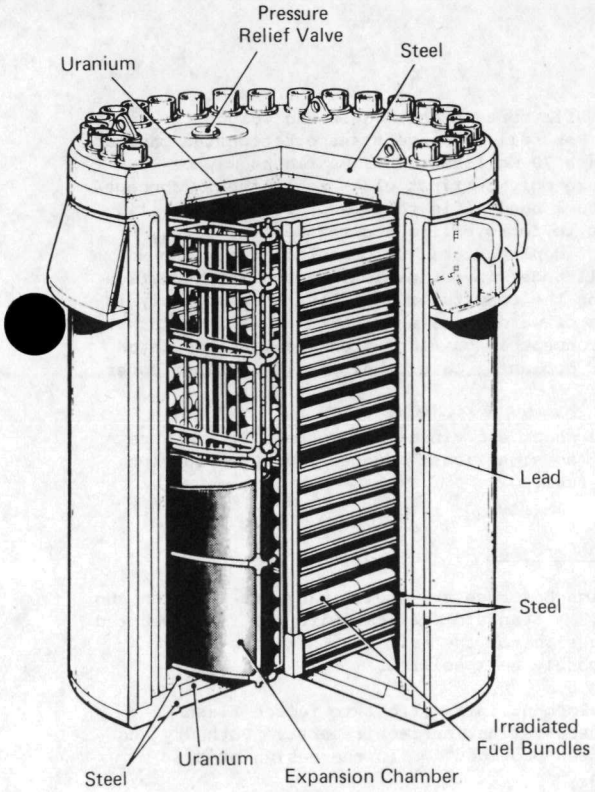


FIGURE 1
Pickering On-Site Flask

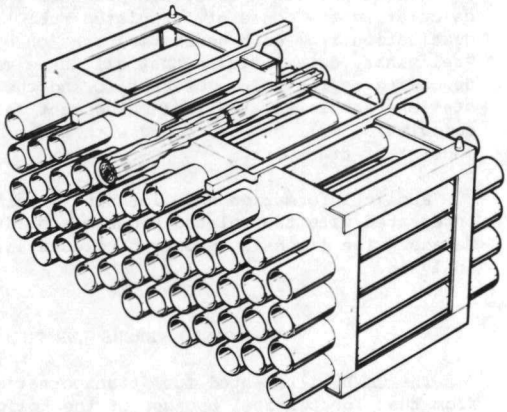


FIGURE 2
Shipping and Storage Module

.2 Flask Design

The road transport flask will contain one module and weigh approximately 30 Mg loaded. For rail or barge transportation the payload will be up to four modules in a 70 Mg flask. Larger flasks may be developed in the future for use only on first class rail lines. Concepts for holding the modules and fuel bundles in the flask, flask shape, lid and seal design, and concepts to improve fire resistance (including thermal shields and the use of depleted uranium or solid steel in place of lead for shielding) are all under investigation. A coolant selection study is under way, drawing on the results of the flask thermal analysis to assess the relative merits of water, air and inert gas filled flasks. The current reference flask concept is of fabricated steel and depleted uranium construction with air coolant. Cast steel flasks are also under consideration.

The coolant selection and shock and vibration programs will ensure that transportation will have no significant adverse impact on the long term integrity of irradiated fuel.

4.3 Other Elements of the Program

Vehicle and tie down design has received preliminary attention in the shock and vibration analysis. A standard flat car with the flask mounted vertically is the reference design for the rail option. Long-stroke hydraulic draft gear will probably be specified.

Dry loading is under development in an effort to reduce flask turnaround time by reducing decontamination requirements. Both dry and wet loading facilities have been provided for in the design of the Darlington generating station.

A mode selection study has produced a computer program for calculating the costs of irradiated fuel transportation to any destination from each generating station by rail, truck and/or barge. Preliminary data indicate that all three modes will be competitive depending on the volume of traffic and the locations of the generating stations and irradiated fuel management facilities. This economic calculation will be used along with other criteria in selection of the mode(s) of transport.

Public information and licensing programs have begun and are integrated with the related programs for irradiated fuel disposal in Canada. The licensing program includes safety and environmental impact analysis.

5. CANDU VERSUS LWR FUEL TRANSPORTATION

The CANDU irradiated fuel transportation system differs significantly from that for LWR fuel because of the following distinguishing characteristics of CANDU fuel:

a) no criticality problem;
b) low burn-up;
c) small size;
d) once-through fuel cycle.

Because a once-through fuel cycle is assumed for planning purposes, there is no economic incentive to transport significant quantities of fuel less than five years cooled. This and the low burn-up permit reduced shielding (25 cm of steel equivalent) and reduce requirements for decay heat removel (less than 2 kW). The small size of CANDU fuel bundles (103 mm diameter x 495 mm length) permits a package shape which maximizes the payload to weight ratio (fuel may comprise over 14% of the package weight). Of course the payload to weight ratio will vary with package size. Larger packages will have higher payload to weight ratios.However, the large number of bundles to be handled (14,000 bundles per year from the Pickering station alone) requires that efficient material handling systems be used.

6. IRRADIATED FUEL MANAGEMENT: SITING OPTIONS

The impact of transportation on irradiated fuel management strategies in Ontario is significant. Transportation cost is a larger part of irradiated fuel management cost than in the case of LWR fuel because of the larger quantity of fuel irradiated and the relatively low CANDU storage costs.

These facts were brought out in a recent major study by Ontario Hydro.[3] Several scenarios (Figure 3) for the siting of irradiated fuel storage facilities were compared in the study:

a) extension of At Reactor (AR) storage;
b) various siting options for construction of a Centralized Storage Facility (CSF)

For each of the scenarios, the following aspects were investigated: a) reference designs for the required facilities and transportation systems; b) irradiated fuel flows between the places of origin (generating stations), storage bays and the final destination (repository); c) feasibility; d) public safety, occupational safety, environmental quality and communities; and e) cost.

Sensitivity analyses were performed for variation in the size of the nuclear generation program assumed, the date at which disposal would begin, and transportation freight rates.

A range of average distances (400 km to 1,600 km) was assumed for off-site transportation to a disposal site. It was assumed that a CSF, if not at the disposal site, would be an average distance of 400 km from the generating stations.

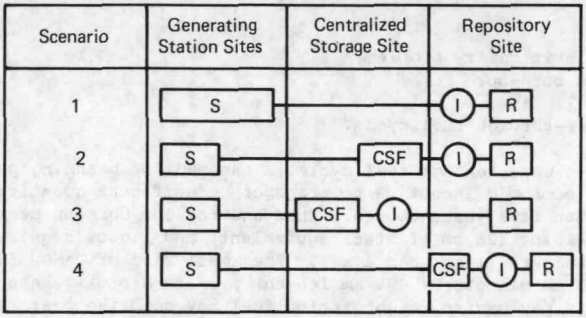


FIGURE 3
Scenarios

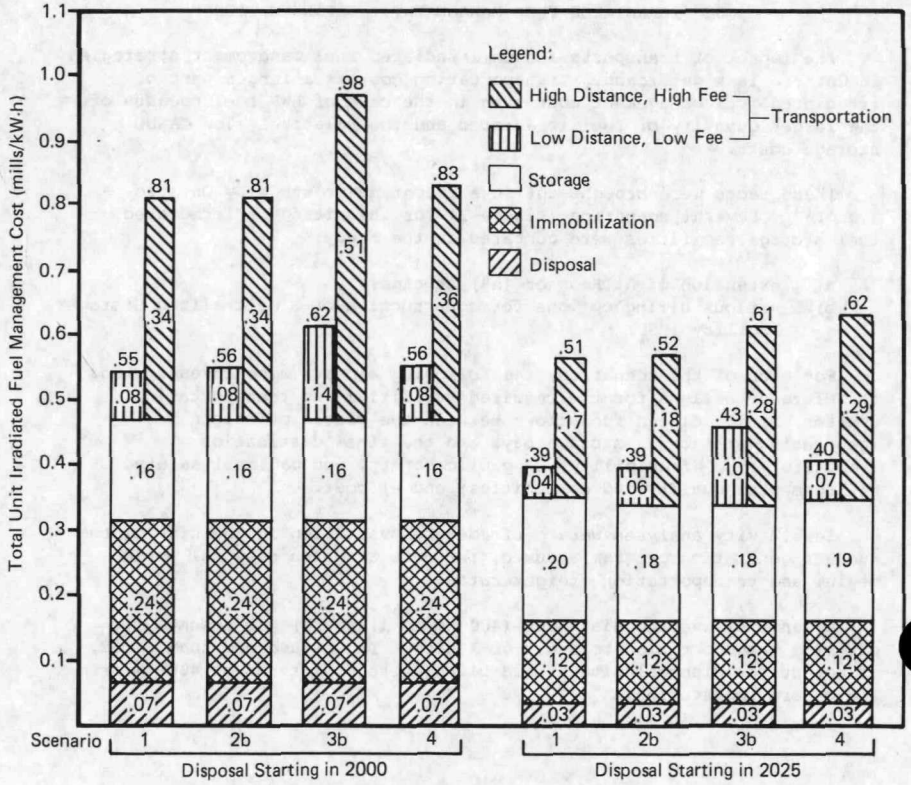


FIGURE 4
Total Unit Irradiated Fuel Management Cost

There are four types of transportation links: a) from primary to auxiliary storage bays at a generating station (GS); b) from GSs to a CSF in scenarios 2, 3 and 4; c) to immobilization; and d) to disposal. Given the large number of storage bays and the many factors to be considered, optimization of the shipping strategy to minimize cost was a complex process.

All scenarios were found to be technically feasible. Only minor differences among the scenarios were found in the areas of safety and effect on the environment.

The cost analysis showed that the cost difference between at reactor storage and the best CSF scenario is small and within the uncertainty of the cost estimates. Total irradiated fuel management cost was found to range between 0.4$/MWh and 0.8$/MWh with transportation constituting 10% to 44% of the cost (Figure 4).

7. CONCLUSIONS

1. Developments in the transportation program have indicated that CANDU fuel can be handled and transported safely and efficiently.

2. An efficient CANDU fuel transportation system could be in place by 1988 at the current pace of development.

3. CANDU irradiated fuel will be stored at reactor sites in Ontario until the fuel is sent for disposal or reprocessing. Therefore, full-scale transportation will not begin until after the year 2000.

REFERENCES

1. D.W. Souther, Transportation of Ontario Hydro Spent Nuclear Fuel, September 1976, in The Management of Irradiated Fuel in Ontario, Volume 2, Ontario Hydro Report No. GP-76014, January 1977.

2. T. Loewen, J.W. Forest, & M.A. Elbestawi, CANDU Irradiated Fuel Transportation System - Dynamic Analysis.

3. P.K.M. Rao, R.W. Barnes, R. Kortright, S.J. Naqvi, Management of Irradiated Fuel: Storage Siting Options, Ontario Hydro Report 79418, December 1979.

DISCUSSION

Question by K. Janberg, FR Germany: Do you only consider disposal in Canada or is reprocessing still an open question to be discussed?

Answer: Reprocessing is still an open question. The Canadian Nuclear Fuel Waste Management Program is developing technology for immobilisation and disposal of both irradiated fuel and reprocessing wastes, with priority on fuel. Due to resource limitations our studies have not considered reprocessing waste. The reason for the low priority on reprocessing is that Ontario Hydro studies show no economic incentive for reprocessing of Candu fuel until well after the year 2000. Extended periods of storage of irradiated fuel (more than 50 years) are under consideration.

EVALUATION OF DRY VERSUS WET UNLOADING OF SPENT NUCLEAR FUEL SHIPPING CASKS*

George C. Allen, Jr.
Sandia National Labortories, Albuquerque, NM, USA

Ray W. Lambert
General Electric Corporation, San Jose, CA, USA

Donald J. Larkin
Exxon Nuclear Company, Inc., Richland, WA, USA

The Transportation Technology Center at Sandia National Laboratories completed an evaluation of unloading methods for spent fuel by sponsoring technical programs at Exxon Nuclear Company, Inc., (ENC) and General Electric Corporation, (GE). The purpose of these programs was to provide a comprehensive assessment of the relative merits, capabilities, and limitations of dry and wet unloading methods. The results of this evaluation, when continued, are expected to impact the development of future spent fuel and waste transportation systems. In addition, final conclusions of the evaluation will provide input to designers of future receiving and shipping interfaces at away-from-reactor (AFR) spent fuel storage facilities and geologic nuclear waste repositories in the United States. The results presented here apply to the case where uncanistered spent fuel from light water reactors is to be handled. The conclusions may be different if uncontaminated canistered waste forms are considered in the future.

All U.S. light-water-reactor (LWR) power plants in service, and most of those planned to go operational in the near future, employ "wet" pool loading of spent fuel shipping casks. For this method, a shipping cask is placed into a water-filled pool and loaded using underwater spent fuel handling operations. While a considerable amount of operational experience exists with wet loading/unloading methods, some facilities that would ship and receive spent fuel and encapsulated waste in the future are planning to use "dry" loading/unloading methods. For this method, a shipping cask would be coupled with a shielded hot cell for loading/unloading operations. Exterior surfaces of the cask are maintained dry and are not in contact with the interior environment of the hot cell. The cask would be unloaded using remote handling techniques and hardware in the hot cell.

The evaluation of alternate unloading methods was accomplished by completing four tasks. These tasks included (1) developing basic assumptions; (2) selecting reference systems for evaluation; (3) analyzing the reference systems; and (4) comparing these systems. Similar basic assumptions were selected for both the GE and ENC evaluations. Sample reference design parameters are shown in Table 1. Agreement on a list of reference parameters enhances the ability to compare output between the two evaluations. Although there are some quantitative differences between the results obtained by the two evaluations, there is good agreement on the trends and conclusions on dry and wet alternatives.

There are some differences in the ENC and GE evaluations. The ENC evaluation of alternate unloading methods was completed by comparing an ENC-designed

*This work sponsored by US Department of Energy under Contract DE-AD04-76-DP00789.

dry concept to reference wet unloading concepts found in a proposed AFR design[1] and a conceptual repository design.[2] The design basis for the dry unloading facilities was developed from existing concepts[3] and from proposed concepts that were generated during the course of the evaluation. The effects of facility through-put were assessed by evaluating both "small" plants with a receipt rate of about 4 to 8 metric tonne of heavy metal per day (MTHM/day) and "large" plants with a receipt rate of about 14 to 16 MTHM/day. For receipt of spent fuel, ENC personnel assumed that standardized truck and rail shipping casks which had been designed for the proposed ENC dry unloading facility were used. This was done in order to provide a common load basis for comparing the throughputs of the facilities evaluated.

For the GE evaluation, advanced wet and dry unloading concepts were developed and used. Because improved wet unloading options were considered, the efficiency of the wet unloading system used in the GE evaluation was higher than the reference concepts used by ENC. In addition, the GE evaluation was based on the assumption that the unloading facility must accept any cask of a design which presently exists in the U.S. This reduces the advantage of dry unloading, which would tend to be more efficient with "standardized" casks. The reference wet and dry designs for the cask unloading segments of AFR storage facilities and waste repositories were analyzed based on GE's eight years of experience with unloading spent fuel shipping casks. Quantitative differences between the alternatives were identified in facility capital costs, operating costs, turnaround times, and personnel radiation exposure. Qualitative differences were reviewed in the areas of accidents and accident recovery, expected on-line efficiency, maintenance requirements, impact on cask technology, and operational safety. These differences were evaluated to compare the advantages and disadvantages of wet and dry unloading systems and some results are presented in the following sections.

Table 1

Reference System Design Parameters

	Exxon Nuclear	General Electric
Average Daily Receiving Rate (MTHM/Day)		
Small Plant	4-8	7
Large Plant	14-16	15
Rail/Truck Ratio by Weight	70/30	70/30
PWR/BWR Ratio by Weight	60/40	60/40
Fuel Weight per Assembly (MTHM)		
PWR	0.46	0.46
BWR	0.18	0.19
Fuel Age		
Years After Removal From Reactor Core	5	5
Cost Basis	1980 US Dollars	1980 US Dollars

The reference designs for AFR storage facilities and waste repositories were reviewed with respect to the functions performed by each portion of the facility. The capital costs associated with the plant and equipment required to handle and unload spent fuel shipping casks may range from 10 to 20% of the total AFR cost, depending on the specific facility features. For a geologic repository, the fraction is less than 10% and is closer to about 5%. Following the identification of facilities and equipment required for shipping cask handling, capital cost differences between unloading alternatives were determined for these parts of the plant alone. These differences amounted to a 40 to 50% increase in the capital costs of a dry unloading portion of a plant over a wet unloading portion. In the cask handling parts of the plant there are features which are common to both dry and wet unloading. These common features, which consist of the building structures and basic service utilities, vehicle receipt and preparation area, and cranes and hoists, account for a higher proportion of the capital cost of the wet unloading facility than of the dry unloading facility. However, for all ranges of concepts, it appears that the capital cost for an entire facility that uses a dry unloading system will be less than 10% higher than for a facility that uses a comparable wet unloading system.

The steps required to unload spent fuel from a shipping cask are shown in Figure 1. These functional steps were divided into operational subtasks for each unloading alternative compared for differences in cost, manpower, and radiation exposure. This comparison showed that from an operational viewpoint, wet and dry alternatives differed in only two of the seven functional areas. These areas, cask unloading and cask decontamination, were then examined in the greatest detail.

A time-and-motion analysis was made for the process steps of each alternative under evaluation. The results showed that the dry unloading shortened the cask turnaround time at the storage facility or repository. The expected cask turnaround times are shown in Table 2. The expected benefit from dry unloading is more discernible and larger for rail casks than for truck casks. In its evaluation, ENC used a computer model which not only simulated one cask going through the receiving facility, but accounted for the interactions and interferences created by multiple casks in the system. These interactions which result from having several casks being handled at the same time increased the required turnaround times by 50 to 100%.

On the average, the cask turnaround time was 10 to 30% less for the dry method than for the wet method. Since a cask spends about 15% of its service life at the unloading facility (assuming continuous demand for casks), this reduction in turnaround time could theoretically result in a maximum 5% decrease

Table 2

Cask Turnaround Times
(Hours)

	Single Cask Path				Multi-Cask Path	
	GE Truck Average	Exxon Truck Average	GE Rail Average	Exxon Rail Average	Exxon Truck Average	Exxon Rail Average
Dry Unloading	16	15	33	27	42	52
Wet Unloading	15	19	37	41	48	67

STEPS REQUIRED TO UNLOAD SPENT FUEL SHIPPING CASK

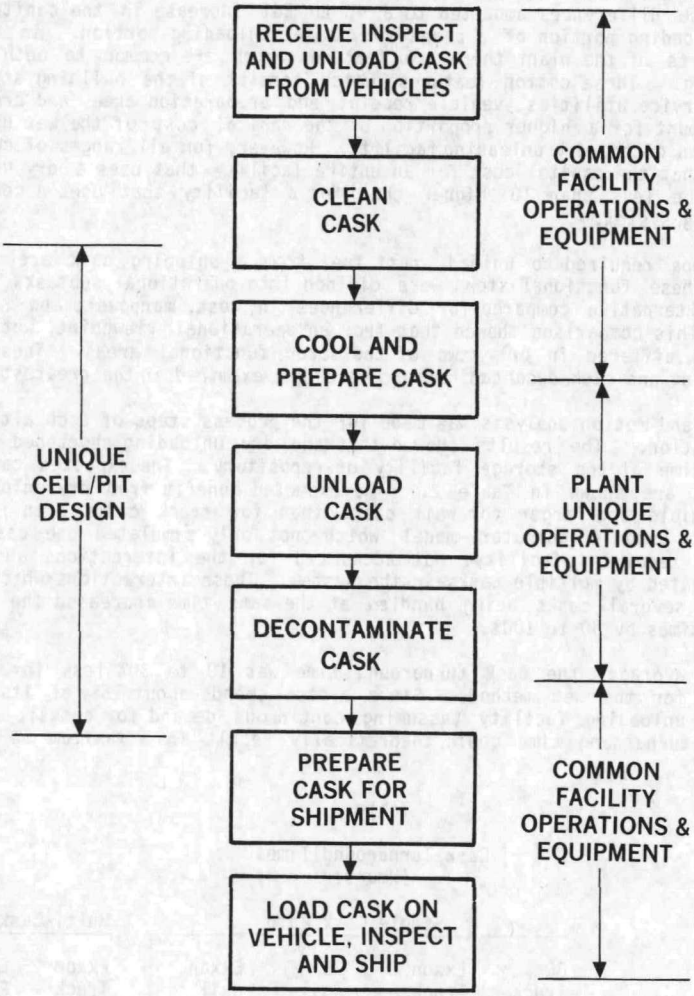


Figure 1

in the required cask fleet size. This amount of time savings is within the cask-to-cask variability of turnaround times based on either differing cask designs or normal scheduling difficulties. In the evaluation of the total operating costs, which includes the effects of turnaround time, the savings in operating cost at the facility for dry unloading when compared with wet unloading ranged from negligible to 20%.

Using the time-and-motion studies which were developed to estimate operating costs, cumulative worker radiation exposure estimates were generated. The analyses indicate that the basic difference between the dry and wet unloading concepts with regard to radiation exposure is in the cask preparation, unloading and decontamination process steps. For dry and advanced wet unloading methods the majority of exposure is not the result of decontaminating an empty cask, but occurs in the preparation of the full cask for unloading. For present wet unloading methods, decontamination does represent a significant addition to total worker exposure. The calculated advantage of dry unloading in reducing exposure appears to be about 20% but could be less than that when non-standard conditions and dry cell maintenance considerations are included. From the GE evaluation, the approximate personnel exposure in millirem per metric tonne of heavy metal unloaded is shown in Table 3.

Table 3
Approximate Personnel Exposure
(MR/MTHM Unloaded)

	Rail Cask	Truck Cask
Wet Unloading*	33	115
Dry Unloading*	25	96

* Low-Level background contributes 30 to 35% to overall personnel

In summary, as a result of assessing the evaluations by Exxon Nuclear and General Electric the following conclusions can be made:

- Neither the incrementally favorable turnaround times and operating costs for dry unloading of spent fuel casks nor the incrementally favorable capital costs of wet unloading constitute compelling factors since these are small parts of the total system costs or turnaround times.
- Reduced personnel exposure is the most attractive feature of dry unloading, but this decrease could be significantly increased by added exposure related to maintenance of the hot cells.
- Based on the incremental and trade-off nature of dry handling benefits and the uncertainties and programmatic problems of implementing a different technology, it is concluded that a shift to dry cask handling is not warranted for the case of shipping uncanistered spent fuel if consideration is given to advanced wet unloading concepts.

The process of evaluating wet versus dry unloading involves analysis of several trade-offs. For each case, the perceived advantages gained are accompanied by some cost or penalty. No single feature of either system has been determined to constitute a compelling case for selection or rejection of either concept. The requirement for cask external surface decontamination has a major impact on the cask turnaround times and, hence, the economics of spent fuel handling. However, these evaluations were based on unloading uncanistered spent fuel. While minimizing the decontamination step is a principal advantage of dry unloading, the requirement to handle material with large amounts of crud and contamination makes the ability to maintain contamination-free operations at all times unlikely.

An additional result of this evaluation which has been documented is that for an unloading facility, receipt of rail casks is preferred to truck casks. There is a 300% to 400% decrease in worker exposure per metric tonne of heavy metal unloaded when rail casks are used instead of truck casks. There is also a significant decrease in the required hours of turnaround time per MTHM unloaded when rail casks are substituted for truck casks.

Dry unloading has the advantage of shorter turnaround times, lower operating costs, and lower personnel exposure. Both concepts were judged equal with respect to feasibility, radwaste generation, safety, and plant maintenance. Wet unloading has the advantages of lower capital cost, greater flexibility in accepting numerous cask designs, being a known technology, and requiring a lower skill level of personnel.

REFERENCES

1. The Away-From-Reactor Spent Fuel Storage Facility, DPE-3547, E. I. du Pont de Nemours and Co., Wilmington, DE, USA, December 1978.

2. National Waste Terminal Storage - Conceptual Reference Repository Description, ONWI/SUB/79#512-01600.16, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, Ohio, USA, September 1979.

3. Preliminary Safety Analysis Report Nuclear Fuel Recovery and Recycling Center, XN-FR-32, PSAR, Docket 50-564. Exxon Nuclear Company, Inc., Richland, WA, USA, February 1978.

DISCUSSION

Question by B. G. Pettersson, Sweden: Could you outline the radiological protection aspects of hot cell maintenance operations (for the "dry" option); in particular whether such maintenace operations will significantly influence the conclusion that, for occupational exposure reasons, the "dry" option would be preferred?

Answer: The dry unloading method is expected to give a lower personnel exposure but the reduced exposure may not be significant even the uncertainty in estimates for hot cell maintenance.

Question by R. Kortright, Canada: What developments in wet loading were assumed in the GE study of advanced wet loading systems?

Answer: The main developments to improve wet unloading in the US that were assumed in the GE evaluation included: 1. methods to reduce the surface areas of the cask that contact the pool water (such as shrouds); 2. methods to reduce the dispersion of contaminates in the pool water during fuel handling; 3. methods to reduce facility contamination from having rigging and handling tools go in and out of the pool water (such as using a cask elevator in the fuel pool).

Question by C. R. Davis, USA: In light of all the papers presented during this symposium on dry shipping and storage in the Federal Republic of Germany, England and the Fort St. Vrain HTGR spent fuel, do you anticipate reconsidering your decision to favor wet cask unloading?

Answer: The results of the GE and EXXON evaluations are based on the condition that uncanistered light water reactor fuels must be unloaded from a cask that had been previously loaded "wet" in a reactor pool. If canistered materials are to be transported or if the cask were not loaded using a wet method at the reactor end, then the conclusion could be different.

SEA TRANSPORT OF SPENT FUEL

B. Gustafsson
Swedish Nuclear Fuel Supply Co., Stockholm, Sweden

T. Milchert
Salén Technologies AB, Stockholm, Sweden

INTRODUCTION

The Swedish Nuclear Fuel Supply Company (Svensk Kärnbränsleförsörjning AB, SKBF), acting on behalf of its owners, is conducting a project aimed at setting up a system for the transportation of spent nuclear fuel from Swedish reactor plants. The owners of SKBF are the three utilities holding licences for nuclear power plants in Sweden, namely:

Statens Vattenfallsverk (Swedish State Power Board)	- SV
AB Sydkraft (South Power Board)	- SK
Oskarshamnsverkets Kraftgrupp AB	- OKG

Before going into operation, this transport system should obtain approval from the following authorities in Sweden, acting within their respective fields of competence:

The Swedish Nuclear Power Inspectorate	- SKI
The National Institute of Radiation Protection	- SSI
The National Administration of Shipping and Navigation	- Sjöfartsverket

To ensure that demands and suggestions made by the authorities can be complied with at an early stage, such approvals would, according to the present schedule, be needed before the middle of 1980. The required approvals were obtained on July 15th, 1980.

As the transport system will operate between Sweden and France, and the vessel may be sailing under the French flag during the initial years of operation, the ship concept should also be approved by "The Central Safety Commission of the French Maritime Authorities" and the "Dangerous Goods Commission" of the same authorities. These approvals are expected to be granted during the month of November, 1980.

SPENT FUEL MANAGEMENT IN SWEDEN

The Swedish nuclear power plants are all of the light water reactor type. The various locations, owners, manufacturers, dates of commen-

cement of operation, net outputs and amounts of fuel disharged are listed in Table 1.

The spent fuel is stored for a period of at least one year in a pool adjacent to the reactor. The fuel pools can store discharged spent fuel corresponding to several years of operation plus one full core in the event that its temporary removal from the reactor is necessary. The storage capacity available for spent fuel is at present being expanded in the six reactors now in operation. This is being done by the installation of new fuel racks with greater capacity, thus making it possible to store spent fuel from 5-8 years of operation.

After temporary storage in the reactor pools, the spent fuel will be transported away from the power plants.

In 1976 and 1979, spent fuel was shipped from Oskarshamn to Windscale under the terms of the reprocessing contract between BNFL (British Nuclear Fuels Ltd.) and OKG. This contract covers 140 tonnes of uranium. Additional shipments are scheduled from OKG to BNFL for the period 1980-1983.

Acting on behalf of Sydkraft and the Swedish State Power Board, the Swedish Nuclear Fuel Supply Co. has contracted with the French company COGEMA (Compagnie Générale des Matiéres Nucléaires) for a total amount of 722 tonnes to be shipped to the reprocessing plant at La Hague. According to present plans, these shipments will take place during the period 1982-1990.

Fuel that is not shipped directly abroad for reprocessing will be sent to the Central Spent Fuel Storage Facility, CLAB, which is under construction at Simpevarp near the Oskarshamn power station. The CLAB facility, which is expected to be in operation by January 1st, 1985, will serve all the Swedish power reactors.

Table 1

SWEDISH NUCLEAR POWER PLANTS

Site/Unite	Owner/ Operator	Type	Manufacturer	Start of regular operation	Net output MW	Reactor core, tonnes U	Fuel discharged/year		Spent fuel in pools -Oct. 1979 Tonnes of U
							Assemblies 1).	Tonnes of U 1)	
Oskarshamn 1	OGK	BWR	Asea-Atom	1972	440	80	90	16	27
Oskarshamn 2	OKG	BWR	"	1974	580	80	100	18	60
Oskarshamn 3	OKG	BWR	"	1986	1060	126	144	26	-
Barsebäck 1	SK	BWR	"	1975	580	80	100	18	47
Barsebäck 2	SK	BWR	"	1977	580	80	100	18	37
Ringhals 1	SV	BWR	"	1976	760	117	120	21	39
Ringhals 2	SV	PWR	Westinghouse	1975	820	71	42	20	34
Ringhals 3	SV	PWR	"	1981	900	72	48	22	0
Ringhals 4	SV	PWR	"	1982	900	72	48	22	0
Forsmark 1	FKA/SV	BWR	Asea-Atom	1981	900	122	133	24	0
Forsmark 2	FKA/SV	BWR	"	1981	900	122	133	24	0
Forsmark 3	FKA/SV	BWR	"	1985	1060	126	144	26	-
				TOTAL	9480	1148	-	257	244

1) Average

TRANSPORTATION OF SPENT FUEL FROM SWEDISH POWER PLANTS

When the Central Spent Fuel Storage Facility, CLAB, is in operation, there will be continuous spent fuel transports to it from the power plants. For this purpose, a domestic transport system, which will operate during the major part of the year, is required.

As all the power plants are located on the coast, it has been deemed expedient that the transports should take place by sea. This will allow several flasks to be carried in the same transport, and the disadvantages of heavy transports on public roads will be eliminated. (Transport by rail is not possible owing to the lack of rail links to the plants.) All the sites have harbours. Some improvements and additions will be made to enable roll-on/roll-off handling.

The only land transportation required is to and from the harbours and from the Oskarshamn power plant to the CLAB facility.

Thus the transport system, in addition to the flasks themselves, will comprise a vessel (or vessels) of approximately 2000 dwt (tonnes deadweight) and terminal equipment, i.e. tractors and trailers, which carry the flasks on their transport frames.

The projected ship described in chapter 4 permits cargo handling both by vehicles driven on board (roll-on/roll-off, ro/ro) and by cranes based on shore (lift-on/lift-off, lo/lo). Normally, the ro/ro technique will be used, thus giving faster and safer cargo handling.

As mentioned in chapter 2, several transports are also scheduled from Swedish power plants to locations abroad. For the transports to La Hague, scheduled to take place during 1982-1990, the same equipment is to be used. The transport system will be put into service in 1982, two to three years before the first transport to CLAB is scheduled to take place.

The transport system thus serves the following purposes:

1. Transportation from Swedish reactor plants to La Hague
2. Domestic transports from reactor plants to CLAB
3. Future transports from CLAB to a reprocessing facility or other facility for further treatment

TRANSPORT VESSEL

For the purpose of designing a vessel suitable for the transport volumes entailed by the Swedish nuclear programme, the Central Spent Fuel Storage Facility, CLAB, and for transports to La Hague, design and safety studies have been carried out in several stages since 1977. Comments from the concerned authorities on the earlier layout of the vessel, as well as international developments, have been taken into account in the current design of the vessel presented here. Limitations in Swedish harbours and terminals regarding draught and size of the vessel have also been allowed for. A basic design requirement has been to permit cargo handling by vehicles driven on board (ro/ro) and by shore-based cranes (lo/lo), the former ensuring faster and safer handling.

Hence, the vessel is a combined roll-on/roll-off and lift-on/lift-off vessel with a deadweight of about 2000 tonnes, a length of about 90 m, a breadth of 18 m and a draught of 4 m. The maximum payload when carrying loaded casks is about 1100 tonnes, equivalent to 10 transport flasks.

The vessel has one cargo deck, one hold, an aft ramp and cargo hatches forming the top of the cargo hold. The hold is surrounded by side tanks and a double bottom. The superstructure containing the crew's living quarters is located forward.

Fig. 1 shows a perspective sketch of the vessel.

Main requirements met by the design include:

- Retained floatability after serious damage
- High reliability of machinery and electrical systems
- Effective fire fighting systems
- Ventilation system in cargo hold
- Bilgewater system for the cargo hold, including holding tank
- Good characteristics with respect to seaworthiness, manoeuvring and navigation in ice
- Radiation shielding for living quarters and working areas
- Instrumentation for measurement of radiation
- Modern navigation and communications equipment
- Special means for facilitating search and salvage of a sunken vessel

The vessel will be built for unrestricted worldwide service in compliance with applicable national and international rules and recommendations, and to the highest classification requirements.

The same vessel or a vessel of similar design may be used for future transportation of radioactive cargo other than spent fuel and spent core components, such as vitrified waste from reprocessing or low- and mediumactive waste from the power plants. The waste products will then be placed in transport flasks and specially designed transport containers, respectively. These possibilities have been allowed for in the design of the vessel.

Some of the above requirements go beyond those that normally apply for merchant ships and may require some explanation.

Floatability

In order to provide protection in the event of accidents such as groundings or collisions, the vessel was designed with a deep double bottom and side tanks, which are expected to protect the cargo hold in virtually all conceivable cases of grounding and in 85-90% of all statistically expected cases of serious collision. Collision calculations have been carried out, showing that the side walls of the cargo hold will be penetrated by a vessel colliding at a speed in excess of 13-19 knots, depending on the size and shape of the vessel. However, even if the hull is penetrated, the vessel will remain afloat due to the subdivision of the hull into watertight compartments. This subdivision was devised on the basis of very strict international regulations for ships carrying dangerous chemicals in

bulk. Extensive floatability calculations were carried out at an early stage in the design work.

Propulsion Machinery and Electrical Systems

The propulsion machinery consists of two independent geared diesel engines with separate auxiliary systems (pumps etc.). The engines drive two propellers equipped with controllable pitch blades, which means that the engines need not be stopped and reversed during manoeuvring.

The machinery is designed for an unattended machinery space 24 hours day and can be controlled completely from the navigation bridge and the engine control room.

The electrical installations and cables for starbord and port side propelling systems will be separated.

Apart from ordinary authority requirements, an emergency generating plant will supply consumers necessary for starting up from "dead ship" as well as the following:

- Engine room instrumentation
- International communication system
- Steering gear
- Cargo hold ventilation

Fire detection and extinguishing system

An extensive system will be installed, as the following table indicates:

Space	Systems provided	Detection/ Alarm on bridge	Activation of extinguishing system		
			Autom.	Bridge	Local
Cargo hold	- Fire detection	X			
	- Water spray			X	
	- Water hydrants on deck outside				X
	- Foam hydrants inside				X
Main engine rooms and pumprooms	- Fire detection	X			
	- Halon 1303 system			X	X
Main generator room	- Water hydrants				X
Engine control room	- Extinguishers				X
Laboratory					

Space	Systems provided	Detection/ Alarm on bridge	Activation of extinguishing system		
			Autom.	Bridge	Local
Deck house					
Cabins, messrooms, dayrooms, offices	- Water sprinkler	X	X		
Corridors, staircase, galley, wheel house	- Fire detection	X			
	- Water hydrants				X
	- Extinguishers				X

Ventilation system in cargo hold

The flasks for spent fuel are designed for a maximum heat emission of 40 kW each. In order to keep the temperature in the hold at an acceptable level, a ventilation system will be installed. The system will also be used for venting exhaust gases from the diesel-driven vehicles during cargo handling.

The ventilation system is designed for a maximum heat emission in the cargo hold of 400 kW. With 30 air changes per hour in the cargo hold, the increase in the temperature of the ventilation air between intake and exhaust is calculated to be 15° C at maximum heat emission.

In the case of lower heat emission, which will often be the case, ventilation may be reduced accordingly.

The air intakes will be equipped with fine filters to remove dust and salt particles from the air. Two centrifugal fans will be installed, each of which is designed for full ventilation capacity. Exhaust fans will be installed in ventilators situated in the aft part of the cargo hold.

The air temperature at the exhaust fans is measured and recorded on the safety console on the bridge, from which the ventilation system is controlled.

Radiation shielding and instrumentation for radiation measurement

In order to limit exposure of the personnel on board to radiation to max. 500 mrem/year as an average, the vessel will be equipped with radiation shields consisting of water-filled tanks and concrete walls. The shields are located between the cargo hold and the living and working spaces, i.e. the deckhouse and the machinery rooms.

Instrumentation for gamma and neutron measurement will be permanently installed in the cargo hold and main engine rooms. Furthermore, a laboratory space on board will contain equipment for measurement of radioactivity in liquid and air samples. Portable instruments for

Special installations facilitating search and salvage

In order to facilitate search for the sunken vessel, a special signal system will be installed.

Two line buoys will be fitted on the vessel, each with a line length of 500 m. These buoys will float up to the surface and mark the vessel's position if it should sink.

In addition, two acoustic transmitters (pingers) will be installed. These transmitters are activated by water when submerged. The signals can be received by passive hydrophones.

For easy connection of lifting cables, eight lifting fittings will be installed in suitable positions along the vessel's sides.

The possibility of lifting the vessel from the sea bottom has been carefully studied.

ACCIDENTS DURING SEA TRANSPORT

The probabilities of various mishaps damaging the transport vessel - such as vessel sinks after collision or grounding, large hole in cargo hold results from collision, flasks exposed to impact forces due to collision and fire accidents - have been analyzed. The potential hazards to the transport flasks resulting from such accidents have also been analyzed. Table 2 summarizes the various cases that have been analyzed. It can be concluded from table 2 that cases 1a (serious collision causing the ship to founder) and 2a (serious grounding causing the ship to founder)will under no circumstances cause failure of the integrity of the flask.

Cases 1b (serious collision involving deep penetration of cargo hold) and 1c (loss of flask through hole in hold wall) also have large failure margins, although leakage cannot be completely ruled our by strictly theoretical methods. The gaskets or penetrating sleeves may be damaged, thus permitting a slow transfer of material from the flask interior to the ambient water (or air). Total rupture of the flask body is not possible.

In order to obtain an upper limit for the total probability of such leaks per year, let us assume that a leak occurs in 10% of the cases 1b and 1c. Bearing in mind the fact that the different cases in table 2 are not independent, we can conclude that the probability of leakage will not exceed 10^{-5} per year.

Case 1d, fire of long duration, is more difficult to evaluate because of the several unlikely conditions that have to be fulfilled. The possible leakage pathway in the event of fire would be through damaged gaskets. The term "long duration" means a fire lasting more than 30 minutes and directly involving the cargo hold. It is obvious that the upper probability limit (10^{-5}) is assumed, only a fraction of such fires will actually cause the flasks to leak. If this frac-

tion is assumed to be as high as 10%, the total probability of a fire that leads to failure of flask integrity is on the order of 10^{-6} per year.

Table 2

PROBABILITY OF SERIOUS ACCIDENTS WITH VESSEL CARRYING LOADED CASKS

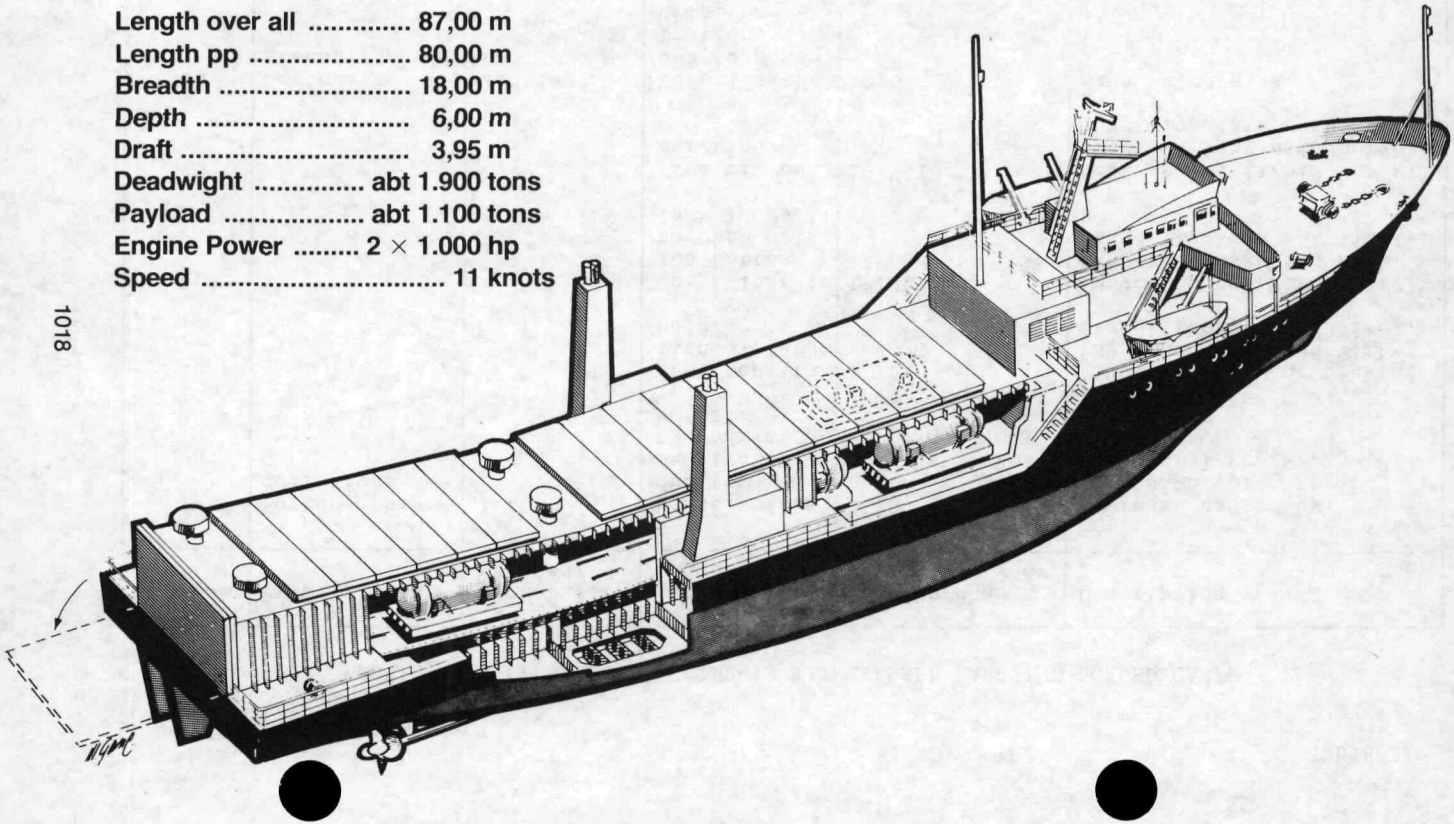
Case	Primary event	Probability per year	Alternative consequences	Total probability per year	Description
1a	Serious ship collisions	$1.5 \cdot 10^{-3}$	Vessel sinks due to penetration by ramming vessel.	$2 \cdot 10^{-4}$	No leaks caused by corrosion for a period of several years.
1b	"	"	Deep penetration of cargo hold.	$1 \cdot 10^{-4}$	Impact forces not greater than the casks can withstand. Leakage not likely.
1c	"	"	Cargo lost to sea bottom through hole in hold wall.	$1 \cdot 10^{-5}$	Impact forces not greater than the casks can withstand. Leakage not likely.
1d	"	"	Fire of long duration.	10^{-6}-10^{-5}	Leakage not likely but will occur if very unfavourable conditions are assumed.
2a	Serious groundings	$2.5 \cdot 10^{-3}$	Vessel sinks due to penetration, cargo undamaged.	$4 \cdot 10^{-4}$	Same as case 1a.

Fig 1

MAIN DATA

Length over all	87,00 m
Length pp	80,00 m
Breadth	18,00 m
Depth	6,00 m
Draft	3,95 m
Deadwight	abt 1.900 tons
Payload	abt 1.100 tons
Engine Power	2 × 1.000 hp
Speed	11 knots

1018



DISCUSSION

Question by K. Janberg, FR Germany: What happens to the ship's hull if a cask falls on to it during the unloading process?

Answer: The case with a cask falling on to the cargo deck is not a serious accident with respect to release of activity, as the cask would withstand the mechanical impact. A penetration of the deck or hull structure would not affect the floatability of our vessel.

Comment by M. S. T. Price, United Kingdom: I believe that the safety record of ro/ro ships is not as good as for more conventional ships and that as a result you have had to take extra measures to provide a ship which is acceptable. Can you please comment.

Answer: It is true that in general terms the damage floatability of ro/ro ships is not as good as for some other types. As explained this special vessel was designed with respect to very strict IMCO criteria of floatability in damaged condition which means a subdivision of the vessel's hull below the cargo deck into watertight sections. Floatability calculations were made before the design was fixed.

Question by H.-J. Möller, FR Germany: What is the weight of the units including support? What is used for securing? Are your calculations regarding collisions at sea carried out theoretically only? I refer to the collisions which have taken place during the last 2 years near the light vessel "Deutsche Bucht." The ships concerned (200 to 250 m length) were hidden to the side, side shells were opened over an area of 30 m length (above water) and 10 x 8 m (below water line). In my opinion a ro/ro vessel of 80 m length will not withstand such a collision.

Answer: The weight of each cask including the transport frame is about 90 tons. The units will be fixed to the vessel's deck during voyage, probably by bolts.

The interaction between colliding vessels is relatively complicated and in a calculation masses, speeds and types of structures must be considered. Where a little ship is struck, the collision energy will be relatively small which compensates for the weaker structure of the smaller ship. In our case their are two main additional factors to be considered, namely that the ship has a double hull and that it will be strengthened for naviagation in ice. Collision calculations made for this vessel indicate that the ship will survive even very serious accidents with large collision partners.

SUBSEABED DISPOSAL TRANSPORTATION SYSTEM*

G. C. Allen, Jr., M. E. Vernon, D. R. Anderson
Sandia National Laboratories, Albuquerque, NM, USA

The emplacement of solidified high-level waste and/or spent fuel in geolo formations beneath the sea floor (an alternative to land-based geologic disposa is an option being considered by several nations, including the U.S., as a permanent solution to the disposal problem of nuclear wastes.[1] The oceans cover more than 70 percent of the earth's surface; while they contain many valuable resources, they also cover some of the most inaccessible and unproductive areas of the planet. It appears that certain oceanic regions (the mid-plate/mid-gyre regions) may offer practical sites for disposal of high-level radioactive wastes and/or spent fuel.

The transportation of this high-level radioactive waste and/or spent fuel from originating sites such as reactors, reprocessors, or storage sites to final subseabed emplacement will involve four major areas: (1) land transportation systems, (2) port handling facilities, (3) interim storage, and (4) sea transportation systems. The relationship of these areas is shown in Figure 1. Land transportation systems will cover the transportation from originating sites to receiving stations of the port handling facilities. Port handling facilities will prepare the shipments for transfer to the sea transportation systems. Interim storage will accommodate the short-term accumulation of waste resulting from

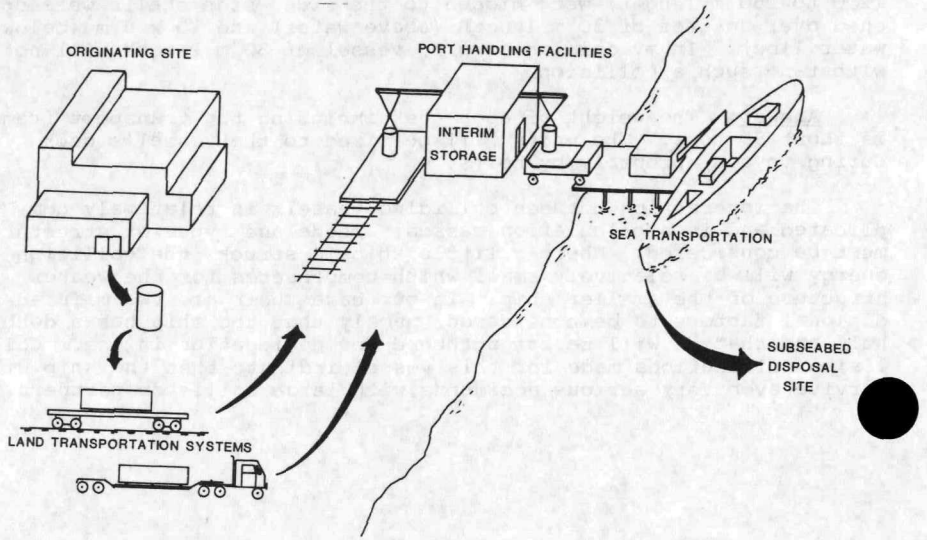


Figure 1. Subseabed Disposal Transportation System

*Work sponsored by the U.S. Department of Energy under Contract DE-AC04-76DP00789.

dissimilar delivery and shipping schedules. The sea transportation systems will be used for movement of the waste material from the port facilities to the final disposal sites. This paper covers the programmatic activities of the Subseabed Disposal Transportation (SDT) task, reviews site selection criteria, discusses transoceanic route densities, and describes oceanic climatic factors which could affect SDT activity.

The Subseabed Disposal Transportation task, which is being coordinated by Sandia National Laboratories (SNL) in Albuquerque, NM, has been divided into three major activities: Technical Direction and Planning, Transportation Interfaces, and Systems Development. Respective objectives of these activities are to provide technical direction and to plan for SDT activities, to establish and coordinate transportation information between subseabed tasks, and to ensure the development of major transportation systems.

Future work in the Technical Direction and Planning task will be aimed at maintaining a proper perspective on SDT activities to ensure the availability of transportation hardware if or when this disposal option is shown to be technically and politically feasible. The present time horizons for planning purposes of the subseabed disposal transportation program are shown in Table 1. This table assumes that the SDT activity is continued into a system operation phase.

Tasks in the Transportation Interface activity include establishing transportation interfaces, assembling information to identify institutional issues relating to transportation, establishing a transportation system design description, and assessing the environmental impact resulting from various subseabed transportation systems. For present planning and environmental assessment activities, it is estimated that the risks and environmental impacts of dock facilities would be no greater than those associated with surface storage and transfer facilities to be used with a reprocessing plant or spent fuel overpacking facility.

For the penetrometer system of emplacing wastes in the subseabed, it is anticipated that the waste would be packaged in a specially designed penetrometer or overpack. While being loaded onto the ship, waste overpacks would be fitted with locating devices and recovery mechanisms. The location devices would be designed for a 5-year operational life and would, upon activation, continuously identify the position of the waste container. Once placed in storage aboard the combination transport/emplacement ship, the waste would be transported to the disposal area

Table 1

Subseabed Disposal Transportation Planning Time Horizons

1980 - 1985	Technical and Institutional Issues Identified Concept Alternatives Evaluated Site Selection Criteria Established
1985 - 1987	Conceptual Ship and Dock Design
1987 - 1994	Detail Ship and Dock Design
1994 - 1999	Construction
1999 - 2000	Operational Tests
2000	System Operation

by a route designed to minimize risks to the general population and environment, with minimal interference to other ocean activities (fishing, shipping, recreation, etc). It is expected that acceptable sites can be identified in both the North Pacific and North Atlantic.

To date, a task which supports site selection criteria development has been the major focus of the Transportation Interface activity. In selecting acceptable sites, the subseabed disposal program must establish site suitability guidelines. The following technical guidelines[2] are currently taken into consideration in the U.S. during the site evaluation process until criteria are established for the U.S. by the Environmental Protection Agency.

1) Sites should be between 40°N and 50°S latitudes.
2) Water Depth at the site should be 4000 meters or greater.
3) Sites should be remote from continental margins.
4) Sites should be away from areas of potential seabed resources.
5) Sites should be away from transoceanic cables in use.
6) Sites should be away from areas that might be subject to catastrophic natural phenomena, such as submarine slides, volcanoes and earthquakes, that would decrease a site's environmental predictability.
7) The area of a site should be defined by precise coordinates, with an area as small as practicable, but no larger than 10^4 km^2.
8) Sites should preferably be in areas covered by precise navigational aids.
9) Sites should be away from areas, such as submarine canyons, which may unpredictably affect rates of exchange of deep waters and organisms with surface waters near the continental shelf.
10) Sites should be chosen for convenient conduct of operations and to avoid, so far as possible, the risk of collision with other traffic and undue navigational difficulties.
11) Ocean bottom current shear stress should not exceed critical erosional shear stress.

By developing some site selection criteria at the present time, not only would the eventual selection of a disposal site be based on well established information, but the areas of the sea which are currently being used for experimental explorations would have a high probability of being similar to the actual disposal sites. The transportation of nuclear waste is one factor which affects the choice of subseabed disposal sites. Guideline number 10 specifically addresses some transportation issues.

During the operational period (through emplacement) of subseabed disposal, probable accidents are primarily transportation-related. Potential risks during ocean transport chiefly concern the release of waste to the water column and hence to the biology of the ocean. The maximum risk would be posed by the sinking of the transporting ship or by loss of waste canisters overboard. To minimize hazards to transport and emplacement, subseabed disposal operations should be carried out in areas of the ocean which are not heavily traveled. Therefore, transoceanic traffic lane information[3,4] is being evaluated to assist in identifying ocean traffic densities. Traffic densities are affected by weather, seasons, and volume of cargo (inclement weather tends to reduce traffic densities, winter tends to supress traffic from the pole areas, and the increasing volume of cargo tends to increase traffic densities). The information assembled to date is being used to develop traffic density distributions for the ocean areas under consideration.

Site selection will also be influenced by climatic effects on the sea-transportation system. Climatic effect considerations will include potential weather damage to transport ships and handling restrictions during transportation

and emplacement. Climatic information[5] has been gathered for a substantial period of time (eg, information on surface winds has been recorded for over 200 years). SNL has incorporated this information onto computer file tapes and has processed it using a computer graphics routine[6] to present isobars on global projections of pertinent climatic data for monthly, seasonal, or yearly averages. Marine climatic information on cyclones, wave heights, wind speeds, and visibility restrictions have been placed in the files. Figure 2 illustrates the output using the graphics program when the data base is used to define the percentage of time that the mean surface wind speed will be less than 5.7 meters/sec. Examples of other means of using the data are shown in Figures 3, 4, 5, and 6. These figures show seasonal and location variations in wind velocities and wave heights. However, this information on climatic conditions is but one step in the development of transportation [redacted]e selection criteria. Additional evaluations are still required to determine how [redacted]se climatic factors affect the overall disposal risk and operational efficiency.

In the System Development activity, future work includes assessing the capability of existing port and transport systems to satisfy subseabed disposal transportation requirements and developing ship and port concepts. Evaluations addressing the generic questions of transporter and emplacement ships have recently been started. These initial evaluations will be carried out at the Massachusetts Institute of Technology in Cambridge, Massachusetts, and will review and analyze present concepts used in ocean going spent fuel carriers to determine applicability to subseabed disposal transport systems. Ship concepts are being developed that will be used as a baseline for future assessments. Evaluations of port facilities have not yet been initiated. They will be started when technical and environmental feasibility of the basic disposal concept has been assured.

In summary, transportation requirements and interfaces are being considered in the evaluation of the subseabed disposal option. Factors which affect the risk and effectiveness of transportation are being included in site selection criteria. However, detail development of port facilities and transport/emplacement equipment is still several years into the future.

REFERENCES

1. D. R. Anderson, Nuclear Waste Disposal in Subseabed Geologic Formations: The Seabed Disposal Program, SAND78-2211, Sandia National Laboratories, Albuquerque, NM, USA, May 1979.

2. IAEA INFCICRC/205/Add.1/Rev. 1, August 1978.

3. L. P. Solomon, et al., Ocean Route Envelopes, TR-036049 for NODA 600, NSTL, Bay St. Louis, Mississippi, April 19, 1977.

4. Robert P. Thompson, A Study of Maritime Mobile Satellite Service Requirements, Frequency Planning, Modulation and Interference Analysis, Vol. 1, DOT-CG-00505A, September 15, 1970.

[redacted] U.S. Navy Marine Climatic Atlas of the World, Vol. 1-5.

6. H. Christiansen, MOVIE.BYU - A General Purpose Computer Graphics System, Brigham Young University, Provo, UT, USA, January 1980.

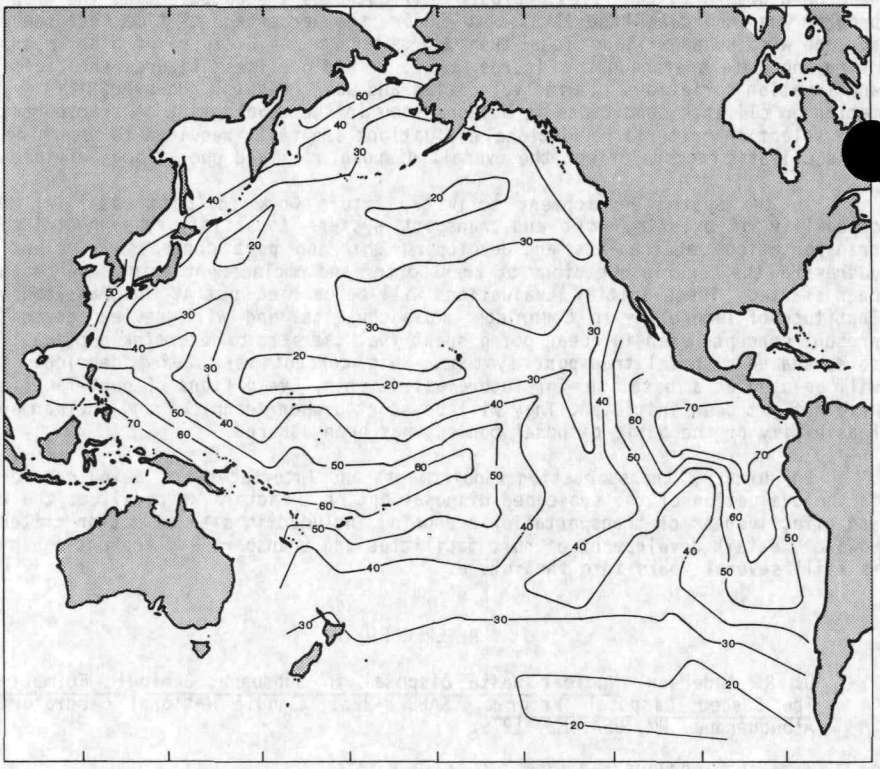


Figure 2. Percentage of Time That the Mean Surface Winds Are Less Than 20.4 km/h for the Month of April

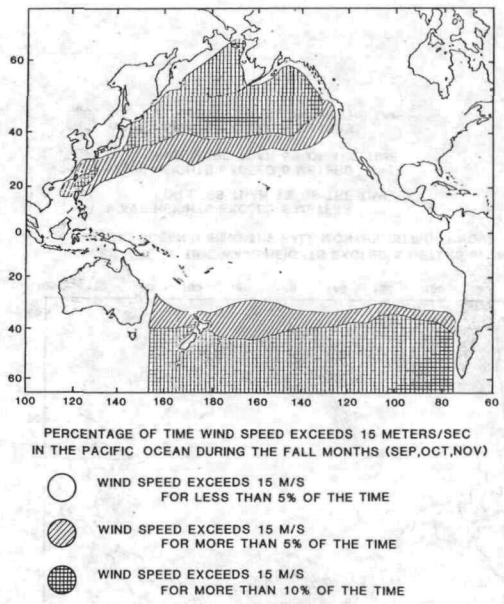


Figure 3. Percentage of Time Wind Speed Exceeds 15 Meters/Sec in the Pacific Ocean During the Fall Months (Sept,Oct,Nov).

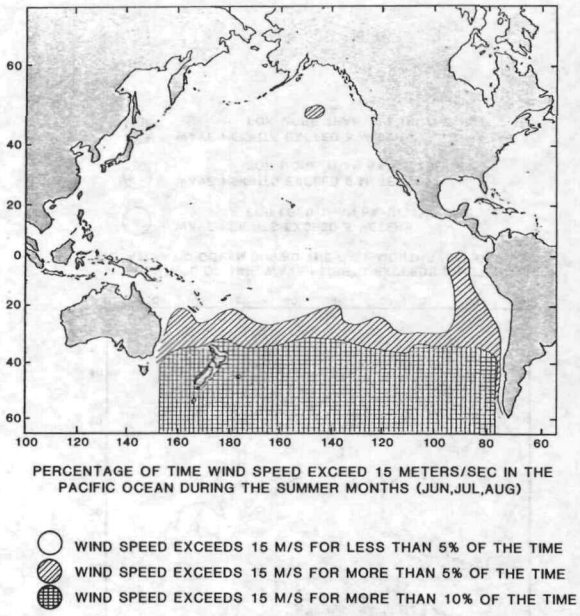
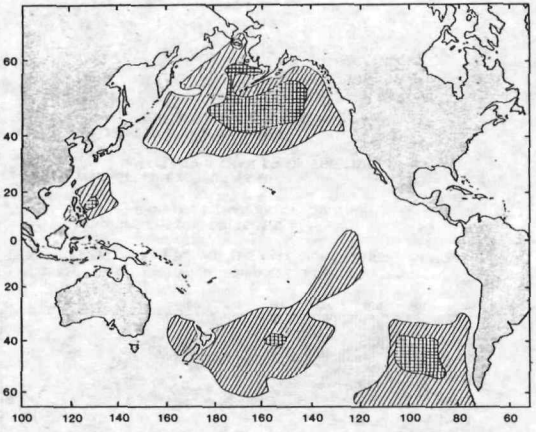


Figure 4. Percentage of Time Wind Speed Exceeds 15 Meters/Sec in the Pacific Ocean During the Summer Months (Jun,Jul,Aug)



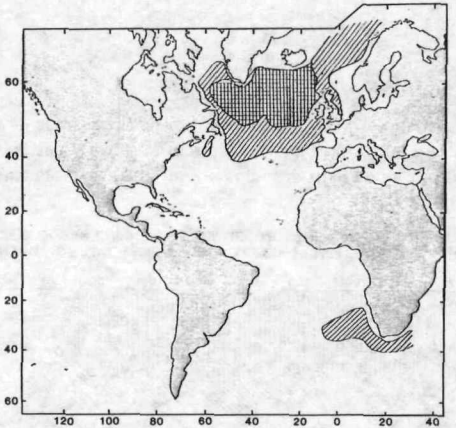
PERCENTAGE OF TIME WAVE HEIGHTS EXCEED 5 METERS IN THE PACIFIC OCEAN DURING THE FALL MONTHS (SEP,OCT,NOV)

WAVE HEIGHTS EXCEED 5 METERS FOR LESS THAN 5% OF THE TIME

WAVE HEIGHTS EXCEED 5 METERS FOR MORE THAN 5% OF THE TIME

WAVE HEIGHTS EXCEED 5 METERS FOR MORE THAN 10% OF THE TIME

Figure 5. Percentage of Time Wave Heights Exceed 5 Meters in the Pacific Ocean During the Fall Months (Sep,Oct,Nov)



PERCENTAGE OF TIME WAVE HEIGHTS EXCEEDS 5 METERS IN THE ATLANTIC OCEAN DURING THE FALL MONTHS (SEP,OCT,NOV)

WAVE HEIGHTS EXCEED 5 METERS FOR LESS THAN 5% OF THE TIME

WAVE HEIGHTS EXCEED 5 METERS FOR MORE THAN 5% OF THE TIME

WAVE HEIGHTS EXCEED 5 METERS FOR MORE THAN 10% OF THE TIME

Figure 6. Percentage of Time Wave Heights Exceed 5 Meters in the Atlantic Ocean During the Fall Months (Sep,Oct,Nov)

SESSION XVII
INSTITUTIONAL ISSUES

THE ARCHIPELAGO OF NUCLEAR MATERIAL EXPORT LICENSING

Mrs. Diane Wright Harmon
Edlow International Co.
Washington, D.C., U.S.A.

The title of this paper, "The Archipelago of Nuclear Material Export Licensing", should be taken literally rather than figuratively, for export cases must pass through six separate government agencies, each which works independently under their own guidelines and perspectives.

The short history of legislative licensing actions which have brought us to the present maze can be traced to the 1974 Energy Reorganization Act. This Act abolished the Atomic Energy Commission and created two new government agencies, the Nuclear Regulatory Commission (part of the legislative branch of government) and the Energy Research and Development Agency (part of the executive branch). It was enacted on October 11, 1974, five months after the explosion of a device by the Government of India.

Prior to 1974, export licenses and retransfer approvals were issued by the Atomic Energy Commission. The AEC verified that the quantity, enrichment and end use fell within the negotiated and approved guidelines of bilateral agreements for cooperation. After this assurance was checked, an export license was issued, or the government to government approval was signed. This exercise took from a few hours to a few days.

The first export license ever issued was on February 1, 1968 for the General Electric Co. to ship .002 KGs of U235 contained in a fission chamber to Magnetic AB of Bromma, Sweden.

In 1975 the NRC found themselves in the untenable position of having to exercise foreign policy and act to ensure for the common defense and security of the U.S. by issuing, holding or denying an export license for nuclear materials and components. Because of this, the NRC commissioners stopped issuing licenses for a painfully long time until this situation could be clarified. On February 2, 1976 President Gerald R. Ford issued Executive Order 11902 which designated the Department of State as the lead agency and the Secretary of State as head of that agency to confer with "the Secretary of Defense, the Secretary of Commerce, the Administrator of the Energy Research and Development Agency, the Director of the Arms Control and Disarmament Agency and the head of any other department or agency which may have an interest therein" about their views on whether and under what conditions licenses should be issued.

Obviously, it is hard for anyone, given an opportunity to exert some authority, not to exercise that authority by at the least raising questions to be answered, and at the most concurring in an action only if it is handled in the manner that they think is right. Human nature is what we are really dealing with in the form of Executive Orders and Public Laws.

The 1976 Executive Order assisted applicants in getting stalled export licenses moving through the approval machinery again, even if at a slow pace. Each agency newly invited into "the nuclear export club" had to create staffs to deal with their new responsibilities and we all waited while they were educated in the complexities of the worldwide nuclear industry.

With Jimmy Carter as President, the Department of Energy (DOE) was established in August 1977 by the Energy Organization Act. DOE took over all aspects of energy and the functions of the Energy Research and Development Agency became the Office of Energy Research within the DOE.

This brings us to the year that graced us with the quote [redacted] Act to provide for more efficient and effective control over t[redacted] proliferation of nuclear explosive capability," better known as the "Nuclear Non-Proliferation Act of 1978." The opening statements of policy and purpose of the NNPA were to actively pursue through international initiatives a mechanism for fuel supply assurances and to establish more effective international controls over the transfer and use of nuclear materials, including the establishment of common international sanctions.

The international nuclear fuel cycle evaluation called for in 1977 by Mr. Carter was already underway when the NNPA preempted its results by precluding reprocessing and requiring renegotiation of nuclear agreements. The NNPA was to have made the U.S. a reliable supplier and to establish procedures to facilitate the timely processing of requests for subsequent arrangements and export licenses. If anything, the NNPA has had the opposite effect. Six separate federal agencies and a multitude of divisions within those agencies were given some authority over export licenses and retransfer approvals. Each agency had to establish procedures which their individual legal staffs had to interpret as being in compliance with the NNPA.

We will now walk an export license application through the archipelago (under the current regulations and procedures).

A. An applicant with a U.S. address submits an application to the Nuclear Regulatory Commission which includes the following information:

1. Name and U.S. address of applicant
2. Name and address of supplier of material or equipment
3. Name and address of ultimate consignee(s)
4. Name and address of intermediate consignee(s)
5. Date of proposed first shipment
6. Date of completion of final shipment
7. Dates of contractual delivery dates if established
8. Proposed expiration date of license
9. End-use of material or equipment by all consignees, with sufficient detail to permit accurate evaluation of the justification for the proposed export.

B. Within one week of application the NRC forwards a copy of the application to the Department of State and the other four

agencies, the Department of Energy, the Department of Commerce, the Arms Control and Disarmament Agency and the Department of Defense, for their views.

C. These executive branch agencies have 15 days to advise the Department of State if they need additional information to make their judgment.

D. If no additional information is required, The State Department has 30 days to draft the Executive Branch views on the export for the other designees to comment on within 10 days after receipt. The executive branch has 60 days in total to advise the NRC of its views. If additional time for consideration is needed, the clock is stopped and the State Department must notify the Senate Foreign Relations Committee and the House Committee on International Relations of that decision.

E. During the 60 days alloted to the executive branch to complete their views, the Department of Energy must request confirmation from the recipient country that:

1. the export is subject to the terms and conditions of the agreement for cooperation.

2. the consignee is authorized to receive the material and

3. physical security measures will be maintained to a minimum standard set forth in document INFCIRC 225/Rev 1. These standards were designed in 1972 and published by the International Atomic Energy Agency in their present form in 1977. (The physical security assurances were subsequently made on a generic basis between individual countries and the United States.)

F. If there is an export application to a country which does not have a bilateral agreement for nuclear cooperation or a trilateral agreement via the IAEA in place, or is a non-NPT signatory nation such as India, South Africa, or Argentina, the Department of State may request the confirmation letter through diplomatic channels.

G. Each executive branch office with responsibilities for concurrence examines the export from their perspective. For example, ACDA examines the recipient country and material usage from the point of view of historical actions, safeguards, arms programs, and NPT compliance. The Department of Defense examines the export against security intelligence information and policy planning. The Department of Commerce examines an export against overall export administration and planning and gets involved more with component cases over which they had jurisdiction prior to the NNPA.

H. In addition to the Agreement for Cooperation assurance letter,the International Security Affairs division of DOE examines technical, contractual and security aspects of each export or retransfer. The State Department looks at the exports and subsequent arrangements from a current legal and political point of view

on behalf of the President prior to expressing the joint executive branch views to the NRC recommending the action.

I. During the 60 days that the executive branch is reaching their formal determination, several divisions of the NRC are simultaneously examining the case for licensing action. The NRC's Office of International Programs prepares the staff policy paper, if the case is to go to the NRC commissioners prior to licensing, and reviews the actions delegated to the staff. The Nuclear Material Safety and Safeguards Office examines the physical security aspects of each case and may recommend more stringent conditions on the export than current applicable NRC physical security measures may require. The NRC staff of the International Progr[redacted] Office, along with the legal staff, examine compliance with [redacted] Nuclear Proliferation Treaties, IAEA Trilateral Agreements, Bila[redacted] eral Agreements or their equivalent, the NNPA and any other current concerns which might have been gathered from the Interagency Subgroup on Nuclear Export Coordination which is part of the National Security Council's Ad Hoc Group on Non-Proliferation.

J. The NRC has an additional 60 days after receipt of the executive branch views to take action on the case or inform the applicant of the reason for delay. (Note however, that if the executive branch agencies delay the licensing action more than 60 days, the time clock stops, and they do not have to notify the applicant, only Congress.) If the NRC has notified an applicant that they are delaying licensing action, the NNPA gives them an additional 60 days to complete their action. The President may authorize the export by executive order anytime after it has sat before the NRC for 120 days.

K. If the President does order the export, Congress has 60 days of continuous session to vote to override the executive order.

The precedent case for congressional review was the 1978 export to India for the Tarapur reactor.

As a result of the Freedom of Information Act of 1966 and the Government in the Sunshine Act of 1978, all license applications and unclassified correspondence relating to them are available in the NRC public document room. NRC regulations describe to the public how to comment, request a hearing, or petition for leave to intervene on an application. The Edlow International Company application for the Indian export to the Tarapur reactor was again the precedent case which led to the formalization of these NRC procedures. While most export license intervention has ceased, there are groups constantly watching for interesting cases.

The NNPA spelled out the procedure and schedule that t[redacted] agencies and Congress were to follow in processing nuclear expo[redacted] and approving subsequent arrangements.

Section 303 of the NNPA, entitled "SUBSEQUENT ARRANGEMENTS", introduced the concept of prior United States approval for almost any action in which the United States had a part.

The following will list transactions defined as a subsequent arrangement:

a. contracts for furnishing nuclear materials and equipment;
b. approvals for the transfer, for which prior approval is required under an agreement for cooperation, by a recipient of any source or special nuclear material, production or utilization facility, or nuclear technology;
c. authorization for distribution of nuclear materials and equipment;
d. arrangments for physical security;
e. arrangments for storage or disposition of irradiated fuel elements;
f. arrangements for the application of safeguards with respect to nuclear materials and equipment; or
g. any other arrangement the President finds to be important from the standpoint of preventing proliferation.

If a subsequent arrangement has been examined and approved in an export license action which is still valid, the Secretary of Energy need not revert to the other agencies for their concurrence. They merely publish the approval in the Federal Register and the action may then take place after 15 days of lying before the public.

An approval for reprocessing or re-transfer for reprocessing must be reported to Congress prior to completion of the action.

Now for the good news: there have been many improvements which are now facilitating licensing on the timely basis called for in the NNPA. For many countries which have signed the NPT, the NRC staff can issue licenses for single reloads or the first core and three reloads, or five reloads per application without commission review. The only delay would be the time it takes for the recipient country to submit the assurance letter against the agreement for cooperation. Theoretically a license could be issued for a 10 year supply of material in 30 days.

The NRC can issue timely licenses on the staff level for all material except:

1. Production and utilization facilities;
2. One effective kilogram or more of high enriched uranium or plutonium;
3. 1,000 kilograms or more of heavy water or nuclear grade graphite;
4. NRC licensed components intended for use in a reprocessing enrichment, or heavy water production facility
5. Any proposed export of any material or equipment, including Section 109 components, items and substances to a country to which the Commissioners have not previously authorized an export pursuant to the Nuclear Non-Proliferation Act of 1978;
6. Any other license application determined by the staff or a majority of the Commissioners to warrant review by the Commission.

There are also many new general licenses for designated quantities of special nuclear material, source material, by-product material, and deuterium which became effective April 21, 1980.

There are proposed regulations in progress now which will eliminate all U.S. import licenses except irradiated material. Further there may be a general license for all components to countries which are a party to the NPT. General licenses are being expanded for certain quantities of tritium, californium, americum, and polonium as well as standard reference materials.

The executive branch procedures published in June 1978 gave a one year time limit for transfers between intermediate and final consignees without requesting prior U.S. approval.

But again, in an effort to reduce export disincentives, an action that has already been approved under subsequent arrangement or export license, the Executive Branch made procedural change which allows any and all actions approved under an export license to be carried out within the life of that license.

For example, assume a license is issued for storage in Sweden of U.S. origin natural uranium prior to its enrichment in France and later for fabrication in West Germany for use in a Japanese reactor. There is no legal or political reason not to grant a license for these actions. During the course of the licensing, three assurance letters would have to be received from Sweden, Euratom, and Japan. If the license had a ten year expiration date, all of the actions could be completed without retransfer requests from Sweden to Euratom and then from Euratom to Japan being required.

The best way to assure a timely processing of a subsequent arrangement or an export license is to submit complete and clear information on the original application. This will prevent the various agencies from raising questions which can cause delays. Although the U.S. government does not look favorably on what appears to be stockpiling, they do allow foreign storage for probable reactor use.

The U.S. Congress now has a greater appreciation of international nonproliferation and security goals. While the power over nuclear matters has been greatly dispersed from the days of the Atomic Energy Commission, the present format is improving and becoming more effective and timely.

DISCUSSION

Question by W. A. Brobst, USA: What practical recommendations do you have to shorten both the time and the administrative work effort to obtain an export licence?

Answer: NPT signature countries can obtain one licence for up to 5 reloads with a 10 year life span. A complete application which does not raise questions can be licensed within one month.

ADMINISTRATIVE PROCEDURES CONCERNING THE TRANSPORT OF RADIOACTIVE MATERIALS.

S. Piermattei- Comitato Nazionale Energia Nucleare
H. Nacfaire- Commission of European Communities
M. Roberti, C. Faloci- Comitato Nazionale Energia Nucleare

The present paper summarizes the results of a contract which was given by the Commission of the European Communities to CNEN (Italian Committee for Nuclear Energy). The aim of this contract was to analyze the various legislations and administrative procedures concerning the transport of radioactive materials in Member States in order to develop any possible action to remove technical obstacles,if any, to the movement of the radioactive substances and in the same time to ensure a greater safety of the general public and workers.

We wish to underline here that the objective of the Commission was not to duplicate the work carried out in this field by the International Atomic Energy Agency, but to try, whenever possible, to make more coherent the existing national procedures.

The transport of radioactive materials is an activity with peculiar characteristics with respect to those of the other activities which involve the use of radioisotopes such as use, production, processing etc., and take place in well definite environments. The transport takes place in any environments: highways, airports, railway stations, main roads etc. and packages have to move safely through the intrinsic guarantee of the packaging without any limitations except those related to the maximum radiation level on the package's surface and at one meter distance, to non fixed contamination and to the total amount of transport indexes which can be transported. The Transport Regulations must assure an adequate degree of safety during normal and accident conditions of transport, adequate containment, safe dissipation of heat and prevention of criticality.

We wish to recall here that the technical bases of Transport Regulations actually in force nationally and internationally were set up by IAEA, which was entrusted with this problem by the U. N Economic and Social council.

The first issue of the IAEA Regulations was published in 1961; the regulations were revised in the period 1964-1967 and 1973. Next revision scheduled for 1983 is now under way.

International Transport Organizations: RID, ADR, IMCO and recently ICAO consider IAEA Transport Regulations as basis for their own Regulations.

In the present study we have examined for each Member State the regulations and codes in force; the principles on which these regulations are based, the administrative procedures to be followed for the carriage of radioactive materials and finally the radiation protection provisions for personnel involved in the transport of radioactive substances and the existing provisions in case of emergency. The situation is summarized in the table.

As it can be seen the technical bases of the National Regulations existing in the nine countries are based on the IAEA Regulations, 1973 edition: Although in two cases Italy and Netherland previous editions are still in force.

The International Transport Conventions have been ratified by almost all the countries and in this case the 1973 edition of IAEA Regulations are applied.

To have the same technical bases means that the concept of type B(U) packages, special form materials, control of criticality, radiation levels at the surface of the packages and at one meter distance, are accepted; however different approaches exist as far as the administrative procedures are concerned.

TABLE I - Situation in the CEC Countries

Country	Shi... Procedures	Transport Authoriz.	Rad protection organiz.	ADR	RID	IMCO	IATA	ADNR (4)	Paris conventi...	...ruxelles convention	Postal shipment
Belgium	IAEA 1973	Required (1)	Qualified expert	in force	in force	in force	Part 2A	in force	in force		not admitted
Fed. Republic Germany	IAEA 1973	Required (2)	Transport workers are liable to receive up to 1,5 rem/year (or agreed organization (5))	in force	in force	in force	Part 2A Part 2B	in force	in force	in force	not admitted
France	IAEA 1973		Transport workers are liable to receive up to 1,5 rem/year	in force	in force	in force	Part 2A		in force	in force	admitted
Italy	IAEA 1973 rail	Required (1)	Qualified expert	in force	in force		Part 2B		in force	in force	not admitted
Luxembourg	Follow provisions ADR and RID	Required (1)	Health physics supervision for personnel involved in routine work	in force	in force		Part 2A Part 2B				not admitted
Netherlands	IAEA 1967	Required (3)	Transport workers are liable to receive up to 1,5 rem/year	in force	in force	in force	Part 2A Part 2B	in force	in force	in force	not admitted
Denmark	IAEA 1973		Transport workers are liable to receive up to 1,5 rem/year		in force	in force	Part 2A		in force	in force	not admitted
Ireland	Contained in transport licence and follow IAEA requirements	Required (1)	Qualified expert		in force	in force	Part 2A Part 2B				not admitted
United Kingdom	IAEA 1973		Transport workers are liable to receive up to 1,5 rem/year	in force	in force	in force	Part 2A Part 2B		in force	in force	admitted

1) Authorization is required for quantities above the CEC exempt quantities

2) Exempt quantities are those appearing in schedules 1-2-3-4 of IAEA transport Regulations, 1973 ed

3) Authorization is required to carry quantities above large sources (marg. A.2.7 IAEA Regs 1967)

4) ADNR follows 1967 edition of IAEA Regulations

5) It is required up to now only for quantities of non fissile materials exceeding 10^{10} times the limits of CEC exempt quantities

With few exceptions the trasport of radioactive materials within the CEC is subjected to a prior of authorization or licence. This authorization can be issued in a form of a general licence which allows, other than the use of radioactive materials, also their transport. The authorization can be also issued as a single transport authorization to carry well defined quantities of radioisotopes. In this case the licence might relieve the holder to apply the competent Authority for the shipment approval certificate. In Netherland, infact, the licence to transport radioactive materials is required only for quantities (large sources) which, according to the provisions of 1967 edition of the IAEA Regulations, need a shipment approval.

Finally the authorization can be issued in form of a general transport licence which allows the transport of certain radioactive substances up to a maximum activity, in certain packages and it is valid for a fixed period of time. In most of cases this licence is required to carry radioactive materials in quantities exceeding the limits of the exempt quantities of Annex I of the CEC Directive. The Authorization is required for all modes of transport, for national and international shipments and for transit; to obtain this authorization the applicant has to satisfy certain prerequisites.

The prerequisities the applicant has to comply with, except those of general nature such as identity, reliability, etc., are: the supervision by a qualified expert or an agreed institution capable to intervene in case of accidents, insurance with national companies and in Italy also agreed vehicles are required. It is worthwhile to point out that some of these prerequisites are rather onerous and, being the issuing of the authorization subject to the existence of these prerequisites, this can give rise to a delay in obtaining it.

The prior authorization or licence for the carriage of radicactive materials originates directly from the CEC Directive, where in art. 2 it is explicitely written that the Directive applies inter alia also to the transport of radioactive substances and therefore, a reporting or a prior authorization regime must be applied. The existence of a prior authorization relieves the Competent Authority from the compliance with the requirements of the Directive being this compliance transferred to the holder of the authorization.

The non uniform application (or the non application at all) of the Directive causes, according to our opinion inconveniences which can produce considerable delay in the exploitation of the transport activities and can favour certain local situations.

Therefore it seems to us of fundamental importance the Competent Authorities have a common approach to the problem if authorizing or not the transport of radioactive materials; it is clear that to adopt a prior authorization regime allows a better control of the activities being developed, estimates of the collective doses to general public and workers can be performed more easily but, on the other hand, it causes a heavy burden of work to the Comptent Authorities.

We wish to stress the fact that if a prior authorization regime is adopted the threshold values above which this regime applies should be the same for each country.

As far as the transport documents and approval certificates such as: shipment, package design and special form material, it appears that the information contained are basically the same, but it should be very useful for the Competent Authorities, carriers, custom people etc. to have the same format where all the information needed are given in the same sequences.

This is only a formal question but it could result in the speeding up of the administrative procedures.

In this context we want to stress also the importance to have multilingual documents expecially in cases when supplementary procedures during shipment could be needed.

As far as the transport workers are concerned it appears in most of cases they are considered a group liable to receive up to 1.5 rem/year. From studies carried out in different countries that it has been demonstrated that it is very un-

likely they will receive more than this figure.

From this study it appears that the Competent Authorities have different attitudes with respect to this problem: in some cases, according to the provisions of the CEC Directive, the presence of a qualified expert or of an agreed institute is required for radiological control of personnel involved, in others the evaluation of radiation exposure is made directly from carriers or Competent Authority. Being the control of radiation exposure of transport and storage personnel one of the task assigned by the Regulations to the Competent Authority it appears to us of fundamental importance to carry out periodic assessments, in order to ensure that the dose limitation system is not exceeded. The work the International Atomic Energy Agency is developing in this field will prove very useful and only in particular cases, such as airports or main railways stations further investigations of the relevant Competent Authority will be required.

As far as the training of personnel involved in transport: drivers, cargo handlers, etc., we noticed certain discrepancies: in some cases no training at all is required, in others general information are given on the risks connected with nuclear energy, in others it is mandatory to follow training courses.

From a general point of view a training of the personnel, adequate to the activities being carried out it is an important goal to be reached. A common programme at Community level could be developed involving carriers, drivers, cargo handlers, custom officers, etc. This would assure a common level of knowledge by all the workers involved, and facilitate the cross of the borders.

Along this line we propose also to study the possibility to coordinate the intervention procedures in case of accidents, taking into account the work carried out in this field by International Atomic Energy. From this survey it

appears that in each country an emergency service exists and therefore a coordination of the intervention procedures could be feasible in case of international transport. In this context it should be also very useful to establish mutual help agreements in case of emergencies between neighbouring countries.

The procedures adopted for package design and special form material approvals follow the requirements of the IAEA Regulations and the Competent Authorities seem to use the same general approach. Where the Competent Authorities use a different behaviour is in the control of packagings during their life, in certain cases it is up to the person who owns the certificates, in others it is up to the Competent Authorities, in other cases it does not exist at all. If in the future, the movements of type B (U) packages will increase we deem very important from the safety point of view that Competent Authorities be aware of the number of packagings of a given design have been built, fabrication date, conditions, etc.

This will be probably an argument to be included in the revision of Regulations, we want simply to draw the attention on this problem.

If the standardization of the administrative procedures is obtained the ultimate goal should be to put into operation within the CEC countries a cooperation agreements for the transport of radioactive substances along the same line of that existing within the Benelux countries.

DISCUSSION

Question by J. Chatelet, France: France is one of the countries requiring training of drivers for the transportation of dangerous goods:

- certification decreed by the Ministry of Transport for all dangerous goods;
- certification decreed by the Ministry of Employment and Ministry of Health for the transportation of gamma radiography equipment.

I would like to know in which countries there is a regulation requiring driver certification?

Answer: Drivers have to be certified in no other CEC country apart from France. Training of personnel is carried out in some countries but it is not mandatory.

Question by B. W. Emmerson, United Kingdom: Did any of the CEC countries examined in your study have administrative procedures which made provision for the training of drivers or their views in dealing with an emergency situation during the course of transport of radioactive materials, in addition to their training for normal transport operations?

Answer: No special training is required for drivers in case of emergencies. They are only informed about the minimal actions to be taken in case of accidents.

Comment by W. Stolz, Austria: In Austria the training of drivers for the transportation of radioactive materials was made compulsory by law in May 1980. This includes theoretical (legal aspects, hazards) as well as practical (measures to be taken in the event of an accident) instruction. This instruction can be restricted to individual ADR classes (e.g. class 3, flammable liquids) or cover all ADR classes.

NUCLEAR MATERIAL SHIPMENT STUDY*

E. W. Shepherd
Sandia National Laboratories, Albuquerque, NM, USA

OVERVIEW

The Transportation Technology Center (TTC) has undertaken a major study effort to collect data on normal shipment volumes of all radioactive materials which move in commerce in the United States. This information will be used for public inquiries, state and local planning, improvement of the risk assessment data base, the International Atomic Energy Agency (IAEA) worldwide collection effort, emergency response planning and environmental assessments. The shipment study described in this presentation is being performed for the TTC by the Center for Urban and Regional Policy, SRI International, Menlo Park, CA. Funding for this study is being shared by the US Department of Energy (DOE), US Department of Transportation (DOT) and the US Nuclear Regulatory Commission (NRC).

There is a continuing need to know the quantities and types of materials and their packagings which move routinely in commerce. In addition to being needed for replies to public inquiries, this information provides the backbone of any environmental assessments on the subject.

In conjunction with the collection of normal shipment volume data there is a need for accurate accident and incident data. The TTC is collecting information on incident frequency and severity. This information is stored in a computer system specifically designed for only radioactive materials reports. All data is entered by technical specialists familiar with accident analysis and radioactive materials. This information plus the shipment data will be made available by the TTC for use in risk assessments.

Currently, the only nationwide data available is the result of an NRC funded study by Battelle (PNL)[1] performed in 1975. This study, done as a voluntary survey, incorporated the inherent shortcomings of a voluntary survey. However, the Battelle study should be a significant building block for this new effort in collecting a current, comprehensive data base as well as devising a system to periodically update this information.

The major objective of this effort is to establish a comprehensive data base on radioactive material shipment volumes in the United States and to use this data to generate quantitative descriptions of the transport of radioactive materials. The major tasks underway are: (1) an audit of the procedures and paperwork used by shippers, generators and carriers; (2) an analysis of how the transport system operates; (3) an analysis of the measures needed to track quantities of radioactive materials on a continuous basis; (4) preparation of the survey plan necessary to conduct the survey for a 1 year period starting in 1981 and (5) establishment of the format for a data base suitable for environmental risk assessment, accident risk and related informational needs.

*Work sponsored by the US Department of Energy under contract DE-AC04-76DP00789.

The pre-survey tasks of this study were started in September 1980 and are scheduled to be completed in March 1981. The major tasks to be completed in the pre-survey phases are: (1) an audit of the transportation system procedures, documents and problems; (2) recommendations on the feasibility of collecting data on a continuing basis and reported annually or other appropriate time frame; (3) an assessment of data already being recorded in material specific data bases; (4) an assessment of data that can be extracted from existing documents and related records and (5) a milestone chart of the survey plan.

The survey itself is scheduled to begin in 1981 and cover a 12 month period. The reporting period is expected to cover the period January 1, 1981 through December 31, 1981. The survey is being designed around shipping documents and other records which will permit backtracking to the beginning of 1981.

OUTLINE OF THE STUDY

Phase I Tasks

TASK 1 Analyze surveys done in the past (e.g., NUREG-0073, Transport of Radioactive Material in the U.S.) and being done now on all radioactive material shipments. Specific points to be included and provided in report form are:

a. The identification of past and probable problems in data collection.
b. The determination of assumptions that can be used for projections of survey data.
c. The determination of the suitability of statistical techniques, survey procedures and data collection methods currently in use.
d. Comments on the usefulness of available data for answering questions and documenting reports.
e. What stratification of shippers and materials should be used.

TASK 2 Audit/analyze all current shipper, carrier and user documents to determine what data are already being collected, how it is collected and how it is reported. Parameters and data to be collected must include, but are not limited to:

a. Shipment Mode
b. Package Type
c. Shipping Distance
d. Material Shipped
e. Amount per Shipment (state units to be used in addition to number of packages)
f. Physical Form of the Radioactive Material
g. Chemical Form of the Radioactive Material
h. Shipment Frequency
i. Intended Use of Material Shipped (medical, research, etc.)
j. Origin, Points in Between, Destination Points (routing)

TASK 3 Investigate existing data bases to determine what data are available and in what format.

TASK 4 Prepare a comprehensive outline of the overall assessment plan. This assessment will include a discussion on the relative merits of various approaches for determining the quantity of radioactive material being shipped in the United States. At least two approaches will be considered.

One approach would be to update the work done for NUREG-0073, with appropriate modifications. Another approach would be to design a new plan starting with 1980 as the base year. Regardless of the approach, the information and data from TASKS 1, 2 and 3 will be incorporated. The plan will specify collection techniques, sampling requirements, analytic techniques for data reduction and preliminary guidelines for report format. The plan will include a milestone chart of the major elements to be accomplished.

Phase II Tasks

TASK 1 Design the assessment forms and consult with major segments of industry to assure maximum cooperation in using these forms.

TASK 2 Propose logical groupings for the nuclear materials so that the collection techniques will take into account significant parameters of a class of material or of a segment of the nuclear material industry (i.e., type of material [medical, spent fuel, etc.] or risk of material or on similarities in shipping procedures).

TASK 3 Incorporate any modifications to TASK 4 of Phase I necessary for collecting the data as specified in Phase I, TASK 2 and Phase II, TASK 2.

Phase III Tasks

Based on the results of Phase I and Phase II, authorization will be provided in the third phase to:

TASK 1 Establish the collection techniques and the implementation procedures to achieve desired statistical and extrapolative results.

TASK 2 Conduct the survey, do the calculations, establish data base, prepare interim reports and a final report.

SUMMARY OF RADIOACTIVE MATERIAL TRANSPORTATION ASSESSMENT PROGRAM PLAN

Method of Approach

An examination of prior surveys and of available source records indicates that the quantitative description of radioactive material transportation can be accomplished with as much precision and reliability as is desired by using a carefully selected, properly partitioned subset of radioactive material shipment data to estimate system-wide quantities. Further, this representative sample can be loaded into a computerized general purpose file management and statistical analysis system to establish a comprehensive data base. Finally, the experience gained through the survey and establishment of the data base should provide information adequate for recommendations for a methodology for updating this data base, either through ongoing or periodic data collection.

The major problem envisioned for the survey is the acquisition of data needed to establish and maintain the system data base. Several design principles will help overcome the data acquisition problem:

- Industry participants who supply input data should benefit from the output. If the output is useful, this will provide an incentive to supply high quality data.

- The confidentiality of data supplied to the system by industry must be respected and preserved to avoid violating the proprietary rights of individual firms. Confidentiality will be guaranteed.

- Data supplied by industry must result as much as possible as a natural spin-off of on-going operations. Although this is a standard system design principle, it is especially relevant in the nuclear and radionuclide industries where there are already extensive regulatory record-keeping requirements.

- Data must be acquired as economically as possible. This means that reliance on time-consuming survey questionnaires must be avoided or greatly minimized. In addition, data need not be collected from all industries that package, transport or otherwise handle RAM. The study design will be based on a scientifically selected sample of existing shipping records available at key points or nodes in the RAM transportation network.

The Sampling Frames

Two possible sampling frames were considered to represent the population. One was the set of entities licensed by NRC to use, ship, carry or otherwise handle radioactive material. Questionnaires administered to an appropriate subset of these licensees would allow estimation of radioactive material transportation quantities according to any logical grouping that is desirable to maintain. Another possible frame was the set of shipping papers generated over some desired time span. Again, an appropriate subset would allow estimation of radioactive material shipment quantities within any logical grouping of interest.

This study will base the assessment of radioactive material transport on shipping papers rather than to rely on questionnaires administered to licensees. The factors determining this decision were:

- Data quality.

- Costs.

- A close correspondence between the minimum data requirements and the data requirements for shipping papers specified in 49 CFR 172.202 and 172.203.

- A concentration of records within a limited number of entities, simplifying and reducing the costs of data collection.

Shipping Document Information Available

DOT regulations specify that shipping papers for recording radioactive material movement must be prepared for every shipment. The data required for shipping papers by 49 CFR 172.202 and 172.203 include:

- Proper shipping name of material.

- Hazard class.

- Quantity (weight, volume or other).
- Radionuclides present (from 49 CFR 173.396).
- Physical and chemical form (if in normal form).
- Activity for each package.
- Category of label applied to each package
- Transport index assigned to each package.
- Fissile class information.
- Package ID marking, if NRC-approved package.

Typically, these papers also include place of origin and destination place and company name, plus transporter company name. These data elements correspond to the data requirements for risk assessment (normal transport, accident and criticality).

A Radiation Shipment Record (RSR) currently used by the largest commercial waste management facility illustrates that all of the information required for the study can be obtained (mode, origin, destination), inferred (intended use of the material from shipper or destination name) or calculated (shipping distance) directly from this shipping paper. There is close correspondence between the requirements of the study and the shipping papers analyzed. Points between origin and destination as well as shipping distance may present some difficulties because routes sometimes include interim ports and more than one transportation mode, but in prior studies, several assumptions about routing have been used to resolve this complication.

Statistical Analysis

Most analysis in this study will deal with explaining the structure of the data population.

The key to selecting a representative sample from a limited number of companies or facilities is a detailed understanding of the radioactive material cycles or networks. Three such networks have been identified: the Nuclear Power Cycle, the Commercial Applications Cycle and the Institutional Applications Cycle. Within each cycle will be identified the major processing, usage and disposal points (called nodes) the major by-products at each node and the paths for each by-product between nodes. Finally, the number and name of each entity involved with the process or activity at each node will be identified. Once the three (or more) networks are completely understood, a sampling strategy can be formulated for collecting a maximum amount of data at a minimum cost.

The data type will determine the statistical tests that will be applied to explore the structure of the population. Classification data (such as radionuclide, chemical and physical forms) are used merely to classify radioactive material transportation. Ordinal data (such as hazard class and transport index) are used to assign a relative relationship (e.g., more or less than) to members of each group. Ratio-measurements (e.g., quantity, volume, weight, activity) correspond to common and consistent units of actual measurement. A variety of statistical tests are required for analysis of radioactive material transportation data. For example, classificatory and ordinal data may require analysis

via nonparametric or distribution-free statistical tests.

Conceptual System Design Framework

The design framework for the study is presented schematically in Figure 1. The heart of the system is the system data base shown between the EDIT program and the "Variable Builder" program. The stream of new data to be added to the system data base is outlined below (each step corresponds to the number shown in the illustration).

(1) Shipping records acquired from industry data sources arrive at the study operations center. These records, probably copies made and coded from industry files, are the primary data source for the prototype system.

(2) Supplementary data such as special surveys or other data bases (e.g., the Hazardous Material Accident Incident Reports) can be entered at this point if they are to become part of the system data base.

(3) CADE (or a software package similar to this UNIVAC program) is used to translate the input data into machine-readable format. An operator keys the data onto a magnetic tape or other storage medium.

(4) Fixed format tape is used to store the data temporarily on magnetic tape.

(5) SAS Loader (Statistical Analysis System Loader program) is used to establish a SAS-compatible data base. SAS is an extremely powerful software package that incorporates a high level user-oriented language. It is SAS that provides the analytical and reporting flexibility.

(6) System specifications are used to instruct the SAS loader on SAS file requirements. The SAS data dictionary is included in the SAS file. The SAS file after EDIT becomes the system data base. The data dictionary includes edit and validation criteria that are automatically applied to data elements.

(7) Data dictionary external to the system data base will be maintained on a word processing system (a Wang system in this case) and be provided to users as reference documents. There is a dictionary for each unique data input format although data elements are common on all formats as appropriate.

(8) SAS file comprises an intermediate stage for new unedited data that is being processed into the system data base.

(9) SAS Edit results in data passing the edit checks going into the system data base and data failing the edit checks noted on an error listing to facilitate manual corrections of inputs (Steps 1 and 2).

(10) System data base stores the edited, clean data plus the SAS data dictionary.

(11) Sample weights resulting from the sampling strategy must be used to estimate the total universe or one or more of its strata (or partitions) from the sample data contained in the system data base. These weights will be predetermined and stored in the data base.

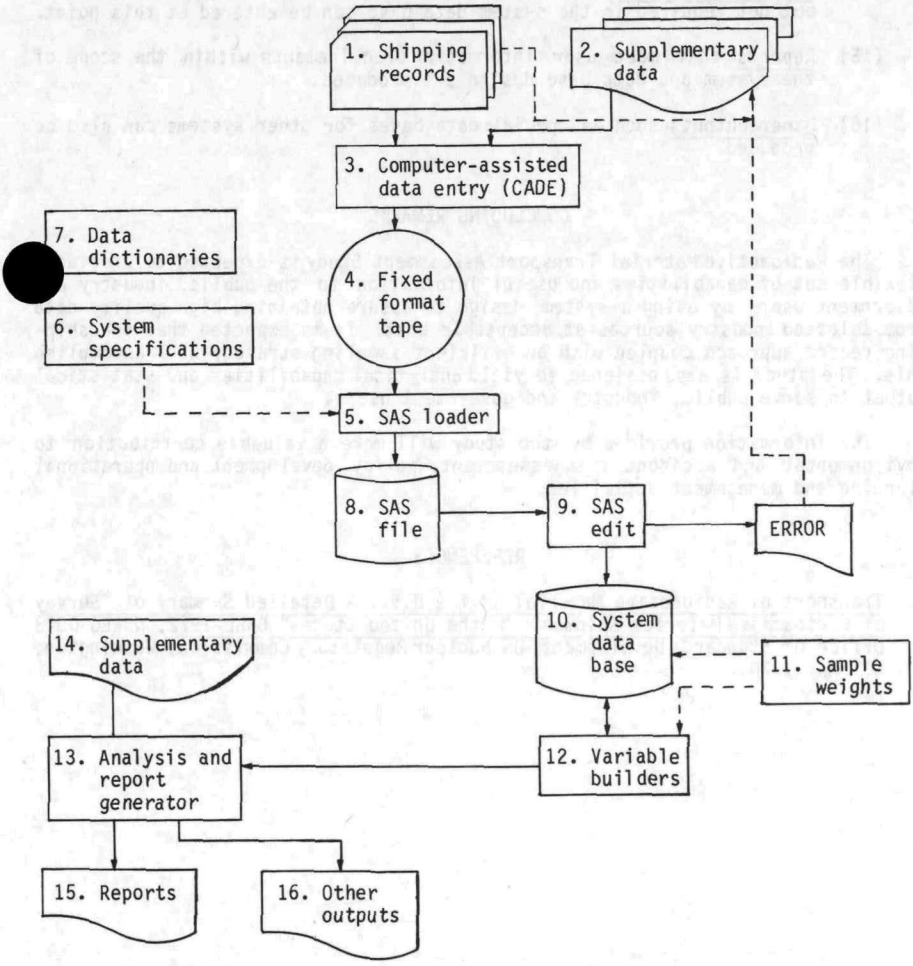


Figure 1. Study conceptual framework.

(12) Variable builders are SAS programs that develop the risk analysis, policy and management variables needed for user analyses and reports. An analyst or programmer trained in the easy-to-learn SAS language can prepare the needed programs. Initially, the SRI systems team will perform this task. This step utilizes the weights needed to perform tabulations within the context of the universe of interest.

(13) Analysis and report generator capabilities of SAS can now be used to perform the desired analysis and generate the desired reports under the control of an analyst (initially SRI).

(14) Supplementary data required for a particular analysis or reporting task but not required in the system data base can be entered at this point.

(15) Reports that meet user information requirements within the scope of the system and data base design are produced.

(16) Other outputs such as special data bases for other systems can also be produced.

CONCLUDING REMARKS

The Radioactive Material Transport Assessment Study is expected to provide a flexible set of capabilities and useful information to the public, industry and government users by using a system design to assure obtaining high quality data from selected industry sources at acceptable cost. It is expected that the shipping record approach coupled with an efficient sampling strategy will accomplish this. The study is also designed to yield analytical capabilities and statistical output to serve public, industry and government users.

The information provided by the study will make a valuable contribution to environmental and accident risk assessment, policy development and operational planning and management activities.

REFERENCES

1. Transport of Radioactive Material in the U.S., A Detailed Summary of "Survey of Radioactive Material Shipments in the United States" BNWL-1972, NUREG-0073 Office of Standards Development, US Nuclear Regulatory Commission, Washington, DC, May 1976.

Comment by A. W. Grella, USA: Your remarks on the new study tend to make me conclude that you have come down rather hard on the 1975 study. I would like to comment in defence of that study that it did provide useful data and its statistical sampling methodology did take account of the groups being sampled. I think your study may have set some over optimistic goals in depending on shipping paper data which shippers are not legally required to maintain.

Answer: You are right. I did not mean to so degrade the 1975 study but instead to stress that in the public mind that study is now outdated. The data provided by the previous study has been widely used and has found the basis of risk studies since its publication. Likewise I recognise that the study I have described is very optimistic

Question by R. F. Barker, Austria: What are your reasons for setting the system up so it can be kept current?

Answer: This is more of a concession to the public than a technical need. Unless there are major shifts in public policy the volume of radioactive materials shipped in the US obviously changes only a little each year. This study is to include up-dating provisions so that the slope of that change can be estimated and so that the public will not assume that the data are not current simply because they are several years old.

Question by W. G. Bevan, USA: In your analysis of transport data will you assume internal routing guides of common carriers as the route of movement for LTL shipments rather than to attempt to trace the actual route of movement?

Answer: Yes. No available information will give exact routing for individual shipments so algorithms are used to predict most likely routes used for shipments.

THE SYSTEM OF INSURANCE AND INDEMNITY COVERAGE FOR PUBLIC LIABILITY ASSOCIATED WITH NUCLEAR MATERIALS IN THE COURSE OF TRANSPORTATION IN THE UNITED STATES

by
Omer F. Brown, II
Office of the General Counsel
United States Department of Energy
Washington, District of Columbia, U.S.A.

INTRODUCTION

As the number of shipments of radioactive materials in the United States has increased, more people have become concerned about public liability and compensation issues. This growing concern and the increasing effects of other institutional factors on the packaging and transportation of radioactive materials both are reflected in the number of papers on "non-technical" matters being presented at this year's symposium. Two years ago at the fifth symposium in Las Vegas, I was one of only a few speakers to address a "non-technical" subject. 1/ This year, there unfortunately are more of us. I say "unfortunately", because institutional restraints that more and more are inhibiting movements of radioactive materials appear inversely proportional to the safety record of such movements.

Instead of imposing unnecessary restrictions on the movement of radioactive materials, concerned persons should consider the regulatory scheme that applies to these materials (in many cases, internationally) as a high standard that shippers and transporters of other hazardous commodities would be well advised to emulate. It is our obligation to better educate them about how effective this regulatory scheme is. One of the most misunderstood aspects of it is coverage for public liability. This presentation will attempt to describe the breadth of that coverage.

Few realize that the regulatory scheme for transportation of radioactive materials in the United States is especially comprehensive not only as a result of adherence to strict safety standards, but also because of the omnibus system of insurance and indemnity coverage for public liability. We long have recognized that, even with stringent packaging and transportation safety standards, the possibility of a transportation accident resulting in some measure of damages to persons or the environment, while remote, cannot be totally eliminated. Therefore, the Federal Government and the nuclear and insurance communities have developed a broad system of insurance and indemnity protection to make funds available when there is public liability as a result of an incident involving the transportation of nuclear materials.

THE BASIS FOR PUBLIC LIABILITY

In the United States with our federal system of government, liability for damages to persons or property can be imposed by either Federal or state statute or case tort law. 2/ In the European countries that have implemented the 1960 Paris Convention 3/, there is a system of strict liability, irrespective of fault, for nuclear damage. In our country, on the other hand, there are a number of

traditional legal theories, including negligence and strict liability, on which public liability may be based.

As in other common law countries, our tort laws also are not static. Therefore, the basis for public liability in the United States may vary from jurisdiction to jurisdiction and from time to time. When not specifically imposed by statute, nuclear tort liability decisions would be derived largely from past judicial precedents with the legal doctrine of stare decisis providing a degree of predictability. For this discussion, it is sufficient to note that the local law of the State that has the most significant relationship to the occurrence and the entities involved usually would govern both as to liability and compensation. 4/ principal concern here is with the sources of funds for compensation when ce is liability.

THE SCOPE OF INSURANCE AND INDEMNITY COVERAGE

The Federal Government by adopting the Price-Anderson Act 5/ in 1957 established a unique system of private insurance and Federal Government indemnity. This system provides broad coverage for public liability associated with nuclear material both while it is at a covered nuclear facility or in the course of transportation to or from such a facility. Congress enacted this statute for the dual purpose of (1) assuring that funds would be available in the unlikely event of a serious nuclear incident, and (2) encouraging private industry to participate in the nuclear field. This statute since has been re-enacted twice and amended several other times by Congress. The most recent re-enactment and amendment was at the end of 1975. 6/ The 1975 amendments provide for the phasing out of the Federal Government's role as an indemnitor of nuclear power plant licensees with the application of the Industry Retrospective Rating Plan described below. When this phase out is complete, government indemnity coverage for transportation activities essentially will apply only to shipments made by nonprofit educational institutions, foreign reactors, and the Federal Government and its contractors.

The application of the Price-Anderson financial protection system to commercial nuclear facility licensees and the government indemnity agreements with those licensees are administered by the United States Nuclear Regulatory Commission (NRC). 7/ The Federal Government self-insures its risk in the area of nuclear materials transportation as it does generally. Some Federal Government nuclear contractors and nonprofit educational institutions also have indemnity agreements with the Federal Government. These agreements are entered into under the Price-Anderson Act by either the NRC or the United States Department of Energy (DOE). The indemnity agreements with nuclear contractors can cover liability up to $500 million for each nuclear incident occurring within the United States or up to $100 million for each nuclear incident occurring outside the United States. The indemnity agreements with nonprofit educational institutions cover liability between $250 thousand and $500 million. The indemnity agreements may be the sole source of funds for public liability where there is not also underlying liability insurance from private sources.

Private insurance, when applicable, is provided by two "pools" of property and liability insurance companies. A pool is a group of companies that pledges assets that together provide resources to insure risks of a size that would be beyond the financial capability of a single company. The pool of 136 stock insurance companies is called American Nuclear Insurers, while the pool of 110 mutual insurance companies is called the Mutual Atomic Energy Liability Underwriters. The insurance industry world-wide participates in the American pools through reinsurance treaties. These non-American reinsurers account for approximately 49 percent of our domestic private insurance capacity. The insurance pools issue

two types of liability policies: the Facility Form, and the Supplier's and Transporter's Form.

Shipments of some materials of low radioactivity (and hence lower risk), such as uranium ore, yellowcake, and radiopharmaceuticals, are not covered under the specific provisions of the Price-Anderson Act. Instead, they are covered by conventional liability insurance issued by individual private insurance companies. A table accompanying this paper indicates the insurance and/or indemnity coverage for public liability applicable for particular types of radioactive shipments.

Most public attention has centered on shipments to or from nuclear power plants. The Price-Anderson Act requires that every operator of a nuclear power plant provide to the NRC evidence of financial protection equal to the maximum nuclear liability insurance available from private sources at reasonable cost terms as a condition of its license. Power plant operators could satisfy this financial protection requirement by a means other than the purchase of a Facility Form policy provided by the pools; however, thus far, they all have elected to purchase the Facility Form of protection to satisfy the Act's financial protection requirements. Facility Form coverage for nuclear power plants amounted to $60 million in 1957 which at the time was an unprecedented amount of liability insurance. In the intervening years, it has more than doubled to $160 million, effective May 1, 1979. 8/

In December 1975, Congress amended the Price-Anderson Act by creating an Industry Retrospective Rating Plan for the purpose of substantially increasing the amount of financial protection to the public provided by private resources. The Retrospective Plan, which went into effect on August 1, 1977, applies to nuclear power plants and to nuclear material in the course of transportation to or from such facilities. This new system provides that, in the event the amount of financial protection provided by the Facility Form is insufficient to pay all claims power plant operators would be assessed up to $5 million per incident retrospectively for each reactor to pay for loss that exceeds nuclear liability insurance coverage (now $160 million). Payment by the operator is guaranteed by the insurance pools or the NRC. With 70 power plants in operation, the Retrospective Plan (as of October 1980) provides at least $350 million more of private financial protection, displacing the Federal Government's indemnity role by that amount. The Act specifies that, if a nuclear liability loss exceeds the private financial protection, the Federal Government is to indemnify the person liable up to a total of $560 million. Today, that means that the Federal Government's indemnity for nuclear shipments to or from power plants amounts to $50 million, for a total of $560 million of coverage. As the insurance pools increase their primary layer of financial protection and as additional power plants are licensed to operate (thus increasing the secondary layer), the total amount of financial protection will grow beyond $560 million. Furthermore, the Price-Anderson Act provides that liability is limited to the amount of coverage provided by the system. This limitation-on-liability provision was upheld unanimously by the Supreme Court of the United States in 1978. 9/

As shown on the table, public liability for some shipments is covered by indemnity agreement alone. These are shipments of nuclear materials other than those to or from facilities covered by the pools' Facility Form or the Industry Retrospective Rating Plan. They include, for example, some shipments to or from Federal Government facilities and facilities of indemnified NRC and DOE contractors. They also include shipments within the United States of spent nuclear fuel from foreign reactors. Transportation coverage under these indemnity agreements is virtually the same as that provided by the Facility Form and the Retrospective Plan.

Transportation liability coverage is afforded by the "omnibus" features of the Price-Anderson insurance-indemnity system. This is the case whether either the Facility Form, the Retrospective Plan or an indemnity agreement applies. All transport modes are covered. The system affords very broad liability protection to anyone liable under applicable law (not just the named insured) for personal injury or property damage arising from the transport of nuclear materials. Liability even for a terrorist act that causes the dispersal of nuclear material is covered as long as the act occurs in the course of the insured transportation. Although the system covers nuclear liability from the first dollar of loss, no one but the nuclear facility licensee or indemnified entity pays a premium or fee, if any, for protection. The system further provides for interim emergency assistance payments to persons or organizations affected by a nuclear incident without requiring them to sign a release or otherwise compromise their claims.

Additionally, transporters of nuclear materials can purchase liability coverage under the pools' Supplier's and Transporter's Form in amounts up to $160 million. This coverage is available principally to protect a carrier's liability arising out of the transport of nuclear material to or from locations other than nuclear facilities where indemnity agreements apply.

TWO MISUNDERSTOOD FEATURES

The Price-Anderson system includes two often misunderstood features that, contrary to the views of critics, actually provide for a higher level of public protection than otherwise might be available in the event of a nuclear incident. These are (1) the Price-Anderson Act provisions that apply when the Federal Government determines that an "extraordinary nuclear occurrence" has taken place and (2) the nuclear exclusions that appear in most conventional property and liability insurance policies purchased by members of the public.

The "extraordinary nuclear occurrence" (ENO) provision was added to the Price-Anderson Act in 1966 for the purpose of further assuring prompt compensation to the public for serious nuclear incidents without at the same time totally displacing State laws by creation of a "Federal tort". The 1966 amendment provides that, in the event of an ENO, certain key State law defenses are waived and all cases can be consolidated in a single Federal district court. Congress did not wish to make these provisions applicable to all nuclear incidents for fear of encouraging nuisance suits. 10/ Determination as to whether an incident was an ENO would be made by the NRC or DOE on the basis of predetermined criteria. 11/ As a result of the waiver of defenses, a person suffering nuclear injury or damage as a result of an ENO need show only a causal connection between his injury or damage and the nuclear incident in order to recover. In other words, when there is an ENO, there essentially is a "no-fault" recovery scheme. Without usually available tort law defenses, such as contributory negligence or assumption of risk, liability should be easier to establish when there is an ENO. Use of the ENO provision therefore would provide greater protection to the public than would exist under conventional tort law.

Some have criticized the fact that most conventional property and liability insurance policies purchased by members of the public exclude coverage for nuclear incidents. They seem to imply that these exclusions deny any coverage for nuclear risks. However, these nuclear exclusions merely are a reflection of the comprehensive system of coverage already provided the public by the insurance pools' nuclear liability policies (for which others pay the premiums). The private insurance industry has channeled its resources through the pools into the Price-Anderson system for the purpose of accumulating large sums of insurance. There are nuclear exclusions on conventional liability policies in order to

avoid cumulation of loss from both the pools and individual insurance policies issued by separate insurance companies. The factor of importance to the public is that coverage is afforded, rather than under which policy it is provided. The basic protection that the insured under a conventional policy requires for liability due to the nuclear activities involved is provided to him not in his conventional policies, but through the Price-Anderson system, while the costs of this coverage are borne by others.

CONCLUSION

The breadth of existing liability coverage for nuclear shipments becomes more evident when compared to the coverage now applied or even proposed for shipments of other hazardous materials. 12/ For example, existing compensation schemes for incidents involving oil spills concentrate mainly on assuring there will be funds for cleanup rather than also for compensating individuals. Even the so-called "superfund" bill that passed the United States House of Representatives in September of this year would not provide coverage as comprehensive as that already applicable to nuclear shipments. 13/ The "superfund" bill, which would apply to spills of oil and hazardous materials, such as propylene and chlorine, for example, generally would limit liability to $50 million per incident and would not provide compensation for personal injuries.

The Price-Anderson system of insurance and indemnity coverage remains the most comprehensive Federal program for compensating persons or organizations when damaged by an incident involving a hazardous material. While there always is room for improvement, the nuclear insurance-indemnity system, together with the stringent nuclear safety regulations, provides a very high level of public protection against the potential hazards of radioactive shipments. As noted earlier, these standards should serve as examples to be emulated by anyone concerned with regulating any hazardous materials.

* * * *

Selected References

1/ O.F. Brown, Private Shippers' Efforts to Transport Spent Nuclear Fuel and Waste By Rail - The Interstate Commerce Commission Cases, Proceedings of the 5th Int'l. Symposium - Packaging and Transportation of Radioactive Materials, Las Vegas, Nevada (May 7-12, 1978), vol. 1 at 186. One of the issues in those cases was the adequacy of liability coverage to protect a railroad in the event of a nuclear incident. For a specific discussion of the liability issues raised by the railroads, see Talk by O.F. Brown, Nuclear Transportation: Insurance Issues, Atomic Industrial Forum Workshop on Insurance, Newport, Rhode Island (June 26, 1979).

2/ See generally 86 C.J.S. Torts §§1 et seq. (1954). Claims against the Federal Government are governed by the provisions of the Federal Tort Claims Act, 28 U.S.C. §§2671 et seq.

3/ The Paris Convention on Third Party Liability in the Field of Nuclear Ene See P.Reyners, Current Nuclear Damage Indemnification Issues in Europe, Atomic Industrial Forum Conference on Legal and Insurance Issues, New York, New York (June 1-4, 1980).

4/ See generally Restatement (Second) of Conflict of Laws §§145 et seq. (1971), and 86 C.J.S. Torts §§24 et seq. (1954).

5/ Pub.L. 85-256; 71 Stat. 576 (codified as section 170 of the Atomic Energy Act of 1954, as amended; 42 U.S.C. §2210).

6/ Act of Dec. 31, 1975; Pub.L. No. 94-197; 89 Stat. 1111. In March 1976, the NRC published a notice inviting advice and recommendations from interested persons with regard to its intent to modify its regulations in light of the 1975 amendments. 41 Fed. Reg. 11327-28. The NRC said it was particularly interested in receiving views with respect to any advantage to the public or the carrier that would result from separate transportation coverage by the Price-Anderson Act as contrasted with present coverage under the "omnibus" features of the Act. Id. at 11328. It asked that deficiencies, if any, in public protection under present coverage be identified. Id. In September 1976, after receiving public comments, the NRC published proposed rules to implement the 1975 amendments. 41 Fed. Reg. 40511-21. It said in its notice that comments received on the issue of transportation coverage generally expressed the view that the "omnibus" provisions of existing licensee financial protection plus indemnity coverage were adequate. Id. at 40515. No contrary views were received. Id. New regulations implementing the 1975 Price-Anderson amendments were published in January 1977. 42 Fed. Reg. 46-54. They went into effect on August 1, 1977, without any new provisions initiating separate transportation coverage.

7/ See 10 C.F.R. Part 140.

8/ 44 Fed. Reg. 20633 (1979).

9/ Duke Power Co. v. Carolina Environmental Study Group, 438 U.S. 59 (1978).

10/ See Excerpts from JCAE Report September 16, 1966, Accompanying Bills to Amend Price-Anderson Act to Provide Immediate Financial Assistance to Claimants and to Require Waiver of Defenses, reprinted in Selected Materials on Atomic Energy Indemnity and Insurance Legislation, Joint Comm. on Atomic Energy, 93d Cong., 2d Sess. (1974) at 299-332. See also Proposed Amendments to Price-Anderson Act Relating to Waiver of Defenses, Hearings Before the Joint Comm. on Atomic Energy, 89th Cong., 2d Sess. 31, 75-76 (1966) (Statement of Rep. Holifield).

11/ The NRC in April 1980 determined that the events at Three Mile Island Nuclear Station in March 1979, while "extraordinary" in ordinary parlance, did not constitute an "extraordinary nuclear occurrence" as that term is defined by the Price-Anderson Act and the NRC's regulations. 45 Fed. Reg. 27590 (1980). The NRC now is conducting a rulemaking to examine the need to modify the current ENO criteria and, if necessary, the statute itself. Id.

12/ See generally L.M. Trosten and O.F. Brown, Energy Law Service Monograph - Transportation of Energy Resources (August 1979) §§14A.60 et seq.

13/ See H.R. 85, 96th Cong., 2d Sess., "the Comprehensive Oil and Hazardous Substances Pollution Liability and Compensation Act", 126 Cong. Rec. H9185 (daily ed. Sept. 19, 1980).

Low-Level Waste	Government or Educational Facility	Disposal Facility	Government Indemnity 3,4/
Radiopharmaceuticals and Radioactive Isotopes	Manufacturer or User	User or Disposal Facility	Conventional Liability Insurance 5/ or sometimes Supplier's and Transporter's Form 6/

1/ The Facility Form is the liability insurance policy issued by the nuclear insurance pools to licensees required by the U.S. Nuclear Regulatory Commission to maintain financial protection to cover public liability claims (now in the amount of $160 million for nuclear power plants). This provides broad coverage for liability arising out of nuclear material while it is at an insured facility or in the course of transportation to or from the facility.

2/ The Industry Retrospective Rating Plan Premium of $5 million each would be assessed against the operators of nuclear power plants in the event the amount of financial protection provided by the Facility Form were insufficient to pay all claims. At the present time (October 1980), this would amount to an additional $350 million of private financial protection (i.e., 70 operating nuclear power plants times $5 million each).

3/ Indemnity for public liability is provided by the Federal Government under agreements entered into pursuant to the Price-Anderson Act. These agreements are entered into between the Federal Government and its contractors (for liability up to $500 million) or nonprofit educational institutions (for liability between $250 thousand and $500 million) as well as between the Federal Government and operators of nuclear power plants. In the case of shipments to or from nuclear power plants, the total coverage is now $560 million (i.e., $160 million under the Facility Form, $350 million under the Industry Retrospective Rating Plan, and $50 million under the Federal Government indemnity agreement). In the case of nonprofit educational institutions, the first $250 thousand usually is covered under a separate Facility Form for that amount.

4/ If the shipment is to or from a private nuclear installation covered by a Facility Form issued by the nuclear insurance pools, coverage would be available under both the Facility Form and government indemnity.

5/ Conventional liability insurance policies issued by individual private insurance companies cover shipments of some materials of low radioactivity (and hence lower risk).

6/ The Supplier's and Transporter's (S&T) Form is the liability insurance policy issued by the nuclear insurance pools to certain entities handling radioactive materials (in amounts up to $160 million). It usually is purchased only by entities shipping or transporting materials with high radioactivity levels. Some S&T Forms have been written to include radiopharmaceuticals and radioactive isotopes, which usually are covered only by conventional insurance policies.

Applicable Insurance and/or Indemnity Coverage for Public Liability Associated with Nuclear Materials In the Course of Transportation in the United States

Nuclear Material	Origin	Destination	Applicable Coverage for Liability
Spent Nuclear Fuel	Nuclear Power Plant	Storage Facility or Reprocessing Plant	Facility Form, 1/ Industry Retrospective Rating Plan, 2/ Government Indemnity 3/
Spent Nuclear Fuel	Government, Educational or Foreign Reactor	Storage Facility or Reprocessing Plant	Government Indemnity 3,4/
High-Level Waste	Storage Facility or Reprocessing Plant	Storage or Disposal Facility	Government Indemnity 3,4/
Unirradiated Fuel	Fuel Fabrication Facility	Nuclear Power Plant	Facility Form, 1/ Industry Retrospective Rating Plan, 2/ Government Indemnity 3/
Unirradiated Fuel	Fuel Fabrication Facility	Government or Educational Reactor	Government Indemnity 3,4/
Enriched Uranium Hexaflouride (UF_6)	Enrichment Plant	Fuel Fabrication Facility	Government Indemnity 3,4/
Uranium Hexaflouride (UF_6)	Conversion Plant	Enrichment Plant	Government Indemnity 3/
Uranium Oxide (U_3O_8 - Yellowcake)	Uranium Mill	Conversion Plant	Conventional Liability Insurance 5/
Uranium Ore	Uranium Mine	Uranium Mill	Conventional Liability Insurance 5/
Low-Level Waste	Nuclear Power Plant	Disposal Facility	Facility Form, 1/ Industry Retrospective Rating Plan, 2/ Government Indemnity 3/

DISCUSSION

Question by W. A. Brobst, USA: Assume a traffic accident involving a shipment of spent fuel covered by Price-Anderson, but not releasing any radioactive material and not resulting in any nuclear injury or damage. In such a case, an evacuation would cause "psychological damage and emotional harm," could cause traffic injuries/deaths and damage, could result in nearby businesses being denied temp arily access to their buildings, and could mean temporary loss of use of a railroad right-of-way. What recourse do the "injured" persons have for compensation for their losses? What if there is ruling of "ENO?"

Answer: The Price-Anderson insurance indemnity system would provide compensation for such injury or damage if anyone is legally liable for them. There does not have to be an actual release of nuclear material for coverage to apply. As I indicated in my paper, a determination by the federal government that the incident was not an "extraordinary nuclear occurrence" does not preclude recovering damages. The Three Mile Island incident of March 1979 is an example of an incident that, while "extraordinary" in ordinary parlance, was determined not to be an "extraordinary nuclear occurrence" within the meaning of the Price-Anderson Act and the Nuclear Regulatory Commission's regulations. In that case, the insurance pools made emergency assistance payments and advanced funds to those in need. These payments amounted to about $ 2 million. Price-Anderson will cover anyone found legally liable for the events at Three Mile Island up to $ 560 million.

Question by B. W. Emmerson, United Kingdom: Can you comment on the application of the Price-Anderson insurance indemnity system in the case of an individual who pursues a claim for damages although there is minimal supporting evidence of actual harm being caused? For example, compensation for the supposed loss of property value caused by a passing vehicle carrying radioactive materials, even though an accident does not occur.

Does the Price-Anderson System provide indemnity against punitive damage claims, bearing in mind that for any particular claim, the punitive damages may greatly exceed those awarded for physical damage?

Answer: The Price-Anderson system provides financial coverage for anyone legally liable for damages. Therefore, the issue is whether there is liability. The answer to the question of liability may vary from jurisdiction to jurisdiction. If a court determined that there was liability for loss of property value caused by a passing vehicle carrying nuclear materials to or from a facility covered by Price-Anderson, that liability would be covered by the system.

Whether the Price-Anderson system provides insurance or indemnity against punitive damage claims is an open question. It ordinarily, however, is considered contrary to public policy to provide insurance or indemnity coverage for punitive damages. This question may be answered in the Karen Silkwood case pending in the Oklahoma courts. That case now is on appeal from a jury award of both compensatory and punitive damages. In that case, the punitive damages were several times larger than the compensatory damages judgment.

LIABILITY AND INSURANCE DURING INTERNATIONAL TRANSPORTS OF NUCLEAR FUEL

Ulrich E. Timm
Transnuklear GmbH
6450 Hanau 11

A remarkable safety record has up to now governed international transports of nuclear fuel. No serious incidents involving nuclear damages arising from nuclear consignments have yet occurred. It is, however, evident that in spite of all high safety requirements there still remains the risk of nuclear accidents in the course of carriage of nuclear material. The possibility of an incident giving rise to nuclear damage to both persons and property cannot be neglected.

The question of civil liability for nuclear damages arising from nuclear incidents in the course of carriage of nuclear material was given attention from the beginning of the peaceful use of nuclear energy.

1. Nuclear Liability during International Transports

In the event of nuclear damage the following decisive questions immediately arise:

- Who is liable?
- Is sufficient financial protection available?
- Which court will rule?
- Which law is applicable?

It is most important, that these questions can be easily answered.

This interest is widespread, because at least three different groups of interests can be affected by a nuclear damage occurring in the course of carriage of nuclear substances.

(1) The public, i.e. persons and property without direct relation or contact with the relevant nuclear material.

(2) Persons and goods on board the same means of transport as the relevant nuclear material.

(3) The means of transport on which the relevant nuclear material is carried.

Especially during international transports it is, however, difficult to give the expected answers: The transport might be passing through several foreign countries and on the high seas, there may be means of transport and persons originating from various countries involved, and the legal systems of these countries may be very different.

It was not possible, up to now, to agree on a third party liability and insurance system acceptable worldwide. The most successful attempt to give an international answer to these problems is the Paris Convention and its Brussels Supplementary Convention.

These two Conventions have to be considered together. They represent a comprehensive nuclear third party liability system in which the Paris Convention represents the main and basic legal instrument.

1.1 The Paris Convention

The "Convention on Third Party Liability in the Field of Nuclear Energy"

was signed in Paris on July 29, 1960. Since then this international agreement has been ratified or acceded to by the following countries:

Germany, Belgium, Denmark, Finland, France, Greece, Italy, Norway, Portugal, Sweden, Spain, Turkey, and the United Kingdom. As the most recent country, up to now, the Netherlands ratified the Paris Convention in December 1979.

The Paris Convention applies to the carriage of all "nuclear substances" (art. 4) which include nuclear fuel other than natural and depleted uranium and all radioactive products or waste (art. 1(a) (v)). It can be assumed that this definition includes all materials for which nuclear damage is conceivable.

The main principles of the Paris Convention of interest for international transports can be summarized as follows:

(1) There is only one person liable - the operator of the nuclear installation from which the nuclear substances have been transported (art. 4 (a), art. 6 (b)). In specific cases it is the operator to whom the material is carried, who is liable (art. 4 (b), (c)). The liability of the operator is absolute - without proof of fault - and limited to amounts between 5 and 15 million European Monetary Agreement units of account (EMA u/a = 1,3 US$ at present) (art. 7).

(2) Full financial protection is granted - by insurance or other means up to the maximum amount of the operator's liability (art. 10).

(3) Only one court has jurisdiction - the one of the country in whose territory the nuclear incident occurred (art. 13 (b)).

(4) Only one law and legislation is applicable - the Paris Convention system (art. 6 (a)).

This short analysis proves that the aforementioned four decisive questions are adequately answered.

Due to the fact that the Paris Convention channels all nuclear liability to the operator of the nuclear installation, the carrier of nuclear substances is normally free of any nuclear liability. He may, however, accept liability under the Convention in place of the operator of a nuclear installation, if this is so agreed between them and approved by the competent public authority (art. 4 (e) Paris Convention). In this case the carrier is then considered as an operator of a nuclear installation for all purposes of the Paris Convention.

This option is exercised by specialized carriers based in the member States because they are generally more experienced in the field of international transport of nuclear fuel including insurance and all other connected problems, and want to offer their clients a comprehensive service.

1.2 The Brussels Supplementary Convention

The "Convention Supplementary to the Paris Convention of July 29, 1960 on Third Party Liability in the Field of Nuclear Energy"

was signed in Brussels on January 31, 1963.

The Convention has been ratified or acceded to by the following countries:

Germany, Denmark, Finland, France, Italy, Norway, Sweden, Spain, the United Kingdom, and the Netherlands.

The reason for this Convention was the common conviction that the upper limit of liability fixed by art. 7 (b) of the Paris Convention to amounts between 5 and 15 million EMA u/a is not sufficient. The Brussels Supplementary Convention sets up in art. 3 an additional system of compensation in two stages:

- A first portion of 70 million EMA u/a which is covered by the Government of the country where the installation of the nuclear operator is situated.

- A second layer of 50 million EMA u/a which is covered jointly by the member States of this Convention.

The total amount of compensation offered by the Paris Convention system therefore amounts to 120 million EMA u/a.

1.3 The Vienna Convention

The Paris Convention and the Brussels Supplementary Convention which have a decisive European character are nevertheless up to now the only successful attempt to find a solution to the civil liability problems of the international carriage of nuclear substances. Under the auspices of the IAEA the

"Vienna Convention on Civil Liability for Nuclear Damage"

was concluded on May 21, 1963 for worldwide interpretation.

The principles of this international agreement are very close to those of the Paris Convention. More negotiating parties being involved the result had obviously to be more general and vague; the maximum amount of liability is limited to the very low figure of 5 million US$. The coincidence with the conclusion of the Paris Convention was also of negative influence.

Argentina, Bolivia, Cameroun, Cuba, the Phillipines, Trinidad, Tobago, and Yugoslavia

ratified or acceded to this legislation so that it entered into force on November 12, 1977. Due to the fact that the main nuclear countries did not follow, the Vienna Convention has been of no practical value up to now.

1.4 The Brussels Nuclear Material Maritime Carriage Convention

The Paris Convention (art. 6 (b) and also the Vienna Convention (art. II (5)) created problems especially for the maritime carriage of nuclear material, because they explicitly do not affect the application of the existing international conventions in the field of transport. Due to the existing Maritime Conventions a dual liability could arise under their rules as well as under the rules of the nuclear Conventions. To avoid this, the

"Convention relating to Civil Liability in the Field of Maritime Carriage of Nuclear Material"

was signed in Brussels on December 17, 1971 and since ratified or acceded to by

Germany, Denmark, France, Norway, Sweden, and Spain.

Its aim is to make sure that the operator of a nuclear installation will be exclusively liable for damages caused by a nuclear incident occuring in the course of maritime carriage, thus exonerating shipowners and other persons involved from liability in maritime law for such nuclear damage.

Following art. 1 of this Convention this exoneration is granted if the operator of a nuclear installation is liable under either the Paris or the Vienna Convention or by virtue of a national law provided that such a law is favourable to persons who may suffer damage as these Conventions.

1.5 The Application of the Paris Convention System

One of the difficulties in the practical application of the Paris Convention is its obvious European character. As the most important nuclear countries of Europe did ratify or accede to the Paris Convention, it can be assumed that most European transports of nuclear fuel are governed by this third party liability and insurance system. This is, however, not true for the worldwide transport of nuclear substances, because the Paris Convention is far from being accepted by all countries concerned.

The aim of the Paris Convention to offer one broadly accepted international system of nuclear liability has therefore not yet been achieved. The United States of America are certainly the most important example, Japan another. The United States having closely participated in the negotiations of that Convention, in the end did not sign it.

Transports to, from or through the United States and other such countries obviously cause legal problems because normally the rules of the Paris Convention do not apply in the territory of non-contracting States (art. 2) so that different national laws have to be considered. This means among other things, that for the time being carriers and all other persons involved cannot be sure in the case of international transports that they are protected by the exclusive liability of the operator of the nuclear installation and his corresponding insurance cover. The application of national laws which do not accept this channelling of liability cannot be excluded.

The right of recourse against the operator which is offered to the person held liable under national laws (art. 6 (d) Paris Convention) only applies to residents of member States and is a difficult legal instrument. For the present, these uncertainties can only be covered by excessive insurance of all persons involved.

Furthermore, the Paris Convention System does not, up to now, function as perfectly as it could, because - for different reasons - not all member States of the Paris Convention have at the same time ratified or acceded to the Brussels Supplementary Convention and the Brussels Nuclear Material Maritime Carriage Convention. The consequence is - among others - that in some member States of the Paris Convention the additional financial protection up to 120 million EMA u/a of the Brussels Supplementary Convention is missing. In such cases only the insufficient maximum amount of the Paris Convention is available, if the member States have not by their own national legislation fixed a higher amount. Even in the case of transports between member States it has, therefore, to be checked in each special case whether these States did also ratify or accede to the Brussels Conventions whether they have fixed higher maximum amounts of liability by national law or whether they stick to the amounts given by the Paris Convention itself.

Another reason for difficulties is the fact that the application of the Paris Convention itself in the different member States is nowhere really identical. In some articles, the Paris Convention explicitly opens the possibility to the member States to go their own way in detail while respecting the main and decisive principles. For instance the limitations of liability in amount and time as well as the local application of the rules of the Paris Convention have been solved by the Contracting States in different ways. Even between the States who have accepted in principle the Paris Convention system, differences have therefore to be respected.

Thus, the German Atomic Law has fixed the maximum amount of nuclear liability to the outstanding figure of 1 billion DM (art. 31). The maximum amount up to which the operator has to provide financial protection during transport of nuclear fuel is limited to 50 million DM (art. 13 (2) 1). The missing amount up to 1 billion DM is covered by guarantee of the German State (art. 34). In this guarantee the amounts offered by the Brussels Supplementary Convention system are taken into account, so that the Paris Convention system is not neglected or altered, but only completed.

2. Comprehensive Insurance Cover

2.1 The Certificate of Insurance

One of the characteristics of the Paris Convention is that the exclusively liable person has to be covered by insurance or other equivalent financial protection up to the maximum amount of its possible liability. In case of international transports, this insurance cover is documented in a special certificate signed by the insurer and the competent authorities and accepted - in principle - by all member States of the Paris Convention (art. 4 (d) Paris Convention). With this certificate the authorities of each member State concerned with a nuclear transport do not have to verify whether the principles of the Paris Convention - e.g. competent liable person an comprehensive insurance cover - are respected.

However, unfortunately this positive system does not work without problems. First of all, the amount of liability is not identical in all member States, the authorities of a State through which the transport passes therefore have the right to ask for a higher amount of insurance protection than the certificate shows, up to the maximum amount of liability applicable in its own territory (art. 7 (e) Paris Convention).

In this case three solutions are possible and followed in practice. Either the certificate is correspondingly altered, or an additional insurance cover on top of the maximum amount covered by the certificate is concluded with an insurer in the demanding member State, or a complete new insurance cover for that special country is concluded. It is not necessary to argue that especially the last solution is certainly not in line with the intentions of the Paris Convention.

Another difficult item in this connection is that for instance art. 4a (2) of the German Atomic Law demands that - also in case of transports in transit - the insurance cover can only be accepted if it is given or at least guaranteed by a German insurer. This means, that in those cases where international transports go to or pass through Germany, an additional insurance cover has to be concluded for Germany or at least a German insurer has to grant "fronting" to the foreign insurance cover.

2.2 Risks outside the Paris Convention

For the territory of countries who are not members of the Paris Convention, this certificate is of no value. It is not accepted by the authorities of those countries and the principles of the Paris Convention to which it refers do not apply.

For the operator of a nuclear installation and the carrier a comprehensive nuclear third party liability insurance is, therefore, absolutely necessary. This insurance cannot be limited to the Paris Convention, but must also include all claims on the basis of other applicable laws. In this case it is also compulsory that not only the operator or the carrier are protected by the insurance, but that also all other persons involved in the transport get full protection up to a sufficiently high amount.

In Germany as in most countries, the insurance policy for the transport risk is separated from the policy for the local or "stationary" nuclear risk. Normally this policy offers the necessary financial protection in accordance with the principles of the Paris Convention and in addition - for those countries where this system does not apply - an insurance following the principles of "economic channelling". This means that the insurance policy includes financial protection for all persons who may be held liable in accordance with the applicable laws of other countries in connection with nuclear damages arising during the transport concerned.

Provided the maximum amount is high enough, this insurance cover should normally be sufficient for everybody, including shipowners, forwarding agents and any other persons involved. Only if national laws ask for it, has this insurance cover to be granted by different national insurance policies.

3. Conclusion

This short analysis proves that people concerned with international transports of nuclear fuel, such as operators, carriers, shipowners and forwarding agents, find themselves in a rather difficult position, because they are still faced with different international and national liability systems and cannot even be sure in all cases which system applies. For the time being, they have to look for a consequent risk analysis and to make sure they have an appropriate worldwide insurance cover with a competent insurer.

In the long run, this attempt to solve third party liability problems in the field of international transports of nuclear fuel on the basis of excessive insurance cover is, however, unsatisfactory.

I would have preferred to present the Paris Convention as a perfect system. The differences in its application I have quoted - and there are more - prove that this is not possible. However, it has to be acknowledged that the Paris Convention is the best existing instrument in the field of nuclear liability during nuclear transports.

Especially in Europe, where very often several frontiers have to be passed by a single transport, the Paris Convention has proved to function and to fulfil its task to make such transports manageable. Thus, it seems appropriate to solve the problems of civil liability in the field of international nuclear transports on the basis of the Paris Convention systems.

It is, therefore, time that the Paris Convention system was accepted by all nuclear countries and was - at the same time - much more standardized in its application as it is now. Another way out of the actual situation would be a new and decisive international attempt to negotiate a new nuclear third party liability system acceptable worldwide.

DISCUSSION

Question by D. Harmon, USA: What can one do to convince a steamship line to carry radioactive material if the ship's owners have a nuclear exclusion clause in their coverage on the high seas?

Answer: One has to provide that nuclear damage to the ship on board of which the relevant nuclear substances are carried is included in the operator's third party liability insurance policy. Normally this should be possible.

The Paris Convention says in principle that such damage to the carrying ship is excluded from the operator's liability, but offers the possibility to the member states to reinclude this risk by their own national legislation. This option has been exercised by nearly all member states (except Belgium, Italy, Spain). Correspondingly, in these states the nuclear risk of the carrying ship is automatically included in the operator's third party liability insurance policy.

These short remarks certainly cannot solve all possible problems, but should be sufficient to indicate the path which should be followed.

TRANSPORTATION OF RADIOACTIVE MATERIALS: LEGISLATIVE AND REGULATORY INFORMATION SYSTEM*

C. S. Fore,† and M. M. Heiskell††
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

The transportation of radioactive materials, as well as hazardous materials in general, has been an issue of ever-increasing concern and a subject of numerous regulations and legislative actions worldwide. The Transportation Technology Center (TTC) of the U.S. Department of Energy's Sandia Laboratories in Albuquerque, New Mexico, is currently involved in developing a national program to assure the safe shipment of radioactive materials. At Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee, this overall effort is being supported in a specialized manner. As part of the Logistics Modeling program at ORNL, the Ecological Sciences Information Center (ESIC) of ORNL's Information Center Complex/Information Division, has developed comprehensive data bases containing legislative and regulatory actions relevant to the transportation of hazardous materials. The data bases are separated according to status level of the legislation, that is, current legislation versus historical legislation. The Current Legislation Data Base includes all legislative actions introduced during the present year (1980) or those bills carried over from the previous year's sessions. The second data file, Historical Legislation Data Base, consists of all legislative actions since 1976 that have passed and become public laws, as well as those bills and regulations that were unsuccessful and were classified as having died or failed during one of the lawmaking sessions.

Currently, the data bases include state-, local-, and federal-level legislation, with emphasis on the transportation of radioactive materials. Also included are laws affecting carrier's operations such as the restrictions on the Baltimore tunnel and the Golden Gate Bridge. Because of their relevance to the transportation issues, related actions involving areas such as disposal and storage of radioactive wastes, moratoriums on power plant construction and siting, radiation safety and control studies, and remedial action studies, (e.g., decommissioning of nuclear facilities and cleanup of contaminated land), are also included in the data bases. The most common actions affectng transportation of radioactive materials involve establishment of specific agencies to regulate transportation of such materials, requests that notification be made

*Research sponsored by the Office of Nuclear Waste Management, U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation, through the Transportation Technology Center, Sandia National Laboratories.

†Ecological Sciences Information Center, Information Center Complex/Information Division.

††Oak Ridge Operations Office, Oak Ridge, Tennessee, formerly of the Chemical Technology Division, Oak Ridge National Laboratory.

prior to shipment, requirements for special permits and escorts, and attempts to ban completely the transportation of nuclear waste materials. Knowledge of all these regulations and legislative actions will be needed by shippers and carriers, as well as management and planning personnel. Consequently, this comprehensive collection of information will be useful to the international community in readily identifying specific restrictions within the United States.

Jointly, the data bases contain approximately 1300 entries that have been abstracted and indexed into a specialized format. This format establishes a system for storing, selectively searching, and documenting specific pieces of legislation. The information stored in the data bases has been grouped into 13 different subject categories. Those categories chosen to highlight the particular transportation restrictions are as follows:

1. Escort - areas where escorts are required and the agency which is to provide the escort;

2. Notify - areas where prenotification of registration is required;

3. Permit - areas where special permits, licenses, or special insurance requirements exist;

4. Prohibit - areas where transportation of all hazardous materials is banned; and,

5. Weight - areas where maximum weights and dimensions for truck and rail shipments are regulated and identified.

Other legislative information being categorized and of immediate relevance to the transportation issues is covered under the following subject areas:

1. Agency - state agency responsible for regulating transportation;

2. Disposal and Storage - legislative actions affecting disposal, storage, and overall management of radioactive wastes;

3. Emergency - emergency response regulations, as well as the agency contact within the state for emergency plans;

4. Legal Aspects - identification of legal and financial responsibilities (i.e., liabilities);

5. Legislative Contact - legislative services contact in each state or county for copies of bills;

6. Power - actions dealing with power plant construction and siting moratoriums, storage pool expansions, and radiation safety and control studies;

7. Regulations - legislative actions establishing the need for additional regulations relevant to transportation issues (e.g., establishment of study groups);

8. Remedial Actions - legislation dealing with decommissioning of nuclear facilities, cleanup of uranium mill tailings, and restoration of contaminated sites.

The subject categories can be easily revised as new technology develops and as needs change.

The following table shows the percentage of distribution for the categories, and the number of areas where legislative/regulatory actions have been implemented. The categories "Agency" and "Legislative Contact" were omitted from this list since they contain reference-type information.

Categories	Distribution of entries (%)	Areas in U.S. having transport restrictions
Escort	2	6
Notify	18	211
Permit	4	21
Prohibit	15	182
Weight	1	
Disposal and Storage	23	
Emergency	2	
Legal Aspects	3	
Power	21	
Regulations	6	
Remedial Actions	5	

Each entry is structured specifically for ease in retrieving all or specific parts of the bill or resolution. In addition to the generic subject categories, 14 basic pieces of information are being extracted on each legislative action and stored in specially designed identifier fields structured in a hierarchical manner. General data about each action are classified in the following identifier fields: state, locale, title of bills, and sponsor. All federal-level bills are grouped under the heading United States and entered in the state field. More specific information about each bill is gathered and organized in the following manner:

1. Mode - This field makes it possible to determine which mode or modes of transportation (water, rail, truck, air intermodal) are affected by the legislation.

2. Status - The status field provides a comprehensive and up-to-date summary of actions affecting each bill. The primary status levels utilized are: date of introduction to legislative body, date forwarded to a committee and name of committee, hearing dates established, date the lawmaking body acted upon the legislation and action taken, date the legislation was implemented into law or the action was reversed by other legislative actions. This field functions hierarchically in providing the current versus the historical status of a bill. The information is divided into two identifier fields distinguishing these differences, <C STATUS> for current status and <H STATUS> for historical status.

3. Bill number - This field includes a retrospective list of numbers that have been assigned to a bill or resolution, that is, House, Senate, or Assembly number given at date of introduction; public law number assigned when bill is adopted into law; and the referencing title and chapter numbers to the act that the bill is amending.

4. Type of Legislation - Legislation on the federal, state, county, and municipal levels is being incorporated into the data base. Also included are actions affecting carrier operations. Classified as operational, these actions specifically involve tunnels, bridges, toll roads, port authorities, major highways, interstates, etc.

5. Comments - The synopsis of each bill provides a brief summary of the legislative actions contained in the bill.

6. Abstract - An informative abstract describes in detail the actual contents of the bill. Where possible, the entire bill is entered.

7. Keywords - The keywords provide a generic means of selecting information. Each entry is indexed to reflect the area covered under selected subject category.

8. Region - Each bill or regulation is assigned a code that represents the geographic region being affected (e.g., SE for southeast and NW for northwest). These codes may serve as a link to the routing programs.

9. Source - This field identifies the initial supplier of the information and any additional source of information.

10. Reference Report - This field identifies any reports generated by study groups formed through previous legislative actions.

The information entered into the data bases comes from a variety of sources. The primary contributors are federal agencies (U.S. Department of Energy, U.S. Nuclear Regulatory Commission, U.S. Department of Transportation), legislative contacts within each state, regulatory bodies within each locale, carrier associations, published literature, and access to other related information systems.

Because legislative actions and regulations imposed on the transportation sector prohibit in many areas the use of an existing highway or rail system, selection of alternate shipping routes is often necessary. Work has begun on developing an interface between the truck- and rail-routing programs and the legislative data base, so that restrictions in specified areas of the country can be identified on route maps generated to reflect specific radioactive material shipments.

Plans are underway to expand the legislative and regulatory information system to cover information on actions that affect the movement of radioactive materials internationally. Such information is intended to include regulations and mode-specific shipping requirements and to identify restricted ports, highways, rail lines, and airfields. Differences in laws among countries, cities, states, provinces, and other divisions may also be noted. So that legislative information on an international level can be entered into the system on a routine basis, a source of information for the data needs to be established. Currently, technical literature published on the subject of transportation technology is being collected by the ESIC. It is envisioned that this literature, both domestic and foreign, will be abstracted, indexed, and entered into a separate specialized data base. The legislative information can then be cross-referenced with the published literature.

The contents of the data base are part of a comprehensive integrated information system serving many research and development projects. This system, called the Oak Ridge Computerized Hierarchical Information System (ORCHIS), is intended to serve the research and development community and its projects with an integrated information-processing, retrieval, and analysis system that can deal with both alphanumeric and digital information. The ORLOOK (Oak Ridge Look) program is an integral part of the ORCHIS system and provides a simple on-line method for retrieving all of the data or only specific pieces of the information requested. The information stored in the data base is available free to those researchers and decision makers representing governmental, academic, industrial, and foreign affiliations.

The Legislative and Regulatory Information System functions as a repository all legislation that immediately affects or could eventually affect the transportation of radioactive materials. Such legislative actions have been escalating at a high rate since 1977, particulary at the local level. As the potential for impacts grows and as the consequences of these impacts become more severe, the system will serve as a means of strengthening an awareness of the regulatory and legislative actions being generated and acted upon. The contents of the data bases could also be used for planning routes and determining trend analyses. The system has been in operation for approximately one year and is capable of meeting these needs.

Requests for additional information about the system or requests for information stored in the data bases should be forwarded to the following contacts:

Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, Tennessee 37830 USA

Attention: Mr. Larry B. Shappert
Mrs. Cathy S. Fore

Phone: (615) 576-2065
(615) 576-7769

Telex: 810-572-1076
USDOE OKRE

Transportaton Technology Center
Sandia Laboratories
P.O. Box 5800
Albuquerque, New Mexico 87185 USA

Attention: Mr. Edward W. Shepherd

Phone: (505) 844-1740

Telex: 910-989-1600

SESSION XVIII
STRUCTURAL AND THERMAL EVALUATION I

RESISTANCE TO WORSE-THAN-PROSCRIBED FIRE CONDITIONS

G. COHENDY (1) V. STADNIKOFF (2) B. DURET (3)
(1) CEA-FONTENAY AUX ROSES (2) CEA-LE BARP (3) CEA-GRENOBLE

1. INTRODUCTION

The specifications of the most recent AIEA recommendations (Ed. rev. 1973) ve not changed as regards the fire-resistance of type B containers : since 1964 these criteria consist of subjecting a model of such a container to a fire of 800°C for a duration of one-half hour.

It must first be remarked that these conditions are, in fact, quite severe ; originally they had been defined for the wort conceivable circumstances occuring in an accident : for example, a vehicle carrying such a container, with a full gas tank, being involved in a collision and subsequently taking fire.

2. SAFETY CRITERIA

Experimental installations have been constructed in France to meet these specified conditions, and have shown that, in practice, it is difficult to maintain a uniform temperative of 800°C about a container for one-half hour. In particular, as soon as a windspeed of 3 to 4 meters per second is attained the temperature of the exposed face descends below 800°C, even if the experiment is performed in a protected cavity, a circumstance which is, in addition, aggravated with respect to the reality. All the same one must demand if, since the 15 years from the adoption of the regulations, conditions yet more severe could possibly arise in practice. The first example which comes to mind is that of an airplane crash. Taking into account the quantity of inflammable material involved and the nature of the airplane's metallic structures, is it possible that one or the other, if not both, of the above limits would be exceeded ? As regards the temperature this seems quite probable if not certain for the container assembly as there in addition the effect of the wind would be unimpeded. On the contrary the possibility of a duration longer than one-half hour appears small if not negligable . In this case, then, one can envisage a fire characterised by a temperature, locally at least, exceeding 1050°C, even 1100°C, of duration possibly surpassing one-half hour although in the majority of cases it would not last beyond one hour.

There is at least one other case for an "extreme" fire one must consider : that is that of a merchant ship, in particular of the type "roll-on, roll-off". By its very construction and utilisation such a vessel constitutes a tunnel where could be found an assortment of dangerous materials in proximity to each other-one can easily imagine a transport of radioactive materials in the neighborhood of a shipment of flammable liquids. In the extreme case it is definitely possible that the temperature in the neighborhood of these elements will exceed greatly 800°C, at least during the combustion of the flammable liquid. In any case it is certain that e fire will last longer than one-half hour, it will last all the time, in fact, t the ship burns.

It is thus interesting, not only to pose the question of the behaviour of a type B container in conditions exceeding those foreseen by the regulations, but to try to determine a response by proceeding with realistic experimental program.

There exists, in France, a containment of recent design for the transport, for experimental purposes, of fuel materials irradiated in a PWR. This container is however limited by capacity and also, in particular, by the limitation that the heat genaration of the contents not exeed 1000W/m.

A container (of the type B) has subsequently been fabricated, based on the above design, and has successfully withstood the testing procedure prescribed by the Regulations. The fire test was conducted with a one-meter, full-sized model with the corresponding heat generation being simulated by appropriate electrical resistance leating.

It was deciced to put this apparatus in functioning order once again, with the exception, for a number of reasons, of the heat generation device. However, it should be understood that the suppression of this device does not invalidate the experiments performed ; for these effects may be easily taken into account by analytic methods, which have, besides, the advantage of being conservative.

The desired objectives for this series of experiments were to obtain a temperature of 1000°C for a duration of one and one-half hours.

3. TEST DESCRIPTION

The experimental apparatus consisted of :

- a metallic tray, 2m x 4m containing kerosene of the type JP1 (initial fuel depth of 420 mm) ;
- a sheet-metal chimney capping the fuel tray in the form of a truncated pyramid of height = 3 m, resting on four tubular legs (D = 200 mm, L = 800 mm). The internal walls of this chimney were lined with a fire-resistant coating and two layers of insulating material ;
- a watercooled tube penetrating the center of the chimney, inside of which are the thermocouple leads ;
- a chassis of steel tubes in the fuel tray, maintaining the model above the kerosene at the desined height (thermally protected by refractory cotton felt) ;
- an extinction device, designed to stop the combustion of the kerosene desired.

A preliminary experiment was conducted, with a cylindrical block of volume close to that of the model (using six thermocouples) to test the total assembly and in particular the chimney and the fuel feed apparatus. Actual experiments were only performed when the windspeed was below 5 ms^{-1} (to avoid perturbations due to an excessive wind). For these tests twenty-six thermocouples were used :

- 20 implanted in the model (of which 3 were on the external wall) ;
- 4 attached to the exterior of the model, close to the hood ;
- 2 attached to the exterior of the chimney.

Each of the 26 thermocouple assemblies consisted of :

- Chromel-Alumel thermocouple, type thermocoax, diam = 1 mm ;
- a compensatory cable ;
- a cold reference junction ;
- one track on a numerical data acquisition system (sampling rate of 1 track/second, period of 1 minute).

Temperatures were recorded for 4 1/2 hours, thus 3 hours after the extinction of the fire.

- All the thermocouples leads were passed, on exiting the container, through the above described water-cooled tube (passing through one of the protective hoods).
- The signals were recorded on digital tape (so as to be able to trace, on the computer, $\theta = f(t)$).

During the fire the entire assembly was periodically filmed by a mobile camera, and photographs were taken both before and after the test.

4. RESULTS

In these tests, not only temperatures of 1000°C were obtained, but even those exceeding 1100°C. In those conditions the problem arose -and had to be decided upon immediately- as to whether to continue the fire for a full one and one-half hours. In the hope that the model would resist -and, if it did not resist a calculation could recontruct the results- the duration was not modified, and the model did in fact rest intact. In fact the maximum temperature obtained in the lead, which constitutes the essential biological protection, never exceeded 173°C (one hour after

extinction). Of course this value must be corrected for the heat generation of the envisaged contents of the vessel : a calculation showned that, one an one-half hours after extinction the internal temperature of the lead did not exceed 273°C, well below the fusion temperature of 327°C.

5. MODELING

The success of this experiment led us to attempt an analytic approach to this problem. Starting with a previously established stealy-state regime, we have simulated an external fire (by unsteady heat-transfer calculations) and evaluated its influence on the internal temperature gradient in the container.

Besides the lead (a protection against γ-radiation) the package consists of a neutronic barrier based on boron and hydrogen compounds and, especially in our case, a thermal protection of moist plaster. In the course of the fire, the [illegible]ess water in the plaster evaporates and escapes by way of fused plugs designed [illegible] this purpose. The plaster, a bi-hydrated lime sulfate, transforms at first to a semi-hydrated form and then to calcium sulfate. It should be noted that the thermal conductivity of the moist plaster is rather good (which is desirable for the normal, low temperature discharge of the decay heat) while that of the $CaSO_4.1/2H_2O$ is small (desirable for safety in accidental conditions). An initial calculation of the different latent heats of transformation of the plaster led to the definition of an equivalent specific heat.

Finally, a simulation of an actual experiment was effected to validate the different hypotheses made (in particular those concerning the moist plaster). The essential calculations are summarized in the three following figures :

Figure 1 shows the thermal gradient within each of the layers of the vessel, in this case subjected to a fire of 800°C lasting 30 mm ; thus the specified conditions of the Regulations. If initially the package is placed in an environnement of 38°C, we see that the lead, originally at 99°C, passes to 102°C at the moment of extinction, to attain a maximum of 152°C after 2 h 40 mm, safely below critical conditions.

For a second analysis we have simulated a fire of 1000°K (727°C) of duration varying from 10 minutes up to the time for the fusion of the lead. This permits us to define the temperature of the lead and that of the exterior steel for a fire of varying duration (Figure 2).

For example, in a fire lasting 60 minutes the lead will obtain its maximum temperature of 170°C after 2 h 50 mm.

Thus we see that the container can resist a fire of temperature 1000°K for 3 h 50 mm, the lead not attaining its fusion temperature (327°C) until 45 minutes later.

Finally, similar analyses were effected for fire temperatures ranging from 800°C to 1500°C. Figure 3 is a summary of these calculations. For each temperature, we have obtained the maximum duration of the fire which would lead to (after extinction) melting of the lead.

From these results one may determine the behaviour of the container as a function of the temperature and duration of the fire. In particular one may conclude that the present design is capable of resisting a fire of 800°C for 3 h 20 mm, which is a considerable margin with respect to the conditions specified in the regulations.

6. CONCLUSIONS

Although one may not formally conclude from the preceding experiments and [illegible]culations that a Type B container, having resisted for one-half hour a fire of 800°C, would equally well resist, for example, a fire of 1100°C lasting one and one-half hours, certainly the preceding analysis shows that this container has quite good fire resistance. In addition, of course, the specific design of each container is critical, and it is difficult to make general statements. It is thus necessary to continue this type of analysis, particularly for lead-scaled containers, where one must distinguish between the behaviour of the different thermal protections used. In the case of packing for fissile products, the thermal properties and

behaviour of the neutron absorbing material must also be studied. However, the method employed here -full-scale experimentation and, ultimately, analytic calculations- appears to be quite good and it seems profitable to continue in this manner.

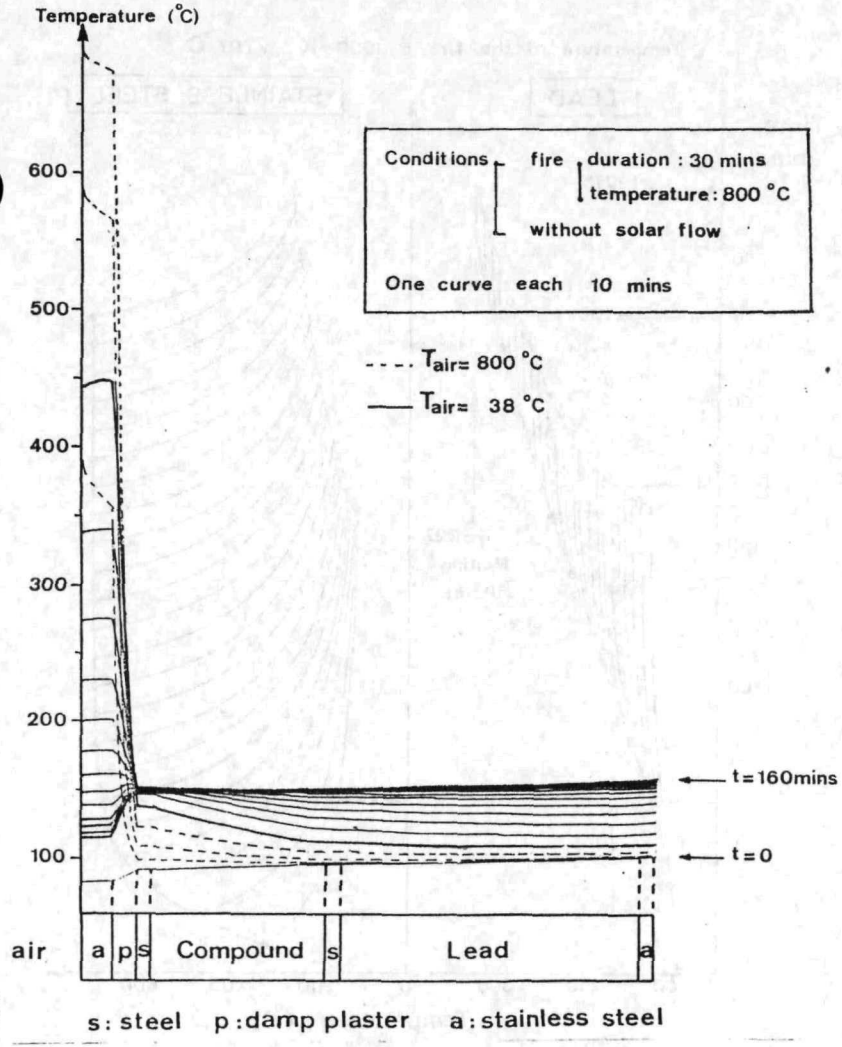


FIG. 1 : EVOLUTION OF THE TEMPERATURE GRADIENT ACROSS THE SHELL DURING AND AFTER A FIRE CONDITION AS SPECIFIED IN I.A.E.A.

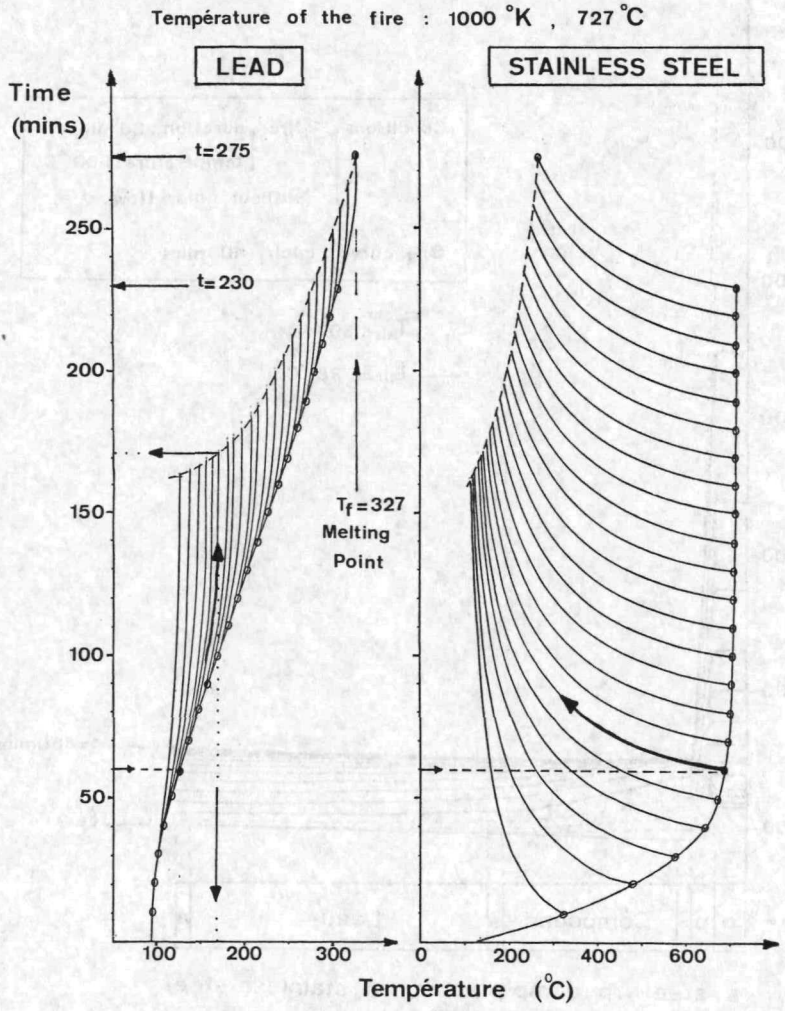


FIG. 2 : INFLUENCE OF THE FIRE DURATION ON THE LEAD AND STAINLESS STEEL TEMPERATURE

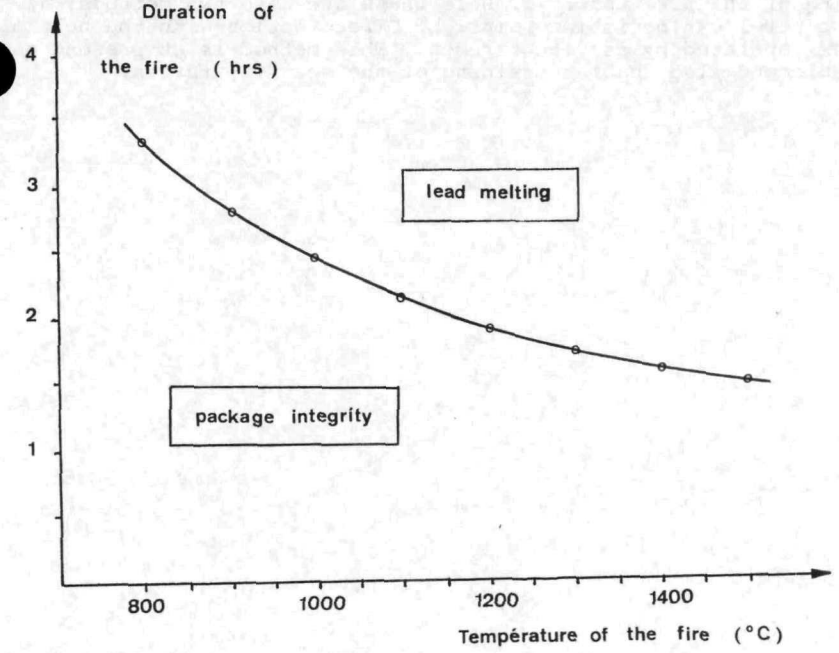


FIG. 3 : INFLUENCE OF TEMPERATURE AND DURATION OF THE FIRE ON BEHAVIOUR OF THE CASK IL46

DISCUSSION

Question by C.P.Haigh, United Kingdom: 1. Was all the fuel used in position at the start of the test or was it pumped in during the fire? 2. What was the method of extinguishing the fire?

Answer: 1. Our fire test equipment has two possibilites. In fact, in the present case, the kerosene was in position from the beginning of the fire test. 2. Here there are also two methods: a) Remote control extinguishing ramps; b) Direct action with the help of lances, operated by service firemen. This method is at present mo reliable and also enables training of the security forces.

MODELLING OF POOL FIRE ENVIRONMENTS USING EXPERIMENTAL RESULTS OF A TWO-HOUR TEST OF A RAILCAR/CASK SYSTEM*

J. E. Hamann and D. E. Klein
University of Texas, Austin, Texas, USA

R. B. Pope and H. R. Yoshimura
Sandia National Laboratories, Albuquerque, NM, USA

INTRODUCTION

During January, 1978, a 67 tonne spent fuel shipping cask in a 68 tonne railcar was exposed to a 122 minute, JP-4 fuel pool fire. This test was intended to simulate an extra severe, but highly improbable, fire-associated accident condition which might be encountered by a shipping cask. The shipping cask, originally constructed in 1962, was a double-walled, lead-shielded cylindrical vessel 3.96 m long and 1.5 m in diameter. The railcar provided a "crash-barrier" surrounding the cask, but was of open steel beam construction providing essentially unlimited air circulation for natural convection cooling.

Prior to the fire test, the cask/railcar system was subjected to a 131 km/h impact into a massive concrete barrier. After the crash test, the cask and railcar were moved to the fire site and placed over a specially constructed fuel pool. The railcar was supported by concrete pedestals located centrally in the pool, with wheel trucks under the impacted end of the car to maintain its post-impact orientation. This positioning subjected the cask to maximum flame temperatures. Details of the tests and preliminary results from the fire test were provided in Reference 1. A final assessment of the response of the cask and resulting structural damage to the long-duration fire is given in a companion paper at PATRAM '80 (Reference 2).

This paper presents the results of analyses performed since the test which used the measured temperature response of the cask/railcar and measured flame temperature and other environmental parameters to assess the thermal input from the fire. Measurements obtained during the test have been used to more clearly define the effects that varying source temperatures, intervening structure, and wind can have on the net thermal input to a large package involved in a fire.

DISCUSSION

The thermal analysis of the cask/railcar system evolved through three distinct stages with each stage more complex and more accurate than the preceding stage.

*Work sponsored by U.S. Department of Energy under Contract DE-AC04-76DP00789.

The first stage consisted of two pre-test predictions of the cask response to two thermal environments. Consistent with the regulations (IAEA Safety Series No. 6), the thermal environments were assumed to be uniform in both space and time. The first prediction was based upon an 800°C radiant source with no convection. The second prediction was based upon a 1000°C radiant source with a convective coefficient ten times that for ambient still air at 1000°C.

The test results showed that the flame temperatures were far from uniform, either in space or time. As a result of this non-uniformity, surface temperatures measured at different locations on the body also varied widely. On the average, the flame temperatures were higher near the top than at the middle or bottom of the cask. Considering only the cask at the axial center, the hottest location on the cask structure was found to be the top, which agrees with the gener[redacted] observed trend in flame temperatures. The flame temperatures were generally hi[redacted] on the leeward side of the cask than on the windward side, early in the te[redacted], whereas the reverse occurred later in the test. However, the cask was hotter on the windward side than on the leeward side throughout the test. The relative magnitude of the differences in the cask surface temperatures is illustrated in Figure 1, which shows surface temperatures one hour after the test began.

The preceding illustrates that the assumptions made in the pre-test calculations were not entirely correct. A comparison between the predicted time response and actual response at three locations on the cask surface is shown in Figure 2. These data further demonstrate the spatial variation of cask temperature response, and reveal that the measured surface temperatures were near or below the predicted response for an 800°C flame temperature during the first portion of the test, but at some locations temperatures exceeded the 800°C prediction in the later portion of the test.

Because the one-dimensional model did not accurately predict the cask behavior, a more detailed post-test analysis was initiated. This analysis was used to assess the impact that factors not considered in the pre-test analysis had on the cask response, and provided a better understanding of the fire environment. Foremost among the factors to be considered was the spatial variation of flame temperatures. The non-uniformity of source temperatures, coupled with its effect on convection heat transfer, required a two-dimensional analysis. Another factor studied was the effect of the railcar structure which partially surrounded the cask. A final factor examined more carefully was the effect of a phase change by the lead on the internal temperature distribution of the cask. Post-test examination had indicated considerable lead melting.

The possible influence of the railcar was first indicated by the nature of the measured and predicted surface temperature profiles. The pre-test analysis predicted a rapid initial temperature increase followed by a relatively slow increase in temperature. This was a reasonable expectation since $(T_{fire})^4 - (T_{cask})^4$ is initially large but decreases dramatically as the cask body increases in temperature. The measured surface temperatures, however, exhibited a profile exactly opposite of that expected. Failure of the analysis to predict the measured temperature variations was tenatatively explained by assuming that the rail[redacted] structure acted as a thermal radiation shield between the cask and the heat so[redacted] until the railcar had heated to a temperature where it became a radiant sourc[redacted].

The second stage of the analysis began as a search for a computer program capable of modeling two or three dimension convective and radiative boundary conditions, and phase change. The heat conduction code HEATING5 written by W. D. Turner, D. C. Elrod, and I. I. Siman-Tou at Oak Ridge National Laboratory satisfied these requirements.

The first two-dimensional modelling was performed by developing a linear representation of the flame temperature at different circumferential locations on the cask. The linearization was performed over the first 90 minutes of the test data. Because of cask/railcar structure movement beyond this time, thermocouple stretching was possible, and the data are suspect. The railcar structure was included as a separate heat sink and radiation source. The top of the car was not located close to the railcar structure, whereas the locations 45° to windward or leeward of the top were directly inside of large steel beam sections of the railcar.

Figure 3 compares, for the top center of the cask, the results of the prediction for the surface temperature with the measured response. The five curves shown are (1) the prediction using a 1000°C uniform fire source and convective input as in the pre-test analysis; (2) the prediction using linearized measured source temperature, not accounting for the railcar structure; (3) the prediction using linearized measured source temperature and including the lower railcar structure; (4) the prediction using the linearized measured source temperature and including both the upper and lower railcar structure; and (5) the actual measured surface temperature variation. For curves (2), (3), and (4), a convective coefficient of 28.4 W/m^2 °C was assumed. As demonstrated in Figure 2, accounting for the actual variation in local flame temperatures yielded good comparison between the measured and predicted values. For this case, where the railcar structure did not locally impede the radiant input, the effect of structure on predicted response is minimal.

Figures 4 and 5 compare the predicted response of the gamma shielding with the measured responses for locations 45° from the top on the windward and leeward sides, respectively. The response of the gamma shield is shown here to indicate that the effects of lead melt were properly accounted for in the modelling. The data shown are for thermocouples located 12.7 cm into the lead region of the cask. The following comments for the lead response also apply to the measured and predicted surface temperature responses at these locations. The predicted response using the linearized measured source temperature, represented by curve (1), did not provide a good a comparison between the predicted and measured values as it did for the top location. Similarly, the predictions did not change significantly by adding just the lower railcar structure to the analysis, as represented by curve (2). However, the cask predicted response after upper railcar structure was added to the analysis, represented by curve (3), provided a significantly better comparison with the measured response, which is shown as curve (4). It is apparent that the presence of the railcar structure locally provided a significant radiant shielding effect.

The preceding demonstrates that accounting for the effects of actual temperature variation and radiant shielding from intervening structure provides significant improvements in the comparison between predicted and measured temperature variations. However, further effort was needed to more clearly understand the importance of the variations in source temperatures and in convective input. Therefore, a more complex two-dimensional modelling effort was undertaken. This third stage model incorporated a piece-wise linearization with time of the source temperatures at each location, and also a circumferentially-varying forced convective-model accounting for wind effects. The complexity of the model is illustrated by Table I.

A plot of the piece-wise linearized source temperatures is shown in Figure 6. The five curves are numbered as shown in Table I.

Table I. Circumferential Variation of Source Temperature, Convective Model and Radiant Shield for Complex 2-D Model.

Circumferential Position*	Source Temperature Model	Convective Model	Radiant Shield
0.0 - 22.5°	1	1	NO
22.5 - 45.0	1	1	YES
45.0 - 67.5	2	1	YES
67.5 - 90.0	2	1	NO
90.0 - 112.5	2	1	NO
112.5 - 135.0	2	2	YES
135.0 - 157.5	3	2	YES
157.5 - 180.0	3	3	NO
180.0 - 202.5	3	4	NO
202.5 - 225.0	3	5	YES
225.0 - 247.5	4	6	YES
247.5 - 270.0	4	6	NO
270.0 - 292.5	4	6	NO
292.5 - 315.0	4	6	YES
315.0 - 337.5	5	6	YES
337.5 - 360.0	5	6	NO

* Circumferential position 0° is at bottom of cask, 90° is at leeward side, and 270° is at windward side.

Each convective model is based upon an application of a forced convective model (Reference 3), and is specified locally based upon the cask location, assumed wind direction, and local source temperature and wall temperature (Reference 4). Thus, the local convective model is cross-coupled to the local temperature model, which, in turn, varies with time. The column in Table I for radiant shield indicates whether or not railcar structure existed locally.

With this complex model, the comparison of predicted response with measured response is much improved over that from the less complex two-dimensional model. Figure 7 shows the comparisons for the top surface and for the gamma shield, 45° from the top on both the windward and leeward sides. These are the same measured data shown in Figures 3, 4 and 5.

CONCLUSION

In conclusion, it has been demonstrated that time and spatial variation the local source temperatures, the radiant shielding of intervening structure the effects of wind can significantly affect the amount of heat input to a large package in a simulated accidental fire. The pool fire provided a significantly non-uniform heat source to the package. Despite these effects, however, the amount of heat input to the package was generally equivalent to that which would be received from a regulatory 800°C uniform thermal source.

REFERENCES

1. Yoshimura, H. R., "Full Scale Simulation of Accidents on Spent-Nuclear-Fuel Shipping Systems," Proceedings of the 5th International Symposium on Packaging and Transportation of Radioactive Materials, May, 1978, pp 463-476.

2. Rack, H. J., and Yoshimura, H. R., "Post Test Evaluation of a Fire Tested Rail Spent Fuel Cask," to be Presented at PATRAM '80, November, 1980.

3. Giedt, W. H., "Investigation of Variation of Point Unit Heat-Transfer Coefficient Around a Cylinder Normal to an Air Stream," Transactions of the ASME, May 1949, pp 375-381.

Hamann, J. E., "Thermal Analysis of a Railcar/Cask in an Engulfing Fire," M.S. Thesis, The University of Texas at Austin (to be published).

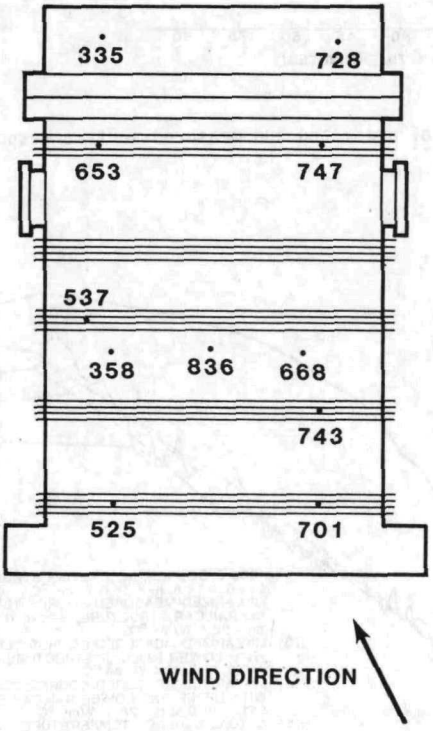


Figure 1. Plan view of cask showing measured surface temperatures in °C 60 minutes into the fire.

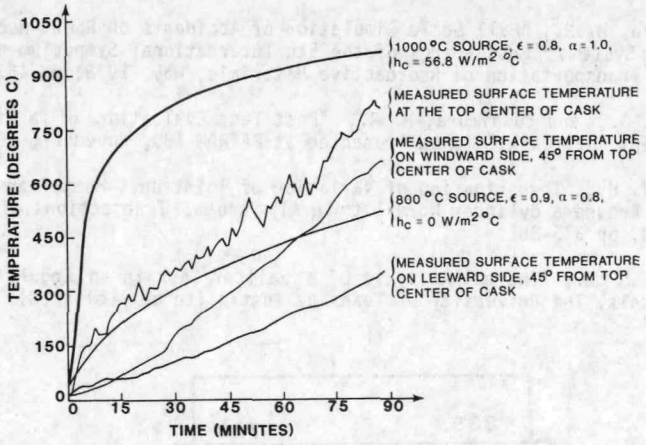


Figure 2. Comparison of predicted and measured surface responses.

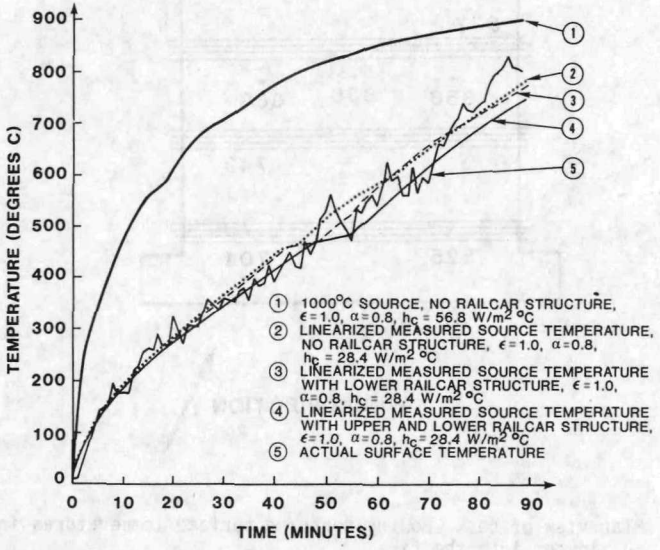


Figure 3. Comparison of predicted and measured surface temperature history at the top center of cask.

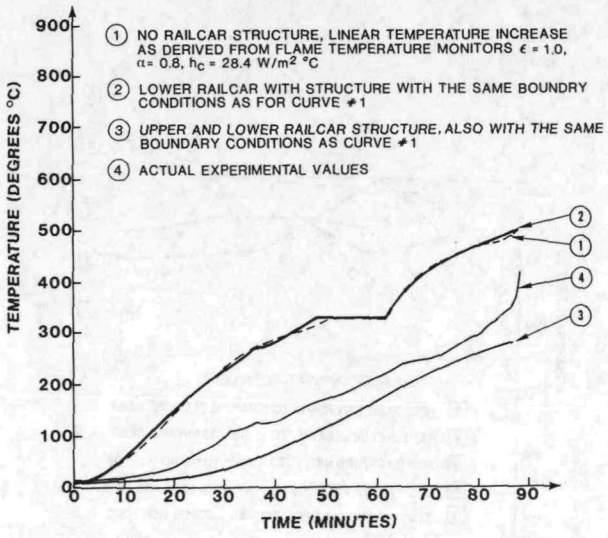


Figure 4. Windward side temperature profiles in gamma shield at a position located 45° from top center 12.7 cm into the lead region of the cask

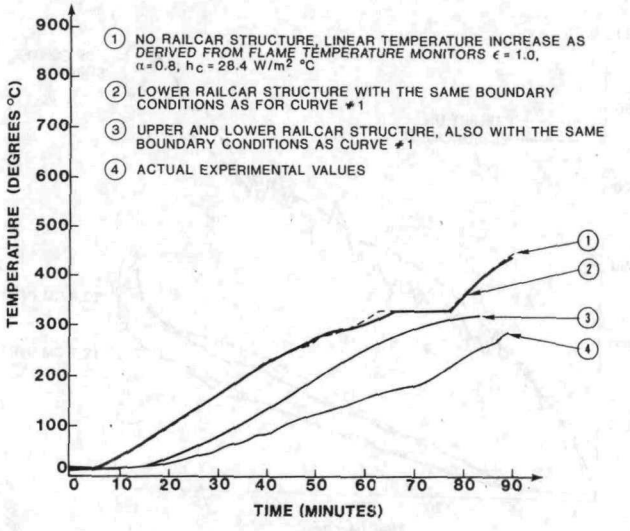


Figure 5. Leeward side temperature profiles in gamma shield at a position located 45° from top center 12.7 cm into the lead region of the cask

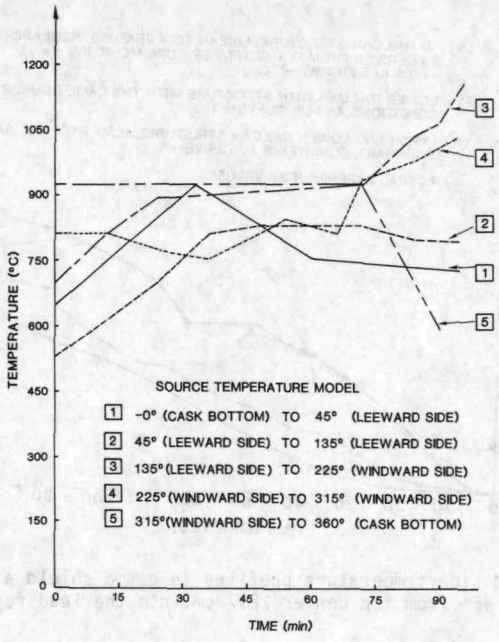


Figure 6. Piecewise linearization of source temperatures

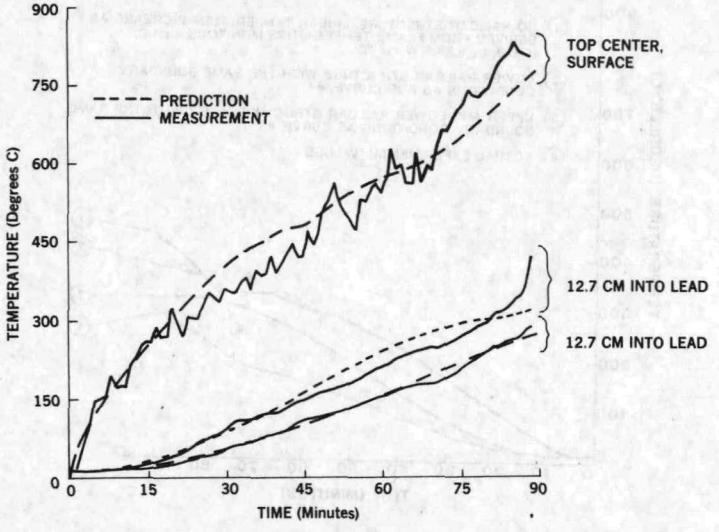


Figure 7. Comparison of predicted and measured temperature response using complex two-dimensional model

Question by G. Malkoske, Canada: Could you explain how the thermocouples were installed 12.7 cm into the lead?

Answer: A hole was drilled through the outer stainless steel shell of the cask into the lead gamma shield to a depth of 12.7 cm. The hole in the stainless steel shell was tapped to allow insertion of a sealing bushing. A stainless steel sheathed thermocouple was inserted into the hole and seated carefully against the lead. The penetration was sealed with a compression type fitting and checked for tightness.

Question by M. Brown, United Kingdom: Did the non-uniformity of temperature have any effect upon the flask seals?

Answer: This is difficult to determine since we did not have a direct method to measure flange distortions or assess seal condition in the fire environment.

Question by S. Williamson, United Kingdom: Could you please give some guide lines with regard to the cost of the fire test and subsequent analysis.

Answer: If I recall correctly, the cost for performing the rail cask fire test and thermal analysis was approximately $ 500,000 (1978 dollars). The actual cost of the fire test and analysis is difficult to determine since many costs such as analytical and technical support were shared between all five cask tests. The subsequent analysis, which is the subject of this paper, was performed by the University of Texas for a cost of approximately $ 20,000.

MODELING THE NORMAL THERMAL ENVIRONMENT

William H. Lake
U.S. Nuclear Regulatory Commission

INTRODUCTION

The normal, hot day, thermal environment is specified in the IAEA Regulations [Ref. 1] by a constant temperature and uniform solar heat flux. This is a simplification of the actual thermal environment in which the ambient temperature and solar flux are cyclic. Real systems attenuate the effect of the periodically varying environment. This paper considers the adequacy of the constant temperature uniform solar heat flux model to represent real thermal environments. Thermal environmental models are presented, attenuation parameters are obtained from two thermal response models, and the limitations and applicability of the models are discussed. Two typical water cooled spent fuel casks are studied under various thermal environments using the models presented. Conclusions are drawn as to the adequacy of the IAEA thermal environmental model to represent real thermal environments.

THERMAL MODELS

Environmental Models

The normal, hot day, thermal environment specified in the IAEA Regulations [Ref. 1] includes a constant 38°C ambient temperature and a total solar heating of 800 gcal/cm^2 for 12 hours for an upward facing, flat, horizontal surface. This solar heating is interpreted as a uniform flux of 66.7 gcal/hr-cm^2 for 12 hours followed by no heating for 12 hours. The simplest analysis assumes a steady 38°C and continuous (24 hour) uniform heating of 66.7 gcal/hr-cm^2.

An upward facing horizontal surface in a real environment is subject to diurnal solar heating and a periodic ambient temperature. The solar model, for the period between sunrise and sunset [Ref. 2] is given by:

$$G_i = G_o \tau_a^m \cos(z) \tag{1}$$

$$\cos(z) = \sin(\lambda) \sin(\delta) + \cos(\lambda) \cos(\delta) \cos(H) \tag{2}$$

$$H_o = \cos^{-1} [\tan(\lambda) \tan(\delta)] \tag{3}$$

The periodic ambient temperature is represented by:

$$T_\infty(t) = T_o + \Delta T_o \cos[(\pi/12)(t - \Delta\tau - 12)] \tag{4}$$

A combined effective ambient temperature model, which includes the temperature increase due to solar heating, is used so that exact solutions to the thermal models may be obtained.

$$T'_{\infty}(t) = T'_{o} + \Delta T'_{o} \cos [(\pi/12)(t - 12)] \quad (5)$$

$$T'_{o} = T_{o} + \Delta T_{s}/2 \quad (6.a)$$

$$\Delta T'_{o} = \Delta T_{o} + \Delta T_{s}/2 \quad (6.b)$$

To estimate the temperature increase due to solar heating an upward facing solar heated flat plate is considered. The heat balance for a flat plate which is heated by the incident solar radiation, and cooled by natural, turbulent convection and radiation heat transfer to the environment is described by the equation:

$$\alpha G_{i} = C_{o} [(T_{\infty} - \Delta T_{s}) - T_{\infty}] + \varepsilon\sigma [(\Theta_{\infty} + \Delta T_{s})^{4} - \Theta_{\infty}^{4}] \quad (7)$$

(η_{o+1})

where G_{i} = incident solar flux, gcal/hr-cm^2

G_{o} = solar constant, 119.5 gcal/hr-cm^2

τ_{a} = atmospheric transmission coefficient

m = relative air mass, approx. sec(z), $0° < z < 80°$

z = angle between the solar and surface unit normal vectors

λ = earth latitude measured from the equator

δ = sun declination angle, $-23\ 1/2° < \delta < +\ 23\ 1/2°$

H = hour angle, traverse angle for sun to body's longitude

H_{o} = hour angles between: sunrise and noon, and noon and sunset

$T_{\infty}(t), T'_{\infty}(t)$ = ambient and effective ambient temperatures, °C

T_{o}, T'_{o} = average and effective average daily temperatures, °C

$\Delta T_{o}, \Delta T'_{o}$ = differences between average and maximum ambient and effective ambient temperatures, °C

Θ_{∞} = absolute ambient temperature, °K

t = time of day measured from midnight, hrs

$\Delta\tau$ = lag of peak ambient temperature from solar peak at noon, hrs

ΔT_{s} = difference between the maximum body surface temperature with and without the peak solar flux, °C

α, ε = radiative surface properties, absorption and emissivity coefficients

C_{o} = convection constant = 0.13 gcal/hr-cm^2-°C [Ref. 3]

η_{o} = convection constant [Ref. 3]

σ = Stefan-Boltzmann constant, 4.88×10^{-9} gcal/hr-cm^2-°K^4

The maximum incident solar flux occurs when the angle between the solar and surface unit normal vectors are minimum (i.e., when the angle, z, is minimum).

This occurs at noon, at the summer solstice outside the tropics. In the tropics it occurs at noon when the sun declination angle is equal to the local latitude (i.e., $\delta = \lambda$). Table 1 is obtained by solving equation (1), and presents the maximum incident solar flux for various latitudes and atmospheric transmission coefficients.

TABLE 1. MAXIMUM SOLAR FLUX VALUES AS A FUNCTION OF LATITUDE AND ATMOSPHERIC TRANSMISSION COEFFICIENT (τ_a)

	Maximum Solar Flux, $G_i$max (gcal/hr-cm^2)			
Latitude(λ)	τ_a=0.62	τ_a=0.70	τ_a=0.81	τ_a=1.0
0-23-1/2*	74	84	97	120
30°	74	83	96	119
40°	70	79	92	115
50°	63	72	85	107
60°	53	62	74	96
70°	41	49	61	82
80°	28	35	45	66
90°	14	20	28	48

*Maximum flux occurs when $\lambda = \delta$ (sun's declination angle) for $0 < \lambda < 23\text{-}1/2°$, and when $\delta = 23\text{-}1/2°$ for all other λ.

The maximum solar flux values are used to estimate the maximum temperature increase due to solar heating of an upward facing flat plate. These values, presented in Table 2, are calculated using equation (7), assuming the radiation surface properties are equal to 0.8 ($\alpha = \varepsilon = 0.8$).

TABLE 2. MAXIMUM TEMPERATURE INCREASE OF AN UPWARD FACING, SOLAR HEATED FLAT PLATE (WITH SURFACE RADIATION PROPERTIES, $\alpha = \varepsilon = 0.8$)

	Maximum Temperature Increase (°C)			
Latitude(λ)	τ_a=0.62	τ_a=0.70	τ_a=0.81	τ_a=1.0
0-23-1/2*	54	60	67	78
30°	54	59	66	77
40°	52	57	64	75
50°	47	53	60	71
60°	42	47	54	66
70°	34	39	46	59
80°	24	29	37	49
90°	14	18	25	38

*Maximum flux occurs when $\lambda = \delta$ (sun's declination angle) for $0 < \lambda < 23\text{-}1/2°$, and when $\delta = 23\text{-}1/2°$ for all other λ.

Thermal Response Models

Two models are considered, one representing a system with negligible thermal gradient, the other a massive system with slow thermal response and a significant thermal gradient.

First, the model with negligible thermal gradient is considered [Ref. 2]. The thermal gradient across a slab of thickness, x(cm), conductivity, k(gcal/hr-cm-°C), and surface heat transfer coefficient, h(gcal/hr-cm^2-°C), is considered negligible if

$$h\ x\ /\ k < 0.1 \tag{8}$$

The heat balance is given by:

$$c\ \rho\ V\ dT = hA_s\ (T_\infty - T)\ dt \tag{9}$$

where

c = specific heat, gcal/g-°C

ρ = density, g/cm^3

V = volume, cm^3

A_s = surface area, cm^2

T = system average temperature, °C

The steady periodic solution obtained for equation (9) with the periodic ambient temperature in equation (5) is:

$$T(t) = T'_o + \left(\frac{\Delta T'_o}{1+(\pi RC/12)^2}\right)\left(\frac{\pi RC}{12}\right)\{\sin[(\pi/12)(t-12)] + \cos[(\pi/12)(t-12)]\} \tag{10}$$

where

R = the surface thermal resistance, $1/hA_s$

C = the thermal capacity, $c\rho V$

The solution to equation (10) for a system with characteristic time, RC (hrs), over time, t, is its response history. The maximum thermal response variation, ΔT_{max}, due to the environmental variation, $\Delta T'_o$ is:

$$\Delta T_{max} = \begin{cases} \dfrac{\Delta T'_o}{1+(\pi\ RC/12)^2}, & RC \le 12/\pi \quad (11.a) \\ \dfrac{\Delta T'_o(\pi RC/12)}{1+(\pi RC/12)^2}, & RC > 12/\pi \quad (11.b) \end{cases}$$

Attenuation parameters, A_n, for the negligible thermal gradient model are obtained from equations (11.a) and (11.b).

$$A_n = \begin{cases} [1 + (\pi RC/12)^2]^{-1}, & RC \le 12/\pi \quad (12.a) \\ (\pi RC/12)\ [1 + (\pi RC/12)^2]^{-1}, & RC > 12/\pi \quad (12.b) \end{cases}$$

The second model represents the system with slow thermal response and relatively large internal thermal resistance. The system may be described by the characteristic time constant, τ (hrs), and internal thermal resistance, R_i (hr-°C/gcal). These parameters may be calculated or obtained from tests. For a uniform flat plate they are:

$$\tau = x^2/\hat{\alpha} \tag{13}$$

$$R_i = x/(kA) \tag{14}$$

where

x = distance into the plate, cm

$\hat{\alpha}$ = thermal diffusivity ρ, cm^2/hr

k = thermal conductivity, gcal/hr-cm-°C

A = area normal to heat flow direction, cm^2

Systems of this type often have a significantly lower surface resistance, than internal thermal resistance. When the value $R/R_i = k/hx$ (the inverse of equation (8)) is low, the surface temperature is approximately equal to the ambient temperature. The steady periodic solution for the semi-infinite slab with a surface boundary temperature of equation (5) [Ref. 3] is:

$$T(t) = T_o' + \Delta T_o' \exp(-\sqrt{\pi\tau/24}) \cos [(\pi/12)(t-12)-\sqrt{\pi\tau/24}] \tag{15}$$

The maximum variation, ΔT_{max}, of the system represented by τ, and the attenuation parameter, A_τ, are:

$$\Delta T_{max} = \Delta T_o' \exp(-\sqrt{\pi\tau/24}) \tag{16}$$

$$A_\tau = \exp(-\sqrt{\pi\tau/24}) \tag{17}$$

The characteristic time constant is expressed in terms of a specified attenuation parameter from equation (17). Table 3 presents time constants for several specified attenuation values.

$$\tau = [\log_e (1/A_\tau)]^2/(\pi/24) \tag{18}$$

TABLE 3. CHARACTERISTIC TIME CONSTANTS FOR SPECIFIED VALUES OF ATTENUATION

Attenuation (%)	1%	2%	5%	10%	15%	20%
Characteristic time constant, τ (hr)	162	117	69	41	28	20

If a desired attenuation, A_τ, is specified the required characteristic flat plate time constant, τ, can be estimated from equation (7). Also the validity of using the semi-infinite slab model for a flat plate of finite thickness, x, can be assessed.

The attenuation parameters in equations (12.a) and (12.b), and (17) represent two different thermal response models. Table 4 presents attenuation values for both models for several values of characteristic time (RC, τ).

TABLE 4. ATTENUATION PARAMETERS FOR SEVERAL CHARACTERISTIC TIMES

Characteristic time (hr)	Attenutation Parameters A_n (for neglibible thermal gradient model)	A_τ (for infinite slab model)
1.00	0.94	0.70
2.00	0.78	0.60
3.00	0.62	0.53
3.82(π/12)	0.50	0.49
5.00	0.48	0.45
7.50	0.40	0.37
10.00	0.33	0.32
15.00	0.24	0.25
20.00	0.18	0.20
50.00	0.08	0.08
100.00	0.04	0.03

APPLICATION TO SPENT FUEL CASKS

The thermal environmental models and thermal response models developed are used to study the response of large water cooled spent fuel casks to diurnal variations of solar heating and ambient temperatures. The thermal response models are both one-dimensional models. The neglible thermal gradient model is a flat plate model, and the infinite slab approximates a flat plate of finite thickness for periodically varying surface temperatures when the characteristic time constant is large. Spent fuel casks are typically long hollow cylinders which are modeled as two-dimensional circular sections (i.e., infinite length is assumed). The use of flat plate models introduces errors due to geometry and dimension. For the curved surface the area normal to radial flow of heat decreases as the cylinder is entered (i.e., flow towards centerline). This decreases the thermal capacity, but increases the thermal resistance; however, the effects do not compensate exactly. The two-dimensional effects will result in lower attenuation than predicted by the flat plate model, because of additional heat penetrating a central location. The spent fuel casks considered have large characteristic time constants, making these effects negligible.

Spent Fuel Cask Models

Two typical water cooled spent fuel casks are considered. One represents a truck cask, the other a rail cask. The pertinent specifications are given below:

1. The truck cask is 22,700 kg, consisting of 17% steel, 70% lead, and 13% water. It has a measured unit conductance of 67 kcal/hr-°C. A thermal capacity and characteristic time constant of 3890 kcal/°C and 58 hours are calculated.

   The rail cask is 63,500 kg, consisting of 18% steel, 73% depleted uranium, and 9% water. It has a measured unit conductance of 326 kcal/hr-°C. A thermal capacity and characteristic time constant of 8522 kcal/°C and 26 hours are calculated.

Both casks are treated as upward facing flat plates having equal surface radiation properties ($\alpha = \varepsilon = 0.8$), and the characteristic time constants calculated above.

Thermal Environments

Three thermal environments are considered: 1) the uniform IAEA model, 2) a moderately hot environment, and 3) an extremely hot (i.e., desert) environment. The environmental models are specified below.

1. A steady ambient temperature of 38°C, and uniform solar heating of 66.7 gcal/hr-cm^2.

2. An ambient temperature of 27°C ± 11°C, and a maximum solar flux of 72 gcal/hr-cm^2 (50°-latitude, 0.7 atmospheric transmission - Table 1).

3. An ambient temperature of 35°C ± 11°C, and a maximum solar flux of 83 gcal/hr-cm^2 (30°-latitude, 0.7 atmospheric transmission coefficient - Table 1).

Peak Internal Temperatures

Using the equations developed and the results presented, the peak internal (i.e., bulk coolant) temperatures are estimated for the two cask models subjected to the three thermal environments specified. The results are presented in Table 5.

TABLE 5. PEAK INTERNAL CASK TEMPERATURES

	Truck Cask	Rail Cask
Time Constant, τ (hrs)	58	26
Attenuation Parameter, $A\tau$	0.064	0.158
Peak Internal Cask Temperatures:		
IAEA Thermal Environment	57.5°C	57.5°C
Moderately Hot Environment	55.9°C	59.4°C
Extremely Hot (desert) Environment	67.1°C	70.9°C

CONCLUSIONS

A method is presented and shown to be applicable for estimating the thermal responses of the internal regions of large water cooled spent fuel casks to periodic variations in ambient thermal environments. These effects are significant for water cooled casks with internal heat (where much higher internal temperatures result) since internal pressures are very sensitive to temperature.

The results presented in Table 5 show that there is little difference between the IAEA and moderately hot environments for the two cask models. The differences between the IAEA and extremely hot environments range between 9.6°C and 13.4°C depending on the cask time constant. Although distinguishable, these differences still do not appear to be too significant. Based on the results considered in this study, it can be concluded that the use of the IAEA uniform thermal environment model is adequate for estimating the thermal performance of large water cooled casks in hot thermal environments.

REFERENCES

1. International Atomic Energy Agency, Safety Series No. 6, Vienna, 1973.

2. Frank Kreith, Principles of Heat Transfer, Second Edition, International Textbook Co., Scranton, PA, 1966.

3. William H. McAdams, Heat Transmission, Third Edition, McGraw Hill Book Co., NY, 1954.

4. P. J. Schneider, Conduction Heat Transfer, Addison-Wesley Publishing Co., Inc., Reading, MA, 1955.

TRANSPORT FIRE HAZARD AND THE CALCULATION OF PACKAGE RESPONSE

R. L. D. YOUNG

UNITED KINGDOM ATOMIC ENERGY AUTHORITY

Safety and Reliability Directorate

Wigshaw Lane

Culcheth

Warrington WA3 4NE

United Kingdom

SUMMARY

The fire hazard to small ($\lesssim$ 500 kg) Type B packages is assessed with particular reference to aircraft crash fires. Container response including the effect of a charring insulation is studied theoretically. It is concluded that an actual fire test rather than a "design" fire or a furnace test is the most satisfactory qualification method.

1. INTRODUCTION

Transport of plutonium by air provides increased speed, safety, security and reliability over other shipping methods. However, air transport accidents pose the most serious threat to RAM packages due to the high speeds and fuel loads typical of modern jet aircraft. The accident environment with particular reference to impact is considered in a companion paper by Brown [1]. Here the fire hazard is assessed; its likelihood, intensity and duration and the response of containers engulfed in a fire are studied.

The work was performed under contract to the Commission of the European Communities as part of their study on the environmental impact of LWRs. Similar work has been performed in the USA [2] but space does not allow a review of this work.

2. THE FIRE HAZARD

We examine here aircraft crash fire statistics and the fire environment.

2.1 Aircraft Crash Fire Statistics

Two recent detailed analyses of aircraft accidents are those by Clarke et a [2], [3] for US military cargo aircraft and Lucha et al [4] for US civil air carrier accidents involving fires between 1963-74.

Table 1: Analysis of the Occurrence and Severity of Aircraft Crash Fires (Percentages)

	Civil [4] overall	Civil [4] cargo	Military [2], [3] cargo
Frequency:*			
Fire	29	37	35
Fire & Impact	19	33	22
Operation mode:			
Ground	5	3	15
Take-off	23	32	21
Flight	24	20	30
Landing	48	45	34
Fire Occurrence:			
Ground	6	0	32**
In-flight	7	3	6
Post-impact	87	97	62
Damage Total* (Fire)			
Destroyed	21(16)	35(20)	not
Severe	63(6)	58(5)	available
Moderate	16(7)	7(3)	

* In all accidents
** This figure includes take-off and landing accidents

From Table 1 it is clear that fire is a significant hazard in any aircraft accident, especially after impact with the ground. The relatively large proportion of accidents in which a cargo aircraft is destroyed shows that RAM packages may well face a doubly hostile environment - first impact then an engulfing fire. The frequency of such an occurrence is assessed at 1.1×10^{-6} per flight from [1] for Western Europe.

2.2 The Fire Environment

In an aircraft crash fire, ruptured fuel tanks or lines allow a fuel mist to wet the aircraft's surface while it is still moving [5] and sufficient heat sources exist to ignite it resulting in the aircraft's envelopment by fire. The aluminium skin can melt in 1 minute [6], enabling RAM packages to be rapidly exposed to the full intensity of the fire.

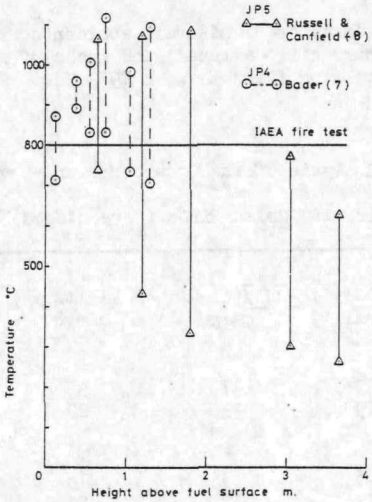


Figure 1: Centre-edge vertical temperature profiles in aviation fuel fires

Within the fire, conditions can be severe. Figure 1 shows typical centre-edge temperatures in JP4 [7] and JP5 [8] fires at different vertical locations. In [7] it was concluded that a black body flame temperature of 1010°C was representative; this agrees with [8] where it was estimated that convective heat fluxes to an object engulfed in a fire could be as high as 60% of the total heat flux.

2.3 Aircraft Crash Fire Duration

Little quantitative data exists for aircraft crash fire duration. By use of statistics of aircraft service years (eg [4]) and fuel loads carried it is possible to arrive at a "mean aircraft". When this crashes according to a low, medium or high impact angle, the resultant fuel pool is assumed to take an area equivalent to wingspan x overall length, ground plan area or half-wing area respectively. Using a surface regression rate of 5.5×10^{-5} m/s [4] for kerosene fuels burn times of 9.4, 44.4 and 155 minutes respectively are obtained; and with estimates of the probability of such impact angles derived from [2] and assumption that take-off accidents involve a full fuel pay load, in-flight accidents $\frac{1}{2}$ payload and landing accidents $\frac{1}{4}$ payload, we can construct a cumulative distribution for fire duration using mode of occurrence data from Table 1. The results are presented below in Figure 2 where most pessimistic spot values for BAC-111 and B747 are shown to illustrate the range by type.

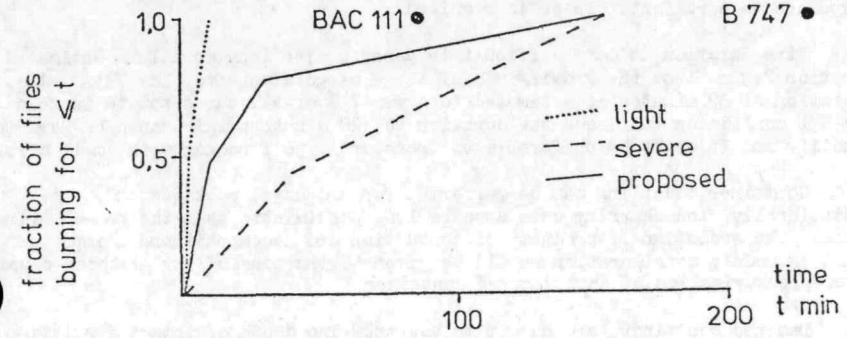


Figure 2: Cumulative Probabilities for Fire Duration

3. CONTAINER RESPONSE

A theoretical assessment has been made of the thermal response of small Type B packages (SL-10 and PAT-1 type) to aircraft crash fires. The transient heat conduction equation for a multi-material medium with internal heat generation was solved, and an attempt made to model the charring of the insulation/shock absorber (wood & water-extended polyester (WEP) in SL-10, wood in PAT-1). Because detailed experimental data on combustion of these materials in vitiated atmospheres is lacking, a simple model based on Carslaw & Jaeger's [9] recommendation for phase-change studies was used; a unique charring temperature was assumed and the effects of evolution of gases, steam etc during pyrolysis were ignored. Assumed fire conditions were black body radiation at 1100°C for 1 h followed by a 1 h cooling period at ambient temperatures. The results presented in Figures 3 & 4 for the SL-10 and PAT-1 show the development of the temperature profile through the central section and demonstrate the insulating effect of a charred medium. The position of the char front after 1 h is also shown. The experimental position shown in Figure 4 is taken from ref [10]; agreement is reasonably satisfactory. The importance of evolved gases and liquids may be assessed for WEP from [11]. WEP heated on a face to 800°C for 30 mins was destroyed to a depth of 6 mm with further detectable damage for 8 mm with evolution of gases and liquids, whereas in Figure 3 a 90 mm char depth is predicted. Thus a simple charring model appears to overpredict the hazard when evolved gases etc are important as in WEP and underpredict for the case of wood. Work is continuing on this study.

4. DISCUSSION AND CONCLUSIONS

The aircraft fire statistics presented in section 2.1 show that a post-impact fire in which the aircraft is destroyed is a non-negligible event. The thermal regime encountered by a RAM package is very severe - temperatures of 1100°C and radiant heat fluxes of 162 KW/m^2 could be encountered. Large RAM packages - eg spent fuel flasks may well not be exposed to such conditions because they tend to be transported in a way in which total engulfment by a fire would be unlikely. However, qualification of small packages which could be so engulfed should allow for such severe thermal regimes: current IAEA "design fires" of 800°C [12] would appear to be inadequate, and in view of the strong convection flux noted in [8], an actual fire test rather than

exposure to a radiant furnace is required.

Fire duration is more difficult to assess. The long durations estimated in section 2.3 reflect the growing use of large cargo aircraft. The IAEA [12] fire duration of 30 minutes is estimated to cover 75% of all cases but to increase this to 95% confidence increases the duration to 120 minutes; this could be hard to justify and it might be preferable to transport type B packages in small aircraft.

Container modelling can be performed for undamaged packages and, computationally, the charring case appears less pessimistic than the non-charring case. The evolution of products of combustion and decomposition is important but hard to model; consideration should be given to the possibility of these causing over-pressurization of an undamaged container.

Damaged containers are harder to assess. The depth of intact insulation surrounding the RAM cannot be known a priori and a punctured outer skin permits the creation of flammable gases from the insulation and the char itself to burn. Such effects cannot be predicted and fire testing damaged and undamaged containers appears to be the most satisfactory method of qualifcation.

5. ACKNOWLEDGEMENTS

Acknowledgements are due to the commission of the European Community for funding this work and to Mrs B Rogowski of the Fire Research Station, Borehamwood, Herts UK for comments on behaviour of packages in fires.

6. REFERENCES

1. BROWN M. L., "Analysis of the accident environment and container response for the air transport of small type B packages" this Conference.

2. CLARKE R. K., FOLEY J. T., HARTMAN W. F., LARSON D. W., LARSON D. W., "Seventies of Transportation Accidents Vol II - Cargo Aircraft" SLA-74-001 (1975)

3. "The parameters of A/c accidents: a listing of the individual accident data used in the aircraft portion of the transportations accident criterion study" Sandia Laboratories 1973

4. LUCHA C. V., ROBERTSON M. A. & SCHOOLEY F. A., "An analysis of aircraft accidents involving fires" NASA CR 137690 (1975)

5. PINKEL I. I., PRESTON M. G. & PESMAN G. J., "Mechanism of start and development of aircraft crash fires" NASA Report 1133 (1975)

6. GEYER G. B., "Effect of ground crash fire on aircraft fuselage integrity" Interim report NA-69-367 (RD-69-46) (1969)

7. BADER B. E., "Heat Transfer in Liquid Hydrocarbon Fuel Fires" Nuclear Engineering 61 p 78-90 (1965)

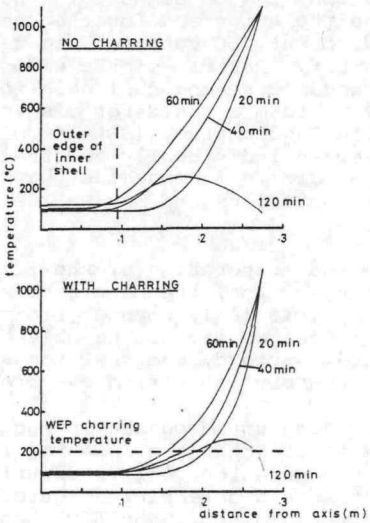
8. RUSSELL L. H. & CANFIELD J. A., "Experimental Measurement of heat-transfer to a cylinder immersed in a large aviation-fuel fire" J. Heat Transfer 95 p397-404 (1973)

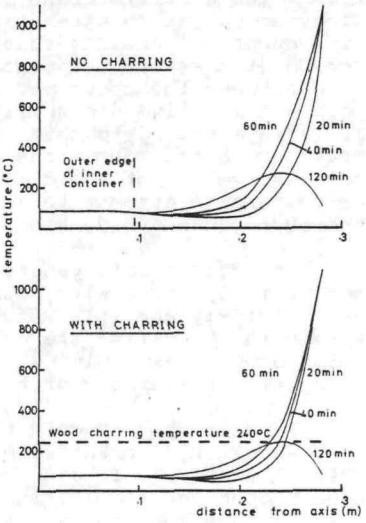
9. CARSLAW H. S. & JAEGER J. C., "Conduction of Heat in Solids" Oxford University Press (1959)

10. ANDERSEN J. A., DUFFEY T. A., DUPREE S. A., NILSON R. H., "PARC (Plutonium Accident resistant container) Program Research, Design & Development" NUREG/CR-0030 (1978)

11. McMILLAN T. S., "Thermal Evaluation of Water-extended polyester" Report DP-1262 AEC (1972)

12. IAEA Regulations for the safe transport of radio-active materials (1973) Safety Series No 6 IAEA Vienna 1979





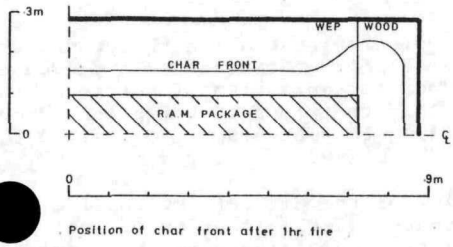


Figure 3: SL-10 type container - temperature profiles in central section and position of char front

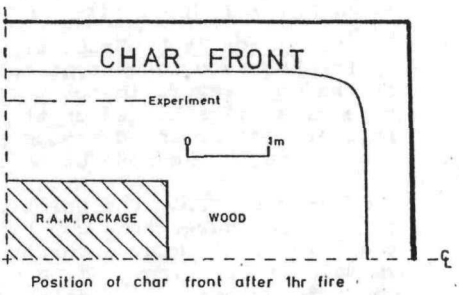


Figure 4: PAT-1 type container - temperature profiles in central section and position of char front

Comment by S. Williamson, United Kingdom: You make assumptions, in the aircraft accident, that there is no loss of fuel into the ground and no evaporation of the fuel. Both these, and particularly the latter, are hardly likely in view of the temperature you assume.

The readers of your paper should therefore have reservations with regard to your suggestion that an aircraft accident fire is more severe than the IAEA regulatory thermal test. If statements like this are to be made at PATRAM '80 they must be fully substantiated and your assumptions are not reasonable.

Answer: It must be stressed that there is no real data base from which aircraft crash fire durations can be assessed. The method I presented was an attempt to define the upper and lower bounds for fire duration, and, in accordance with our CEC contract, to take into account the severe accident environment. Because of the wide difference between the upper and lower bounds it seemed desirable to propose a cumulative probability curve to form a basis for discussion. It is interesting to note that in the Sandia study (SLA-74-001) aircraft crash fire durations were assessed individually and their proposed curve is not too different from mine. I emphasise that such curves are an attempt to define fire durations and may well err on the over-conservative side.

As regards soil penetration by and evaporation of the fuel, I do not entirely agree with your comments. First, the nature of soils varies widely and in a safety case, particularly when we are still attempting to define the problems, I feel we may not be able to claim soil porosity as a mitigating feature. Second, the fuel tends to burn from the surface of the liquid thereby inhibiting evaporation.

Finally, you suggest that I am being unreasonable in suggesting that an aircraft crash fire is more severe than the IAEA regulatory thermal test. As I have indicated the quesiton of fire duration is a very open one and I think 30 minutes is a defensible figure. However, I have personally strong reservations about the 800°C condition in view of the significantly higher temperaturesthat have been attained in open pool fire tests.

Question by R. Brissier, France: You have shown the distribution of the kerosene concentration after different impacts. In the case of the most severe impact the maximum concentration of kerosene, therefore fire, is situated on the level of the wings of the plane. Would it therefore be an advantage according to your study, to position containers in the nose or tail of the plane?

Answer: From the point of view of the aircraft crash fire is hard to answer this question because of the location of fuel fuel lines. In many aircraft fuel tanks are placed in the fuselage as well as the wings. In a crash a container placed near the nose could be pushed rearwards towards the tanks, while if placed in the rear could shunt forwards, again towards the tanks, due to inertia. Also, for rear-engined aircraft fuel lines run from the wings and central fuselage to the engines and if ruptured could feed a fire anywhere in their run. Thus without constraints designed to restrict container motion I do not think a simple absolute answer can be given.

VALIDITY OF SCALING SHIPPING CASKS EXPERIENCING LARGE DEFORMATIONS

R. J. Burian
BATTELLE'S COLUMBUS LABORATORIES
Columbus, Ohio U.S.A.

INTRODUCTION

An increasing awareness of the public to the potential hazards of radioactive materials has quite properly caused the nuclear transportation community to critically examine the designs and design methods used in developing transportation packages. Testing of packages to hypothetical accident conditions is one commonly accepted way of demonstrating the integrity of the package and its ability to contain the radioactive contents during a severe loading incident. Where large packages weighing several tons are involved, it has proven economically advantageous to test models of the package and extrapolate the results to the full-size package. It is necessary, however, to verify that such extrapolation accurately will predict the response of the full-size package to a similar load. Moreover, relatively few studies have included tests at extra regulatory conditions.

Battelle's Columbus Laboratories recently completed a program sponsored by the United States Department of Energy[1] to assess the validity of employing scaling techniques to determine the response of lead and uranium shielded shipping containers to extreme impact conditions. For this program we developed two simplified package designs for the truck shipment of light water reactor fuel elements. One design used lead shielding and the other used depleted uranium (DU) shielding. Since our objective was to study scaling validity we did not feel constrained to exactly reproduce an actual truck cask in current use. Thus, the models were not replicas of any particular container but rather they incorporated the basic structural features common to many containers in current use including non-LWR packages. Only the primary structural components were modeled as shown in Figure 1 which is a section of the DU shielded model. These included the outer shell, heads, cover, shielding and inner cavity shell. Other features such as a neutron shield, trunnions, and valves were not included in the models. The lead shielded models are similar differing only in the shielding thickness and, therefore, the outer diameter.

The noteworthy feature of the DU models shown in Figure 1 is the way in which the uranium shield was made. The side shield consisted of eight segments machined to interlock with adjacent axial segments. The bottom uranium shield fit in a counterbore in the lower most side shield segment.

CASK MODELING

Seventeen modeling parameters were identified as having possible significance. They included nine geometric dimensions, four material properties and four impact parameters. These were combined into fourteen dimensionless terms

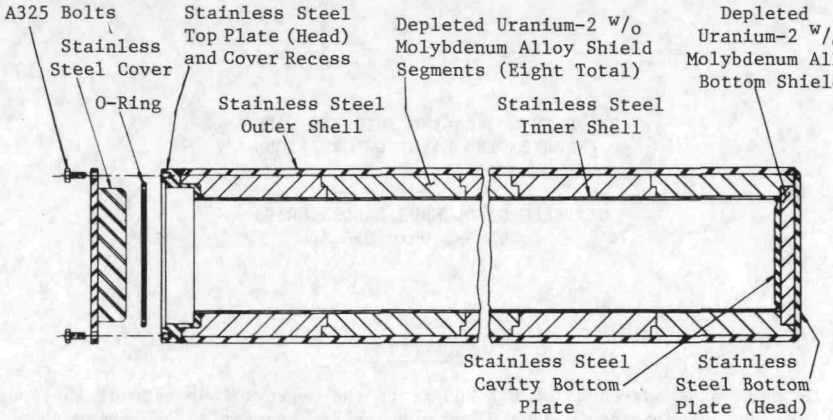


FIGURE 1. CONFIGURATION OF DEPLETED URANIUM SHIELDED MODEL CASKS

using the Buckingham Pi Theorom. From these fourteen dimensionless terms we arrived at the following relations. If we scale the model dimensions linearly, then

- The same materials should be used in the models as in the prototype
- The strains, both elastic and plastic, occurring during impact are identical in the model and in the prototype
- When the same materials are used in the model as in the prototype, the impact velocities and hence the drop height should be the same
- For equal impact velocities, the response time for the impact event varies directly as the linear scale of the dimensions
- For equal impact velocities, the deceleration (impact g-load) varies inversely as the linear scale of the dimensions
- For equal impact velocities the angular velocity varies inversely as the linear scale of the dimension.

TEST PROGRAM

A total of eighteen models were fabricated and tested in three scale sizes, 1/8, 1/4, and 1/2 scale. Thirteen of the models were lead shielded and five were uranium shielded. To achieve our objective of studying the validity of scaling relationships for cask models undergoing large deformations we planned a testing program in which the models were dropped at heights from 4.5 meters to 42.7 meters (15 ft. to 140 ft.). In order to produce the maximum deformation and damage possible, the casks were unprotected, in other words, no impact limiters were used.

The cask models were dropped in three orientations, axial, corner center gravity, and 30-degree with the horizontal (slap down). The axial and corner center of gravity orientations were chosen to produce the maximum local damage. The slap down orientation was selected to examine the effect of secondary impact. The 30-degree angle was chosen based on calculations performed at Los Alamos Scientific Laboratory(2) which indicated that for these models, the 30-degree angle would produce the greatest impact velocity at the second end, approximately twice the initial impact velocity.

PRETEST EXAMINATIONS

The casks were made as nearly as possible according to the fabrication procedures and using the same quality assurance procedures as would be used on a full-size cask. Prior to testing, the casks were subjected to a series of pretest examinations.

It was expected that the planned tests would produce high stresses in the weld areas between the heads and the outer shell both at the top and bottom end. Therefore, these welds were subjected to dye penetrant and ultrasonic nondestructive testing beyond that performed as part of the fabrication quality control procedures. The primary objective of the examinations was to establish a reference [illegible] a base for comparison with post-test examination and to aid in establishing a [illegible]se and effect for any unusual or unexplained weld performance and not to serve [illegible] a higher level fabrication quality-control tool.

The DU-shielded casks were radiographed using a 70-curie Co-60 source to establish reference condition and mating fit of all DU segments in each of the four DU-shielded model casks. Varying degrees of porosity were observed in many segments of three models. Since one of the objectives was to evaluate cask models fabricated by typical methods, the porous segments were not replaced.

Pretest measurements of the cask models included length, circumference, and straightness. To aid visual evaluation of outer shell distortion, the cask models were painted bright yellow with a grid pattern approximately 32 mm square painted about 300 mm in from each end.

RESULTS

Post-test examinations included a study of the high-speed motion picture films to determine transient response characteristics during the impact event; non-destructive examinations of the cask models to study external cask deformations and seal and weld integrity; and destructive examinations to study internal deformations, metallurgical characteristics and mechanical properties of the structural and shielding materials.

Impact Response

The six (five 1/4 scale and one 1/8 scale) casks that were impacted in an axial or a corner center of gravity orientation rebounded a short height before landing a second time on the same end. The energy absorbed/dissipated during initial impact was determined to be about 99 percent of the kinetic energy. All twelve (two 1/2-scale, six 1/4-scale, and four 1/8-scale) cask models impacting in a 30-degree slap-down mode experienced a slight bowing upon initial impact. The models then entered into a rotational as well as translational mode which caused the end opposite the impact to accelerate downward and the initial impact [illegible] to lift free of the target. The second end struck the pad with the initial [illegible] elevated about 5 to 10 degrees. At second impact a more severe bow occurred; [illegible]ith the maximum about 1/3 of the length from the second impact end. Subsequent impacts at alternate ends occurred as many as eleven times before the models went into a rocking mode. The angular velocities after the initial impact and the fraction of energy absorbed during initial impact show good agreement with scaling relationships.

Impact of lead-shielded models directly on the end or at the corner C.G. angle resulted not only in deformations in the region adjacent to the impact point but also throughout the length of the cask. At the impact point the outer shell bulged outward. The inner shells of models dropped from 21.3 and 42.7 meters were bulged noticeably inward near the impact end (Figure 2). The diameter of the inner shell was decreased more or less uniformly along almost the entire length of the cavity. One cask model was inadvertently dropped axially on the wooden block. Its outer shell bottom plate and the cavity bottom plate were significantly concaved inward.

For axial impacts from heights above 21.3 meters, the slumping lead pulled the center cavity tube downward producing a severe deformation in the cavity top plate, Figure 3. The initially flat top plate was formed into a cone. The light seal welds at the circumferential edge of the plate cracked but the structural welds remained intact thus maintaining isolation of the lead shield region from the cavity. No DU-shielded models were tested in the axial orientation.

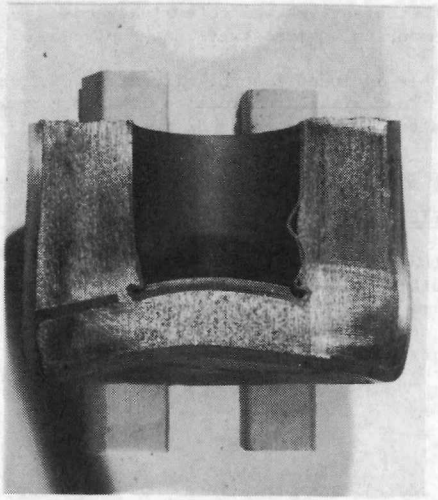


FIGURE 2. DEFORMATION AT BOTTOM END IMPACT POINT OF CASK MODEL DROPPED AXIALLY ON A BLOCK OF WOOD

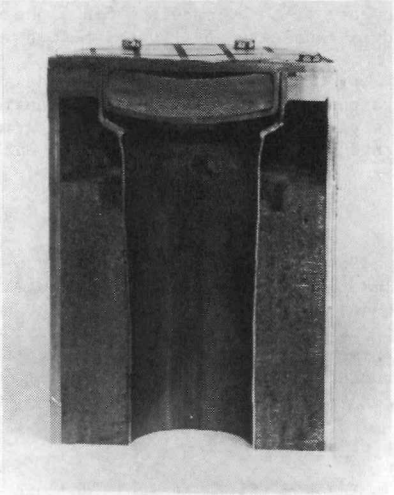


FIGURE 3. EXAMPLE OF LEAD SLUMP AND CONING EXHIBITED BY CASK MODELS DROPPED IN AXIAL AND CORNER C.G. ORIENTATIONS

All the lead-shielded cask models tested in the 30-degree slap down orientation exhibited similar deformations varying in degree with the drop height. The greatest local deformation was at the initial impact end. The deformation at the second impact end was distributed over a larger area and deformation of the cavity was more severe at this end. Although the cavity at the initial impact end had a significant bulge, it was very localized and at a short distance from the end of the cavity it was essentially circular. In contrast, the cavity at the second impact end was significantly oval shaped for over 1/3 the cask length.

All four DU-shielded models were tested in the 30-degree slap down orientation. As with the lead-shielded models, the DU-shielded models exhibited bowing and ovality. However, localized external deformation was less.

Both the lead and DU-shielded models exhibited about the same degree of axial bowing. The bowing was not axially symmetrical, however. A significant portion of the bow occurred approximately 1/3 of the length from the second impact end, Figure 4. The bowing of the lead-shielded models was continuous but the bowing of the DU-shielded models was produced by the localized stretching of the stainless steel shells at the interface between the DU segments.

Structural Damage. About half of the lead-shielded models exhibited cracks he welds, the most severe being in those dropped from 21.3 and 42.7 meters in axial or corner center of gravity orientations. Subsequent destructive examination revealed incomplete weld penetration in these welds. Significantly, the one model dropped axially from 42.7 meters onto a wooden block did not exhibit any weld cracks although it also had poor weld penetration and was highly deformed in the weld region. The stainless steel shells of all four DU models tested did not exhibit any cracks.

All casks dropped axially or in the corner center of gravity orientation from all heights onto the bottom end maintained seal integrity. All bolts were tight after testing but could be readily loosened and the cover removed.

With one exception, all 1/2- and 1/4-scale models, both lead- and DU-shielded, dropped in the 30-degree slap down orientation on the bottom (closed) end from heights of 9.1 meters and greater exhibited leaking seals. The one exception was a 1/4-scale uranium shielded cask model. This cask, dropped from 9.1 meters onto the bottom apparently leaked at the moment of second impact but the seal recovered and further leakage did not occur after 24 hours.

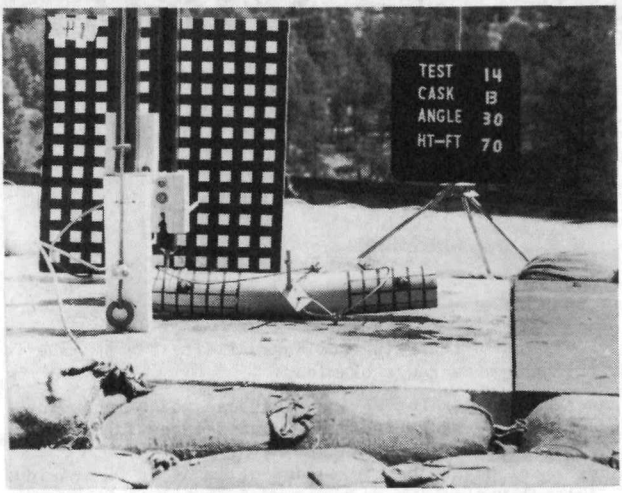


FIGURE 4. BOWING IN MODELS DROPPED IN A 30-DEGREE SLAP DOWN ORIENTATION

Shielding. The lead slump observed agreed with predictions from established correlations. Comparison of a limited number of density measurements of lead archive samples and samples from highly deformed and undeformed regions of tested models indicated an apparent densification of lead during impact. Lead as poured in the models had a porosity of about 1.4 percent which was reduced to as little as 0.5 percent after impact.

The individual DU segments retained their structural integrity. The individual segments did not exhibit any cracks or spalling due to impact and bowing of the individual segments was not observed. Calculations based on the observed ovality indicated average strains of about 18 percent; the local strains were undoubtedly greater.

Radiographs of the DU-shielded models after testing clearly showed the structurally intact DU segments separated at their interfaces in a fan-like manner. By longitudinally sectioning the 1/4 and 1/8-scale models it was determined that because of the interlocking design of the uranium segments, the separations were not fully open gaps but only regions where a decrease in shielding occurred, Figure 5. Calculations indicated that the radiation streaming through these "gaps" would be increased only about 10 percent.

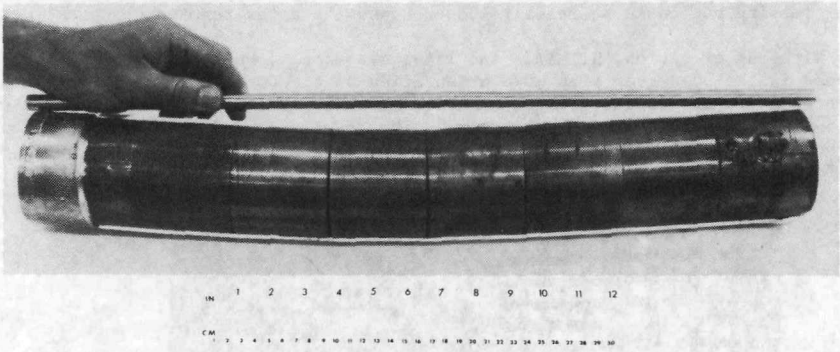


FIGURE 5. DU-SHIELDED MODEL B AFTER DROP TEST

CONCLUSIONS

The following conclusions regarding scaling validity can be made for shipping container models within the size range examined:

(1) Qualitative agreement exists for several response modes including
- Rebound for both lead- and DU-shielded models impacting axially and in a 30-degree slap down orientation,
- Localized deformations at the closed end of lead- or DU-shielded containers initially striking on that end in axial or 30-degree slap down orientation,
- Deformation at the cover end of lead- or DU-shielded containers when that end is the second impact end for a 30-degree slap down orientation,

- Bowing in the lead- and DU-shielded models impacting in a 30-degree slap down orientation,
- Seal leakage for lead and uranium models impacting at a 30-degree slap down orientation.

(2) Quantitative agreement indicating the validity of the scaling relations for both the lead- and DU-shielded models exists for
- The fraction of kinetic energy absorbed during initial impact at a 30-degree slap down orientation, and
- The angular velocity of the models after initial impact at a 30-degree slap down orientation.

(3) Probable scaling validity of axial bowing exists in lead- and DU-shielded models dropped in the 30-degree slap down orientation.

addition the following significant observations were made:

(1) For the shielding segment design employed in these DU-shielded models, impact in the 30-degree slap down orientation can produce fan-shaped gaps between the DU segments. Scaled to a full-size cask, the decrease in shielding for a fall from 21.3 meters would result in about a 10 percent increase in the dose at 1 meter.

(2) Although the results of material characterization tests indicate the DU alloy used in the shield is a brittle material, it responded during impact in a very ductile manner even at high strain rates.

(3) Maximum bowing during an impact event at the 30-degree slap down orientation occurs during the second impact.

(4) Even minor impact protection significantly reduces structural damage.

REFERENCES

(1) Burian, R. J., Black, W. E., Lawrence, A. A., and Balmert, M. E., "The Validity of Scale Modeling for Large Deformations in Shipping Containers", BMI-2040, Battelle's Columbus Laboratories, October 19, 1979.

(2) Straight, J. W., "Nuclear Shipping Container Secondary Impact Analysis", LA-8081-MS, Los Alamos Scientific Laboratory, October, 1979.

DISCUSSION

Question by S. Williamson, United Kingdom: Your tests were done on flask models using lead or uranium as gamma shielding. Do you consider that your observations and conclusions on the validity of sealing would be applicable to flasks using thick steel bodies or cast iron?

Answer: I do not know why these conclusions would not apply to the heavy, all steel or cast iron casks. This could easily be confirmed with a few tests.

Question by W. Gibson, Canada: 1. Did you attempt to provid contact and/or adhesion of the lead to the inner and outer steel jackets? 2. What effect(s) would this contact/adhesion have on the drag of the slumping lead, on the inner cavity tubes?

Answer: 1. The models were produced by the same techniques as a full scale cask would be; however, no special surface preparation was done on the steel jackets. 2. The effects of surface preparation would have to be determined by test. Any comment about the effect would be purely speculation

Question by A. Olivieri, Italy: 1. I would like to know the reason why you did not take into account, regarding your drop test program, side drops. 2. But with regard to fuel integrity, side drop seems to be the most severe test.

Answer: 1. Minimum localised damage is obtained in a side drop. The purpose of the tests was to determine cask scaling validity and not necessarily cover all drop orientations. 2. The purpose was to look at cask damage not fuel or contents damage. However, a dummy, surrogate fuel assembly was included in two tests. These showed some deformation but no failure.

Comment by K. Goldmann, USA: My comment is prompted by your reply to Mr. Williamson's question in whcih you indicated that scaling may not be valid for steel casks.

Buckingham's Pi theorem, if properly applied, represents a law of nature for drop tests, except for strain rate effects on material properties. The results which are contained in the paper appear to confirm that the laws of nature have not been violated. Rather the differences in behaviour between the scale models appear to be due to slight differences in constructing, welding, test orientation, etc. which might also have been observed between two or more models of identical size, strain rate variations of less than one order o magnitude should have been negligible for these tests. Thus, I would predict a similar program of testing steel casks would also demonstrate the validity of the scaling laws for such casks.

Answer: I think you misunderstood my comment. The scaling laws would apply but the exact impacts on the steel or cast iron type containers would have to be confirmed by tests. I disagree with your comment that the differences observed in the tests are due to slight differences in construction. The results showed some very unique differences between uranium and lead shielded containers and good consistant results for different scale sizes of the same cast type.

EXPERIENCE WITH FUEL DAMAGE FROM ABNORMAL CONDITIONS IN HANDLING AND TRANSPORT

W. J. Bailey
Pacific Northwest Laboratory

D. C. Langstaff
Columbia Engineers Services, Inc.

INTRODUCTION

The operational features for domestic commercial spent light-water reactor (LWR) fuel in a once-through fuel cycle are shown in Figure 1. It is anticipated that spent fuel bundles (the terms "fuel bundles" and "fuel assemblies" are used interchangeably in the nuclear industry) would be shipped from the spent fuel storage pool at the reactor site (a) to an interim spent fuel storage facility such as an away-from-reactor (AFR) facility and then shipped to the packaging facility or (b) directly to the packaging facility. From the packaging facility, where the fuel bundles are readied for repository emplacement, the spent fuel bundles are shipped to the repository. One option being studied involves disassembly of the fuel bundles and the handling and transport of suitable quantities of fuel rods rather than discrete fuel bundles.

The design of receiving, handling and storage facilities at a nuclear waste repository requires information on the expected condition of the spent LWR fuel when it arrives at the repository. In programs sponsored by the United States Department of Energy (DOE), a considerable effort is underway by several contractors on spent fuel characterization in the accident/sabotage environment.

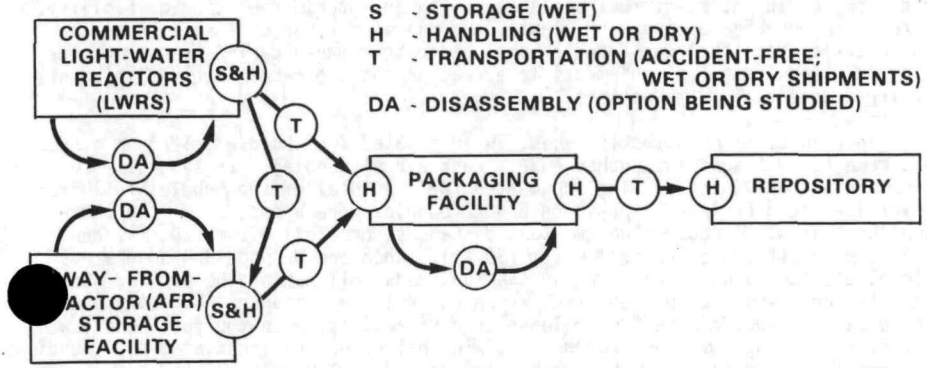


FIGURE 1. Operational Features for Spent LWR Fuel in a Once-Through Fuel Cycle (PNL's Study is Directed Toward Evaluating Damage to Spent LWR Fuel as a Result of Handling/Transport Operations)

DOE's National Waste Terminal Storage (NWTS) Program also has need for similar fuel characterization information under normal or slightly abnormal conditions. The Office of Nuclear Waste Isolation (ONWI) at Battelle Columbus Laboratories (BCL) has described the need in the NWTS Program for the best possible definition of the expected "arrival condition" of spent LWR fuel (after long-term storage in water, handling, and "accident-free" transportation) at a repository to support repository and packaging facility design efforts.

OBJECTIVE OF PNL STUDY

The Pacific Northwest Laboratory (PNL), operated for the DOE by Battelle Memorial Institute under Contract DE-AC06 76RLO 1830, is performing research under the Nuclear Materials Transportation Studies Project to meet the objects of specific elements of the Nuclear Materials Transportation Program, which is being conducted at the Transportation Technology Center (TTC) at Sandia National Laboratories.

The potential arrival condition of spent LWR fuel at a repository is being studied by PNL in one of the project's tasks. The objective of one particular part of that task is to survey and analyze the available information on the damage to irradiated LWR fuel as a result of normal handling/transport operations (there have been tens of thousands of fuel moves), which entail some abnormal occurrences. This paper describes the findings from that part of the studies. This paper also includes pertinent data from earlier studies performed by PNL for the USAEC (see NUREG-0032) and the USNRC (under FIN B2151 in 1978). The material in this paper is discussed more fully (includes detailed reference list) in a report, PNL-3582 Vol. 1 (TTC-0165).

RESULTS TO DATE

As shown in Table 1 the PNL study to date includes an evaluation of 75 cases (58 domestic, 17 non-USA) where damage to fuel occurred or may have occurred during handling/transport operations at reactor sites and at spent fuel storage facilities. Irradiated fuel is involved in almost all cases (four deal with fresh fuel). Of those cases where damage to fuel occurred during fuel handling, 39 cases were associated with fuel moves at reactor cores, 15 with fuel handling and cask loading at spent fuel pools, one with an interim fuel storage facility, three with non-USA reprocessing plants, and one with a hot cell facility. In three cases where fuel was found or thought to be damaged during transport, one case involved fresh fuel shipments to a reactor and two cases involved shipments of irradiated fuel from reactors.

In each of three domestic cases, an irradiated fuel bundle tilted or tipped and came to rest against another object (not a fuel bundle). In 14 cases (10 domestic, 4 non-USA) involving irradiated fuel, a total of two bundles fell from a vertical to a horizontal position and 12 bundles were dropped. Of the three bundles that were dropped the greatest distances, one fell 6.1 m (20 ft), one fell 8 m (26 ft), and one fell 9.1 m (30 ft). When one dropped BWR bundle was picked up, the channel fell off and the fuel rods fell out of the bundle. So far, two cases have been identified where gas releases occurred, but only one of these cases showed a momentary release of radioactivity when the fuel bundle was moved, indicating some fuel rod damage. In that case, one irradiated fuel bundle was dropped 9.1 m (30 ft) onto another irradiated fuel bundle and the radioactivity release came from the latter bundle.

Damage to fuel during bundle handling operations also occurred in other ways. In ten cases involving over 68 fuel bundles the grid spacers were damaged during handling operations (e.g., by catching on some portion of an adjacent

fuel bundle). With 32 of these bundles, the grid spacers were damaged enough that they provided less than full support for the fuel rods. Two cases involve damage caused by twisting the fuel bundles. In one case with fresh fuel, handling operations resulted in bent fuel rods. Other cases involve these kinds of damage: top nozzles separated from cans (broken tack welds), nut capture devices bent, lower nozzles and tie plates deformed, bails deformed, channels deformed, hold-down springs deformed or broken, and fuel loss from two bundles with failed fuel rods. One case involves fuel rod handling: an unaccounted-for spent fuel rod (upper shank of this tie rod was broken) was found when the spent fuel pool was drained.

In the cases involving transport incidents, three cases (one with fresh fuel two with spent fuel) involve fuel damage during transport. In the fresh fuel e, excessive lateral loads during shipment and/or handling resulted in damage to grid spacers. In the two cases (one non-USA, one domestic) with spent fuel, radiation releases to the casks were encountered after stainless steel-clad fuel was shipped dry. In the case of the non-USA spent fuel, the four bundles had appeared intact prior to shipment; however, the fuel was from a lot that had experienced fuel failures. In the case of the domestic spent fuel, the one fuel bundle was known to contain failed fuel rods.

SUMMARY AND CONCLUSIONS

Very few fuel bundles have sustained major mechanical damage because of handling operations. Little is known about damage to fuel during shipments; however, such damage, if any, is apparently minor.

In most cases involving damage from handling, it appears that the result was minor degradation of fuel bundle components and no breaching of the fuel rod cladding or release of radioactive gases, even in those cases where fuel bundles were dropped as much as 1.5 to 9 meters through water and impacted on other fuel bundles or the bottom of the spent fuel storage pool.

TABLE 1. Summary of Experience in Handling/Transport Operations With Irradiated LWR Fuel*

Receiving of New Fuel at Reactor

- Fresh PWR fuel bundle dropped 1.5 m (5 ft) to floor. No fuel rods ruptured; some fuel rods bowed; some spacer grids broken; fitting at lower end bent. (Docket 50302-209)
- Non-contact between some Zircaloy grid spring fingers and fuel rods was noted with fresh PWR fuel. Non-contact believed to be caused by excessive lateral loads applied to fuel rods during handling or shipping. (Am. Power Conf. 35:215-236, 1973).

Refueling of Reactor Core

(a) Fresh Fuel Bundles:

- Fresh PWR fuel bundle dropped 0.10 or 0.13 m (4 or 5 in.). Bundle skel[illegible] replaced before bundle loaded into core. (Docket 50251, letter, April 20, 1973).
- Two fuel rods in a fresh PWR fuel bundle damaged when bundle loaded into core between two slightly distorted bundles. One rod was out of upper end fitting; one rod was bent; no cladding failure observed. (Docket 50313, RO No. 77-7).

(b) Irradiated BWR Fuel Bundles:

- One fuel bundle ("A"), which had a burnup of 291 GJ/kgU (3365MWd/MTU), dropped 9.1 m (30 ft) onto another bundle ("B"). Lower tie plate cage on "A" was deformed upwards toward tie plate. Bail handle on "B" was deformed almost horizontally; channel driven downward so that its lower edge flaired over lower tie plate shoulder; movement of "B" caused temporary increase in airborne radioactivity. Channels on two fuel bundles adjacent to "B" were dented on top edge. (Docket 50331-518)
- A channeled fuel bundle dropped about 3.7 m (12 ft) to transfer pool floor. No gaseous release noted. No apparent damage to bundle. (Docket 50010, RO No. 78-034/03L-0).
- Irradiated fuel bundles discharged included one bundle that was a grappling problem and one dropped bundle. (Docket 50249-404; NEDO-10505; NEDM-10735, Suppl. 6-8).
- Fuel bundle bumped against reactor vessel. (Nucl. Safety, p. 334, May-June 1964).
- One fuel bundle damaged during refueling outage. (Docket 50237-420).
- Spacer grid damage to one fuel bundle was extensive and beyond repair. No evidence of damage to fuel rods. (Docket 50409-240, -236, -230, and -276).
- Three fuel bundles not fully seated in core due to spring clips being hung up on core upper grid. (Docket 50254-642 and -551).
- Bowing of fuel rods observed. Shroud locking rings unlocked during previous operations, causing twisting and stressing of bundles. (Docket 115-5-99).
- When changing orientation of fuel bundles at a non-USA BWR, general purpose grapple sometimes gets entangled with bundle bale and is difficult to release. (IAEA- SM-178/43).
- Twice, grapple caught bale of dummy fuel at a non-USA BWR; each time fuel slipped off while being lifted. (IAEA-SM-178/43).
- Many freely supported spacers on top grid of core (present day channels have spacers welded to them) found to be bent or gall on fuel bundles at a non-USA BWR. (IAEA-SM-178/43).
- Mechanical interference between fuel bundle channels and control blades noted at a non-USA BWR. (Docket 50245, letter, Jan. 30, 1973)

TABLE 1. contd

- Fuel bundles at a non-USA BWR (pressure tube) found lodged at bottom of pressure tubes. Damage to bundles was slight. (BNES Conf. On "Nuclear Fuel Performance", pp. 51.1-51.6, Oct. 1973)

(c) Irradiated PWR Fuel Bundles:

- A PWR fuel bundle fell several centimeters (a few inches) to top of core adjacent to its original position. Upper nozzle and upper fuel bundle wrapper sheet damaged. (Docket 50029-216; Docket 50029, letter, Nov 20, 1973; Docket 50029, letter, Mar. 14, 1973)
- A fuel bundle tipped about 0.6-0.8 m (24-30 in.) onto core baffle. No visible damage to fuel bundle. (Docket 50304-262; Docket 50304, letter, Dec. 5, 1973)
- One fuel bundle tipped against a corner of the core shroud. (Docket 50255-241)
- One peripheral fuel rod of a fuel bundle was bowed outside envelope of bundle. Rod had an apparent cladding failure. No evidence of damage due to bowed fuel rod observed in adjacent fuel bundle. (Docket 50317, LER 78-11)
- Some trouble in refueling due to bowed fuel rods in PWR fuel bundles. (NUREG-0032)
- Two spacer grids were damaged on a fuel bundle. (Docket 50309-79)
- Bowed fuel rods resulted in some fuel loading problems; damaged spacer grids noted on 29 fuel bundles. (NUREG-0032; Docket 50309, semiannual report, Aug. 27, 1974; Docket 50309-262; EPRI NP-218; Nucleonics Week 15(37):8, Sept. 12, 1974, and 15(50):4, Dec. 12, 1974, a McGraw-Hill publication)
- Sections of a spring clip grid spacer from fuel bundle found in steam generator. Six fuel rods partially unsupported at that one grid location. No apparent damage to other bundles. (Docket 50261-294; Docket 50261, Report No. 8, Aug. 29, 1974)
- Grid strap damage noted on 31 fuel bundles. Pieces missing or grid material laid over. No damage to fuel rods noted. (Docket 50272, LER 79-044/03L-1; NRC IE Circular No. 80-13)
- One fuel bundle had damaged grid. Two fuel bundles registered high loads when removed from core. (Docket 50344, letter, July 9, 1980)
- Lower nozzle on one fuel bundle jarred during core loading. (Docket 50213, letter July 16, 1980)
- Top nozzles on two PWR fuel bundles became separated (tack welds failed) from can during refueling operation and at cask loading operations at spent fuel pool. (Trans. Am. Nucl. Soc. 16:102-103, 1973)
- Bent nozzle springs noted in three PWR fuel bundles. (Docket 50261, Report No. 8, Aug. 29, 1974)
- Seven failed hold-down springs found on PWR fuel bundles. Several other potentially failed springs also noted. (Docket 50346, LER 80-040/031-0)
- Two of four hold-down springs on top nozzle of a fuel bundle damaged by another bundle. (Docket 50281-355; Docket 50280-419)
- Twenty bent nut capture devices found on fuel bundles. (Docket 50255, letter, Dec. 16, 1971)
- One irradiated PWR fuel bundle was slightly damaged because top of bundle was twisted with respect to the bottom. (Docket 50309, LER 78-018/99X-0)

TABLE 1. contd

- Spent fuel bundle mispositioned, causing damage to bundle and refueling equipment. (Docket 50029, Report No. 104, Sept. 23, 1969)
- Difficulties encountered with fuel handling equipment at one PWR. (NUREG-0032)
- Non-USA fuel bundles damaged by leveling bolts and shims. (Nucl. Safety, p. 83, Jan.-Feb. 1975)
- One non-USA PHWR fuel bundle failed because of mechanical damage due to a handling incident. (BNES Conf. On "Nuclear Fuel Performance", pp. 53.1-53.4, Oct. 1973)
- Five non-USA PHWR fuel bundles failed because of maloperation of fuel-loading equipment. (Reactor Tech. 14-1:68-98, Spring 1971)
- One potential failure mechanism for non-USA PHWR fuel bundles was mech cal damage during fuel handling. (BNES Conf. On "Nuclear Fuel Performance", pp. 56.1-56.5, Oct. 1973)
- Some fuel failures at a non-USA PHWR caused by fueling machines during unusual maneuvers; a few bundles may have been damaged by handling equipment only after discharge. (AECL-4520; Trans. Am. Nucl. Soc. 16:102, 1973)

Storage and Handling of Fuel at the Spent Fuel Pool

(a) Fuel Rods

- Spent tie rod from BWR fuel bundle unexpectedly found on pool floor. Rod upper shank broken. Rod apparently misplaced. (Docket 50155-181 and -247)
- Experience with fuel bundle rebuilding at a non-USA BWR has been good. Not a single fuel rod has been dropped (1085 bundles rebuilt). (RB 78-29; BNWL-TR-320)

(b) BWR Fuel Bundles

- An irradiated fuel bundle fell about 6 m (20 ft) in pool. No measurable release of radioactivity. Nose piece and nose piece end of fuel channel crushed; no indications of broken fuel rods. (Docket 50293, AO No. 74/3)
- Irradiated fuel bundle dropped about 1.8 m (6 ft) to pool floor and then tipped over into corner of pool in 3 m (10 ft) deep spent fuel cask pit. When bundle lifted to vertical position, channel fell off and fuel rods came out of bundle. (Docket 50133-326 and -264)
- One fuel bundle fell from vertical to horizontal position. Bundle that fell not visibly damaged; however, that bundle did bend bail on another bundle. (Docket 50324, letter, April 16, 1976)
- An unchanneled spent fuel bundle fell about 1.5 m (5 ft) to floor of pool. No release of radioactivity measured, even though bundle was damaged (bowed over its whole length). (Docket 50245-421)
- A non-USA fuel bundle with a burnup of 950 GJ/kgU (11,000 MWd/MTU) fell 8 m (26 ft) to bottom of pool. No release of radioactivity noted. (RB 78-29; BNWL-TR-320)
- A non-USA fuel bundle dropped several centimeters (a few inches) when grapple rope snapped. Bundle came to rest on top of spent fuel storage rack. Spacer grids on bundle displaced; cone-piece on bottom tie plate dented. (IAEA-SM-178/43)

(c) PWR Fuel Bundles

- Lifting frame struck one fuel bundle. Two fuel rods on periphery of bundle were bent. Cladding on two fuel rods sound but rod deformation extensive enough to render them unacceptable for further use. One grid spacer on bundle also sustained damage. (Docket 50251-352)

TABLE 1. contd

- One fuel bundle bumped slightly on containment side during transfer to spent fuel pit. No damage noted. (Docket 50344, letter, July 9, 1980)
- Test weight dropped; upper end of weight came to rest against a fuel bundle. No apparent material damage. (Docket 50302, LER 78-031/01T-0)
- After loading shipping cask and while trying to disengage the loading tool, top nozzle of spent fuel bundle broken off. Fuel rods not damaged. (Docket 50003, letter, Aug. 27, 1970)
- Top nozzles on PWR fuel bundles became separated (tack welds failed) from can during cask loading operations with spent fuel (see entry in "Refueling...", above). (Trans. Am. Nucl. Soc. 16:102-103, 1973)
- Broken hold-down springs found on four fuel bundles. (Docket 50269, LER 80-015/01T-0)
- Damage to one fuel bundle caused by handling at the spent fuel pool. (Docket 50255, Report No. 7, Aug. 30, 1974)

Transporting of Irradiated Fuel**

- Truck with two casks containing irradiated fuel went over embankment; fuel remained safely in casks. No statement made as to condition of fuel. (AECU-3613, p. 25)
- Truck shipment of 25 irradiated fuel elements in 7 casks involved in collision with another vehicle. Fuel elements were undisturbed. (TID-16764, p. 7)
- During truck shipment of irradiated fuel, trailer with cask became unhitched from tractor. Trailer tilted; cask slid to and broke through end of trailer. No damage to cask. No statement made as to condition of fuel. (TID-16764, p. 17)
- Railroad cask car with irradiated fuel elements derailed. No damage to cask car; no contamination or increase in radiation intensity at cask surface. No comment made as to condition of fuel. (TID-16764, p. 22)
- Switch engine backed into railway flatcar loaded with cask containing irradiated fuel elements and jounced flatcar about 91 m (300 ft). Cask was not displaced. No radiation release. No comment made as to condition of fuel. (TID-16764, p. 25)
- Railroad shipment of irradiated fuel rods involved in two incidents. No radiation release. No comment made as to condition of fuel. (TID-16764, p. 29)
- As a result of improper practices (cask retaining chains relaxed and truck brakes applied suddenly in attempt to shift cask to meet axle weight requirement), cask loaded with irradiated fuel elements moved about 4.6 m (15 ft), damaging trailer and truck. No evidence of physical damage to cask; however, second trailer, onto which cask was loaded, was found to be contaminated. No comment made as to condition of fuel. (TID-16764, p. 51)
- During truck shipment of cask loaded with irradiated fuel elements, blocks holding cask tie-down cables pulled loose and cask shifted inside sealed exclusive-use trailer in transit. Cask lid and trailer bed were contaminated. Source of contamination was apparently residual water in cask that splashed out while cask was shifting about during transit. No comment made as to condition of fuel. (TID-16764, p. 53)
- Leakage of contaminated cooling water occurred during railroad shipment of cask loaded with irradiated fuel (included some failed fuel in aluminum cans). Cask and car were contaminated. No comment made as to condition of fuel. (TID-16764, p. 67)

TABLE 1. contd

- During truck shipment of cask with irradiated plutonium fuel elements, shipping container leaked. Trailer and street (one location only) were contaminated. No comment made as to condition of fuel. (TID-16764, Suppl. 1, p. 28)
- Cask containing irradiated fuel samples became contaminated during rail shipment. No comment made as to condition of fuel. (TID-16764, Suppl. 2, p. 9)
- Truck with loaded spent fuel cask overturned. Cask assembly thrown into ditch and traveled over 30.5 m (100 ft). No release of cask contents or increase in radiation occurred. Fuel found to be undamaged. (IAEA-SR-10/5)
- During a shipment by truck, trailer carrying cask containing spent fuel underwent a structural failure (it buckled) and cask moved forward about 0.46 m (18 in.). There were no radioactive releases. No comment made as to condition of fuel. (Docket 50010, letter, Feb. 28, 1978)

Handling of Irradiated Fuel After Shipment

(a) Interim Spent Fuel Storage Facility

- A domestic facility has handled 1196 fuel bundles without dropping any. In early operations, it was difficult grappling bundles if they sat far down in cask cavity. Grapples sometimes get caught in the spring clip on top of PWR fuel bundles. A cask tipping incident led to a delay of several days. (NUREG/CR-0956; PNL-3065)

(b) Reprocessing Plant

- Four PWR fuel bundles shipped dry from non-USA reactor to non-USA reprocessing plant. Bundles appeared intact prior to shipment; however, they were from lot that had experienced fuel failures during reactor operation. Radioactive release from bundles contained within cask during shipment. Transfer of fuel from cask to storage pool led to higher than usual activity of pool water; purification system soon permitted normal pool operations. Release probably due to damage to fuel bundles during shipping. (PNL-3171; Nucl. Tech. 43:163-172, 1979)
- Non-USA spent fuel bundles with burnups to 3370 GJ/kgU (39,000 MWd/MTU) have been stored, disassembled, and reprocessed in non-USA facilities. Very few fuel rods were broken during disassembly operations. (Proc. NEA Sem. Storage of Spent Fuel Elements, pp. 19-29, Spain, 1978)
- One fuel bundle at a non-USA plant fell from vertical to horizontal position.

(c) Hot Cell Facility

- One PWR fuel bundle (with known defective fuel rods) shipped dry to a hot cell facility. Radioactive release was contained within cask during shipment. Airborne contamination release during underwater unloading of bundle from cask described in another PATRAM '80 paper (Klingensmith, Paper No. 225).

* Entries in the table involve irradiated fuel in almost all cases. A few pertinent entries involving fresh (unirradiated) fuel have also been included.
PWR = pressurized water reactor; BWR = boiling water reactor;
PHWR = pressurized heavy-water-moderated reactor.

** Accidents involved in some cases.

CALCULATIONS AND EXPERIMENTAL TESTING OF A NEW CONCEPTION IMPACT LIMITER FOR IRRADIATED FUEL SHIPPING CASK

A. Olivieri - G. Pochini - AGIP NUCLEARE S.p.A. - Milano

1) Use of semiellyptical shells as impact limiters

If we go through the history of radioactive material transportation, we notice that the first containers were generally protected against impact accidents with fins specially designed for heat transfer.

Deformation modes of such systems and also their strain energy absorption capabilities was noticed to be quite varied and highly dependent on the angle of the fin axis compared with the vertical line perpendicular to the impact surface.

As the international regulations impose to consider all the possibilities, as regards a container drop on an unyielding surface, and as finned systems are generally highly anisotropic, it is difficult to obtain a satisfactory acceleration reduction using these systems. In fact, anisotropy requires to dimension the impact energy absorption system for those impact directions which otherwise would involve a too low energy absorption, with a consequent acceleration increase in the other directions.

In addition, this phenomenon worsens because, as regards buckling phenomena (either elastic or plastic), the highest load is originated when displacement starts; this means that even considering displacement quite lower than the highest allowed by the system, no consistent peak acceleration reduction occurs.

To overcome these difficulties and, therefore, to obtain low accelerations regardless of the impact angle, the energy absorption system requires to approximate a spherical shaped structure: a system consisting of two standard ellyptical shells welded to each other meets these requirements.
In addition, it provides the cask lid, valves and seals with an effective protection against fire and puncture.

These impact limiters have been studied for AGN-1 irradiated fuel transport cask.
This cask has been designed to transport ENEL's irradiated fuel to a "central" storage pool.
As it is equipped with an iron "gamma" shield, this cask is extremely rigid and the only critical points are: the lid and its seals; the valves, positioned 2 on the cap and 1 on the lower end; the basket and the fuel, which, in any case, shall have a subcritical configuration.

The semiellyptical shells (one on the lid and the other on the lower end) protect both seals and valves.
In addition, since they guarantee low accelerations values, they set a limit to stresses to both the basket and the fuel.

304 type stainless steel is the ideal material as regards these impact limiters; it shows a very good resiliency at any temperature, even -40°C, as provided for by IAEA regulations: however, as the use of common carbon steel cuts costs down, Fe 42 D carbon steel, widely used for low temperature service, is now being studied.

2) Verification of the ellyptical shell energy absorption capabilities through the "MARC" program

2.1. MARC code description

The MARC program was set up by the MARC corporation for the study of complex structural and thermal problems.
It uses the "finite element" method which has already been widely tested in previous codes such as SAP and NASTRAN.
However, compared with the above two codes, MARC can analyze mono, bi and tridimensional structural problems also in plastic field, considering any stresses and strains relation given in input; the code breaks down each element into a certain number of layers (this depends on the type of element used) and gives each layer the necessary elastic properties in accordance with the stresses which it is subjected to.

In addition, MARC allows the study of problems connected with "large displacements", i.e. when the stress characteristics are affected by the system strained geometry.
Therefore, this code can update the stiffness matrix after each calculation, and thus it allows to consider geometric variation at any step.

MARC allows the study of the plastic strain of a system using the method of load increments.

Sometimes it might be convenient to apply a displacement increment to one or more points to better check any possible buckling phenomena; however, from a conceptual view point, the method used by the code to perform the analysis is the same.

2.2. Calculation methods

2.2.1. General

All calculations have been carried out by increasing the

displacement (in the impact direction) of one ore more points of the structure to be analyzed.
It was used the "LARGEDISP" option (large displacement) which allows to consider the strained geometry as far as each load (or displacement) increment is concerned.
MARC program performs a first calculation to determine the load under which all parts of the structure are below elastic limits; load (or displacement) is then progressively increased following the structure deformation.
As regards drops, three standard modes have been analyzed, i.e.:

a) axial drop along the cask simmetry axis ;
b) "side drop", symmetry axis being parallel to the impact surface ;
c) "corner drop", vertical line coming from the cylinder barycenter and going through the contact area between the impact limiter and the impact surface.

2.2.2. Axial drop

The first part of the deformation of an ellyptical shell in the axial direction can be studied by replacing the surface under consideration with a spherical surface.
The axial-symmetric shell surface (element ## 15) has been divided into 5 elements and 5 calculation runs have been performed.

During the first run, point 1 (which is the first point touching the impact surface) undergoes a displacement increment until the axial coordinate of this point becomes the same as in point ## 2.
Then point 2 is displaced and so on up to point 5 included; point 6 has been considered as fixed.

When the first 5 calculations are performed, the remaining part of the shell is analyzed after having been replaced with a toroidal surface.
This surface too has been divided into 5 elements (with 6 nodal points), but only the displacement of point 1 has been studied (see fig. ## 1); as the structure is symmetrical in respect of the circumference weldings, only a 90° angle has been considered.
Including the output data relevant to each calculation in the obtained diagrams we obtain (see fig. ## 2) the load vs displacement diagram relevant to the axial strain of the elly ptical shell under consideration.
The energy relevant to a 60,000 kgs cask (for which the impact limiter has been dimensioned) is absorbed, axial strain being equal to about 700 mm and the highest load

(at strain end) to about 1,200,000 kgs corresponding to a maximum acceleration equal to 20 g approximately.

2.2.3. Side drop

Considering this type of drop, to study the impact limiter behavior, we considered only that section of the shell (1/4 of the whole component) which is affected by strain.

We assumed that strain occurs symmetrically both as regards the vertical plane, which includes the largest circumfere[illegible] and is common to both semiellyptical welded shells, and a[illegible] regards the vertical plane perpendicular to the first and going through the cask symmetry axis.

During the first calculations we used the MARC double curvature shell element (element 24).
The shell deformation has been studied imposing a radial displacement to the point where the first contact, between the impact limiter and the unyielding surface, occurs.

The whole energy of 60,000 kgs cask is absorbed by the two impact limiters with a maximum displacement equal to about 250 mm.
The highest load is equal to about 1.92x106 kgs and corresponds to a maximum acceleration of 64 g.

Another calculation has been performed in axi- symmetrical geometry and the part of shell affected by strain has been simulated using a spherical surface.
Thecurvatureof the above surface has been determined by using always the same length as of the lines of plastic hinges.
Practically, ellipses have been replaced with circumferences, having the same perimeter, which are considered as intersections of a spherical surface with a plane parallel to the horizontal one simulating the impact surface.
This consideration allows to define an "equivalent sphere".
This calculation has been carried out using the axi-symmetrical shell element (element ## 15) as in the case of axial drop.
Also in this case, at first we considered a displacement of the first contact point.
Then, after the next point reached, the incremental displ[illegible] ment was imposed to this point and so on up to the exhaus[illegible] of the spherical shell strain capacity.
By merging the diagrams relevant to each calculation run, we obtain the diagram shown in fig. ## 3, which concurs with the diagrams obtained by the previous calculation carried out with element ## 24.

2.2.4. Corner drop

We consider the drop of a cask showing an axis bending equal to about 20°, as to the vertical line, and assume that the whole drop energy is absorbed by one impact limiter only i.e. that no overturn occurs after the first impact.

The shell surface has been divided into 6 elements using the double curvature shell element (element 24).
Because of symmetry reasons, only half the structure has been considered.
The load vs displacement diagram (see fig. ## 4) has been obtained by imposing a small axial displacement to those points which are very close to the impact surface.
Because of symmetry reasons, the resulting load has then been multiplied by 2.

We noticed that the only impact limiter involved absorbs the whole energy of a 60,000 kgs cask with a maximum displacement equal to about 250 mm.
The highest load is equal to $3{,}3 \times 10^6$ kg and corresponds to an acceleration value equal to 55 g.

3. EXPERIMENTAL TESTING OF ELLYPTICAL SHELLS ENERGY ABSORPTION CAPABILITIES

3.1. Foreword

Several experimental tests, both static and dynamic, were performed in order to verify the impact limiter capabilities and to assess the adopted computing methods.

These preliminary tests were performed on scaled models (1:20 and 1:9); in static compression tests, loads vs displacements were measured; in dynamic drop tests, accelerations vs time diagrams were obtained.
Both 1:20 dynamic tests and 1:9 static compressions tests were carried out at NUOVO PIGNONE, Firenze: this company also fabricated all the testing models and is expected to act as manufacturer as regards all the future tests models and AGN 1 shipping cask prototype.

As regards 1:9 scaled models drop tests, they were performed at the SCALBATRAIO Laboratory of the PISA University, where a cask testing facility was equipped with the following capabilities: drop tests, fire tests, puncture tests and water immersion tests.

So far, only carbon steel models have been used; a typical stress-strain diagram for this kind of steel is shown in the following picture slide.
In order to obtain accelerations vs time diagrams, a measurement device consisting of: some piezoelectric accelerometers, some charge amplifiers, an oscilloscope, was prepared.
Models were fixed to the target and hit by a semplified scaled cask model dropped from the tower using explosive bolts.

3.2. Tests relevant to axial drop

3.2.1. Static compression tests

As regards this kind of test, it was noticed that at first load went up to about 9,000 kgs (this value corresponds to a displacement of about 30 mm); then it started to decrease while displacement reached a 40 mm value (Fig. 5).
The strain energy absorbed in these two phases was about 65 % of the energy corresponding to a 1:9 scaled cask model dropped from 9 m.
Afterwards, the load increased again, with a slope slightly smoother than the previous one, up to 16,500 kgs corresponding to a 62 mm displacement.

As a 1:9 scaled cask model corresponds to 82 kg, at the end of a 9 meters drop test it hits the target, energy being equal to 738 kgm.
By integrating the diagram shown in fig. 5 till this energy value is matched, a 90 mm maximum displacement and a 16,500 maximum compression load were obtained.
In the full scale these values correspond to a 800 mm displacement and to a 22 g acceleration.

3.2.2. Axial drop tests

At the SCALBATRAIO Laboratories of the PISA University two end drop tests were performed on 1:9 scaled models made out of the same carbon steel as used for static tests.
Each model was fixed to the target existing at the NUCLEAR PLANTS INSTITUTE facilities of the PISA University.
Then, a 41 kg guided weight was dropped from 9 meters.
Two acceleration vs time diagrams, very similar to each other, were obtained (see fig. 6); by integrating these curves twice, a load vs displacement diagram was obtained.
This diagram is similar to the static one, but the maximum load value is twice greater (about 17,500 kg).

Tests results are synthetized as follows :

- Test number	1	2
- Impact velocity (m/sec)	13,5	13,6
- Drop height (mts)	9,3	9,4
- Impact time (sec)	6,3	6,0
- Max. acceleration (g)	520	540
- Max. 1100 Hz filtered acceleration (g)	360	390
- Max. load (kg)	-	17.500
- Final displacement (mm)	42	40

It should be considered that for these tests 41 kgs were used. This weight is equal to about half the weight of a 1:9 scaled cask model; so, the total energy absorption was only half the energy that should be absorbed in a 1:9 scale test. An extrapolation of the diagram obtained, on the basis of the static curve, shows that load increases up to about 20,000 kgs, before the whole energy absorption, and that displacement increases up to 65 mm.
It follows that a maximum acceleration value of about 27 g and a displacement value of 550 mm should be considered for a full scale test.

3.3.2. Side drop tests

Some side drop tests on 1:20 scaled carbon steel models were performed at the NUOVO PIGNONE, Florence.
As regards such tests a cask model, equipped with the two impact limiters, was dropped onto a target consisting of a very thick and tough steel plate.

The following figure shows a typical acceleration vs time (fig. 7).
The overimposed continuous line shows a curve in which the vibration waves have been cut off.
From this filtered diagram, a maximum acceleration value of about 1,100 g can be deduced.

3.4. Future tests

This program aims at the following :

- a better definition of the acceleration values as regards the three drop models (first of all, the "corner drop" should be analyzed, as so far no tests for this kind of drop have been performed).

- Scaled model tests validation: it should be assessed that an extrapolation from scaled model tests data is correct.

- A better definition of impact limiters material, taking into account the problems arising from low temperature conditions (reference temperature = -40° C).

In order to verify the correlations between the models geometric scale and the measured or computed parameters, several tests on 1:6 and 1:2 scaled models will be carried out in the next three weeks.

4. CONCLUSIONS

Unfortunately, the experimental program is not yet finished and we are not in the position to draw final conclusions: however, on the basis of the calculations and of the preliminary tests, we expect to have our assumption confirmed.

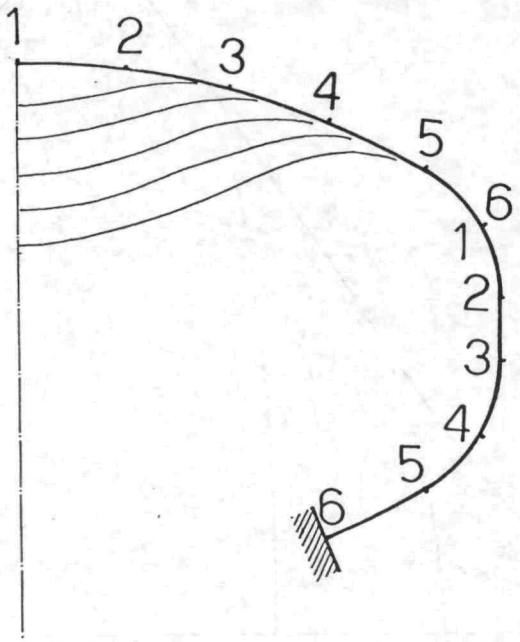


FIG. 1

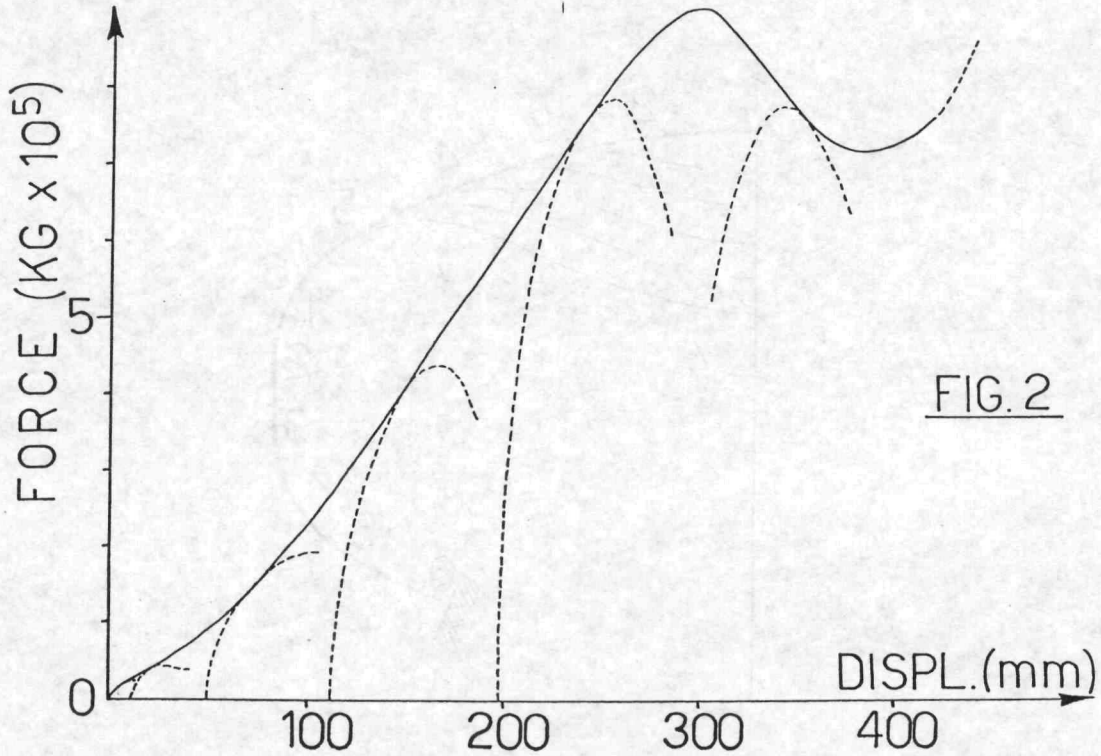


FIG. 2

1130

FIG. 3

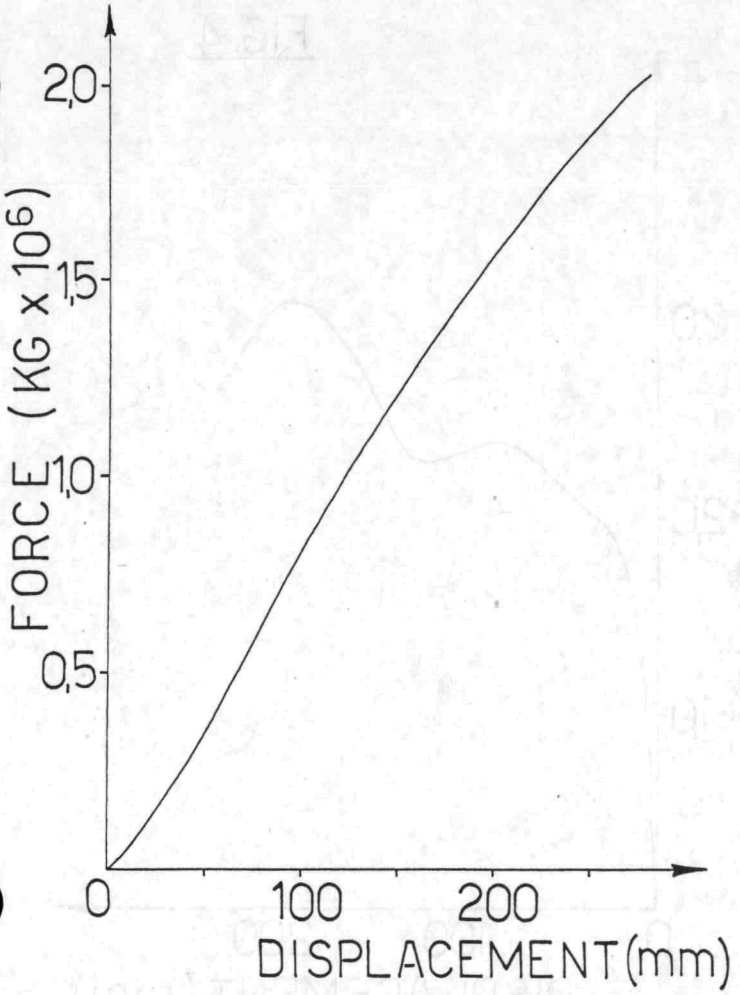


FIG. 4

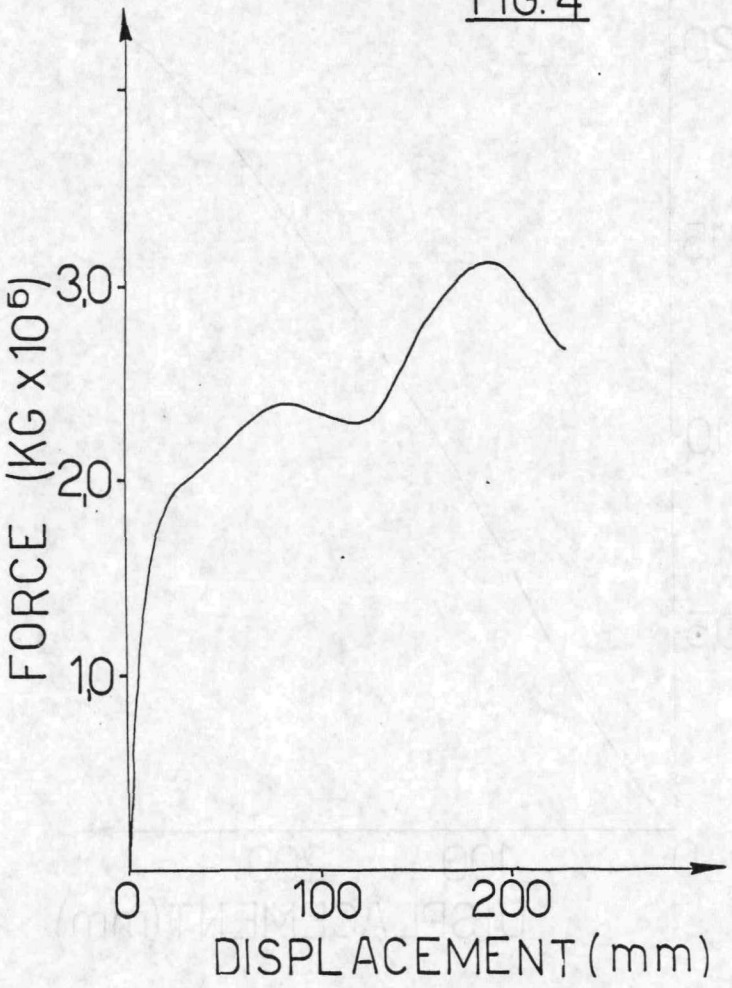
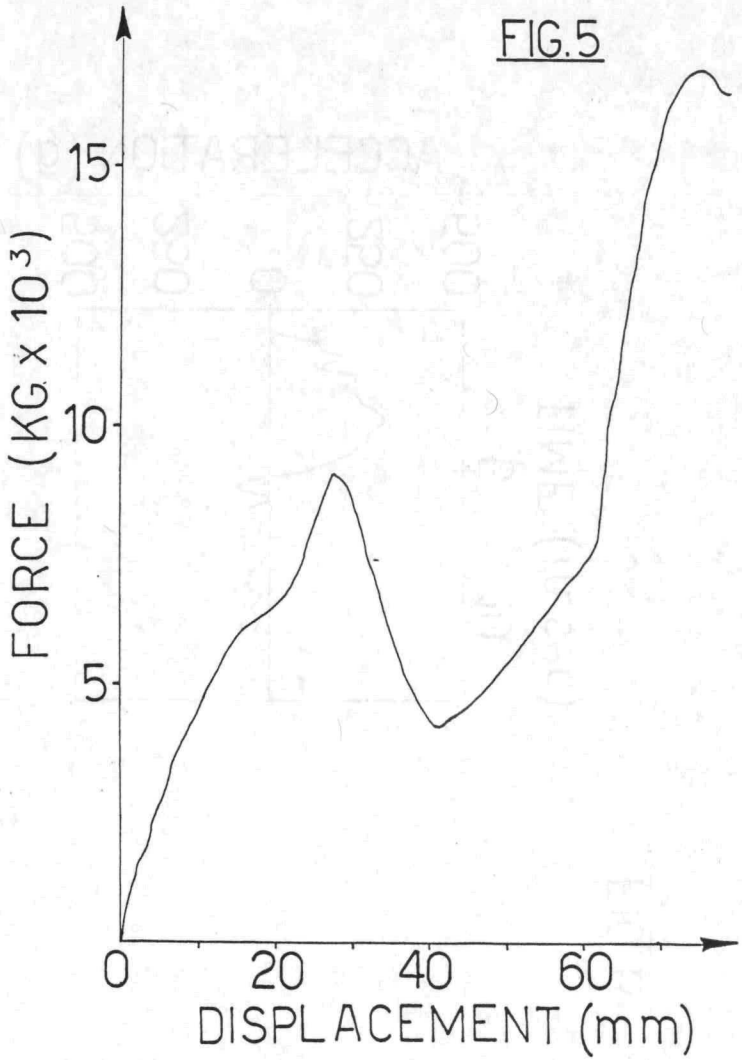


FIG.5



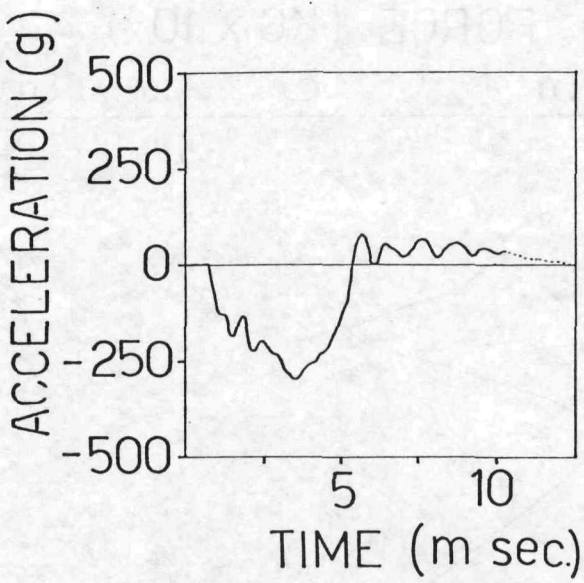
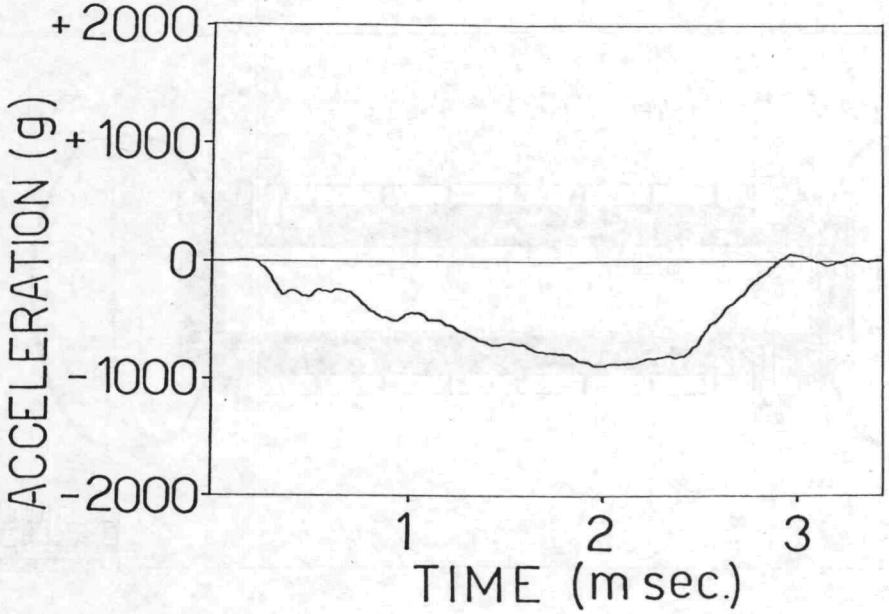
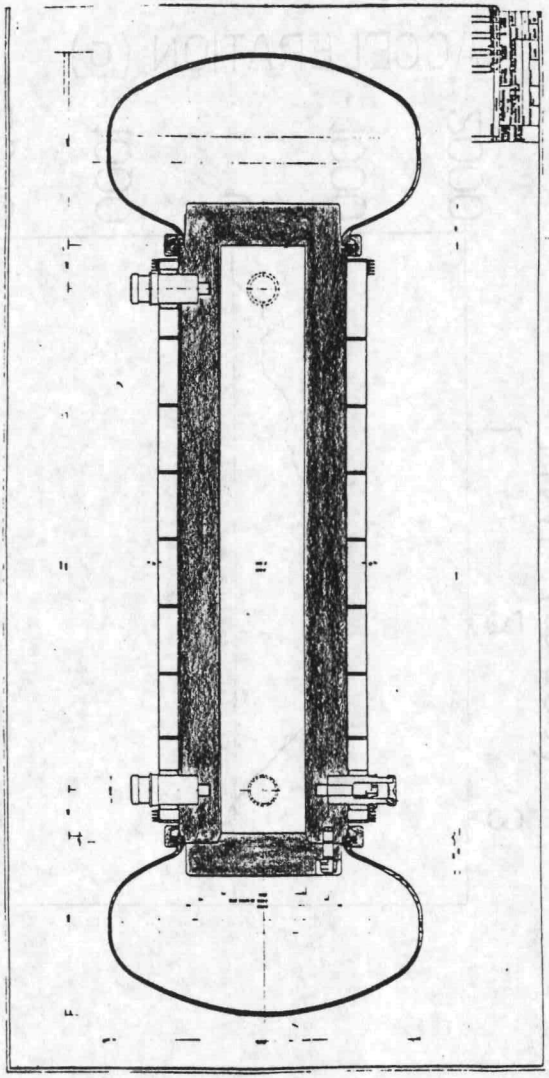


FIG. 6

1134

FIG. 7





DISCUSSION

Question by K. Wieser, FR Germany: What is the difference in cost between the performed calculations and the tests?

Answer: I am not in a position to give an exact evaluation, but you must consider that fin impact limiters tests are very expensive both because of the varying behaviour of fins and because of the presence of a great quantity of welding joints; on the other hand, it should be considered that semi-elliptical shells can be easily obtained as standard heads.

Question by W. Gibson, Canada: Did you derive a dynamic stress-strain curve for the steel impact limiter material which you used in the "MARC" model; in other words, did you take into account the "strain rate" effects in a dynamic load model?

Answer: We know these relations, but we believe that in this first phase, a good result could be obtained even without taking into consideration the strain rate effects.

TEST FOR DEMONSTRATING THE RELIABILITY OF CONTAINERS USED FOR (TRANSPORTING) SPENT FUEL – RESULTS OF 9-METER DROP IMPACT TESTS CONDUCTED ON 50-TON CLASS SPENT FUEL TRANSPORTING CONTAINERS

T. Nagakura, H. Takano, K. Sato, Y. Maeno, H. Shiojiri, N. Kishi, and S. Aoyagi
Central Research Institute of Electric Power Industry

1. Preface

Commissioned by the Science and Technology Agency, the Central Research Institute of Electric Power Industry has been conducting the reliability demonstration test on the spent fuel transporting containers (hereinafter referred to as casks) since 1977. This reliability demonstration test consists a series of tests to demonstrate the soundness of the cask against various conceivable accidents to cask in transit and is scheduled to be conducted according to the regulations for transportation formulated by the IAEA and the Government of Japan. The test program contains; 1) Heat transfer test, 2) Shielding test, 3) Immersion test, 4) Drop impact test, 5) Fire-proof test, and 6) Pressure-proof test.

This reports describes the results of the drop-impact tests conducted in March 1980 as part of the reliability demonstration test by dropping 50-ton class casks arranged in horizontal and vertical postures from a height of 9 meters. The purpose of this test are: 1) To verify the validity of the design procedures regarding the impact load of the cask, 2) To establish the procedures for the numerical simulation to analyze the drop-impact behavior of the cask by checking the result of numerical analysis against the results of test, and 3) To estimate the drop-impact behavior of 100-ton class casks whose test is scheduled to be conducted in the near future.

2. Specimen Cask

The fundamental design concept of the cask to be tested is that it should meet the regulations of IAEA and the Government of Japan and that the result of its demonstration test should be useful for the reliability assessment of several cask types and its structure should be as general as possible. With this design concept in mind the following design conditions were adopted:

(i) The total weight of the cask should be about 50 ton.
(ii) The cask should be natural cooling type and accommodate as many fuel assemblies as possible.
(iii) It should be of 3-layer type (stainless steel-lead-stainless steel).
(iv) The lid should be airtight with using high strength bolts with O ring packing.
(v) The shock absorber should be wooden.
(vi) It should resist a hydraulic pressure of 300 kg/cm^2 exerted by sea water, which may be caused by accidents during its transportation by sea. This is a design condition peculiar to Japan.

After comparing with one another those casks proposed in conformance with the above conditions by five co-operating firms in Japan, the specimen cask was designed and manufactured and it can contain 3 PWR fuel assemblies or 7 BWR fuel assemblies.

The structural concept of the cask is shown in the schematic diagram of Fig. 1.

3. Test

The test was conducted by using the drop-impact testing apparatus installed in the Takeyama Test and Research Center in Kanagawa Prefecture. Figure 2 shows the bird's-eye view of the Takeyama Test and Research Center.

3.1 Drop-impact testing apparatus and measuring apparatus

3.1.1 Drop-impact testing apparatus

The drop-impact testing apparatus, which provides the function of dropping a 100-ton weight class cask arranged in any desired posture from a height of 9 meters and also the function of dropping it from a height of greater than 1 meter onto round mild-steel bars arranged perpendicularly on a drop testing platform, consists of the following devices; (1) a travelling crane (capacity: 120 tons; span 23 m; lift 20 m above testing platform); (2) a drop testing platform (reinforced concrete block: 12 x 12 x 4 m; 2 steel plates each 7 m x 7 m x 100 mm in size); and (3) Appurtenances.

3.1.2 Measuring apparatus and data processing device for drop-impact test

The major measuring apparatuses for drop-impact test are as listed below.

(1) Measuring apparatus for dynamic strain and acceleration (47 ch).

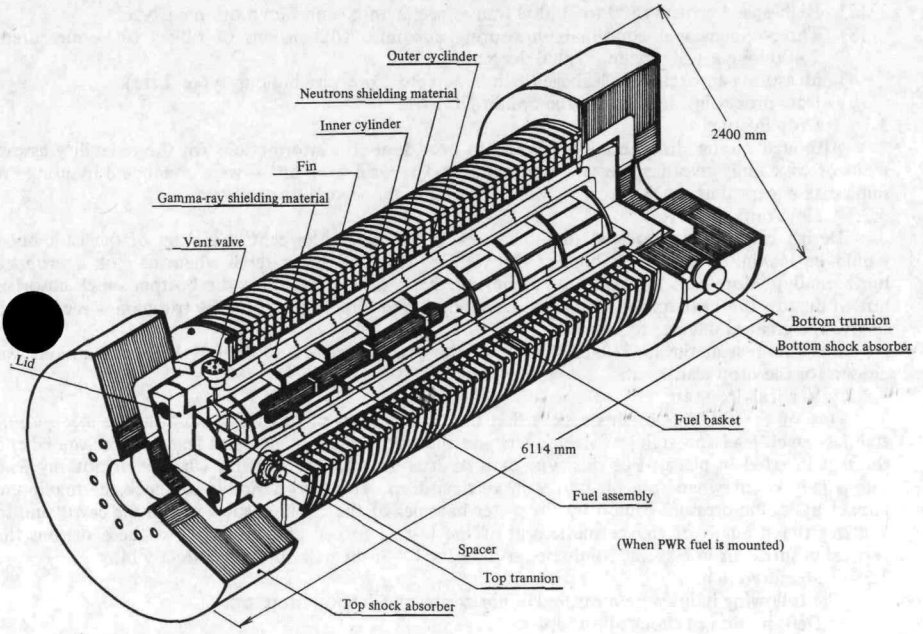
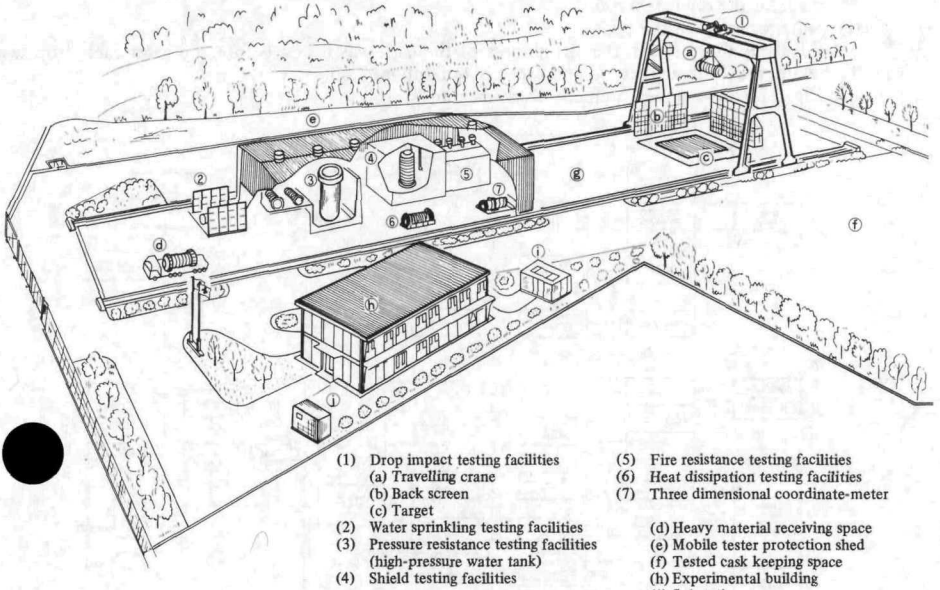


Fig. 1 Illustration of 50-ton class cask for demonstration test



(1) Drop impact testing facilities
(a) Travelling crane
(b) Back screen
(c) Target
(2) Water sprinkling testing facilities
(3) Pressure resistance testing facilities (high-pressure water tank)
(4) Shield testing facilities
(5) Fire resistance testing facilities
(6) Heat dissipation testing facilities
(7) Three dimensional coordinate-meter
(d) Heavy material receiving space
(e) Mobile tester protection shed
(f) Tested cask keeping space
(h) Experimental building
(i) Substation
(j) Boiler room

Fig. 2 Bird's-eye view of spent nuclear fuel transport containers' reliability demonstration test yeard

(2) High-speed camera (500 to 10,000 frames/sec; 2 units) and film motion analyzer.
(3) Three-dimensional coordinate measuring apparatus (dimensions of object to be measured: 2,400 long x 1,500 wide x 7,000 deep mm; 1 unit).
(4) Measuring apparatus for leakage (helium detector: pressure buildup type; 1 set).
(5) Data processing device (mini-computer NOVA).

3.2 Drop Posture

Although corner drop test is expected to provide useful informations for the reliability assessment of cask, only two drop tests – horizontal and vertical drop test – were conducted by using the same cask except that shock absorbers were renewed in the second, vertical test.

3.2.1 Horizontal drop test

Design calculation suggested that the stress produced at the central section of the cask body would be maximum and be as high as the yielding stress of the material when the cask is dropped horizontally. Moreover, the bottom trunnion of this cask is attached to the bottom shock absorber not to disturb the heat transfer from the cask body, and direct impact on the trunnion was expected to give the greatest damage to the cask.

For this reason the horizontal posture of the cask, which would directly hit the trunnion, was chosen for the drop test.

3.2.2 Vertical drop test

One of the features of this cask is that the cask body is of three-layered structure made up of stainless steel, lead and stainless steel layers and the lead layer is made up of lead rings machined and thermal inserted in place. For this type of cask, it is necessary to confirm whether or not any lead slump may occur when subjected to 9 m vertical drop. It is also desirable to check the maximum impact hydraulic pressure caused by the water hammer of the cooling water within the cavity and to confirm the strength of the hermetic seal of the O-ring for lid and valves. For these reasons the vertical posutred drop test was conducted so that the lid of the cask might be directly hit.

3.3 Measured items

The following items were measured in horizontal and vertical drop tests.

(i) Deformation of shock absorbers.
(ii) Acceleration and dynamic strain of the cask body and the basket.
(iii) Acceleration of the imitation-fuel rod which has the same weight as a fuel assembly but is made of the mild steel rod.
(iv) Water pressure in the cavity.

Besides the above items, the airtightness of the cask was tested before and after each drop test by the helium pressurizing method and pressure buildup method.

In Fig. 3 is shown a measurement block-diagram.

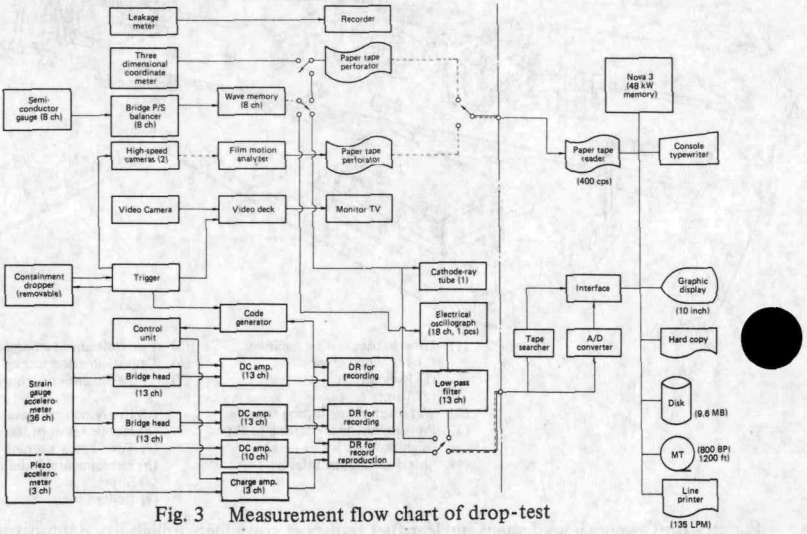


Fig. 3 Measurement flow chart of drop-test

4. Results of 9-Meter Drop-Impact Test

The results of the 9-meter drop-impact tests conducted with the cask horizontally and vertically postured are as summarized below. Figure 4 shows the positions at which the acceleration and dynamic strain of the cask were measured.

4.1 9-Meter Drop-Impact Test of Horizontally Postured Cask

1) Residual deformation of shock absorbers

Since the bottom shock absorber is provided with a trunnion, its residual deformation caused by the drop-impact was different from that of the top shock absorber. After the trunnion reached the target the top side part of the cask came to be accelerated by the moment of rotation until the trunnion which acted as a support sank completely into the bottom shock absorber.

As a result the deformation of the top shock absorber was greater than that of the bottom one, the fomer being 192 mm and the latter 117 mm. The average value of the two was approximately equal to the design value.

2) Maximum acceleration of the cask body

Figure 5 shows the variations with time of the vertical accelerations of the outer cylinder of the cask. Magnitude of the maximum acceleration and the time at which the acceleration reached its maximum value were somewhat different from point to point of the cask. The average of the maximum accelerations at each point on the cask body, when the time lag is ignored, was about 67 G and the mean value of the accelerations of each point at the time when maximum strain was produced in the cask was about 41 G. This difference is considered to orginate in the fact that the bottom shock absorber was provided with a trunnion and as a result its behavior at the time of impact was different from that of the top shock absorber. Each of the above test values was smaller than the design value.

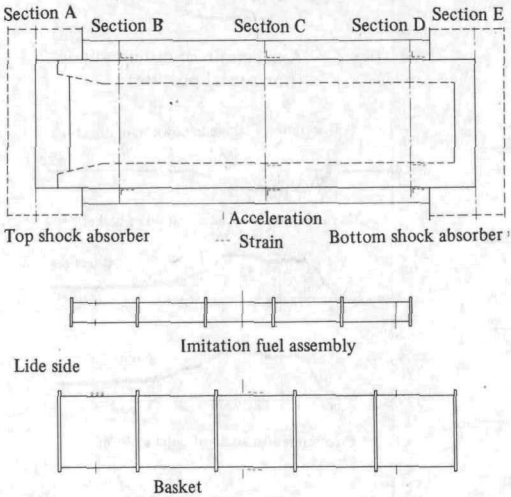


Fig. 4 Measuring positions for acceleration and strain of cask body, imitation fuel rod, and basket

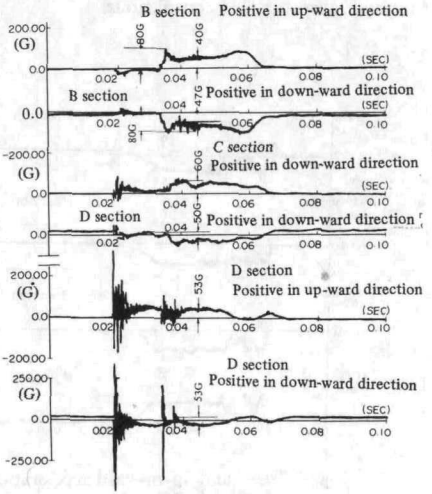


Fig. 5 Up- and down-ward acceration (horizontal posture)

3) Acceleration of basket and fuel rod

The design value of the acceleration of the basket and that of the fuel rod were both 100 G, which proved larger than the test values of them. The maximum test value of the former was about 56 G and that of the latter, about 73 G.

4) Maximum stress developed in the cask body

Figure 6 shows the variation with time of the axial strain developed in the outer cylinder of of the cask. The variation with time of the strain of the upper and lower edges in section C was analogous to the behavior of a beam's imapct bending,amounting to a maximum value of 960 μ.

As for the stress computed from the measured value of strain, the test value of the axial stress was 21.0 kg/mm² as against the design value of 30.9 kg/mm², the former representing about 70 % of the latter.

On the other hand, the circumferential stress was insignificant.

5) Internal impact hydraulic pressure

The test value of the internal impact hydraulic pressure except for the high-frequency component approximated to the design value.

The cask conditions after the drop tests are as shown in the photos bellow (horizontal posture).

4.2 9-Meter Drop-impact Test of Vertically Postured Cask

1) Residual deformation of shock absorbers

The test value was smaller than the design value. The deformation behavior of the shock absorber exhibited in the test was different from that considered at the design time and the residual deformation caused by the cask body, which penetrated the shock absorber, was conspicuous.

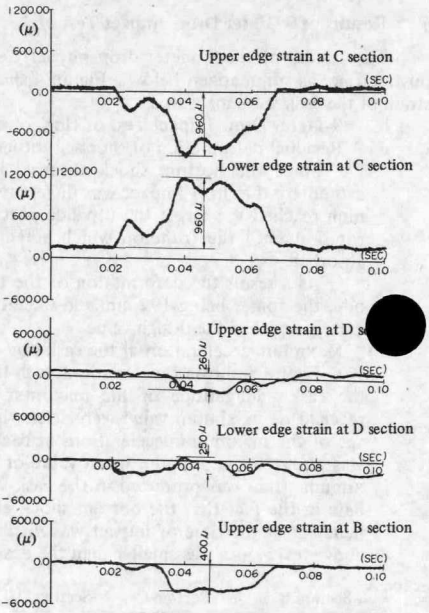


Fig. 6 Axial strain of outer cylinder (horizontal posture)

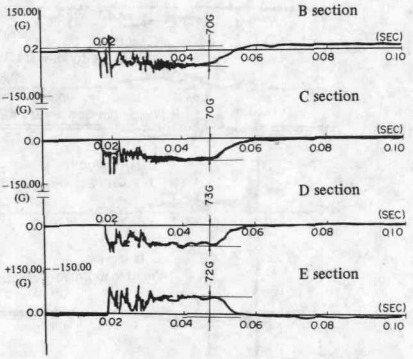


Fig. 7 Up- and down-ward acceleration (vertical posture)

2) Maximum acceleration of cask body

The variations with time of the accelerations measured at various points on the cask body were approximately analogous to one another and the mean value of the maximum values was about 70 G. Also, the mean value of the acceleration of the various points at the time of maximum strain was about 68 G. These two mean values were smaller than the design value of 85 G.

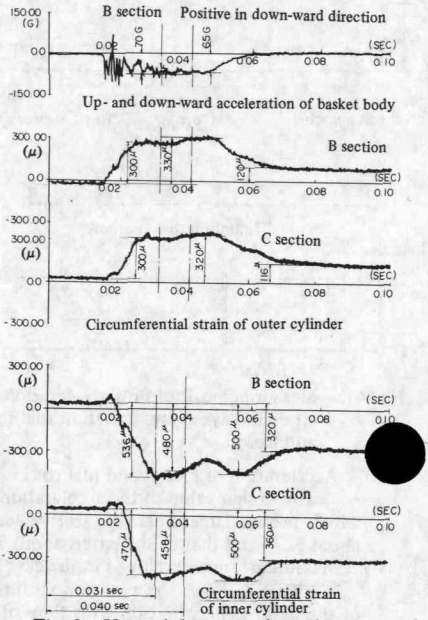


Fig. 8 Up-and down-ward acceleration and axial strain (horizontal posture)

Since the behavior of the residual deformation of the shock absorber is different from that considered at the design time it is difficult to accurately relate the residual deformation to the maximum acceleration. It is known, however, from the analysis of the high-speed camera films and other data that the deformation (including elastic deformation) of the shock absorber due to the initial impact was about 190 mm (equal to the design residual deformation) and in the case of the vertically-postured cask drop the amount of energy absorbed by elastic deformation and fracture of absorber's steel cover was cmparatively large.

This is the reason why the maximum acceleration was smaller than the resign G-value although the residual deformation of the shock absorber was only about 70 % of the design value.

3) *Acceleration of basket and fuel rod*

The design value of the acceleration of the basket and that of the fuel rod were both 100 G, which proved larger than the test values of them. The test value of the acceleration of the basket was about 87 G and the of the fuel rod, about 70 G.

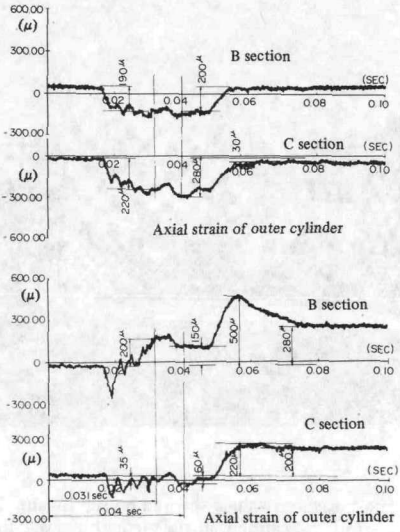


Fig. 9 Axial strain (vertical posture)

4) *Maximum stress developed in cask body*

Figures 7 and 8 illustrate the variations with time of the axial and circumferential strains measured on the cask body. In the case of the circumferential stress the design value for the outer cylinder was 25 kg/mm^2 and that for the inner cylinder, -37 kg/mm^2, but the test value was 6 kg/mm^2 for the former and -11 kg/mm^2 for the latter. On the other hand both the design and test values of the axial strain were insignificant as compared with those of the circumferential strain.

5) Circumferential residual stress

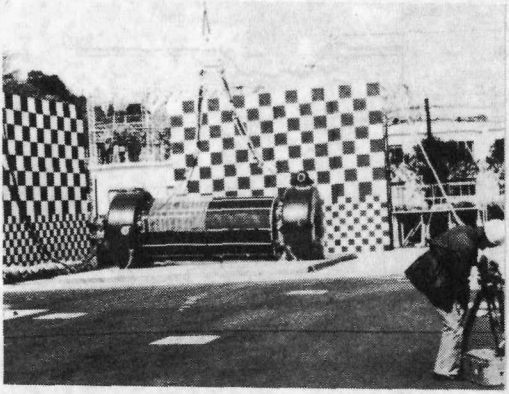
In the case of the inner cylinder the axial residual stress was 4 to 6 kg/mm^2 and the circumferential one was 6 to 7 kg/mm^2. These residual stresses are considered to be due to the lead slump. The amount of lead slumps was computed to be 22.3 mm at the design time, but it was infeasible in the recent test to quantitatively determine the amount of lead slumps. To establish the measuring method of the amount of slump is a future problem.

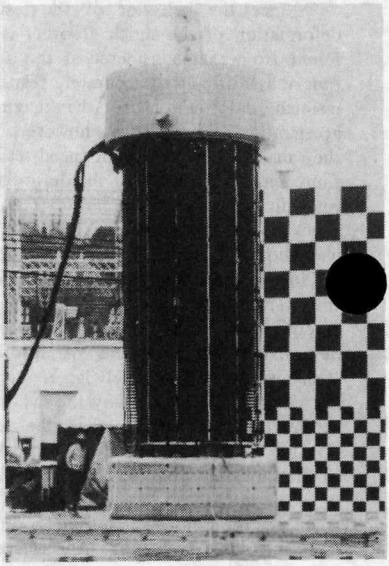
6) Internal impact hydraulic pressure

The test value of the internal impact hydraulic pressure except for the high-frequency component approximated to the design value. The cask conditions after the drop tests are as shown in the photos bellow (vertical posture).

4.3 Airtightness

After conducting the drop impact test, the airtight section of the cask body was checked for disturbance of airtightness by two methods, viz. The helium pressurizing method and the pressure increasing method.





The measured values showed no significant changes before and after the test and were lower than the criterion, indicating that the airtightness was thoroughly maintained (see Table 1).

1) The results obtained from the testing methods, i.e. the helium pressurizing method and the pressure increasing method, showed no significant changes before and after the test and no damage was detected in the airtight section.
2) Possibility of instantaneous discharge of the cooling water (an aqueous solution of copper sulfate with a strength 50 ppm) from the O-ring section of the lid at the time of drop-impact was checked by measurement using the Cu detecting method. Results indicated that there was no discharge of the cooling water.

Table 1 Comparison of design values and test values

Dropping posture / Item of comparison		9m drop in horizontal posture		9m drop in vertical posture		Remarks
		Design value	Test value	Design value	Test value	
Residual deformation of shock absorber		144 mm	192 mm 117 mm*	190 mm	131 mm	* Amount of deformation of shock absorber having trunnion.
Maximum acceleration	1) Cask body	81 G*-1	67 G*-1 41 G*-2	85 G*-1	70 G*-1 68 G*-2	*-1: Average of maximum values for each point. *-2: When maximum response strain occurs.
	2) Basket	100 G	56 G	100 G	87 G	
	3) Fuel rod	100 G	73 G	100 G	70 G	* Imitation fuel assembly.
Maximum stress of cask body	1) Axial stress	Lower edge of outer cylinder 30.9 kg/mm²	+21 kg/mm²	Outer cylinder −8 kg/mm² Inner cylinder −3 kg/mm²	−4 kg/mm² −5 kg/mm²	+ (plus sign): tensile stress; − (minus sign): compressive stress; In design calculation a slump of 22.3 mm was taken into consideration.
	2) Circumferential stress	Lower edge of outer cylinder ———	+ 3 kg/mm²	Outer cylinder 25 kg/mm² Inner cylinder −37 kg/mm²	6 kg/mm² −11 kg/mm²	
Internal hydraulic pressure upon impact		6 kg/cm²	7 kg/cm²	43 kg/cm²	48 kg/cm²	

5. Comparison Between Design Value and Test Value

Table 1 lists the comparison between the design value and the test value as to the residual deformation, maximum acceleration, and the maximum stress developed in the cask body. The test values were all smaller than the design values. Table 2 lists the results of tests on the airtightness for leakage. The measured values before and after the test met the criterion and no significant variation was noted.

Table 2 Results of leakage test after drop-impact test

Item	Measuring point	After drop test of cask in horizontal position	After drop test of cask in vertical position
Helium pressurizing method (atm. cc/sec)	Lid sealing section	5.1×10^{-6}	1.6×10^{-6}
	Vent valve section	1.0×10^{-6}	4.3×10^{-7}
	Drain valve section	8.7×10^{-7}	1.1×10^{-6}
	Socket section	4.9×10^{-8}	1.4×10^{-7}
Vaccum method (atm. cc/sec)	Lid sealing section	1.9×10^{-4}	3.2×10^{-4}
	Vent valve section	5.4×10^{-4}	3.1×10^{-4}
	Drain valve section	5.6×10^{-4}	2.4×10^{-4}
	Socket section	2.6×10^{-4}	2.8×10^{-4}
Criterion (atm. cc/sec)		1.2×10^{-3}	1.2×10^{-3}

6. Numerical Analysis for the Drop-Impact Test

Before conducting the 9-meter drop-impact test of the 50-ton cask, an analysis was made to predict the behavior of the cask subjected to the drop-impact test and the results thereof. This was done to facilitate drawing-up of the specifications for the measuring apparatus, and to aid in selection of measuring points. Also a post-facto analysis of the test was done by examining the test results to verify the applicability of the analytical procedures, and to develop procedures for making a numerical simulation of the drop-impact behavior of the cask.

The analytical procedure employed in this analysis is as follows:

i) The STEALTH computation code based on the finite difference calulus

By analyzing the drop-impact behavior of the shock absorber, the amount of deformation of the shock absorber and the acceleration which the cask body receives are found.

ii) One-dimensional non-linear analysis of the concentrated mass-spring system

By converting the entire structural system of the shock absorber, cask body, etc., into a concentrated mass-spring system, and by making a dynamic response analysis by means of a computation model, the time records of the acceleration, strain, etc., are found.

iii) Analysis of stress by the finite element method

By using the acceleration (G value) obtained from the results of the analyses (1) and (2), and the results of the test, a detailed stress analysis is made for the cask body, basket, etc., in order to find the values of maximum stress.

7. Summary

The result of the recently conducted 9-meter drop-impact tests of the 50-ton class horizontally and vertically postured cask showed no conspicous residual deformation of the cask body, and the results of the airtightness leakage test conducted before and after the drop-impact tests proved a satisfactory degree of airtightness. The soundness of the cask was adequately proved by the recent test, in which the cask, both horizontally and vertically postured, was subjected to a 9-meter drops.

DISCUSSION

Question by J. D. Hart, United Kingdom: I have understood that you are using the 50 t drop tests to gain information on the prediction of performance of a 100 t flask. Do you expect to be able to predict the seal leakage from the 100 t flask after a 9 m drop without actually dropping a full size flask?

Answer: The 50 t class cask is not a scale model of the 100 t class cask to be constructed in the near future. The basic design conditions of the 100 t class cask are not fixed yet. The impact and thermal behaviour of the 100 t cask will be predicted by the same method by which the 50 t cask was designed. But we do not expect to be able to predict the seal leakage behaviour of 100 t class by 50 t cask test results because the scale effect of seal leakage is not clear.

POSTER SESSION B

MOCK-UP TEST ON PLUTONIUM NITRATE LIQUID PACKAGING

Taizo Iwasaki
Power Reactor and Nuclear Fuel Development Corporation

Nagao Niomura, Akira Qnodera
Nobuyoshi Mori, Masaharu Furudera
Ryoji Asano, Tomio Ebisu
Ichiro Kawahara
Hitachi Shipbuilding and Engineering Co., Ltd.

SUMMARY

The plutonium nitrate liquid shipping packaging, PUL-4 was designed and developed for on-site shipment of 4 liters of $Pu(NO_3)_4$ liquid with a maximum decay heat of 10W, and physically tested to demonstrate that it meets the criteria specified in Japanese Regulations which are equivalent to the IAEA Regulations.

The packaging is approximately 1,500 mm in height and 800 mm in diameter and consists of containment vessel and outer shell. The weight of the packaging is about 600 kg. The main features of the packaging are double containment system and balsawood for the impact energy absorption.

Mechanical tests were performed to the criteria except water spray, compression and water immersion tests. Deceleration, deformation and stress were measured and compared with the analytical results. The conclusion was that both deceleration and stress measured were fairly close to the analytical values but deformation measured was as twice as that estimated.

As to the thermal test, open fire test and furnace test (method II) were carried out on the Packagings after the mechanical tests. The severity of the two kinds of the test method was compared and evaluated by the data obtained and found that the furnace test imposed severer condition to the packaging than the open fire test.

The integrity of the packaging was examined by helium leak test and the packaging was demonstrated to survive the test conditions.

DESCRIPTION OF TEST PACKAGING

The packaging is designed to transport 4 liters of $Pu(NO_3)_4$ liquid with a maximum decay heat of 10W. The package is qualified as equivalent to Type BM class II.

Figure 1 shows the outline of the package. It is approximately 800 mm in diameter, 1,500 mm in height and 600 kg in weight and consists of a containment vessel and outer shell. The containment vessel consists of primary vessel and

secondary vessel, and was designed to the criteria of ASME Sec. III class 1 component. The design conditions and service limit are shown in Table 1. The pressure and impact force were determined to withstand the test conditions and radiolytic decomposition. The primary vessel is 807 mm in height and 150.9 mm in inner diameter as shown in Figure 2, and contains the $Pu(NO_3)_4$ liquid bottle in the cavity. Double O-ring system is used between the flange and lid, on which sampling valve is provided. The secondary vessel consists of two cylindrical shells. It is 1,020 mm in height and 315 mm in outer diameter, as shown in Figure 3. The annulus between the two cylindrical shells is filled with water extended polyester (WEP) for neutron shielding. Double O-ring system is used between the flange and lid. Both primary and secondary vessels are made of stainless steel Type 304L.

Table 1 Design Condition and Service Limit

Test Condition	Load	Service Limit by ASME Sec.III(NB-3220)
Normal	Internal Pressure . primary vessel: 9.0 kg/cm^2 . secondary vessel: 2.1 kg/cm^2 Inertia Load: max. 235G Temperature . primary vessel: 60°C . secondary vessel: 53°C	Level A (NB-3222)
Accident	Internal Pressure . primary vessel: 80 kg/cm^2 . secondary vessel: 80 kg/cm^2 Inertia Load: max. 320G Temperature . primary vessel: 63°C . secondary vessel: 59°C	Level B (NB-3224)

The outer shell is made of stainless steel Type 304. Approximately 200 mm thick balsawood is packed into the annulus between the secondary vessel and outer shell for impact energy absorption and fire resistance.

MOCK-UP TEST

Four packagings were tested as shown in Table 2. No.1 packaging was subjected to the normal 1.2 m drop and penetration tests in 3 directions, that is, bottom vertical, top corner and horizontal as it had been estimated that previous test results would not affect the other part of the packaging.
No.2 packaging was subjected to the accidental 9 m drop and puncture tests of bottom vertical and top corner directions on the same reason. No.3 packaging was subjected to the horizontal 9 m drop and puncture tests and furnace test. No.4 packaging was subjected to the normal and accidental mechanical tests of horizontal direction and then open fire test.

Helium leak test was carried out before and after the test to ascertain the integrity of the packaging.

Table 2 Test Items for Mock-up Test

test \ packaging	No.1	No.2	No.3	No.4
mechanical test				
-1.2 m free drop	*			*
-penetration	*			*
-9 m free drop		*	*	*
-1 m puncture		*	*	*
thermal test			*	*
leak test	*	*	*	*

* shows the tests carried out.

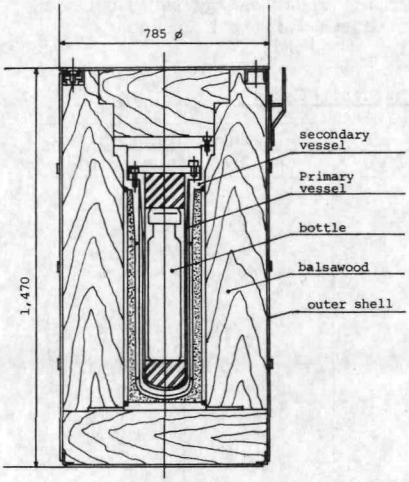


Figure 1 PUL-4 Package

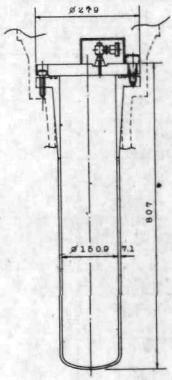


Figure 2 Primary Vessel

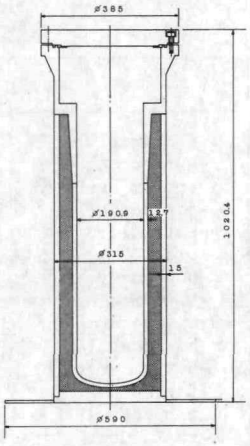


Figure 3 Secondary Vessel

RESULTS AND CONSIDERATIONS

Mechanical Test

(1) Normal Condition

Photo 1 shows the result of 1.2 m corner drop test. About 10 mm height of buckling wave was induced at the outer shell corner. Vertical and horizontal 1.2 m drop tests were also carried out and it was found that outer shells were deformed slightly but not damaged. Maximum deformation and deceleration at 1.2 m drop tests was 33 mm and 188 G respectively.

Photo 2 shows the result of penetration test on the cylindrical part of the outer shell. The outer shell was not penetrated and the depth of the indented recess was about 5 mm.

(2) Accident Condition

Photos 3 and 4 show the results of bottom vertical and top corner 9 m drop tests. At the top corner drop, the outer shell corner buckled and that deformation was 113 mm. The outer shell also buckled at the bottom virtical drop in half wave form, about 130 mm above the bottom plate throughout the perimeter and its height was about 12 mm. Photo 5 shows the result of the horizontal 9 m drop test. The maximum deformation and deceleration at the drop were 113 mm and 287 G.

Photo 6 shows the result of 1 m horizontal puncture test. The deformat[redacted] was 32 mm and no damage was occured on the outer shell. Maximum deformation 39 mm and maximum deceleration of 78G were obtained at top corner puncture.

(3) Comparison

Results of the mechanical test were shown in Table 3 with the analytical results, which were obtained on the assumption that impact energy is absorbed by the plastic deformation of the outer shell and collapse of the balsawood.

Table 3 Results of Mechanical Test

	deceleration (G)		deformation (mm)	
	measured	calculated	measured	calculated
1.2 m free drop				
-horizontal	188	235	7	25
-bottom vertical	-	158	4	15
-top corner	-	129	33	34
penetration				
-side	-	-	3.5	6.1
-top cover	-	-	2.0	5.1
-bottom plate	-	-	2.5	6.1
9 m free drop				
-horizontal	287	330	25	48
-bottom vertical	176	158	28	57
-top corner	146	158	113	42
1 m puncture				
-horizontal	43	48	32	27
-bottom vertical	58	48	20	21
-top corner	78	115	39	30

Measured and calculated values were compared and it was found that the decelerations were in good agreement but the deformations measured were less than a half of those calculated. Difference in deformations is attributed to the f[redacted]ts that only external deformation was measured, resulting in the neglect of int[redacted] deformation, and no elastic deformation was included in the measured results.

Stress analysis was carried out on the assumption that the inertia caused by the deceleration was imposed to the containment vessel. Stress obtained from the 9m drop test was compared with that calculated and it was found that those stress distributions were in good agreement as shown in Figure 4.

Impact of bottom vertical 9 m drop was analyzed using a finite difference equation continum mechanics computer program, PISCES. In the analytical model, containment vessel and target were considered as regid body and expressed balsa-wood with Eular ordinate, outer shell with Lagrange ordinate and impact speed of 13.3 m/sec was given for initial condition. The result is shown in Figure 5, the outer shell buckled in half wave form, about 150 mm above the bottom plate which corresponds fairly to the test result.

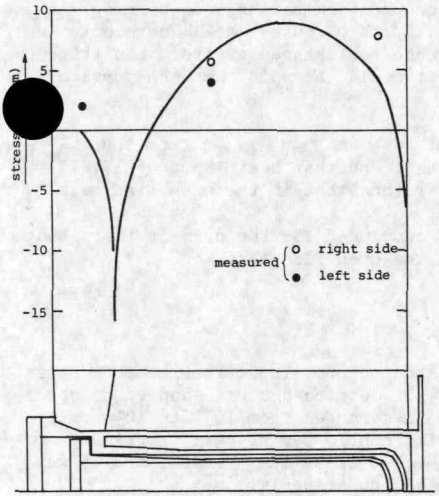


Figure 4 Axial Stress of 9m Drop

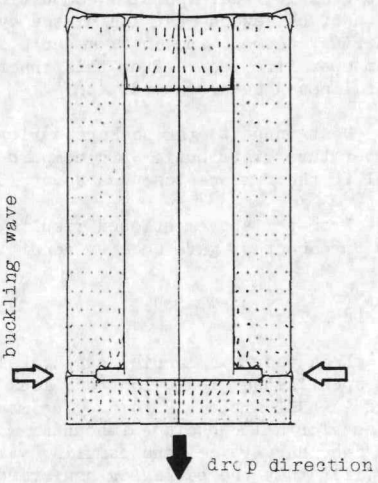


Figure 5 9 m Vertical Drop (Analytical Result)

Thermal Test

(1) Test procedure

Open fire test was performed on an oil burnpit of 2,300 mm long by 1,700 mm wide and 450 mm high and surrounded by water jacket set in the field. About 500 liter of oil with gross heating value of 11,000 Kcal/kg was used for the test. Photo 7 shows the open fire test. Furnace test was conducted in the heat-treatment furnace of 5,000 mm long by 3,000 mm wide and 3,000 mm high, which was preheated to 900°C, 70°C above the preheat temperature recommended. In spite of the preheat, as the temperature decreased below 800°C when the door of the furnace was opened to set into the package, 30 minutes of test time was counted just after the temperature of the furnace rose above 800°C. Photo 8 shows the furnace test.

(2) Result and Evaluation

Figures 6 and 7 show the temperature during open fire test and furnace test The ambient temperature of the furnace test ranged from 800°C to 850°C, on the other hand that of the open fire test changed rapidly and maximum temperature difference amounted to 250°C. It was very hard to keep the test conditions constant because of the wind and such condition should be regarded as inherent to open fire test.

Comparing both temperatures, even though higher temperatures were obtained at some parts in the open fire test, it was found that higher temperatures were kept around the package in the furnace test than in the open fire test. For this reason, it was found that the furnace test imposed severer condition.

The charring of balsawood after the thermal tests were compared to judge which gave severer test condition. Figure 8 shows the results of charring after thermal tests. In both tests, though charring occured around the closure head most greatly, where partial combustion also occured, comparing the charring depth of balsawood just inside the cylindrical part of outer shell where only charring occured, the charred depth from furnace test ranged greater than that from open fire test. From this observation it could be said that the furnace test imposed severer condition.

Heat input to the package during thermal test was estimated from the temperature distribution and compared. It was found that heat input of 17,000 Kcal in the furnace test was about 10% greater than that in the open fire test.

From the abovementioned results, it was concluded for the present tests that the furnace test gave severer condition than the open fire test.

3. Leak Test

Leak test was carried out by helium leak detector. The criteria of the test was specified as leak tight and leak rate of 10^{-7} atm.cc/sec was adopted as specified in ANSI Article N 14.5. The measured values ranged from 10^{-9} to 10^{-10} atm. cc/sec for all containment boundaries, that is, double O-ring part of primary and secondary vessel and sampling valve on the lid of primary vessel, and it was verified that the packaging conformed to the design criteria.

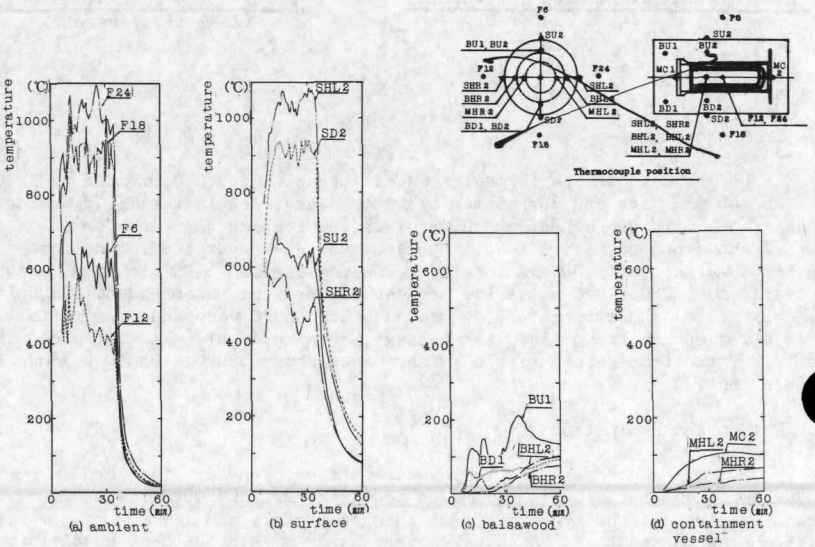


Figure 6 Temperature Change during Open Fire Test

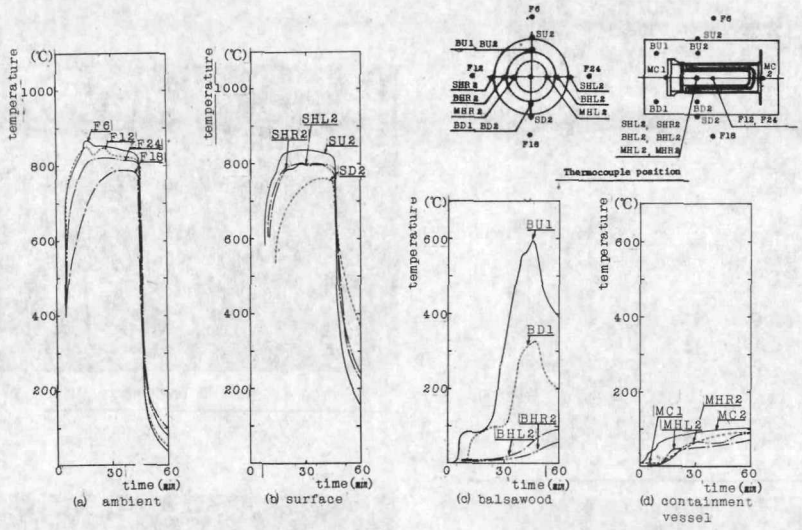


Figure 7 Temperature Change during Furnace Test

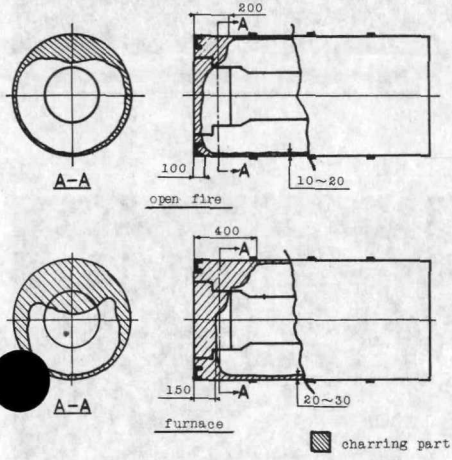


Figure 8 Charring of Balsawood

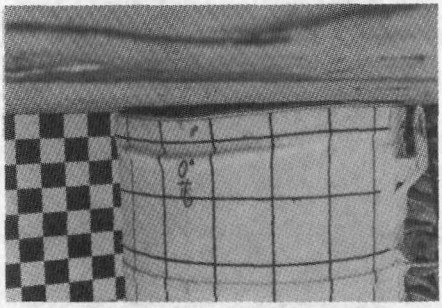


Photo 1 1.2 m Corner Drop Test

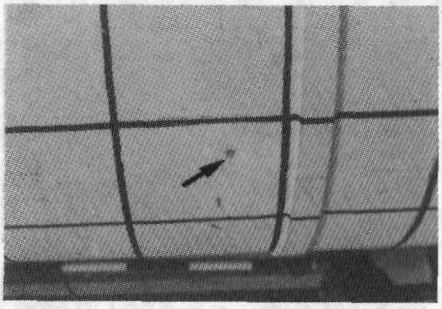


Photo 2 Penetration Test

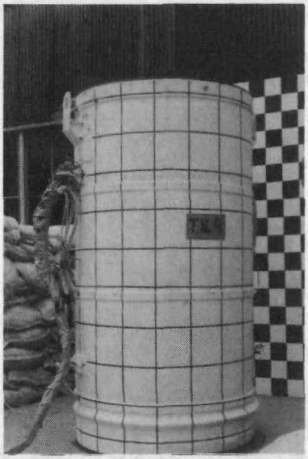


Photo 3 9 m Bottom Vertical Drop

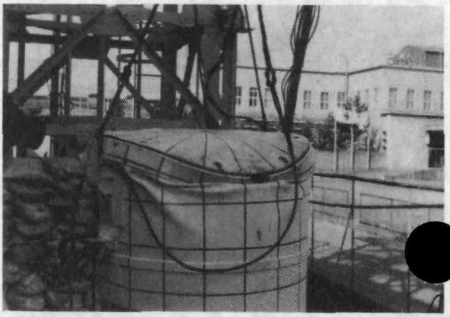


Photo 4 9 m Top Corner Drop

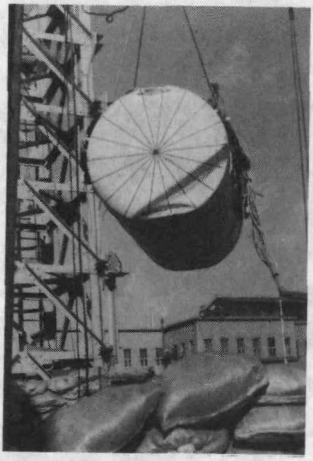


Photo 5 9 m Horizontal Drop

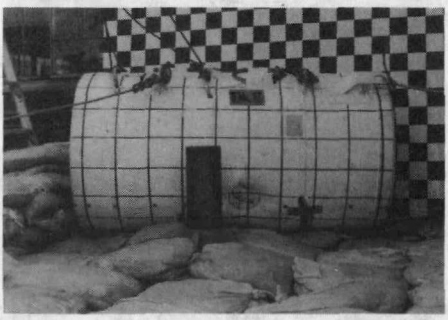


Photo 6 1 m Horizontal Puncture

Photo 7 Open Fire Test

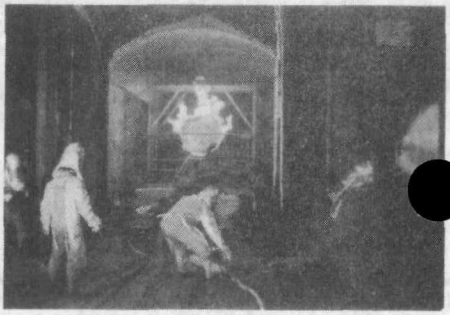


Photo 8 Furnace Test

IMPACT RESISTANCE OF SHIPPING CONTAINERS FOR RADIOACTIVE MATERIALS UNDER FREE FALL TEST

Tsutomu SASAKI, Yoshioki KANAE & Shoji SHIMAMURA
MECHANICAL ENGINEERING LABORATORY,
Agency of Industrial Science and Technology, MITI, JAPAN

Abstract

The object of the present study is mainly two. The first is to investigate an applicability of so-called similitude law in estimating behaviors of a prototype cask under a drop test using test results with a series of scale models. The second is to elaborate testing techniques such as the measurement of deceleration, strains and residual deformations under the test, respectively.
The prototype cask is of three layered sandwich construction weighing approx. 80 ton, 5.8 m long and 1.6 m in diameter. 1/8 and 1/4 scale models of the prototype were fabricated and tested under 9 m drop according to IAEA requirements, where tests were carried out without impact limiters so as to observe behavior of the cask at the most severe condition. Horizontal, vertical and corner drop tests with these models were performed. Drop test of 1/3 models is scheduled in the middle of 1981.
Decelerations at several points (outer and inner) of each cask were determined and residual deformations were measured after the test. The motion of the casks were recorded by a high-speed cinematography.
An analysis of data obtained in the tests suggest an applicability of the similitude law in estimating behavior of the prototype cask under the free fall test from the model test data when careful considerations are made in the design and the fabrication of the model and also in the techniques of the drop test.

1. INTRODUCTION

The present IAEA Regulation requires that any container (cask) for shipping nuclear fuels shall keep the integrity after 9 m drop onto the solid and smooth target. Practical procedures usually employed to examine its integrity of a cask in this category are as follows.
(1) Drop test of the cask itself
This is of course an ideal method. However, when the cask to be examined is larger one such as for shipping LWR spent fuels, the test may be too costly to perform.
(2) Structural analysis with an aid of computer
In order to clarify the structural integrity under 9 m drop at various falling conditions, a relatively large computer system with an elaborated code shall be used. However, suitable and world-wide approved computer codes for this purpose are now in a developing stage.
(3) Drop test of a scale model of the real cask followed by an estimation of the behavior of the prototype with an aid of so-called similitude law
This is much more economical than the method in (1) and also needs quite less numerical work than (2). But in this case, careful attention shall be paid in

the design and the fabrication of the scale model (especially important is the minimum scale ratio) and also in the determination of deceleration, strain and residual deformation due to 9 m drop, respectively. Internationally approved guide for the scale model testing and the interpretation of the result for 9 m free fall test is not yet made.

(4) A combination of the scale model test and the computational analysis

In this case, the scale model test is used as a bench mark one to check the validity of the computation code to be employed. After the reliability of the code is confirmed by comparing with the scale model test data, the structural integrity of the real cask may be examined computationally. At present, this procedure seems to be the most practical one to check the structural integrity of a cask according to the IAEA Regulation. But this is usually somewhat time-consuming and troublesome. The authors have an opinion that this procedure is a transitional way until a reliable and versatile computer code will be complete or detailed technique in the scale model testing and also interpretation of the data is authorized.

In the previous paper [1]~[3], the authors in cooperation with their colleagues presented a certain guide of the scale model testing of spent fuel shipping containers under 9 m drop. The present paper at first aims to show a practical guide line of the scale model test and second to elaborate various testing techniques employed in this test.

2. TEST

2.1 Cask Model

The prototype cask is of three-layered sandwich construction as shown in Fig. 1. The outer layer is made of carbon steel, the intermediate is lead shielding layer and the inner is of stainless steel, respectively. A plug-type lid is mounted with a set of bolts. 1/8, 1/4 and 1/3 scale models of the prototype were designed and made. Weight, overall dimension and fastening bolts of the prototype and three scale models are shown in Table 1, respectively. As found in Table 1, correct scaling down of the bolt size was impracticable in the 1/8 model due to unavoidable interference of the wrenching device with the head of bolt.

Table 1. Design of cask models (Fig. 1)

Scale	Weight	D	d	L	ℓ	t_1, t_2	t_3	t_4	t_5	Bolt
1/1	80	1,624	1,000	5,824	5,168	56	16	48	240	M64×24
1/3	2.97	541	333	1,941	1,723	18.7	5.3	16	80	M22×24
1/4	1.25	406	250	1,456	1,292	14	4	12	60	M16×24
1/8	0.17	203	125	726	646	7	2	6	30	M10×12

Remark

Weight: in metric tons

D, d, L, ℓ, t_1, t_2, t_3, t_4, t_5: in mm

Bolt: ex., M64×24 means 24 sets of Metric bolt of nominal size (outer diameter) 64 mm are mounted.

2.2 Drop Test Facility

Drop test setup shown in Fig. 2 was used which is same as in the previous experiment [1]. This facility was removed since MEL moved to the new site (Tsukuba) from Tokyo last year. However, the new drop test facility of an improved design will be completed before summer of 1981.

2.3 Instrumentation

To evaluate impact behavior of the cask, followings were measured. Decelerations were determined with some sets of acceleration sensors (piezo type and strain gage type). Fig. 3 shows location of sensors at three cases of drop tests, respectively. Measuring systems employed are shown in Fig. 4, where the wireless method (Transmitter to Receiver System) is under improvement. Deformations after the drop test were coarsely measured as indicated in Fig. 5. Detailed determination of three-dimensional deformation of each cask will be made after the completion of 1/3 scale model test.

2.4 Drop Test

In the previous research [1], the horizontal drop was mainly selected. In the present study, drop tests were made in three cases, horizontal, vertical and corner, respectively (Fig. 3). All tests were carried out without any impact limiters so as to observe the cask at the most severe condition.

3. RESULTS AND ANALYSIS

1/8 and 1/4 scale models have been tested under horizontal, vertical and corner drop, respectively. Fig. 6 shows a scene of the corner drop of 1/4 scale model cask. Drop tests of 1/3 scale models will be carried out in the middle of next year after the establishment of new drop test facility in Tsukuba Research Center (new site of MEL).

3.1 Deceleration

In Fig. 7 are shown entire records of deceleration for 1/8 and 1/4 scale models under horizontal, vertical and corner drop, respectively. Peak values of deceleration in these cases are also summarized in Table 2, where these values were determined with the original (unfiltered) records.

Table 2. Deceleration (Fig. 3)

Case	Location	Deceleration (g)	
		1/8 Model	1/4 Model
Horizontal	Outer shell (near lid)	*	960 (S)
	Outer shell (near bottom)	*	800 (S)
	Inner shell (near lid)	20,000 (P)	10,600 (P)
	Inner shell (near bottom)	——	——
Vertical	Bottom, inner	11,500 (P) 12,350 (P)	7,500 (P)
	Lid, outer	*	1,200 (S)
	Lid, inner	2,850 (P)	*
Corner	Bottom, inner	1,800 (P)	620 (P)
	Lid, outer	*	1,200 (S)
	Lid, inner	1,450 (P)	*

* not measured, —— failed

P: piezo type sensor, S: strain gage type sensor

3.2 Residual deformation
Residual deformations observed after the horizontal, vertical and corner drop tests are shown in Table 3.

Table 3. Deformation of cask after drop test (Fig. 5)

Unit: mm

Case	Location	Model	
		1/8	1/4
Horizontal	ΔD	2.8 ∿ 4.2	6.5 ∿ 11.3
Vertical	ΔL	0.2 ∿ 0.7	0.6 ∿ 0.7
	ΔD	0.7 ∿ 0.9	1.7 ∿ 1.8
Corner	ΔL	10.9	21.6
	ΔD	5.8	13.4

3.3 Cut-off Frequency for Deceleration Records
Deceleration waves are usually superposed with unavoidable high modulation waves as clearly found in those of Fig. 7. In order to determine correct peak values of deceleration, entire records such as those in Fig. 7 shall be modified through suitable low-pass filter. As an example, in Fig. 8 are shown the original record of deceleration at the corner drop with 1/8 scale model, those modified through low-pass filters of 10,000, 5,000, 2,000, 1,000 and 750 Hz, respectively. In this case, LPF approx. 2,000 Hz seems to be preferable. Detailed discussion on this problem will be presented in the next opportunity.

3.4 Effect of Strain Rate Dependency of Material Property to be considered in Scale Model Test
Simple law of similitude is based on an assumption that the material property is independent of the strain rate. However, constituent materials for the usual spent fuel shipping container as treated here change in a certain degree their mechanical properties with speed of impact. Effect of the stress dependency of the materials in the present cask model on the similitude of deformation between the prototype and the scale models was analysed. In this analysis, stress-strain curves of structural steel and lead at different strain rate reported by Malvern [4] and Lindholm [5] were useful. Deformations of casks under the horizontal and the vertical drop were calculated by Shappert's simplified formula, respectively [6]. Results are as follows. In case of the horizontal drop, $\delta_p/\delta_m \times S^*$ are 1.023 and 1.035 for 1/4 and 1/8 model, respectively, where δ_p: deformation of prototype, δ_m: that of model, S*: scale factor (4 or 8 in the present case). In case of the vertical drop, those are 1.034 and 1.051, respectively. To keep the error of estimated deformation of the prototype from its scale model test data less than 5 %, models smaller than 1/4 scale shall not be preferable.

4. CONCLUSION

After the completion of the drop test with 1/3 scale models scheduled to be conducted in the middle of next year using new drop test facility, the authors' recommended guide of the scale model testing will be presented. Here, the authors would like to emphasize again that models larger than 1/4 scale shall be used in order to estimate correctly the behavior of the prototype cask under 9 m drop test.

REFERENCE

[1] S. Shimamura et al., IAEA-SM-147/1
[2] S. Aoki, IAEA SM-147/3
[3] S. Aoki et al., CONF-710801
[4] E. Malvern, Introduction to the Mechanics of a Continuous Medium, (Prentice Hall), 1969, p. 329
[5] U. S. Lindholm, J. Mech. Solids, 12(1964), p. 317
[6] L. B. Shappert, ORNL-TM-2410(1969), p. 48

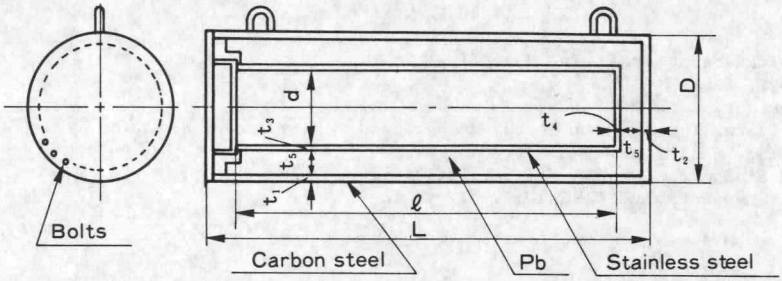


Fig. 1 (upper) Cask model

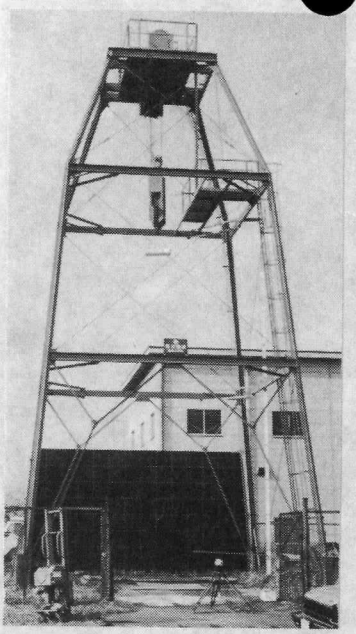
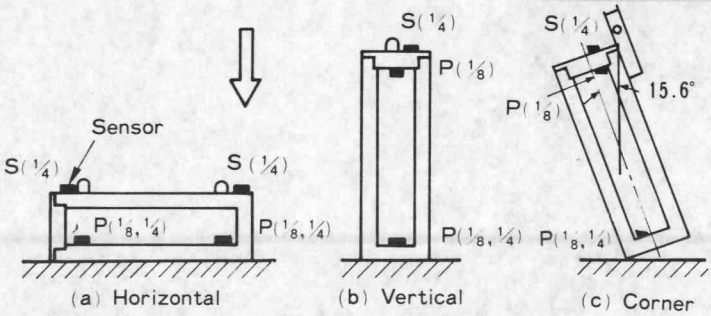


Fig. 2 (middle) Drop test facility

Fig. 3 (lower) Three cases of drop test, a fraction in paranthese shows scale of model.



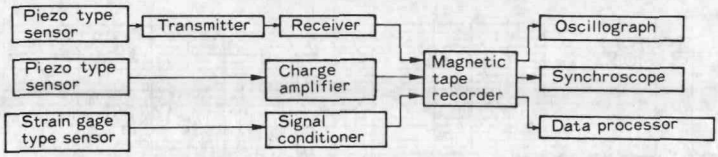


Fig.4 Deceleration measuring systems

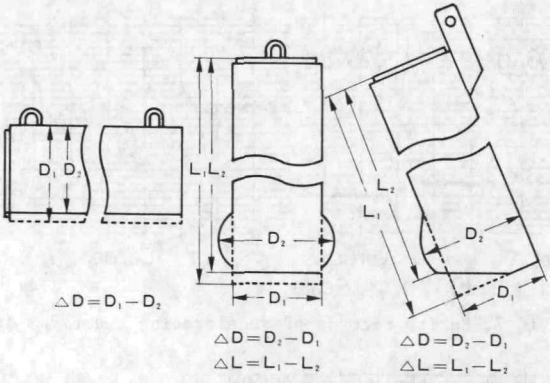


Fig.5 Measurement of dimensional change of cask after drop test

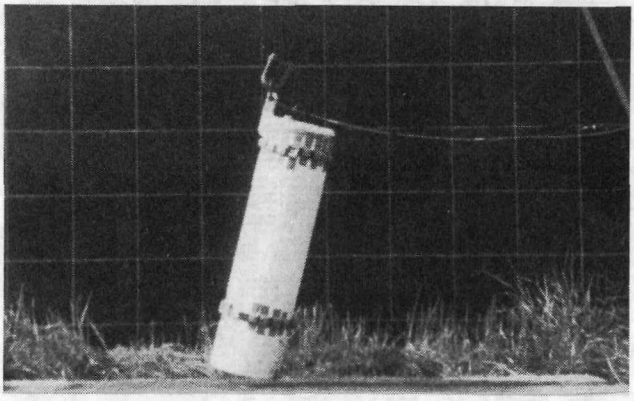


Fig.6 Corner drop of a 1/4 scale cask model

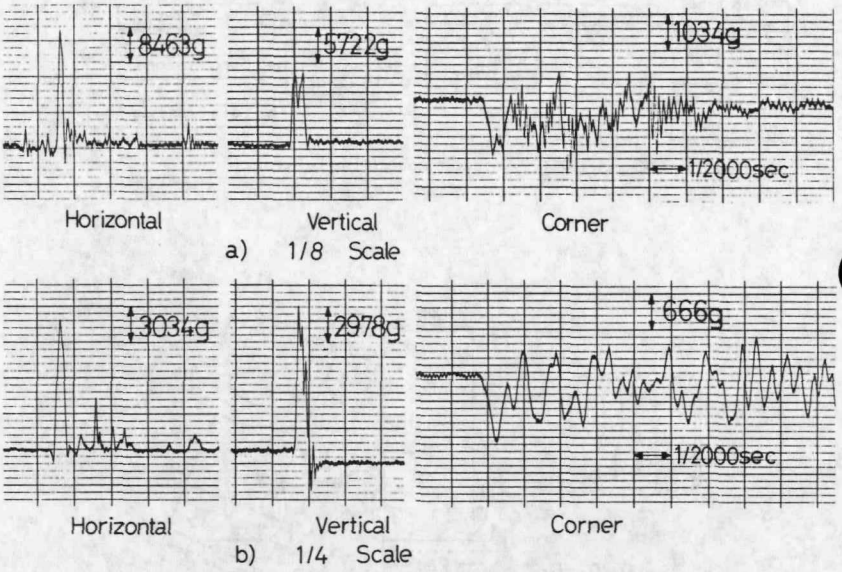


Fig.7 Entire records of deceleration under 9 m drop

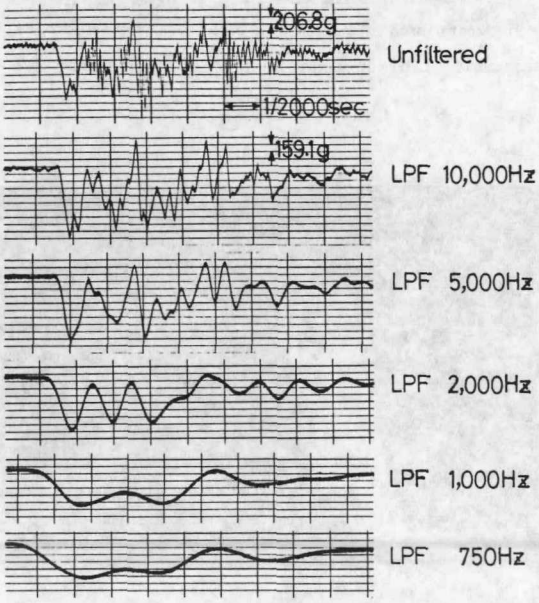


Fig.8 Modification of the entire record of deceleration through low pass filters in case of corner drop of 1/8 model

A SIMPLE CALCULATION TECHNIQUE USING THE SOLID-ANGLE METHOD FOR RADIATION DOSE RATE DISTRIBUTIONS AROUND LARGE PACKAGES

A.Sekiguchi, M.Nakazawa;
University of Tokyo,

K.Ueki, H.Yamakoshi, K.Takeuchi, Y.Kanai;
Ship Research Institute,

T.Yamada;
Hitachi Shipbuilding & Engineering Co., Ltd.,

E.Kudo;
Ship Bureau, Ministry of Transport,

S.Fukuda;
Electric Power Research Institute.

INTRODUCTION

In these days, radioactive large packages like as spent fuel shipping casks have been transported by a cargo ship in Japan. The estimation of radiation dose rate distributions around casks and on board, in advance of the transportation, is not only important for radiation protection but also nesessary in confirming the integlity of the packages. Accordingly, it may be required that a calculation method for the estimation of dose rate distribution is simple so that the person concerned for the cargo ship and the competent authorities can use like, for example, the r^{-2} rule for a point isotropic source located in vacuum.

MEASUREMENT OF DOSE RATE DISTRIBUTIONS AROUND A SHIPPING CASK

In order to find out gamma-ray dose rate distributions and the attenuation tendencies of the distributions, measurements were performed with a scintillation-type surveymeter around a spent fuel shipping cask named "EXCELLOX-III". The shipping cask was contained two PWR spent fuel assemblies of 21300 MWD/MTU in 500 days of cooling time and three of 28600 MWD/MTU in 1220 days of cooling time. The gamma-ray source intensity was estimated to be 3.5×10^6 photons/sec in the shipping cask with the ORIGEN [1] code. Fig. 1 shows a plane view of the quay area and the measured directions around the shipping cask. The Direction No. 1 is an axial direction for the bottom side, the Direction No. 6 is an axial direction for the lid side, and the Direction No. 4 is a radial direction in the mid cask. The measured dose rate distributions for each direction are summarized in Fig. 2. From the measured

dose rate distributions, the attenuation tendency in the Direction No. 1 is approximately same as the r^{-2} rule for a point isotropic source, however it is evident that the tendencies are more slower than the r^{-2} rule in the Direction of No. 2, 3, 4, and 5. Especially the tendency is slow within 2.5 meters along the directions. As Fig. 2 includes the natural background, the dose rates of above 10 meters are mostly of it.

INTRODUCTION OF THE SOLID-ANGLE METHOD

The detectors located in the Direction of No. 1 glare at the bottom surface from which the penetration radiations emerge. On the other hand, the detectors located in the Direction of No. 3, 4, and 5 glare at the side surface or both the side surface and the lid or bottom surface from which also the penetration radiations emerge. Accordingly, at the same distance from the cask surface, the glaring solid angle to the cask surface on the line of the Direction No. 2, 3, 4, and 5 are larger than that of the Direction No. 1 for usual large cylinder-type shipping casks. Usually both the solid angle and the dose rate are decrease as the distance between the detector location and the cask surface becomes far off. Therefore, it may be considered that there is some relationship between the attenuation tendencies of the dose rate distributions along each direction. By the way, the estimated dose rate at 1 meter distance from the cask surface have to state as the transport index in the application of getting the transport permission from the competent authority. To estimate the dose rate at 1 meter distance, the discreate ordinates codes have been used for the most large shipping cask in Japan. Therefore, the solid-angle method with the glaring solid angle and the dose rate at 1 meter distance may make clear the dose rate distributions around a large packages like as a spent fuel shipping cask.

From the above discussions, following equation is introduced.

$$D = Tid \cdot \frac{\Omega x}{\Omega 1} , \qquad (1)$$

where D = the dose rate (mrem/hr),

T_{id} = the dose rate at 1 meter distance from the cask surface (mrem/hr),

$\Omega 1$ = the glaring solid angle at 1 meter distance,

Ωx = the glaring solid angle at x meters distance.

Equation (1) is the basic form of the solid-angle method.

The glaring solid angle is calculated by the following procedures. As an example calculation problem, Fig. 3 shows the geometry of a cylinder side surface. The differential solid angle $d\Omega$ is expressed as:

$$d\Omega = ds \cdot \cos\theta / \ell^2 , \qquad (2)$$

where ds = the differential area element,

ℓ = the distance between the detector location P and Q at the area element,

$\cos\theta = \bar{n} \cdot \overline{QP}$

$\bar{n}$ = the normal vector.

The co-ordinates of the location of P and Q in Fig. 3 can be written as:

$$P = P(0, L, 0), \text{ where } L = R + D,$$

$$Q = Q(R\sin\psi, R\cos\psi, h).$$

The angleψ is restricted between 0 and $\cos^{-1}(R/L)$. The differential area element ds is written as:

$$ds = R \cdot d\psi \cdot dh \qquad (3)$$

$\cos\theta$ in Eq. (2) is derived as follows. The direction cosine of the normal vector $\bar{n}$ is described as:

$$\bar{n} = \bar{n}(u', v', w'),$$
$$= \bar{n}(\sin\psi, \cos\psi, 0). \qquad (4)$$

Also the direction cosine of $\overline{QP}$ is written as:

$$\bar{Q} = \bar{Q}(u'', v'', w'')$$
$$= \bar{Q}(-R\sin\psi/\ell, (L-R\cos\psi)/\ell, -h/\ell), \qquad (5)$$

where $\ell = \sqrt{R^2\sin^2\psi + (L-R\cos\psi)^2 + h^2}$.

The $\cos\theta$ is expressed as the inner product of $\bar{n}$ and $\bar{Q}$.

$$\cos\theta = \bar{n} \cdot \bar{Q}$$
$$= -R\sin^2\psi/\ell + (L-R\cos\psi) \cdot \cos\psi/\ell. \qquad (6)$$

Substitute Eq. (6) into Eq. (2), then following equation is derived.

$$d\Omega = ds \cdot \cos\theta/\ell^2,$$
$$= \frac{1}{\ell^3} \{-R\sin^2\psi + (L-R\cos\psi) \cdot \cos\psi\} \, Rd\psi dh. \qquad (7)$$

Accordingly, the solid angle Ω is finally written as:

$$\Omega = \int_{-\psi_{max}}^{\psi_{max}} \int_0^H \frac{1}{\ell^3} \{-R^2\sin^2\psi + (L-R\cos\psi) \cdot R\cos\psi\} \, d\psi dh, \qquad (8)$$

where $\psi max = \cos^{-1}(R/L)$.

Equation (8) is employed only to calculate the solid angle to the side surface of a cylinder, however the solid angle to other configuration may be obtained with the almost same manner. Therefore, when a figure or a table in which describes the values $\Omega x/\Omega 1$ as a function of distance between the detector location and the cask surface in a radial and in an axial direction for typical casks is presented, the dose rates are easy estimated with the figure or the table.

APPLICATION OF THE SOLID-ANGLE METHOD

1. To a Shipping Cask

To verify the usefulness of the solid angle method for the estimation of dose rates around a shipping cask, Fig. 4 summarizes a comparision of the measured and calculated dose rate distributions around the shipping cask "EXCELLOX-III". For the simplicity of an application of the solid-angle method, typical dose rate distributions of radial and axial direction in the mid cask are calculated, making the iso-dose lines just as shown in Fig.4. (Figures in the parenthesis are calculated values.) The Tid is employed the measured value of 0.28 mrem/hr in the radial direction of No. 2, 3, 4, and 5 and of 1.7 mrem/hr in the axial direction of No. 1, respectively in Fig. 4. The calculated dose rate distributions agree well with the measured ones in the radial direction in the mid cask and in the axial on the cask. Only in the direction of No. 3, it was observed that the measured dose rate is larger than the calculated value on a maximum of 50 %. The discrepancy the direction of No. 3 is considered to be influences of the local peak of the dose rate appered on the cask surface, the non-uniform burn up of the spent fuels loaded in the cask cavity, and the characteristic of the cask structures. The dose rate at 1 meter distance from the bottom surface is 1.7 mrem/hr and the value is larger than that of one at 1 meter distance from the side surface by a factor of 5.

2. To a Spent Fuel Shipping Boat

The gamma-ray dose rate distributions were also measured in the spent fuel shipping boat "HINOURA MARU" loaded the shipping cask named "NH-25" in the No. 2 hold. The other two casks were also loaded in the No. 3 hold. However the dose rates on the side surface of both the casks were lower than that of the cask "NH-25" by a factor of 100, so that it is reasonable that the dose rates at the detectors located in the stem side from the No. 2 hold are contributed only from the cask "NH-25".

The averaged dose rate at the side surface of the cask "NH-25" was 4.0 mR/hr and the Tid was estimated as 1.48 mR/hr.[2] The side view and the measured locations in the "HINOURA MARU" are shown in Fig. 5. The Table 1 summarizes the comparision of gamma-ray dose rates between the measured with TLD's (Thermo-Luminescence Dosimeter) and the calculated values using the solid-angle method at the locations of 34, 33, 32, 31, and 30 in Fig. 5. The calculated value at the location 34 agrees well with the measured one, however as becomes more distance, the discrepancy between the measured and calculated values is getting large along the locations of 34 33 32 31 30 in Fig. 5. This reason may be regarded as the solid-angle method dose not consider the shielding effects of materials located between the radiation source and the detector but only considers the distance between them as the r^{-2} rule dose.

It is assumed that the representation of gamma-ray dose rat with unit of mrem/hr is equal to mR/hr in this paper.

CONCLUSIONS

Through the applications and the discussions, the solid-angle method may be regarded as a simple calculation technique so as to estimate the dose rate distributions around large packages like as spent fuel shipping casks and also in a spent fuel shipping boat.

(1) Fairly good values will be expected of dose rates around a shipping cask with the solid-angle method when there is no severe distortion of the dose rate on the cask surface.

(2) The calculated dose rates with the solid-angle method may be larger than those of the measured ones as becomes more distance and as the shield increases in thickness, and always safety side for the competent authorities.

(REFERENCES)

(1) M. J. Bell: ORIGEN-The ORNL isotope generation and depletion code, ORNL-4628, (1973).

(2) K. Ueki, et al.: Application of solid angle method to the estimation of radiation dose rate distributions around large packages, Report of Ship Research Institute, Vol. 17, No. 4, 279 (1980), in Japanese.

Table 1 Comparision of gamma-ray dose rates between TLD's and Solid-Angle method in the HINOURA MARU.

Locations		34	33	32	31	30
Distance from the cask surface	At 1 m	2.65 m	3.35 m	9.85 m	11.15 m	16.55 m
TLD's mR/hr	1.48	0.59	0.19	0.051	0.009	0.006
Solid-Angle method mR/hr	1.48	0.51	0.36	0.056	0.044	0.022

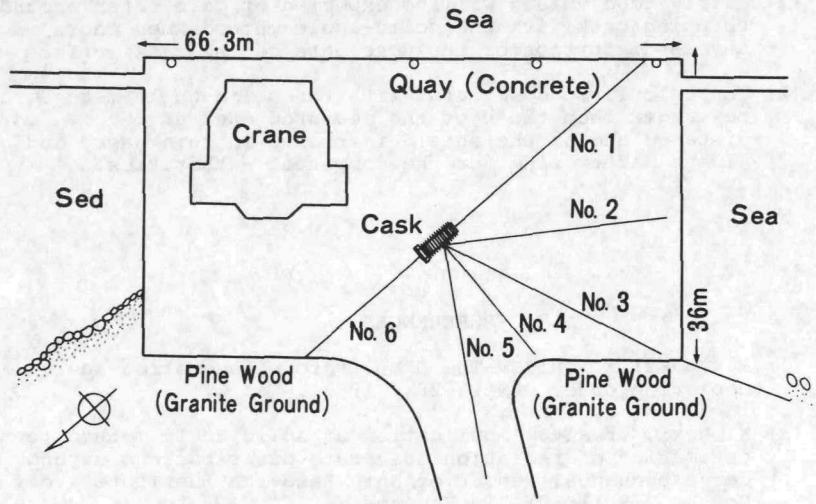


Fig. 1 Experimental arrangement for radiation measurements around a spent fuel shipping cask

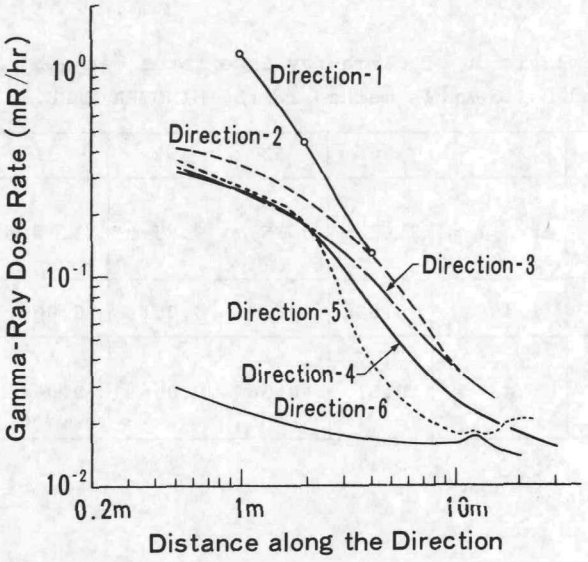


Fig. 2 Gamma-ray dose rate distributions around the cask including the background

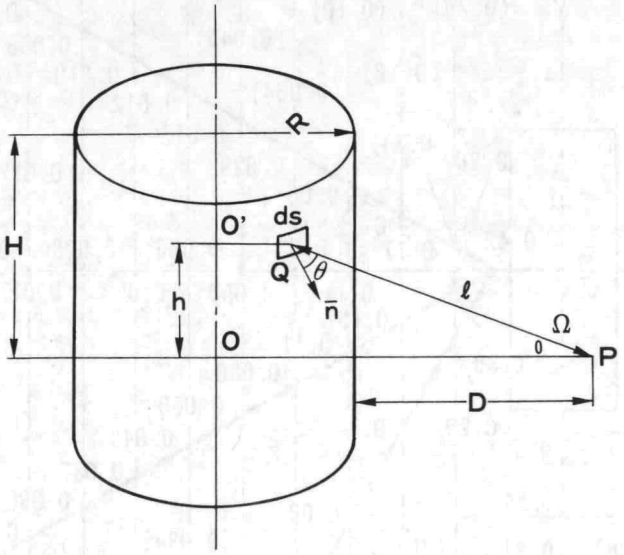


Fig. 3 Geometry for calculation of solid angle of a cylinder

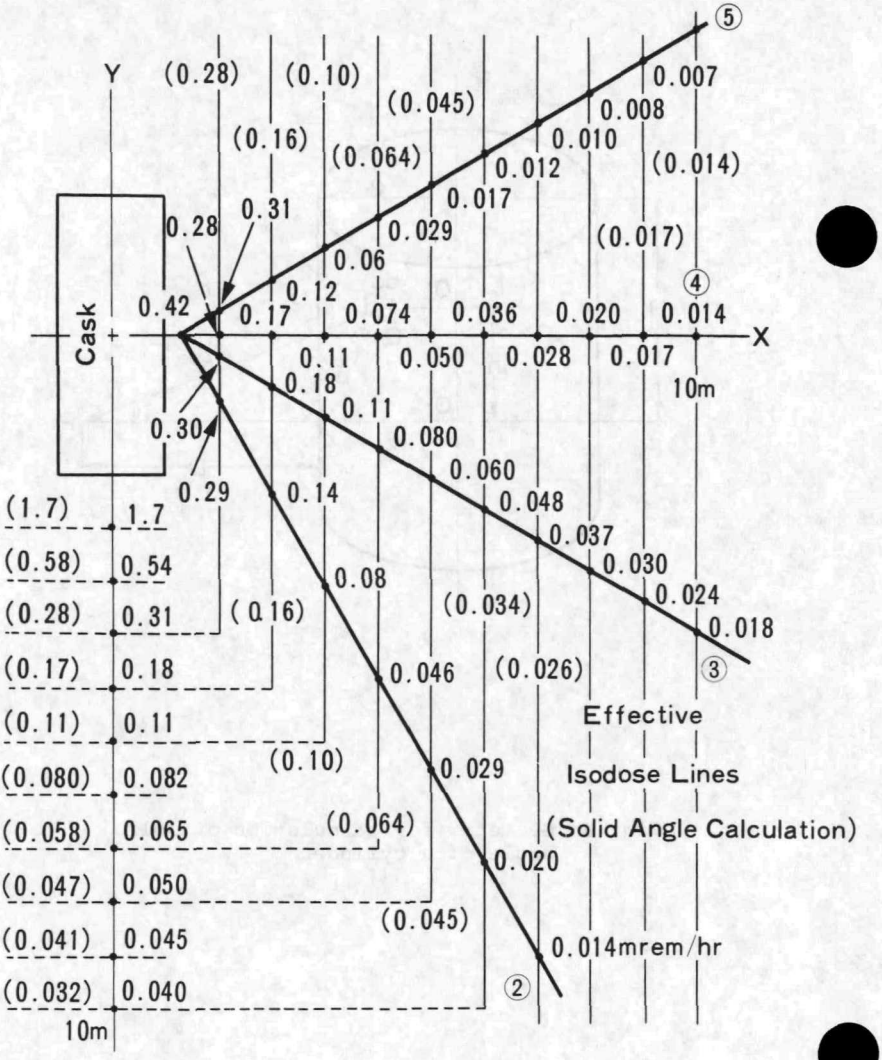


Fig. 4 Gamma-ray dose rate distributions around the shipping cask

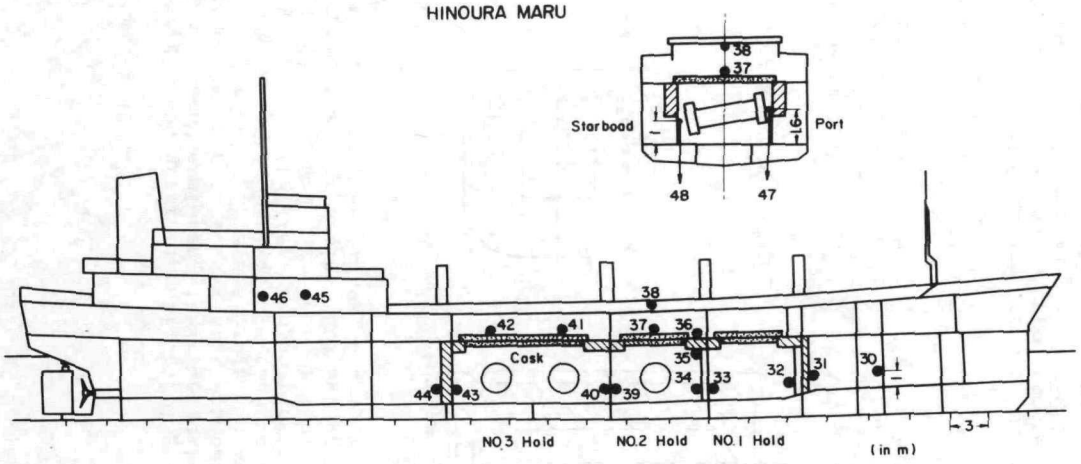


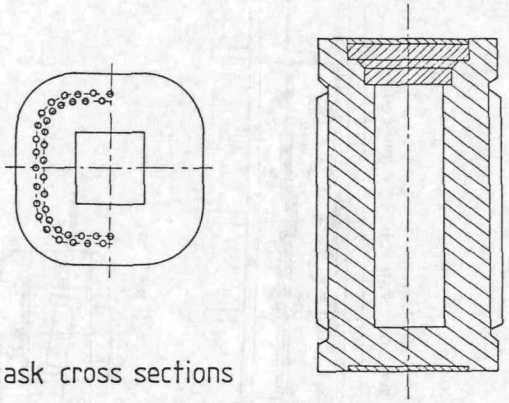
Fig. 5 Side view and the measured locations in the "Hinoura Maru"

SHIELD DESIGN FOR A NEW GENERATION OF SHIPPING CASKS

Wolfgang Denk
Gesellschaft für Reaktorsicherheit, 8046 Garching
Federal Republic of Germany

1. PROBLEM AND DESIGN LIMITS

In the last years a new generation of shipping casks for irradiated fuel elements has been developed with a capacity for 9 PWR and 16 BWR fuel elements, respectively. As these casks should also meet the requirements for intermediate storage, it was felt to be appropriate to have considerably lower dose rates at the cask surface than allowed by the IAEA transport regulations. Therefore a total dose limit of 20 mrem per hour was initially chosen for design. With respect to increasing fuel burnup special considerations had to be made for neutron shielding to stay within the above limit.



cask cross sections

As the type of materials (cast iron, stainless steel and polyethylene) were preset by technical and economical considerations, only the shell thickness, size and location of the neutron absorbers remained as free parameters for the shield design calculations.

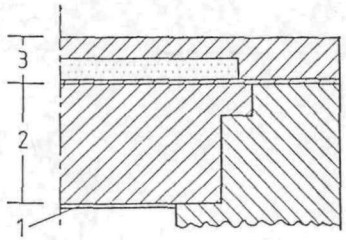
2. MODEL DESCRIPTION

For radial shielding calculations the cask was mocked up as an infinite cylinder. Fuel and basket were homogenized over the interior of the cask. The holes in the wall filled with polyethylene were replaced by an equivalent polyethylene layer, so that these calculations could be done in one dimensional cylindrical geometry. Bottom and lid, respectively, were mocked up in two dimensional(r-z)

geometry including the head of the fuel element, but neglecting the void between fuel element and lid in a conservative manner.

3. CALCULATIONAL PROCEDURE

First a series of parametric calculations has been performed in order to determine the overall wall thickness optimizing amount and location of the polyethylene needed for neutron shielding. The computations typically started with the determination of the neutron and gamma sources taking into account the history of the fuel elements. These calculations were done with the ORIGEN code. All the shielding calculations were carried out with the S_n-codes ANISN and DOT in S_8 P_3-approximation using a coupled 22-group n-γ-library. The radial calculations were performed with the ANISN code starting with a volume distributed source. The effective polyethylene layer thickness was determined comparing ANISN and DOT runs in slab and (x-y) geometry, respectively, where in DOT the holes were represented in square geometry. The shielding calculations for the lid were done with DOT, the neutron and gamma computations carried out in separate runs due to different



1 fuel element head
2 primary lid
3 secondary lid

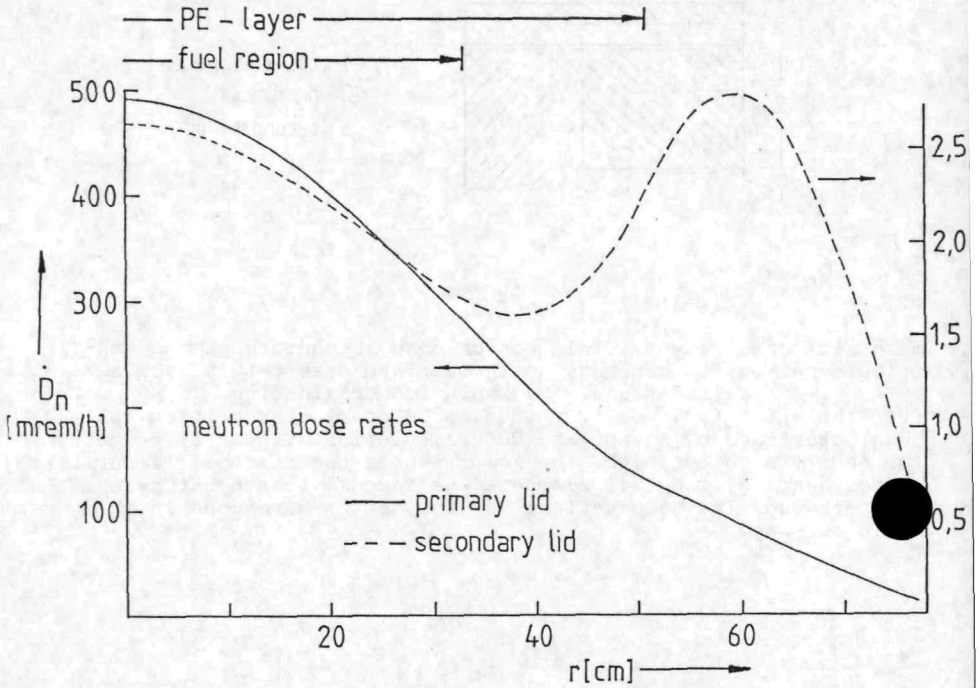
mesh size requirements. This can be done as capture gammas contribute only a few percents to the surface dose rat as is shown by A N I S N calculations. The radiation originating from the spent fuel was treated as a boundary source which has been determined by a separate DOT calculation with (r-z) geometry for an infinite cylinder. The gamma source due to cobalt impurities in the heads of the fuel elements was taken into account by a separate run. The computations for the bottom were done in an analogous way.

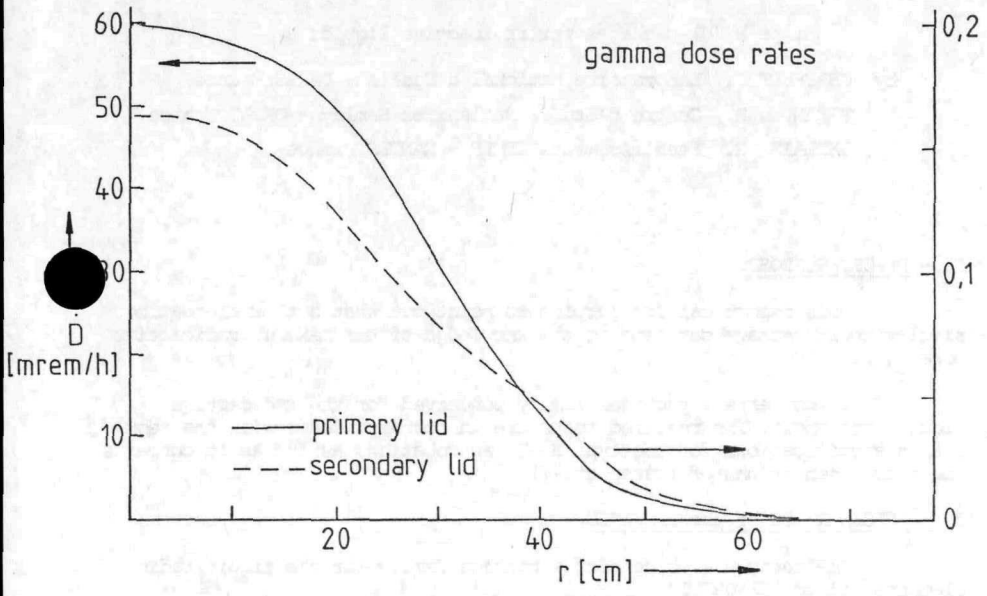
4. VALIDATION

The calculational procedure and the cross section set were checked against an experiment done at the KfA Jülich using a cask prototype with similar geometry and the same materials. In this experiment an Am-Be-neutron source was used. Measured and computed dose rates at the cask surface agreed within 15 percents.

5. RESULTS

The calculations have shown the possibility to meet the design limits with economical expense. Because of the lower density of the cast iron thicker walls are required than for stainless steel casks. The size of the polyethylene rods depends on the fuel type. They were found to be most effective when placed in the outer third of the shell.





Strong cobalt impurities in older fuel element heads caused larger lid thicknesses as originally planned. The radial distribution of the neutron dose rate at the lid surface also heavily influenced the design, so that greater modifications were necessary. As a typical example for the results of the shielding calculations for the cask lid the neutron and gamma dose rates at the lid surface are shown in the figures. They refer to a BWR cask containing 16 fuel elements with a burnup of 27 000 MWd/t layer in the secondary lid clearly can be seen.

THERMAL TESTS CONCEPTION OF

type B (U) packages for radioactive liquids

by CHATELET J. Laboratoire National d'Essais - PARIS France
SERAN R. Centre d'Etudes Nucléaires Saclay - ORSAY France
LEBLAIS R. Etablissements LEMER - NANTES France

1. - REPORT PURPOSE.

This report has for purpose to point out what a thermal heating simulation of package can lend to the knowledge of her maximum radioactive power (1).

It concerns a packages family conceived for the radioactive liquids transport. The realized tests are in correspondence with the regulation specifications for the type B (U) approbation, as far as it concerns the normal conditions of transport. (2)

2. - PRESENTATION of the PACKAGES.

The tests are concerning a package family for the liquid radioelements called "CENDRILLON".

These are all build according to the same principle and they form a scale with the following extreme characteristics :

. lead thickness protection : from 5 cm to 30 cm

. capacity for radioactive solution : from 3 l to 200 l.

This represents a total number of 16 packages distributed between three groups by her dimensions.

The joined diagrams are giving the essential elements :

2.1 - The Cendrillon container (fig. 1).

He has two parts :

- *a containment vessel* of stainless in the form of a flatten revolution ellipsoide ; it contains the radioactive liquid. It is

---

(1) - we use te term "maximum radioactive power" corresponding to the AIEA regulations, to make a distinction with the "full load transportation" used too by the AIEA regulations.

(2) - This paper is standing out of the "compact report" on the following points :
- correction of included material errors ;
- insertion of the "solar heat load" paragraph ;
- change of the diagrams for a better agreement with the analysed model.

fixed by the top to the whole of the Cendrillon, by a chimney containing the transfert pipes, and empted with lead for radioprotection.

- *a lead shielding* encased in a carcasse that surround wholly the containment vessel. Her thickness is determined by the needs of radioprotection. To the top, he has a conical outlet for the passage of the confinment vessel chimney. An upper plate is supporting the mechanisms for the transfert controls. During the transports, the containment vessel chimney and the lead shielding are together in mechanical contact.

2.2 - The shipment shell (fig. 2).

She is withstanding to the free drop an to the thermal test, and she contains the Cendrillon container. She is composed with two hemispheres of steel, containing a proper thickness of thermal insulating material, and shock absorbers. The Cendrillon is upwedged in her shell by some mouldings of rubbers.

3. - TESTS ORGANIZATION.

The general characteristics of the Cendrillons have been determined to answer the various imperatives :

. capacity

. health protection

. withstanding to the free drop and to the thermal test up to 800°C (AIEA Regulations § 719-720).

The tests were intended to know the thermal characteristics of these packages for the normal transport conditions. They were realized in December 1978 and Januar 1979 with two types of Cendrillon :

. a standard series called 5/50 (lead thickness 5 cm, capacity 50 l) ;

. a giant series called 25/8 (lead thickness 25 cm capacity 8 l).

For the sake of commodity, here are only described the results concerning the "standard Cendrillon 5/50".

The containment vessel is empted with oil (2) to the maximum capacity, this oil is heated by an electrical resistance, immersed (thermocoax type) and the temperature is noted to the different critical points of the cited element.

(2) - in order to outpass, to the liquid state, the thresold of the water ebullition.

Three levels of heating have been used (350-237-56 W) with some periods of increase or decrease the temperature.

4. - EQUILIBRIUM STATE At 350 W.

The table 1 is giving the measurement results for a determined power : 347 W. In account of the results obtained with the two other levels of power, it has been possible to deduce the numerical values of the thermal parameters of the componing elements.

So we state that :

4.1 - Between the containment vessel and the lead shielding, the greater part of the exchanges take place by conduction : as the vessel chimney is empted with lead, and the thermal contact correct between the chimney and the lead shielding, the conduction is favourable at this level. We can admit that if the sole source of heating is the radioactive solution, the confinment vessel and the lead shielding are almost taking a thermal balance ($\Delta T = 20°C$).

4.2 - Between the Cendrillon and the shell, the most part of the thermal exchanges take place by radiation : the rubber mouldings send on less than 4 % of the released heat. The temperature fall between the two metallic faces is not very hight ($\Delta T = 20°C$).

4.3 - A preferential zone of transmission is formed in the shell, by the flanges joining the two half-shells. The transmission by this channel (335 W) seems over estimated.

4.4 - Radiation and convection of the shell external face. A gap of 30°C has been noted between the ambient air and the shell surface in the laboratory conditions. It would doubtless not the same during a transport : the air convection would be greater than in a test hall.

The exchange coefficient e, deduced of the measurement results is 0.36, while he is usually of 0.9 for the painted steel. This can be explained by the thermal break going from the shell flanges : the most part of the heat is transmitted through the shell by the flanges and her surrouding.

So we can have a representation of the shell thermal emission, by supposing that 40 % of the external surface in the neighbouring of the flanges radiates with an emissivity coefficient 0.9, and that 60 % of the surface are not radiating.

5. - MAXIMAL RADIOACTIVE POWER CALCULATION.

The table 1 gives for each one of the four exchange zones, the part of the power for the conduction, the radiation and the convection. The corresponding thermal equations are given in the full report. The imposed boundary conditions are the following :

a) - Maximum temperature of the radioactive solution : 152°C or 5 bars (0.5 MPa) for the vapor pressure. This limitation is due to the characteristics of the containment vessel elements.

b) - Maximum temperature of the shell surface : 82°C for the "full load transport" and for an ambient temperature of 38°C (§ 240).

c) - The case of the "not full load transport" has been too considered, but is not here developed.

The analysis of the results and of the corresponding equations shows that the more critical conditions are given by the shell radiation. Therefore, the calculations have been taked for this surface, and we have after controled step by step, that the thermal exchanges in the three other zones would not provoque any impossibility (3). The maximum radioactive power calculated is then : 330 W for the "full load transport" of a type 5/50.

6. - SOLAR HEAT LOAD.

From the experimental results, two extrapolations are necessary, to deduce the maximum radioactive power :

a) - Ambient temperature conditions : the tests were realized with 18°C of ambient temperature, and led to a maximum temperature of 135°C in the radioactive liquid. For the ambient temperature of 38°C (§ 240), the liquid would not outpass the temperature of 152°C, corresponding to an vapour pressure of 0,5 MPa (5 bars) for aqueous solutions (§ 5.b. of this paper) ;

b) - solar heat load : the regulation conditions (§ 232-233) are the heating of 50 % of the shell area during 12h each day, giving a thermal power of 392 w, while the radiation from the radioactive liquid correspond to a permanent radiation of 330 w on 40 % of the same area.

Taking account of the shell thermal inertia, and of the geometrical distribution of the thermal exchanges on the shell area, we can estimate that under the sun, the radioactive liquid temperature would not outpass 165°C, corresponding to a vapour pressure of 0,7 MPa (7 bars), what is the maximum designed pressure from the transport regulation (§ 122 and 239).

. - CONCLUSIONS.

7.1 - To the contrary of previous estimations, the thermal exchange between the containment vessel and the shielding are developed essentially by conduction, an not by radiation : this has permitted the raising of the authorized radioactive power from 56 W to 330 w.

./.

7.2 - The new radioactive power limitation concerns essentially the shell external surface radiation. Then this limitation has the same value for all the series of Cendrillon which are transported into the same size of shell.

7.3 - A shell radiation measurement by "infrared thermography" would doubtless given a better knowledge of her distribution, and per to deduce some design improvements.

7.4 - In the course of this experiment, the time constants of the thermal exchanges have too been measured.

7.5 - The realization of relatively simple tests on two series of Cendrillonshas lent to the knowledge of the thermal parameters for all the Cendrillons in the series standard and giant.

(3) - The calculations uses the experimental values of the temperatures, and the technical values of the thermal characteristics of the components.

EXCHANGE ZONES	TEMPERATURE DIFFERENCES	EXCHANGED THERMAL POWERS (1) (SUM : 347 w)		
		CONDUCTION	RADIATION	CONVECTION
1. - Containment vessel $\rightleftarrows$ shielding	20°C	250 w (75 %)	97 w (25 °)	0
2. - Shielding $\rightleftarrows$ shell	20°C	28 w (6.6 %)	325 w (93.3 %)	0
3. - Between faces of the shell	38°C	68 w isolant	0	0
		335 w flanges	0	0
4. - Shell $\rightleftarrows$ ambient air	30°C	0	← 347w	(100 %) →

(1) - Thermal powers deduced of the measured temperatures of the constituents materials thermal characteristics

Used formules : Conduction : $w = \gamma\, s/l\, [\, T_2 - T_1\,]$

Radiation + convection : $w = \lambda\, S\, [\, T_2^4 - T_1^4\,]$

S = exchange surface (m^2)

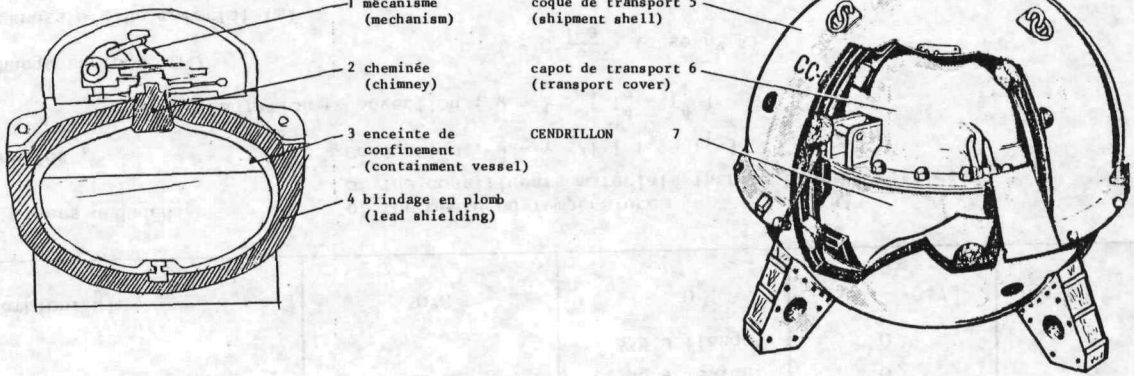
l = thickness of the material (m)

T = surface temperatures (°C)

$\lambda = C \frac{e}{1-e}$ (case n° 4)

$\lambda = C \frac{e}{2-e}$ (case n° 2)

TABLE I - THERMAL EXCHANGES IN EQUILIBRIUM STATE



COQUE DE TRANSPORT CENDRILLON
(CENDRILLON shipment shell)

ACCELERATION AND STRAIN MEASUREMENTS PERFORMED ON CASKS FOR RADIOACTIVE MATERIALS UNDER IMPACT CONDITIONS

K. E. Wieser, B. Schulz-Forberg, M. Jais
Bundesanstalt für Materialprüfung (BAM)
Unter den Eichen 87, D-1000 Berlin 45

During drop tests carried out by the Bundesanstalt für Materialprüfung (BAM) as part of licensing tests, acceleration measurements - and to a certain extent strain measurements - are usually performed on the test object in the impact phase.

Whether these measurements are actually needed is often called in question. Therefore, as an introduction, the main reasons for such measurements should be mentioned. The IAEA type B tests are internationally harmonized and have similar effects to heavy transport accidents. As the prototypes are submitted to destructive tests and the results are valid for all packagings of the same type, these tests have the character of demonstration tests. The applicability to all packagings of the test results (which strictly-speaking are only valid for the test object itself) has to take into account the permissible spectrum of materials and parameters for machining.

In addition, model factors (if models of packagings are tested) and the influence of low temperatures (-40°C) have to be considered. The applicability of the test results to all packagings requires knowledge of the loads produced during the impact phase; they can be calculated by means of acceleration measurements. These data, if unsatisfactory, provide the necessary basis for improvements of the design.

In contrast to other kinds of tests (hydraulic tests, leakage tests, etc.) drop tests are expensive and usually cannot be repeated. Besides this, the most interesting aspect, the impact phase, takes place within some milli-seconds. Both require a reliable measuring technique and careful preparation. In order to determine the forces initiated by the impact the following are used by BAM:

1. Measuring of the deceleration/time function by piezzoelectric accelerameters.
2. Measuring of the strain/time function by means of strain gauges.
3. High speed films of the deformation/time function.

These methods can be described as follows:

1. Measuring of the deceleration/time function by piezzoelectric accelerameters.

This method uses the effect of charge movements in a quartz crystal under compression. Instruments of this kind are available for a great range of decelerations. As additionally their natural frequencies are rather high, very short processes can be measured. In the BAM, decelerations are recorded by means of a so-called transient recorder and then documented by x-y recorders, punch tape or dual beam oscillographs. A separate light barrier is used to trigger the recording; the falling test object hits the light beam some centimeters before impact (fig.1). BAM equipment enables the simultaneous recording of six functions. This extremely efficient instrumentation causes, at the same time, certain difficulties. As all natural frequencies of all parts of the test objects are stimulated by the shock of the impact, the deceleration/time function at the centre of gravity is superimposed by strong vibrations, the amplitudes of which are often higher than those of the basic function, according to the deformation character of the test object. This means that neither determination of a realistic deceleration maximum nor a control of the measurement by integration of the curve is possible (fig.2). The need for filtering is then inevitable.

To filter signals from piezzoelectric instruments BAM uses two methods. Firstly, using an electric method by implementing low-pass filters before the recording takes place, usually in the range of about 1 kHz. Secondly, using an analytic filtering based on a Fourier analysis of the deceleration signal. The advantage of the first method is its immediate, uncomplicated availability. On the other hand a falsification is possible due to suppression of the expected signal and more especially by reduction in the signals peaks.

This disadvantage can be avoided by the analytic method used by BAM. By this method the vibrations corresponding to the natural frequencies can be filtered selectively. Disadvantages are the need for a computer program and for calculating the natural frequencies of all parts of the test object.

This is based on a Fourier-analysis of the unfiltered decceleration curve. The information is a punch tape containing the signal of the transient recorder digitalized in 1024 values. The Fourier-analysis is an approximation of the signal by 512 superimposed sinus and cosinus-functions of different frequencies:

$$a = a(t) = A_o + \sum_{i=1}^{512} A_i \cos i\omega t + B_i \sin i\omega t$$

Dominant frequencies like strong natural frequencies may be distinguished by large values of the factors A_i/A_o and B_i/A_o

Fig. 3 shows an example of the function $\frac{A_i}{A_o} = f(\omega)$,

in which the natural frequencies are easily recognizable. In addition to this figure, it is useful to calculate the natural frequencies approximately including their harmonic vibrations.

Above all very high frequencies may be erased, similar to filtering by electric low-pass filters.
Now the remaining frequencies can be composed and plotted as a filtered curve.
Fig. 2 shows a signal before and after such a filtering procedure. Two different checks have been inserted in the procedure giving an idea how far a falsification of the original function has been caused by the filtering. After several calculation steps it is possible to compare both the known impact velocity

$$v_e = \int a(t)\, dt$$

and the deformation energy during the impact phase

$$E = m \int a(s)\, ds$$

In fig. 4 the calculation procedure is summarized.

2. Measuring of the strain/time function with strain gauges.

The strain gauage technique allows direct measuring of the strain on material during impact, the stress of bolts, tension and crushing strains of different locations of a test object. Measurements of this kind may be desirable if the integral behaviour of a packaging under impact conditions is essentially known. The recording and the analytic processing of the strain signal which may be superimposed by vibration effects is possible using the above mentioned method. Unfortunately the information content of strain measurements is limited as the impact forces cannot be calculated directly.

Strain measurements of this kind were used for instance to prove model laws concerning stresses on models of different scale factors.

3. Acceleration measurements by high speed films

The procedure of high-speed film analysis is based on the fact that the deformation event during the impact phase can be recorded by a multitude of pictures, so that by evaluation a deformation time-function in a digital form can be obtained. A mathematical smoothening of the function before differentiation is advisable to eliminate reading errors.

Apart from a high film frequency, this procedure requires knowledge of the time distances between two picutres. The deformation rate between two picutres can be calibrated with the help of known properties of the test object or marks on it in relation to cross wires in the camera. In fig. 5 the deceleration time -function as a result of such a procedure in relation to the signal of an accelerometer is shown. The differences are negligtible small.

The significant advantage of this method is the fact, that, given the selection of an appropriate point on the test object, the integral deceleration will really be obtained. There is no problem of superimposed vibrations and filtering. There are certain reasons why this method has its rivals; principally because it is practicable only for test objects with rather large deformations (several centimeters). Additional disadvantages are the comparable time spent and difficulties which are connected with the dust and dirt which is blown around in the impact phase.

The summing-up of these three methods forms a sufficient instrument to master the given task. Compared with the expense and uncertainty of fixing the boundary conditions and parameters of the analytic handling of the impact effects, BAM believes that instrumented demonstration tests are, even for expensive packagings, preferable.

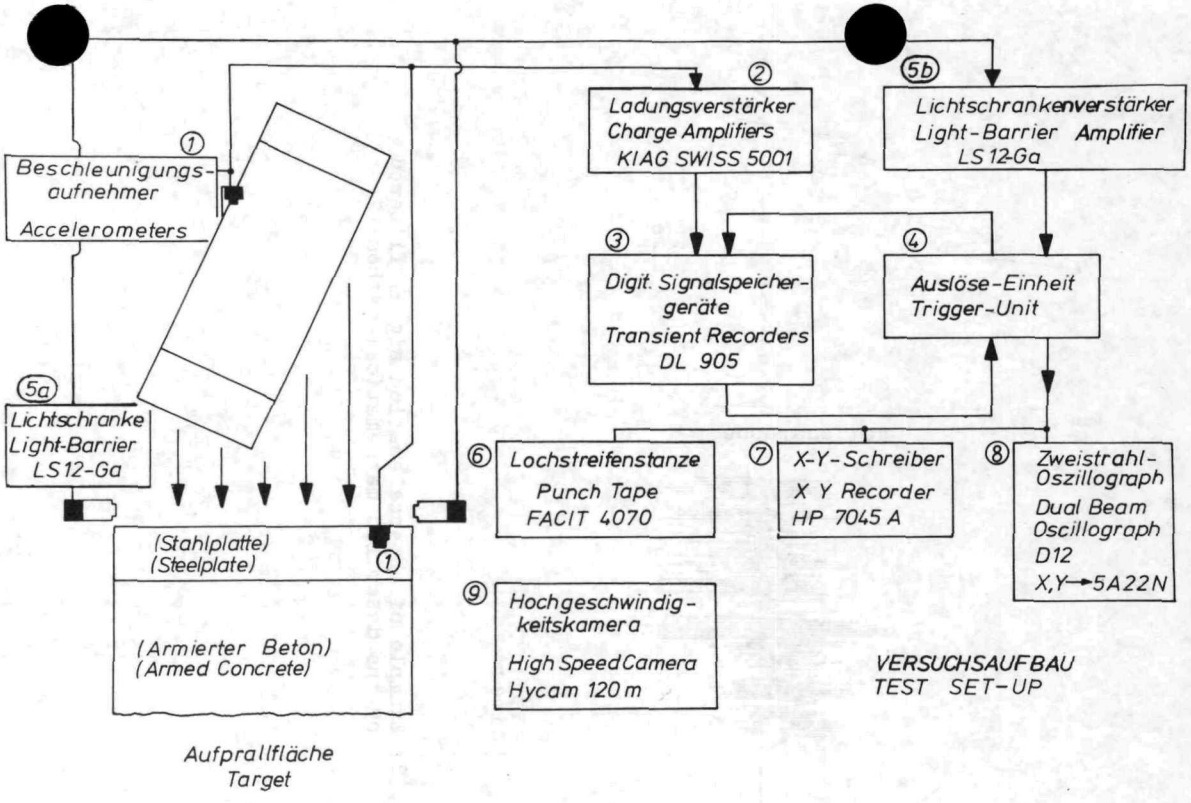


Fig. 1 : Test set-up for acceleration measurements

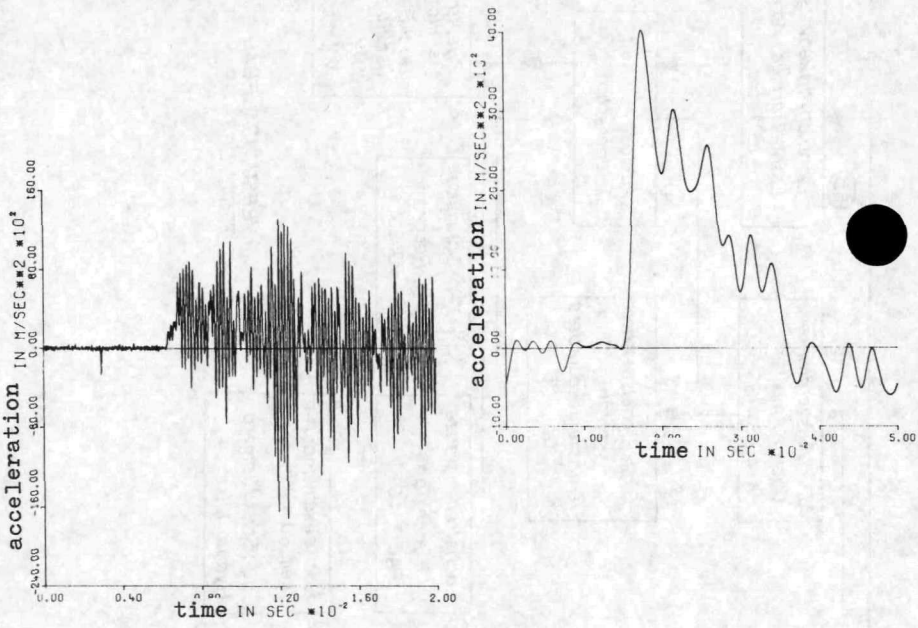


Fig. 2: Example of an acceleration signal filtered by the described mathematical method

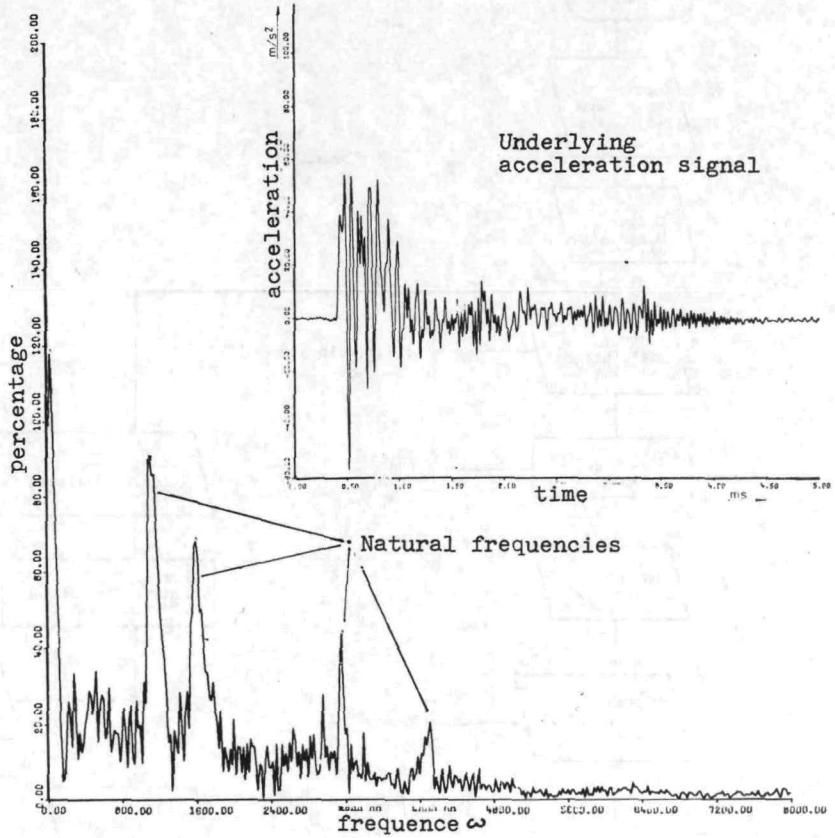


Fig. 3: Example of the function $\frac{A_i}{A_o} = f(\omega)$
of an acceleration signal

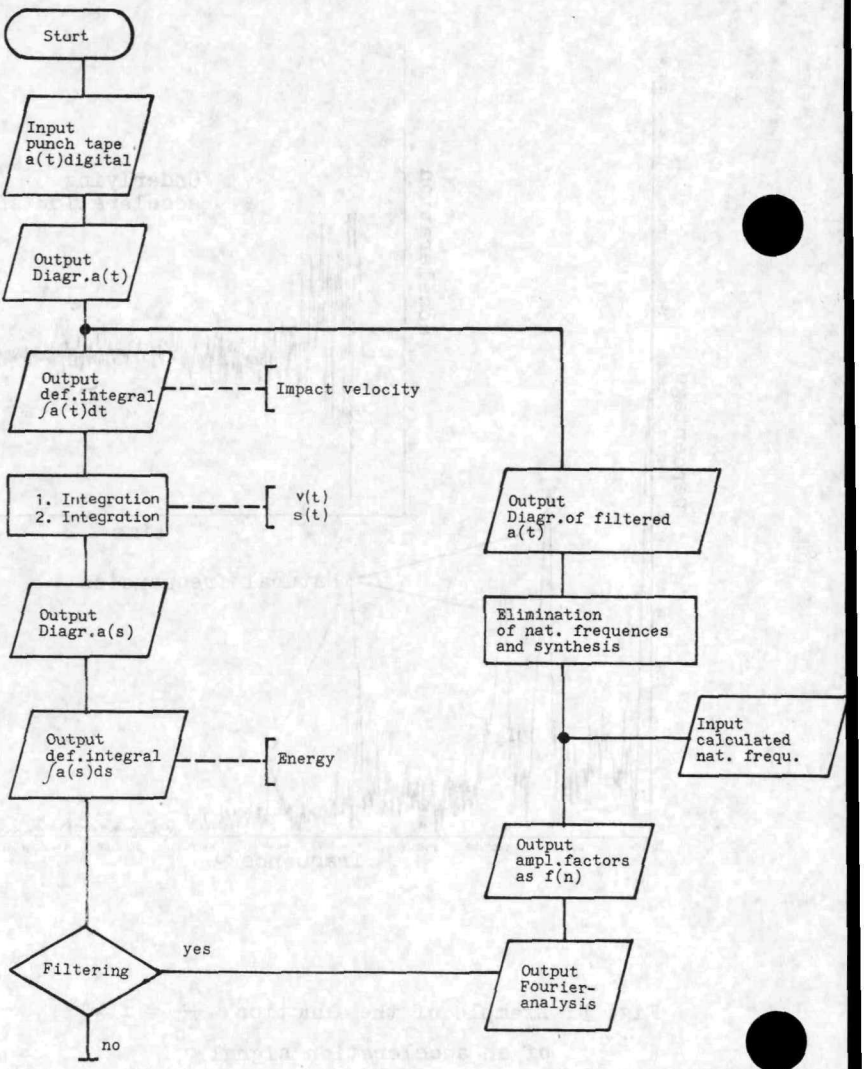


Fig. 4: BAM-Calculation-programm "Mathematical filtering of acceleration signals" - flow diagramm

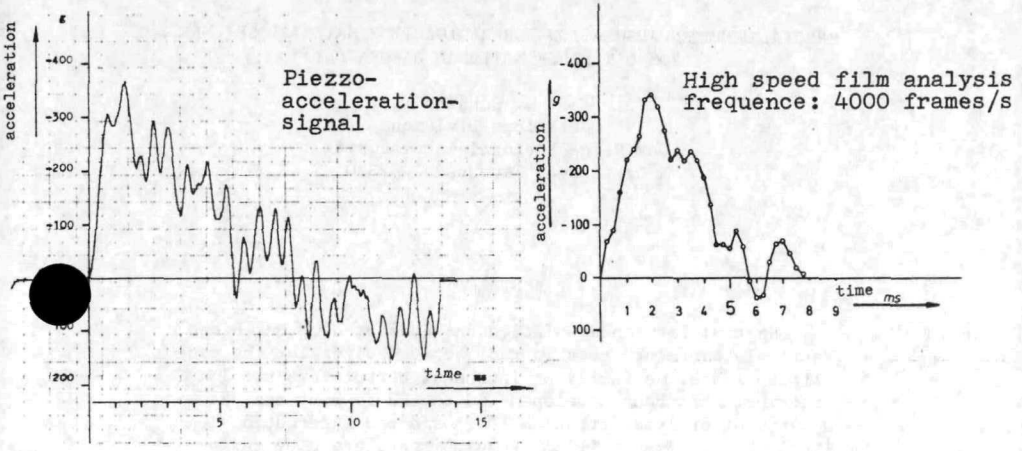


Fig. 5: Comparison of an acceleration signal with a high speed film analysis

SPECIAL FORM ENCAPSULATION FOR RADIOACTIVE MATERIAL SHIPMENTS FROM OAK RIDGE NATIONAL LABORATORY

R. W. Schaich
Operations Division
Oak Ridge National Laboratory*
Oak Ridge, Tennessee 37830

ABSTRACT

Special Form encapsulation has been used at Oak Ridge National Laboratory to ship radioactive solids for the past fifteen years. A family of inexpensive stainless steel containers has been developed and tested to meet the USA Department of Transportation (DOT) and the International Atomic Energy Agency (IAEA) regulations concerning radioactive material shipments as Special Form.

INTRODUCTION

Oak Ridge National Laboratory (ORNL) under the direction of the U.S. Department of Energy (DOE) distributes radioactive material for medical, industrial, and research activities on a worldwide basis. Approximately 5 MCi of radioactive gases, solids, and liquids are shipped each year from ORNL in 3,000 individual packages. The major radioisotopes involved in these shipments include ^{90}Sr, ^{137}Cs, ^{147}Pm, ^{192}Ir, ^{85}Kr, tritium, ^{47}Ca, ^{244}Cm, ^{238}Pu, ^{241}Am, ^{243}Am, ^{237}Np, and small quantities of short-lived fission products.

In 1965 ORNL instituted a regulation requiring all radioactive liquid shipments containing greater than 3 Ci beta/gamma or 1 mCi alpha to be converted to a solid and shipped in welded stainless steel containers which could be opened by the recipient with a standard tool. This action resulted in the design of a family of stainless steel powder containers that meet the present day DOT and IAEA regulations for Special Form. All Special Form containers described in this paper passed the tests prescribed in *Safety Series No. 6, Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition*, International Atomic Energy Agency, Vienna, 1973.

DISCUSSION

A Type I Special Form powder container (Fig. 1) was developed for the shipment of millicurie quantities of alpha materials and curie quantities of beta/gamma materials. These containers are very adaptable to hot cell or glove box operations and are used for the majority of the ORNL Special Form shipments. The Type I container consists of an inner 304L stainless steel cylinder 50.8 mm (2.0-in.) long by 15.9 mm (0.625-in.) diam. The container wall is 0.9 mm (0.035-in.) thick and the inside bottom of the container has a conical depression to aid in the removal of small quantities (milligrams) of radioactive material.

*Operated by Union Carbide Corporation under contract W-7405-eng-26 with the U. S. Department of Energy.

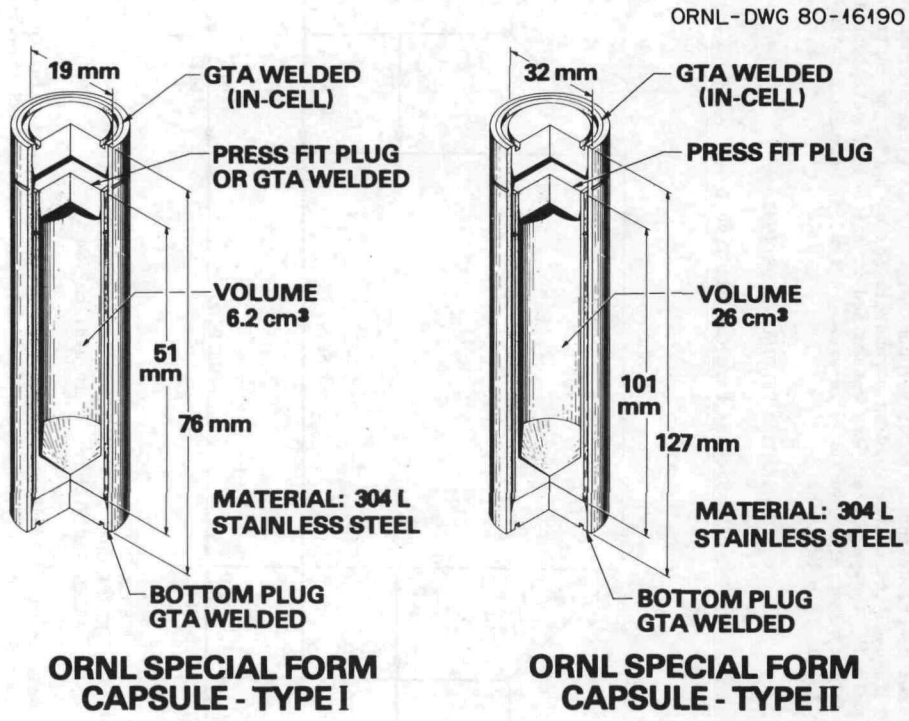


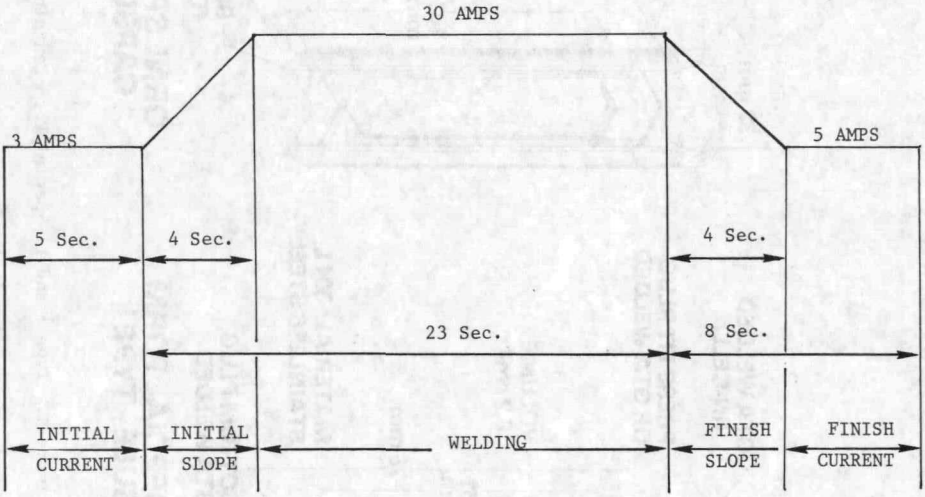
Figure 1. Type I and II Special Form Containers

The internal volume of the inner capsule is 6.2 cm^3. A press-fit cap seals the radioactive powders in the liner and allows the inner capsule to be decontaminated to less than 0.005 microcuries/100 cm^2.

The Type I outer container is fabricated from 19 mm (3/4-in.) 304L stainless steel tubing which has a 0.9 mm (0.035-in.) wall thickness. The outer shell is 76.2 mm (3.0-in.) long and is sealed with a press-fit plug. This unit is inert gas tungsten arc (GTA) welded by a Quality Assurance approved welding program using a fixed electrode and a rotating chuck (Fig. 2). A Vickers or Auto Arc Weld Programmer is used to set the welding parameters for each type of container that is to be welded. The parameters are determined in the weld development laboratory to give maximum penetration in the weld area. A typical welding program is shown in the following illustration.

PROGRAMMED WELDING CONDITIONS

ORNL Special Form Capsule Type I



CHUCK SPEED	—	1 Rev. — 18 Sec.
ELECTRODE	—	1.59 mm dia (2% Tungsten)
ARGON GAS FLOW	—	118 std. cm^3/Sec.

The weld is visually inspected and then tested for leaks using the ethylene glycol-vacuum leak test or the helium leak test described in ANSI N14.5-1977. If the container passes the leak test, it is then cleaned to a smear tolerance less than 0.0005 microcuries/100 cm^2 and placed in approved shipping container for transport to the customer.

If greater integrity is required for the shipment of plutonium or for the fabrication of radioactive sources, the inner container is GTA welded, cleaned and leak tested before insertion in the outer capsule. This system provides double containment and an additive leak test rate for the assembled package of greater than 10^{-10} STD cm^3/sec.

ORNL Photo 1414-80

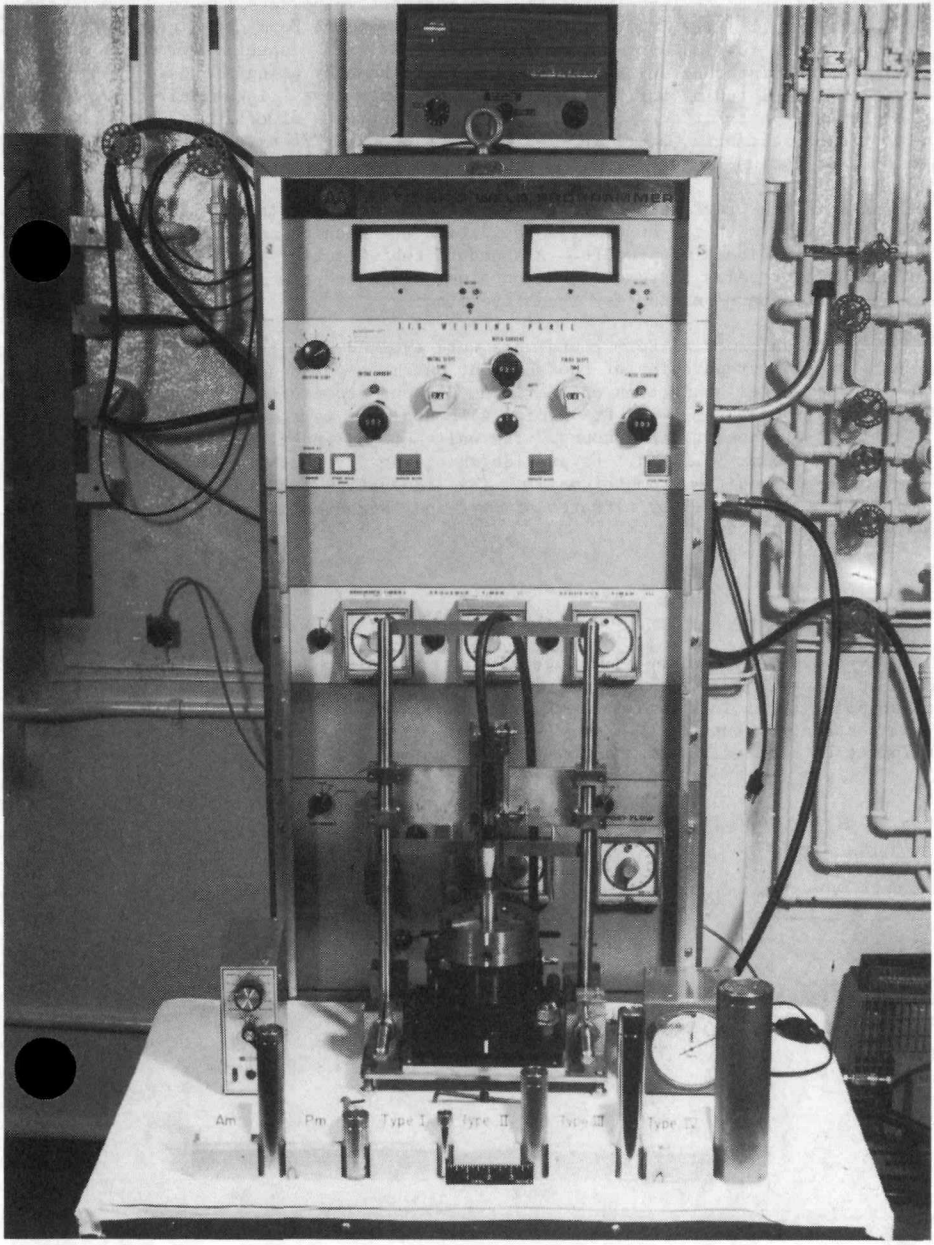


Figure 2. Special Form Welder and Programmer

For shipments of large quantities of radioactive powders (up to 0.5 Kg), the Type II (Fig. 1), III (Fig. 3), and IV (Fig. 3) Special Form containers were developed with internal volumes of 26, 70, and 326 cm^3 respectively. This design maximized the internal volume and minimized the cost by using standard 304L stainless steel tubing for fabrication of the containers. These units have been used for the past fifteen years to ship gram quantities of cesium-137 chloride, strontium-90 titanate, americium-241 oxide, plutonium-238 oxide, neptunium-237 oxide, and curium-244 oxide.

All of the previously described containers have been tested in service and are compatible with the customer's facilities for opening the container and dispensing the radioactive powders. A standard tubing cutter is used in glove box or manipulator cells to remove the container plugs and an experienced operator can open the container in less than five minutes.

Additional Special Form containers were evolved from the original design when glove box preparation of the shipment became a personnel exposure problem due to the larger quantities of americium-241 and promethium-147 required by the customer. Special Am-1 and Pm-1 (Fig. 4) containers were designed to ship gram quantities of these radioisotopes. The walls of the inner container were increased to 6 mm (0.24-in.) to provide shielding for the beta component of the americium-241 and the promethium-147. This innovation reduced the hand exposure to operating personnel by a factor of five and increased the integrity of the package for shipment.

CONCLUSIONS

Approximately ten thousand Special Form containers have been shipped by ORNL to domestic and international customers with no adverse results reported due to shipping conditions or customers processing the containers. The ORNL Special Form containers continue to meet the changing requirements for the packaging and shipment of radioctive material.

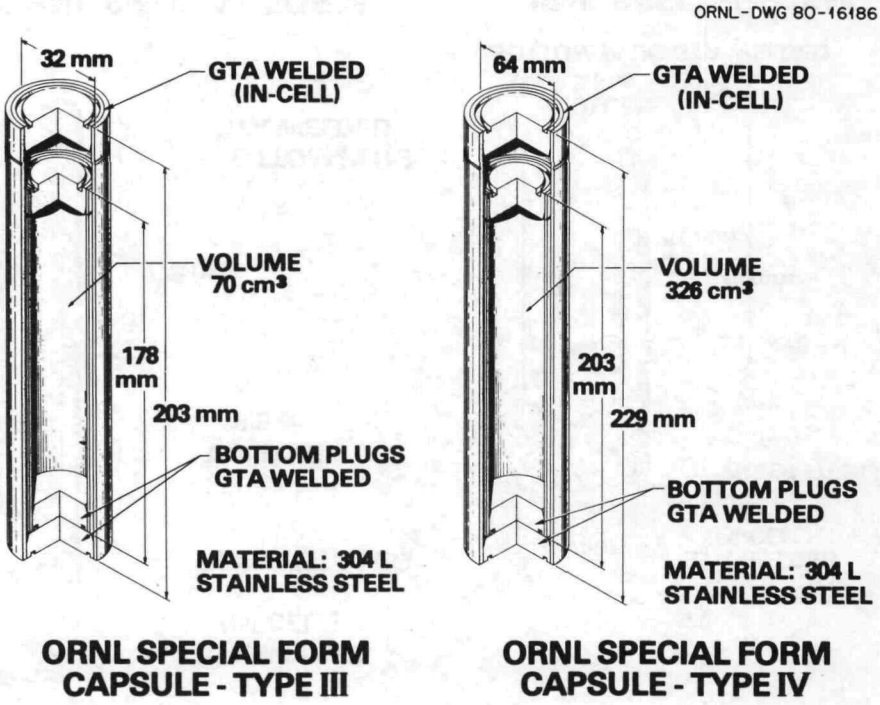


Figure 3. Type III and IV Special Form Containers

ORNL-DWG 80-16185

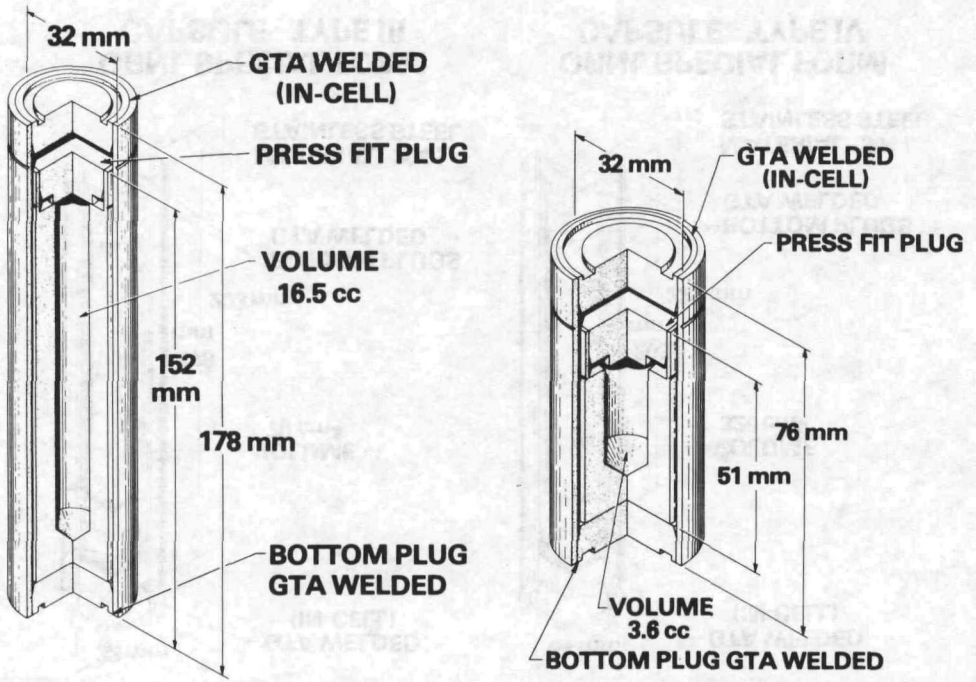


Figure 4. Special Am-1 and Pm-1 Containers

CONTAINMENT SEAL RELIABILITY TESTING

William H. Lake
U.S. Nuclear Regulatory Commission

INTRODUCTION

Reliability analysis is a quantitative, probabalistic method used to predict the performance expectation of a device. Reliability is the absence of failure, and may be defined as: the probability of meeting performance requirements for a designated period under specified conditions. The application of reliability in design and maintenance of containment systems, using elastomer seals, has been considered; the need for component reliability data for radioactive material packaging has been identified [Ref. 1].

The paper develops the reliability testing model. It discusses the general mathematical and physical models, and the Weibull and exponential distributions are presented. Reliability testing is developed on the basis of the reliability model. A general procedure is suggested for reliability testing and the treatment of test data. Containment seals for radioactive materials packaging are addressed. Consideration is given to their environment and performance criteria. A general reliability test plan is proposed.

THE RELIABILITY MODEL

Mathematical Model

A binary model is assumed, in which the device may exist in either of two states: success (operating) or failure. The cumulative failure distribution function, F(t), is stated in terms of a random variable time parameter, t.

$$F(t) = \text{Probability } \{\text{failure occurs at time} \geq t\} \tag{1}$$

For the binary model, the reliability, R(t), which is the probability of success is given by:

$$R(t) = 1 - F(t) = \text{Probability } \{\text{success at time} > t\} \tag{2}$$

The cumulative failure distribution function, F(t), has an associated failure probability density function, f(t). The hazard rate, z(t), is the instantaneous conditional probability of failure, given that failure has not yet occurred. These functions are related as follows [Ref. 2]:

$$F(t) = \int_0^t f(s)\,ds \tag{3}$$

$$R(t) = \exp\left[-\int_0^t z(s)\,ds\right] \tag{4}$$

$$z(t) = f(t)/R(t) \tag{5}$$

Physical Model

Failure is the inability to meet a specific performance requirement. Failure is design specific. The physical model begins with a set of performance criteria which define the failure state. Once performance criteria are established and failure defined, the mechanism of failure is addressed.

The failure of a nonrepairable device is often represented by three time periods. The initial period, in which the hazard rate is decreasing. The useful life, with a constant hazard rate, and wearout, with an increasing hazard rate.

Alternatively, one can characterize the failure of a device by the nature of the failure. Two primary types of failure are: (1) catastrophic and (2) grad[illegible]. The catastrophic or chance failure may be attributed to random, excessive loa[illegible] within the operating environment. It usually has a constant hazard rate. Gra[illegible] failure may be attributed to a constant, sometimes extreme, degrading environment. It has an increasing hazard rate.

The Distribution Function

To complete the quantitative reliability model, a distribution function is chosen which satisfies the physical model. The failure probability density function, f(t), determines the other functions (i.e., F(t), R(t) and z(t)) that comprise the mathematical model.

A distribution model, proposed by W. Weibull [Ref. 3], has found wide application in reliability. The model, known as the Weibull model, specifies the failure probability density function in terms of the continuous random variable, t, and the scaling and shape parameters, α and β. Depending on the choice of the shape factor the hazard rate, z(t), is: 1) decreasing as t increases ($dz/dt < 0$, $\beta < 1$), 2) constant ($dz/dt = 0$, $\beta = 1$), or 3) increases as t increases ($dz/dt > 0$, $\beta > 1$). The mathematical model is ($\alpha > 0$, $\beta > 0$, $t \geq 0$):

$$f(t) = \alpha \beta t^{\beta-1} \exp[-\alpha t^{\beta}] \tag{6}$$

$$F(t) = 1-\exp[-\alpha t^{\beta}] \tag{7}$$

$$R(t) = \exp[-\alpha t^{\beta}] \tag{8}$$

$$z(t) = \alpha\beta t^{\beta-1} \tag{9}$$

The constant hazard rate ($\beta = 1$) Weibull model, reduces to the exponential distribution, with constant hazard rate, λ. The mathematical model for the exponential distribution is ($\lambda > 0$, $t \geq 0$):

$$f(t) = \lambda e^{-\lambda t} \tag{10}$$

$$F(t) = 1-e^{-\lambda t} \tag{11}$$

$$R(t) = e^{-\lambda t} \tag{12}$$

$$z(t) = \lambda \tag{13}$$

The constant hazard rate implies no change in a component over time (i.e., no degradation and no improvement). A component with a constant hazard rate is just as reliable, whether new or old.

Linearized forms of the Weibull and exponential models can be obtained. The linearized forms, used to determine distribution parameters (i.e., α, β, λ) from test data are:

$$\log_e[\log_e(1/(1-F))] = \log_e \alpha + \beta \log_e t \tag{14}$$

$$\log_e[1/(1-F)] = \lambda t \tag{15}$$

RELIABILITY TESTING

A reliability test program may range from a planned collection of field test and inspection data to well-controlled laboratory tests. The data collected (i.e., number of failures observed) must be sufficient to provide statistical confidence. It is also desirable to obtain the test data quickly. These goals are difficult to achieve with highly reliable components which do not fail quickly, and are expensive. Expense often limits the numbers tested. Difficulties are minimized by careful test design, and the use of accelerated tests [Refs. 2, 6] when possible.

Test Plans

Two basic test schemes are used: 1) tests with replacement, and 2) tests without replacement. Testing with replacement involves the same number of components under test throughout the prescribed test period, with replacement upon failure. Testing without replacement involves an initial test population which decreases as components fail. Such a test may be run until all components fail, or to some prescribed number of failures, which is less than the entire population under test.

Failure data (i.e., time to failure) are sometimes recorded in a continuous fashion, but more often at discrete time intervals; corresponding to tests and inspection intervals for field data, or observation intervals for laboratory tests.

Failure Data

Experience has shown that useful information is obtained from either the failure probability density function or the hazard rate presented in terms of time [Ref. 2].

For N items tested without replacement, and placed in service at time $t = o$, with $n(t)$ survivors at time, t_i, the empirical failure probability density function and hazard rate (assuming they are piecewise-continuous functions,) $f_d(t)$ and $z_d(t)$ are given by [Ref. 2]:

$$f_d(t) = \frac{[n(t_i) - n(t_i+\Delta t_i)]/N}{\Delta t_i}, \qquad t_i > t \geq t_i + \Delta t_i \tag{16}$$

$$z_d(t) = \frac{[n(t_i) - n(t_i+\Delta t_i)]/n(t_i)}{\Delta t_i}, \qquad t_i > t \geq t_i + \Delta t_i \tag{17}$$

The functions $f_d(t)$ and $z_d(t)$ are typically presented as histograms. The time parameters, t_i and Δt_i are chosen in some consistent manner. For example, Δt_i may be the interval between failure i-1 and i, with t_i being the time of failure i.

Interpretation of Data

The histogram plots are compared with various mathematical distributions of f(t) and z(t). The comparison of the histogram behavior with that of the analytic function leads to a choice of distributions. Once the distribution is chosen, various quantitative procedures are available to determine the constant parameters that define the mathematical reliability model.

The characteristics of the distribution functions are available from several sources [Refs. 2, 4]. Graphical methods are also described [Refs. 2, 5]. Analytic curve fitting procedures, including; least squares, moment, and maximum liklihood estimates, are also described [Refs. 2, 6, 7, 8].

A data analysis procedure, with examples, is given by H. F. Rondeau [Re[redacted] for the exponential and Weibull models. The method uses the linearized expre[redacted]ns presented in equations (14) and (15), and the estimator given below for F(t) (where: i, is the i-th failure which occurs at time, t_i, for N components tested without replacement).

$$F(t_i) = (i - 1/2)/N \qquad (18)$$

APPLICATION TO CONTAINMENT SEALS

Transportation Environment

There are a number of different types of seals used in packaging for radioactive materials. They include elastomers and metallic seals, and may be simple off-the-shelf items or complex special design items. What they all have in common, however, is their environment and the requirement that they maintain adequate containment. The transportation environment contains three distinct regions. They are: (1) handling, (2) normal conditions of transportation, and (3) accident conditions.

The handling environment includes the loading and unloading of a package, and is design specific. The exposure of the seal to harmful elements will depend on tools needed to perform opening/closing operations, complexity of the operations, skills of the operators, complexity and materials of the seal, etc.

The transportation environment includes normal conditions and accident conditions. Tests and performance requirements are specified in the IAEA regulations [Ref. 10] to cover these conditions. The normal conditions are generally characterized by continuous exposure to radiation, and continuous, but cyclic heating, pressure and vibrations. The accident environment is best characterized by the random occurrence of loads from impact, crushing, heat and pressure. The packaging must perform under normal conditions throughout its service life. The accident conditions may never occur, but the packaging must be able to survive a single accident event. Although the environments are specified the seal's performance is design dependent, since the packaging offers protection for the seal. [redacted]lso, the performance under accident conditions depends on the present condition o[redacted] seal due to its normal service history.

Performance Criteria

The physical performance criteria is specified to provide adequate containment under normal and accident conditions of transport. Since handling occurs within

the framework of normal operations normal conditions criteria should apply. Allowable releases for radioactive materials for normal and accident conditions are given in the IAEA regulations [Ref. 10]. The packaging is assumed to continue to meet the design specification while in service (i.e., no degradation) which in practice may not always be achieved.

Reliability criteria offers a means of quantifying the requirement of continued performance of containment seals over a specified service life. Reliability requirements are not set by the regulations [Ref. 10]. The designer may, therefore, set any desired reliability criteria. The guidance, which follows, is offered toward accomplishing that task. For normal conditions, one is interested in reliability of the seal for the duration of a shipment. Failure, if it occurs i[redacted]obably gradual. In addition the toxicity and form of the contents should be [redacted]ered. Single elastomer seals are capable of achieving reliabilities of 0.[redacted]500 for a 10-week period: redundancy could increase this to 0.99998 [Ref. 1]. For accident conditions one would not want unreliability of the seal to significantly increase the chance of release for this single event. So, reliability should be related to the probability of occurrence of an accident.

Test Plan

A general plan is proposed for designing a reliability test program for containment seals. The proposed plan is based on concepts that are developed or discussed in this paper. The suggested test program design elements are:

1. Establish physical performance criteria.

2. Establish reliability criteria.

3. Define the physical environments to be considered.

4. Determine the order of significance of each combination of physical and reliability criteria and physical environment (i.e., order by difficulty of meeting each combination, if possible).

5. Assuming limited resources and/or time, and based on the order of significance decide which criteria-environment set(s) to include in the test program.

6. Decide on a test scheme and data collection procedure (i.e., with or without replacement and continuous or discrete time interval observations).

7. Determine the number of items to be tested (laboratory) or observed (field), the total test time, observation interval, etc. This step requires a preliminary estimate of the component reliability to ensure that: 1) number under test is sufficient to produce timely results, and 2) the observation interval is sufficiently short to detect failures from the test set uniquely.

8. Reduce the test data to histograms of f(t) and z(t), and decide on an [redacted]appropriate distribution.

9. Using a linearization procedure, determine the constant parameters for the distribution chosen.

10. Document the results of the tests, specifying all assumptions, test conditions, and the applicability.

REFERENCES

1. W. H. Lake, "Reliability in Maintenance and Design of Elastomer Sealed Closures," Proc. 5th Int. Sym, Packaging and Transportation of Radioactive Materials, Las Vegas, NV, May 1978.

2. M. L. Shooman, Probabalistic Reliability: An Engineering Approach, McGraw-Hill, Inc., NY, 1968.

3. W. Weibull, "A Statistical Distribution Function of Wide Applicability," J. Appl. Mech, V18, March 1951, pp 293-297.

4. N. D. Singpurwalla, "Statistical Fatigue Models: A survey," IEEE Transa[redacted]s on Reliability, August 1971, pp. 185-189.

5. R. H. W. Brook, Reliability Concepts in Engineering Manufacture, John Wiley & Sons, NY, 1972.

6. C. O. Smith, Introduction to Reliability in Design, McGraw-Hill, Inc., NY, 1976.

7. M. R. Sampford, "Distribtuion Fitting For Reliability Estimation," Advances In Reliability Technology, UKAEA, February 1977.

8. J. E. Freund, Mathematical Statistics, Second Edition, Prentice-Hall, Inc., NJ, 1971.

9. H. F. Rondeau, "Short-Cut Reliability Analysis," Machine Design, September 21, 1978, pp 108-112.

10. International Atomic Energy Agency, Safety Series No. 6, Vienna, 1973.

EVALUATION OF MAXIMUM IMPACT DAMAGES ON TYPE B(U) SHIPPING PACKAGES

Wayne Gibson and Jiri Kotler
Atomic Energy of Canada Limited - Commercial Products
Ottawa, Ontario, Canada

INTRODUCTION

The IAEA Regulations[1] specify accident conditions for the mechanical and thermal tests to which Type B packages must be subjected. The package must be dropped in such an attitude that "maximum damage" will occur and the order of the two drop tests shall be such that the package will suffer such damage as will lead to "maximum damage" in the thermal test. The analytical and licensing engineers are thereby faced with the task of determining, and justifying, the drop attitudes which will produce these "maximum damages", in terms of the integrity of the radiation shielding and containment functions of the various types of shipping containers. As part of a program to demonstrate compliance of its shipping containers with the Regulations, Atomic Energy of Canada Limited - Commercial Products (AECL-CP) has drop-tested many of its Type B packages to evaluate impact damages. Representative test packages were chosen from three "families" of containers. Comparison of damages and predictions to the rest of the "family" forms the safety evaluation for demonstrating compliance.

AECL-CP CONTAINERS

The compact version of this report[2] describes the basic "families" of AECL-CP packages for transporting solid radioactive materials. Design, construction and materials are similar within a family, and therefore, damages incurred during the drop tests are also similar. Figures 1 and 2 illustrate two types of containers.

The containment systems are defined as follows: (i) Family 2 - the double-encapsulated sealed source capsule which is held in a lead-shielded "drawer"; (ii) Family 3 - the single- or double-encapsulated sealed capsules which are carried in a metal "basket", in the cylindrical cavity; (iii) Family 4 - these packages incorporate one of the previous containments. The capsule tubes are typically sealed by fusion-welded end caps. Dye penetrant weld inspections and ANSI tests for sealed sources[3] confirm leaktightness of the capsules. They are ultrasonically decontaminated and wiped clean prior to shipment.

There are two exceptions to the above. One model is designed specifically to carry mixed fission products from the "Slowpoke" research reactor. The F-231 carries only Cobalt-60 capsules from the power reactors to the AECL-CP production facility. The containment system, in these cases, constitutes the outer closure plug seal and bottom drainline, as well as the capsules themselves.

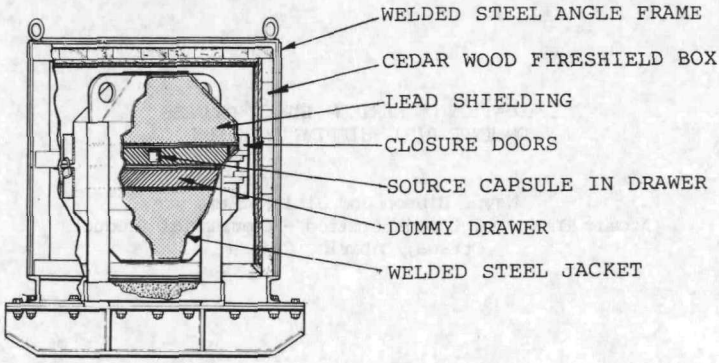


Figure 1. Family 2 Container

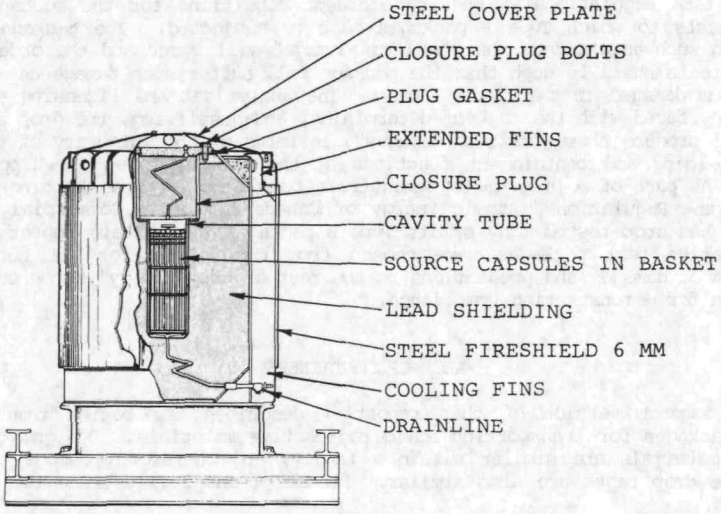


Figure 2. Family 3 Container

DAMAGE ASSESSMENT

There are many types of damage which can be incurred during impacts with unyielding target and the steel pin. Those pertaining to the AECL-CP shipping containers are listed in Table 1, according to three causes and which function they affect.

We are particularly concerned with demonstrating "retention of the integrity of containment and shielding, to the extent required by these Regulations[1]..." (para. 131). The integrity of the two main functions, shielding and containment, is assessed after each package has undergone the drop and fire tests, using the criteria in Table 2.

Table 1. Potential Modes of Failure for AECL-CP Containers

EFFECTED FUNCTION	CAUSE OF FAILURE		
	HIGH G-FORCES	LARGE DEFORMATIONS OF ENERGY ABSORBERS	DIRECT IMPACT OF COMPONENTS
Shielding Components	• Brittle Fracture of Shield Jacket • Lead Slump • Large Strain Loads on Closure Bolts	• Loss of Fireshield Effectiveness • Reduction of Distance Shielding	• Weld Fractures • Lead Shielding Crushes • Closure Bolts Shear • Distortion (Buckling of Cavity and Closure • Puncture/Tearing/Ripping/Crushing of Fire Protection • Detachment of Major Components
Containment Components	• Bending or Buckling due to Impact Loads • Brittle Fracture of Capsules		• Buckling Capsules • Closure Seal Distortion • Drainline Seal Distortion

Table 2. Assessment Criteria

	FUNCTION	
	RADIATION SHIELDING	CONTAINMENT
METHOD	• measurement surveys of radiation fields with source inside containers, after tests	• vacuum liquid bubble tests of inactive source capsules (carried inside container during tests) • gas pressure tests or helium tests with soap bubble tests of cavity, closure seals/drainline
LOSSES	• identification of increases over "before test" levels and locations	• air bubbles indicating failure of welds, tube • soap bubble leaks, loss of internal pressure or helium detection
SERIOUSNESS (PASS/FAIL)	• % increases related to Regulations (para. 229)	• equivalent ANSI[4] leakage rates related to Regulations leaktightness limits (para. 230)

DROP TESTS

The performance of these packages depends primarily on the ability of the outer structure to absorb energy, limit shock loads, and prevent direct impact of critical components and on the remaining fireshield effectiveness, after the drop tests and throughout the fire test. Consequently, they were dropped in such an attitude that the maximum damage would occur to the protective fireshield (and closure components) thereby maximizing the possibility of direct impingement of the fire flames and hot gases on the steel-encased lead container inside the damaged fireshield.

The impacts were recorded by a set of go/no-go "OMNI-G" shock indicators, mounted to the bottom skid. Most of the drops were photographed with 16 mm high speed movie film (1000 and 500 frames/second) and 35 mm black and white stills. The results are tabulated in the compact paper[2]. It should be noted that the g-levels recorded by the OMNI-G's are relative only, and do depend on the method of mounting and location of the gauges.

Photographs from actual high speed film frames of the 9 metre drops are shown in Figure 3, for the F143, and Figure 4, of the F231 containers. As expected, the wooden fireshield initially provides cushioning to the F143 lead container, but it easily crushes onto the body, at which time the highest g-loads are experienced by the package and its contents (200 g's). The package then bounces. Virtually insignificant damage was incurred during this drop test. Unfortunately, in the thermal test, the original two wood layers of the fireshield continued to burn long after the 30 minute furnace fire. The fireshield was therefore modified, by replacing the inside wood layer with a 12 mm thick layer of kaowool insulation. This improved the performance in the fire test greatly, but allowed the container to hit the pad with a much harder impact (>500 g), without benefit of the second wood layer.

The original F231 (Figure 4) did not have any protective fins or top cover. As a result, the top plate weld, encasing the lead, fractured (Figure 5). Two closure boltheads sheared off. The plug was pushed in about 1 cm, buckling the cavity tube wall. The vertical steel cooling fins were subsequently extended over the top shoulder and a steel cover plate was bolted over the plug, on all Family 3 containers. The substantial improvement in sustaining impacts is seen in the photograph of the F168 (Figure 6) after a similar inverted top edge drop. Although the fins are flattened, the outer jacket welds remain sound. Six fins loaded at 10° or less (foreground), have folded plastically in the double-hinge buckling mode, while the fins at the outer edge of the impact zone have folded in the single hinge bending mode, as found in the work of Irvine et al[5].

A GC220 research irradiator (Family 4) was dropped 9 metres onto its top end. The top shipping plate, retaining the "drawer" and source cage, sheared off and the source cavity tube was severely buckled. This package was considered to have failed the accident drop tests.

The simple and economical solution was to provide a transport "crush shield", fabricated with 16 steel fins welded to upper and lower load spreader plates, as shown in Figure 7. Curves for energy absorption of steel fins vs fin heights and impact angles, produced by Irvine et al[5], were used to design the shield. The assembly is bolted to the GC220 via its lifting eyebolts.

Not only did the crush shield prevent direct impact of the closure plate and direct loads on the drawer/cavity, but the deceleration g-loads measured by the OMNI-G gauges were much less than without the shield.

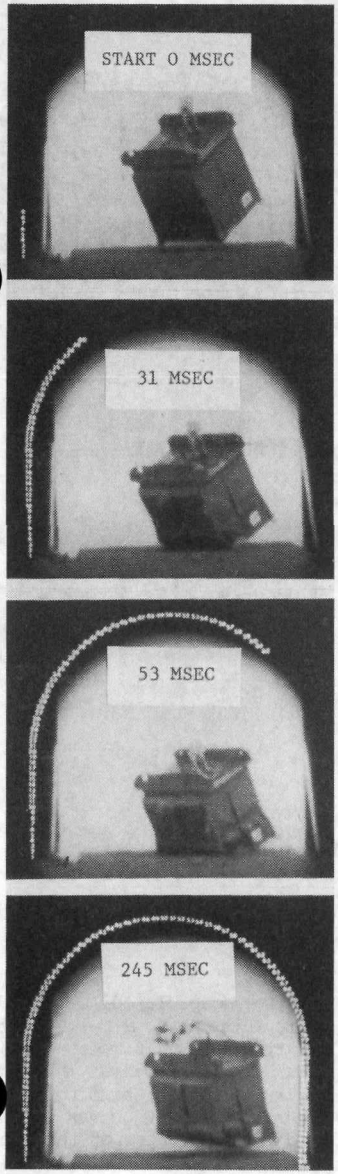


Figure 3. F143 9 Metre Drop

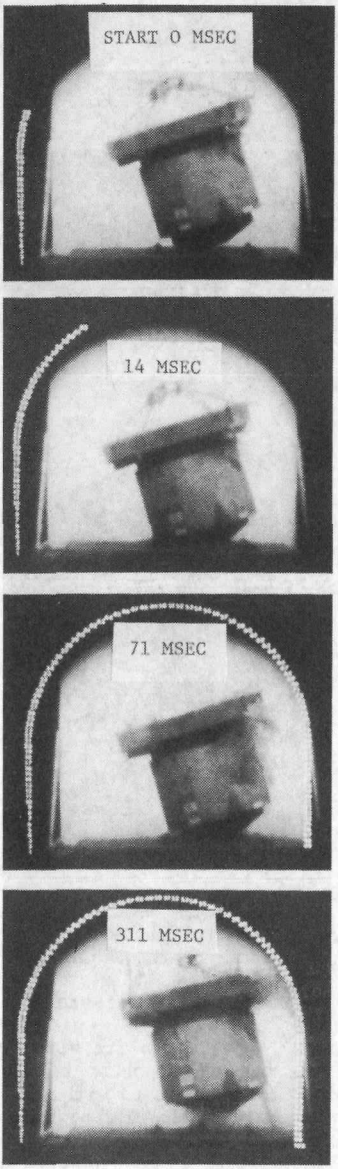


Figure 4. F231 9 Metre Drop

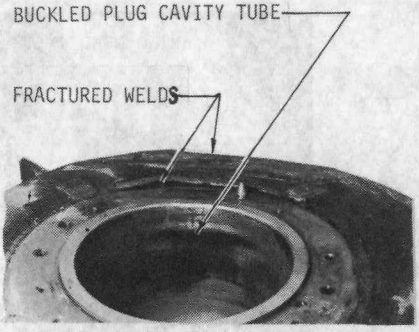


Figure 5. F231 Damage

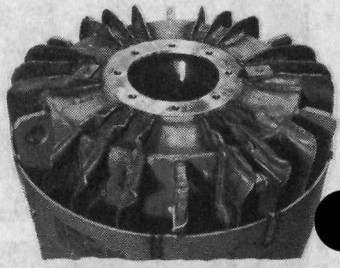
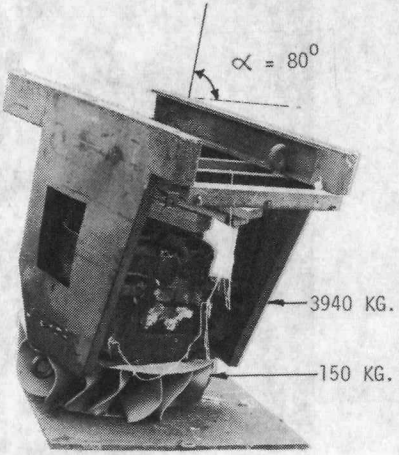


Figure 6. F168 Damage



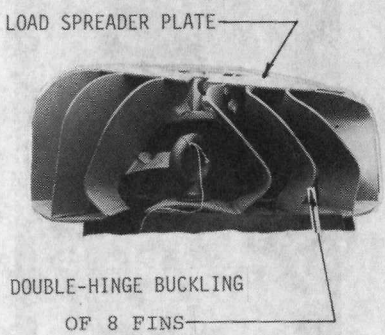


Figure 7. GC220 Crush Shield

Radiation surveys of all the test packages, with Cobalt-60 source capsules inside the cavity, proved that the integrity of shielding remains well within the allowable limit of the Regulations.

The sealed capsules were also tested. Ten single-encapsulated AC-191 "dummy" capsules, 21 cm long, in a metal carrier basket were dropped in a F-127 container (Family 3 type), onto its side from 9 metres. As the ends of the capsules overhang the basket supports, the horizontal drop attitude was chosen to test the ability of the tubes to withstand bending. There was permanent bending in [redacted] capsules, both ends. The highest g-loads recorded by the OMNI-G gauges were [redacted] ured in this attitude (750 g), which is considered a severe test of the containment integrity. It is estimated that the capsules experienced 2000-3000 g's. Two double-encapsulated C-188 "dummy" capsules, 45 cm long, in a basket, were dropped in the F-168 package, onto its top, to test and impacts. No noticeable buckling occurred. Vacuum liquid bubble tests of these capsules in an ethylene glycol liquid bath found no ruptures or leaks. The integrity of leaktightness and material containment was fully retained, for both capsule types.

DYNAMIC MEASUREMENTS

The analysis and predictions of impact damages require a knowledge of the variation of impact forces with time, to understand the processes observed in the high speed movies. The deceleration g-forces indicated by the shock gauges are not the complete forces. Durations of the forces acting on various parts of the packages are not recorded. There are a number of studies[5,6] of measurements of weights dropped onto steel fins, tube clusters and other crushable energy absorbers. However, the composite construction of the AECL-CP wood/steel shields and welded lead/steel containers introduces added complexity. We need to differentiate between the contribution of the various impact components and materials to cushioning the package.

We therefore are recording continuous measurements of g-force response with time, by use of an instrumentation system incorporating piezo-electric accelerometers mounted on two types of shipping containers with different impact and thermal protection and different closures (F143 and F168). The deceleration forces in three axes are recorded at two locations on each container, when dropped in three different attitudes, from a height of 9 metres onto the unyielding target. The output from the accelerometers is recorded on magnetic tape for the six impacts.

Prior to performing these drop test measurements, a 5-inch steel ball was instrumented with the tri-axial accelerometers and dropped repeatedly from various heights up to 0.5 metres onto the unyielding pad. The data from this test specimen is used to check and calibrate the instrumentation system. The 2500 g maximum output of the accelerometers was achieved when the steel ball was dropped from 0.5 metres.

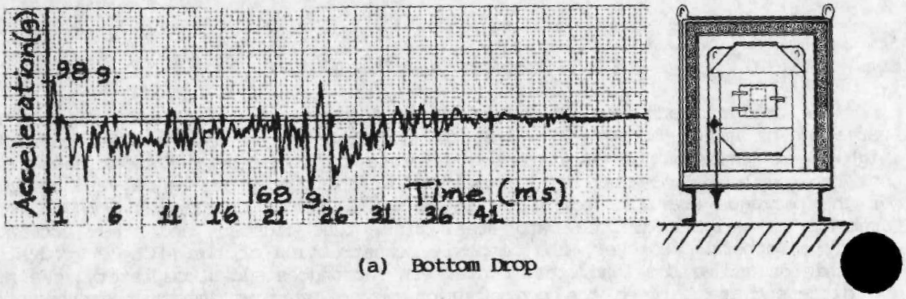
Plots of deceleration g-force vs time, for several axes on the F143 are presented in Figure 8. High speed films and a set of OMNI-G shock indicators simultaneously recorded the effects of these impacts. As can be seen, the peak g-accelerations are well below the OMNI-G levels, with 295 g's being the maximum recorded by the accelerometers in the 9 metre side drop. It is also evident that the significant cumulative deformation of the package somewhat reduces its structural rigidity for each following drop.

The significant amount of information accumulated in these drop tests will require much more interpretation than has been done to date, but it first appears that high g-loads are not as much a cause of damage, with the Family 2 packages, as direct impact.

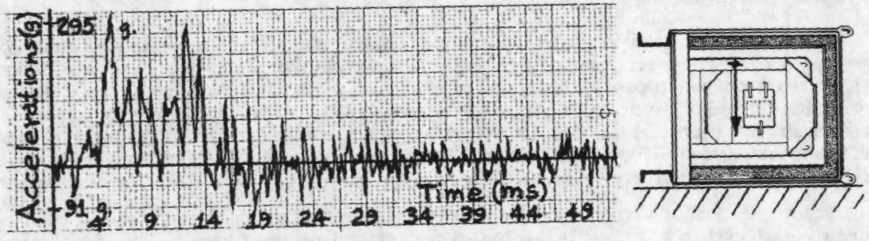
The comparable measurements of the F168 package are not available at time of writing this paper, but this analysis is also continuing.

SUMMARY AND CONCLUSIONS

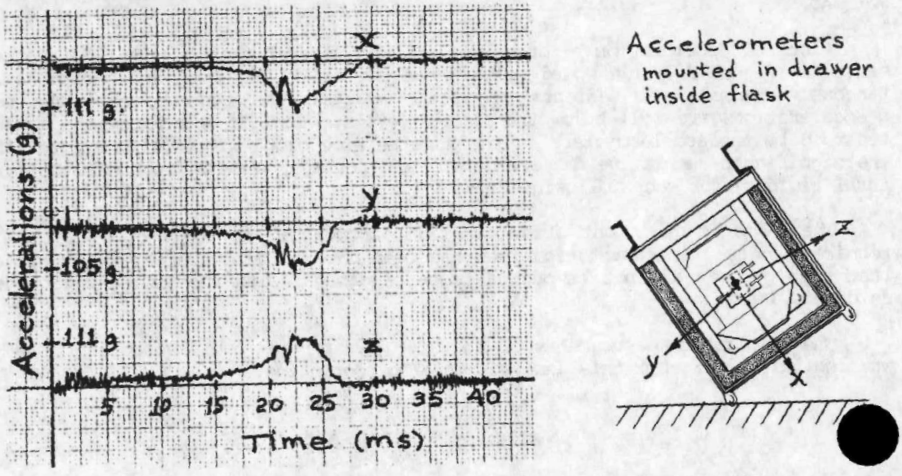
The prediction of damages to existing complex packages after the accident tests is difficult. The drop test program has provided valuable information about the AECL-CP Type B containers. Almost all of the outer protections have been modified for improved performance in the drop and fire tests and re-tested successfully. This has increased confidence in the useful safe life of the AECL-CP shipping containers.



(a) Bottom Drop



(b) Side Drop



(c) Top Edge Drop

Fig. 8. Dynamic Recordings of 9 Metre Drops of F143

An effective sacrificial energy absorber designed to limit impact g-loads and eliminate deformations of critical package components by direct contact (refer Table 1), in different drop attitudes, can minimize the burden on the design and licensing functions of proving that the package did sustain "maximum" possible damages. These "impact limiters" should be designed with pre-determined deceleration load-time response characteristics, to gradually slow the body to rest over a specified crush distance.

The on-going analysis of the dynamic measurements of the F143 and F168 drops will provide an understanding and the quantitative data necessary to apply in the design of crush shields, for new and existing containers.

REFERENCES

1. IAEA Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition, Safety Series No. 6.

2. "Evaluation of Maximum Impact Damages on Type B(U) Shipping Packages"; W. Gibson, J. Kotler; Compact of Patram '80 Symposium.

3. ANSI Standard N43-3 "Classification of Sealed Radioactive Sources"; August 1967.

4. ANSI N14.5-1977 for Leakage Tests on Packages for Shipment of Radioactive Materials.

5. IAEA-SM-147/21 "Destructive Tests Related to Development of LMFBR Fuel Shipping Casks"; A.R. Irvine et al, ORNL; Proceedings - Tests on Transport Packaging for Radioactive Materials, 1971; pp 587-602.

6. "Impact Protection Devices for Radioactive Material Shipping Containers", E.C. Lusk, R.J. Burian; 4th Proceedings of International Symposium on Packaging and Transportation of Radioactive Materials, 1974; pp 436-448.

CONTACT-HANDLED TRANSURANIC TRANSPORTATION SYSTEM STRUCTURAL ANALYSIS*

G. H. Lamoreaux, L. E. Romesberg, and S. H. Sutherland
Sandia National Laboratories, Albuquerque, NM, USA

T. A. Duffey
Private Consultant, Albuquerque, NM, USA

The TRansUranic PACkage Transporter (TRUPACT) is a Type-B overpack under development at Sandia National Laboratories / Transportation Technology Center (SNL/TTC), Albuquerque, New Mexico, USA. The TRUPACT is to be used in the transportation of contact-handled transuranic (CH-TRU) waste for which the United States Department of Energy is responsible.

The leading concept for the transportation system calls for both a truck and rail transported version of the TRUPACT. The rail version, as currently envisioned, is shown in Figure 1. Its predominant features include inner and outer tubular steel frameworks which are sheathed with steel plate and separated by rigid polyurethane foam. Both frames are fitted with hinged doors which utilize elastomeric seals and which are bolted in place during transport. The entire structure weighs approximately 11 tonne, carries a maximum payload of 19 tonne, and measures 7.3 x 2.7 x 3.0 m (L x W x H) externally. The truck version, which is geometrically similar to the rail version, has external dimensions of

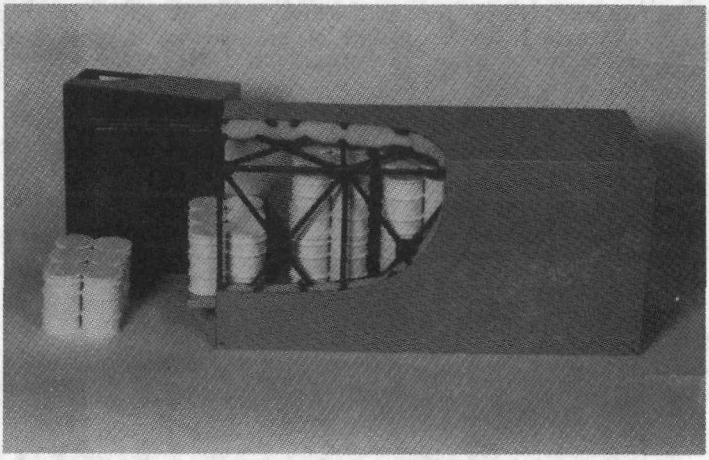


Figure 1. Photograph of a TRUPACT Model, Partial Cutaway Showing Internal Structure

*Work supported by the US Department of Energy under Contract DE-AC04-76-DP00789.

7.5 x 2.3 x 2.7 m, weighs 10 tonne, and has a maximum payload capacity of 13 tonne. The wall thickness of both the truck and rail versions is 36 cm on the sides and 91 cm on the ends.

Emphasis of the TRUPACT development program is currently on the rail version. However, since the two versions differ only in size and weight, analysis techniques, test procedures, and design concepts originated with the rail transported TRUPACT will also apply to the truck transported package.

The TRUPACT is being designed to provide structural and thermal protection to its contents in the simulated accident environments prescribed for Type-B packaging in transport regulations, such as IAEA Safety Series No. 6. Verification that the TRUPACT will survive these environments is being obtained through analysis and testing.

To date, end-on, side-on, and corner impacts of the loaded TRUPACT due to a 9 m drop onto an unyielding surface have been analyzed. In each case the analyses progressed from simplified hand approaches to successively more complex finite element calculations. The series of analyses for each of the three impact conditions are summarized in Table 1. The first analysis of each series represents the hand calculations which were carried out to obtain initial thicknesses of foam. The remaining analyses were performed using the dynamic and nonlinear analysis capabilities of ADINA, a structural analysis finite element computer program.[1]

TABLE I. TRUPACT Analyses Series for Three 9 m Drop Impact Orientations

SIMULATED IMPACT	ANALYSIS	MODEL COMPONENTS			
		Outer Frame	Foam	Inner Frame	Cargo
END-ON	1	-	Nonlinear Spring	Rigid Mass	Rigid Mass
	2	-	Truss Elements	3-D Truss Structure	Nodal Masses
	3	-	Truss Elements	3-D Truss Structure	Truss Elements
	4	3-D Truss Structure	3-D Solid Elements	3-D Truss Structure	Truss Elements
SIDE-ON	1	-	Nonlinear Spring	Rigid Mass	Rigid Mass
	2	-	Truss Elements	Nodal Mass	Truss Elements
	3	3-D Truss Structure	3-D Solid Elements	3-D Truss Structure	Truss Elements
CORNER	1	-	Nonlinear Spring	Rigid Mass	Rigid Mass
	2	3-D Truss Structure	3-D Solid Elements	3-D Truss Structure	Nodal Masses

The end-on impact analysis was initiated under the assumption that the inner and outer frames were structurally independent. That is, the outer frame would absorb its own impact energy by deforming plastically and could therefore be neglected when investigating the response of the inner frame. Initial foam dimensions and crush strengths were selected from the results of the hand calculations (analysis 1). The response to the 9 m end-on drop test was predicted assuming that the cargo was rigid and the frames were independent (analysis 2). Following this, a cargo model which allowed deformation and energy dissipation was included in the finite element model and the response of the foam and frame were reassessed, again under the assumption that the frames were not structurally coupled (analysis 3).

A comparison of the results from these two analyses revealed that the loads imparted to the inner frame were more severe for a deformable cargo model than for a rigid cargo. The foam crush for the flexible cargo was roughly half that for the rigid model. Further, the frame members nearest the impacting end yielded when the flexible cargo model was used but remained elastic for the rigid cargo calculation. This can be explained by the fact that while the cargo does not load the frame directly, it does affect the way in which the foam crushes. This, in turn, directly affects the deceleration of the inner frame. Since a rigid cargo will cause more foam crush than a deformable cargo, the rigid cargo will also cause smaller decelerations. Therefore, the loads transmitted to the inner frame during impact will be smaller for the rigid cargo model than for the deformable case.

To determine the extent of coupling between the two frames, a 1/4-scale model of the TRUPACT was fabricated and the inner frame was forced to move longitudinally relative to the outer frame in an essentially static manner. Since the magnitude of the force required to cause this motion was more than twice that theoretically necessary to crush the foam at the end of the model, it was evident that considerable shear forces were acting on the sides of the inner frame. This indicated that the coupling between frames was significant and should not be neglected. The final analysis of the series was then conducted using a model that consisted of both frames coupled together with three-dimensional solid elements which represented the foam (analysis 4). The foam crush versus time as predicted with this model is plotted in Figure 2. Also shown in this figure is the final foam crush predicted by an actual 9 m drop test of a quarter scale model TRUPACT which used a .345 MPa foam. Note that the foam only crushed 15 percent and 9 percent of its total available thickness of 0.9 m for the 0.345 MPa and 0.690 MPa cases, respectively. The comparison between analysis and test is within 3 percent for total crush.

While the analytical model produced excellent results with respect to the foam crush, it did not do as well when predicting strain levels in the inner frame. As an example, the strain history of the most severely stressed element of the inner frame, element 125, is plotted in Figure 3. This element was located nearest the impacting end and oriented parallel to the TRUPACT's motion prior to impact. The peak strain as taken from Figure 3 is approximately 1.5 percent. Since the truss member is 185 cm long this represents an overall reduction in length of the member of 2.8 cm. Although strain measurements were not taken during the quarter-scale 9 m drop test, inspection of the post-test model produced no visible evidence that the inner frame had yielded. The discrepancy can probably be attributed to the interface way the frames were coupled together and the fact that the inner steel skin was neglected in the analytical model. Evaluation of the finite element results revealed that the inner frame had moved approximately 2 cm relative to the outer frame at peak foam crush. This indicates that the inner frame was, to some extent, driving into the frontal foam and thereby generating forces large enough to yield the members nearest the impact surface. However, the trace from a linear variable differential transformer (LVDT) used to measure actual relative motion between the frames during the 9 m drop test showed that essentially no movement took place. Thus it appears that in the analytical model, the coupling is not as rigid as it

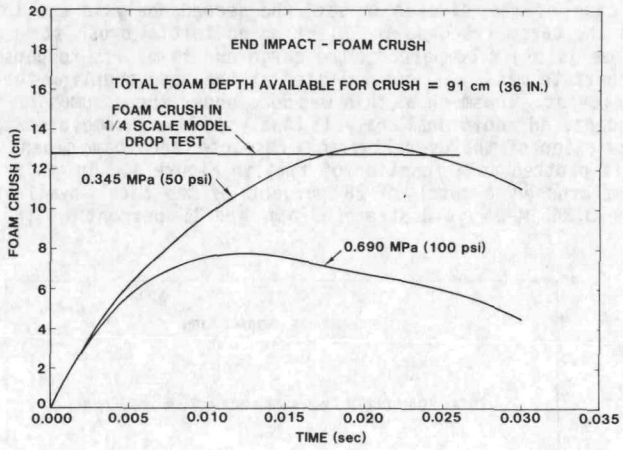


Figure 2. Plot of foam crush versus time as predicted by ADINA for the end-on impact 9 m drop test of the TRUPACT, for 0.345 and 0.690 MPa strength faoms.

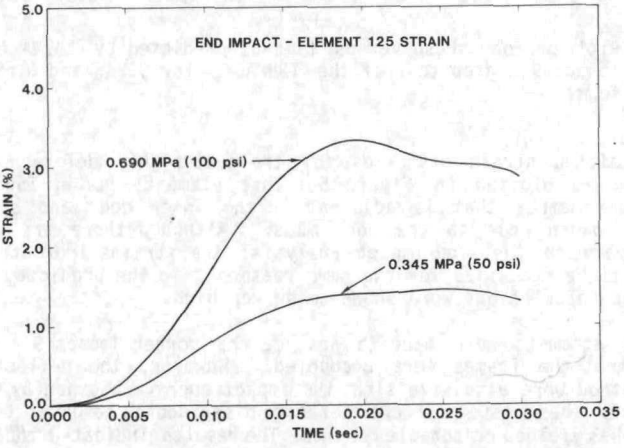


Figure 3. Plot of strain versus time as predicted by ADINA for the most severely stressed element of the inner frame due to the end-on impact 9 m drop test of the TRUPACT, for 0.345 and 0.690 MPa strength foams.

should be and artificially high loadings on the inner frame are predicted as a result.

In the case of the side-on impact, the second analysis was initiated to take into account the cargo response in selecting an initial crush strength of the foam. Since the frame is rigid compared to the cargo and foam, its response was neglected and the appropriate mass was concentrated at the node coupling the cargo elements to the foam element. Inasmuch as this was done under the assumption that the frames were independent, an additional analysis (Analysis 3) was completed to obtain a more accurate prediction of the overall frame response and foam crush. The predicted foam crush is plotted as a function of time in Figure 4. In this impact orientation the foam crushed a total of 28 percent of the total available thickness of 20 cm for the 0.345 MPa yield strength foam and 21 percent of the stronger foam.

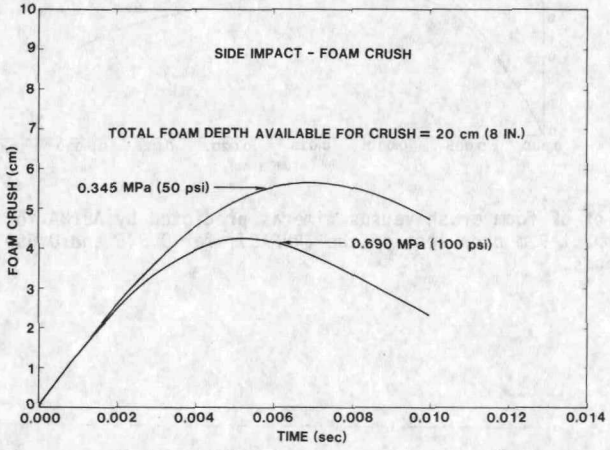


Figure 4. Plot of foam crush versus time as predicted by ADINA for the side-on impact 9 m drop test of the TRUPACT, for 0.345 and 0.690 MPa strength foams

The predicted strain histories for the most highly deformed element of the inner frame are plotted in Figure 5. This element (number 155) represents a diagonal frame member that is adjacent to the inner door and suffers a severe compressive loading due to the door mass. Although there are no experimental data to compare to the side impact analysis, the strains indicated in Figure 5 are thought to be excessive for the same reasons that the predicted strain levels in the end-on calculations were shown to be too high.

Several attempts were made to analyze the corner impact 9 m drop under the assumption that the frames were uncoupled. However, the deflections predicted with this method were excessive since the impact energy absorbed by the outer frame was neglected. These attempts were therefore discarded in favor of the more complex model which has yielded reasonable results. The results indicated that the impacting end of the TRUPACT in the 9 m corner drop would sustain considerable deformation. The degree of deformation is illustrated with the data shown in Figure 6. In this figure the approximate displacement of node 1 into the TRUPACT during impact is plotted as a function of time.

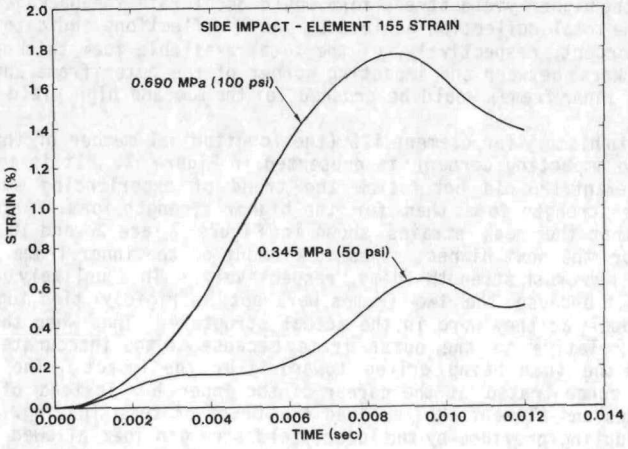


Figure 5. Plot of strain versus time as predicted by ADINA for the most severely stressed element of the inner frame due to the side-on impact 9 m drop test of the TRUPACT, for 0.345 and 0.690 MPa strength foams.

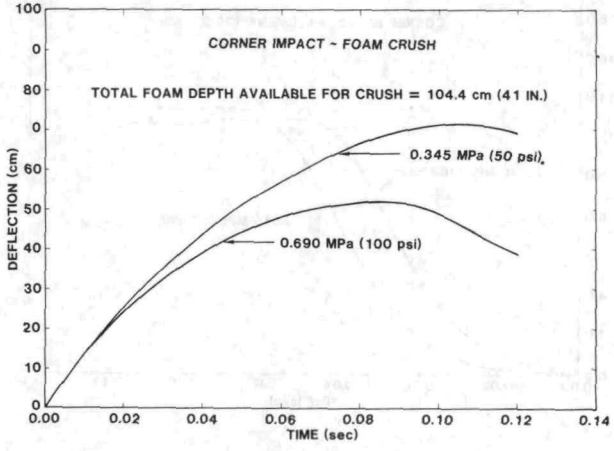


Figure 6. Plot of foam crush versus time as predicted by ADINA for the corner impact 9 m drop test of the TRUPACT, for 0.345 and 0.690 MPa strength foams.

For 0.345 MPa yield stress foam a maximum total displacement of node 1 is approximately 71 cm occurring 0.105 sec after impact. A maximum total displacement of 52 cm at 0.083 sec is predicted when 0.690 MPa foam is considered. As anticipated, the higher yield stress foam would decelerate the package more rapidly and lessens the total deflection achieved. These deflections indicate that 68 percent and 50 percent, respectively, of the total available foam thickness of 104 cm (the foam thickness between the impacting corner of the outer frame and the nearest corner of the inner frame) would be crushed for the low and high yield stress foams.

The strain history for element 125 (the longitudinal member of the inner frame nearest to the impacting corner) is presented in Figure 7. It is interesting to note that element 125 did not follow the trend of experiencing smaller strains for the lower strength foam than for the higher strength foam. Also of interest is the fact that the peak strains shown in Figure 7 are 5 and 10 times greater than those for the next highest stressed element of the inner frame for the 0.690 MPa and 0.345 MPa crush strength foams, respectively. This unlikely behavior probably came about because the two frames were not as rigidly tied together in the analytical model as they were in the actual structure. Thus when the inner frame began moving relative to the outer frame because of the inaccurate coupling, it collided with the foam being driven toward it by the impact. The impact forces were thereby concentrated at the corner of the inner box instead of being spread along its sides and element 125 received the brunt of this loading. Furthermore, the weaker coupling provided by the lower yield strength foam allowed larger forces to be concentrated at the corner of the inner frame and this, in turn, resulted in the higher strain levels seen in element 125 for the lower strength foam than for the higher.

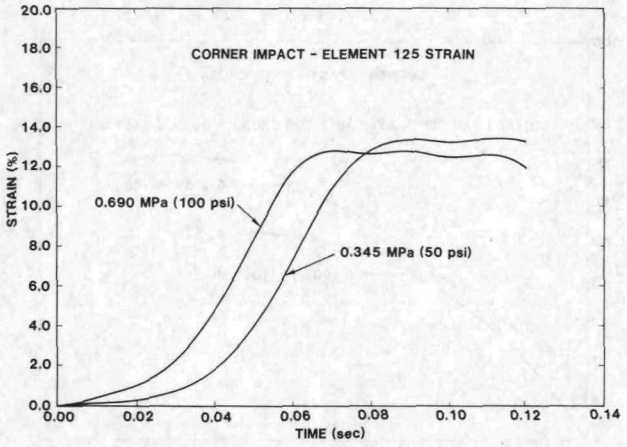


Figure 7. Plot of strain versus time as predicted by ADINA for the most severely stressed element of the inner frame due to the corner impact 9 m drop test of the TRUPACT, for 0.345 and 0.690 MPa strength foams.

Preliminary investigations concerning the resistance of the TRUPACT to the IAEA Safety Series No. 6 puncture test (a drop of 1 m onto a 15 cm diameter steel pin) have also been completed. Static analyses based on the small deflection theory of laterally loaded triangular and rectangular plates on elastic foundations were used to estimate the effect of plate shape and load location on the plate deflection achieved. The results indicated that plate geometry and load location were of minimal importance if the load center was more than one local diameter from the plate boundary.

A series of more appropriate dynamic analyses (based on energy conservation) were performed for centrally loaded circular plates on plastic foundations. In this approach it was necessary to postulate a deformed shape for the plate. In Figure 8 a plausible plate profile is shown. Also shown in the figure is the predicted peak deflection, W_o, of the plate for this profile as a function of plate and foam yield stresses for a plate 0.6 m in radius (r_b) and 0.35 cm thick. With this approach unacceptably large deflections are predicted (available thickness of 200 cm) and penetration of the TRUPACT containment is probable. The dynamic analysis has not yet been extended to include the puncture resistance offered by the plate on the inner frame of the TRUPACT or the possibility of additional plates in the package walls. A comprehensive test program is underway to confirm the validity of the current analysis approach in considering the puncture problem.

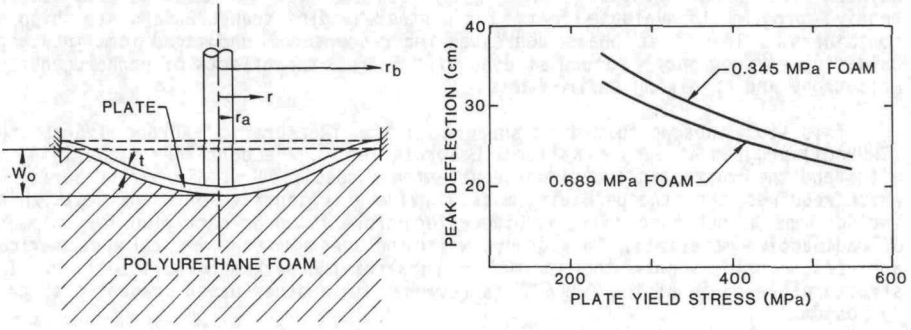


Figure 8. Plate profile assumed and predicted peak deflection of plate in TRUPACT puncture test analysis.

REFERENCE

K. J. Bathe, "ADINA - A finite Element Program for Automatic Dynamic Incremental Nonlinear Analysis," Report 82448-1, Massachusetts Institute of Technology, Cambridge, MA, September 1975.

ONE-DIMENSIONAL THERMAL RESPONSE MODELING OF A TRANSURANIC FOAMED OVERPACK SYSTEM TO A FIRE*

K. E. Suchsland, K. C. Kwong, E. F. Fretter
Acurex Corporation/Aerotherm Aerospace Systems Division, Mountain View, CA, USA

R. D. Boyd, I. Auerbach, H. R. Yoshimura
Sandia National Laboratories, Albuquerque, NM, USA

INTRODUCTION

To date, full scale testing has been used for qualifying foamed overpack containers to a prescribed thermal environment. For large containers such as those that might be used for the shipment of contact-handled transuranic wastes, costs for the procurement of test containers and performance of full scale tests may be substantial. Hence, an accurate thermal model is needed for predicting the response of the container wall which consists of polyurethane foam sandwiched between two steel plates. This study reviews the first phase of a comprehensive program to evaluate foams for contact-handled transuranic waste shipping containers. The first phase addressed the response of undamaged foam in a non-oxidizing environment.[1] Future studies will address the effects of nonhomogeneity, anisotropy and oxidizing environments.

This study was performed in support of the TRansUranic PACkage Transporter (TRUPACT) program at Sandia National Laboratories.[2] The container is designed to withstand the hypothetical accident environments specified in IAEA Safety Series 6, which requires that the packaging must survive a sequential series of tests which include impact, puncture, fire, and immersion before it can be certified for shipment of radioactive materials. This study, which includes both experimental and numerical analyses, considers only the thermal response of the packaging wall section. The structural response of the TRUPACT is reviewed in another paper presented at this symposium.[3]

The TRUPACT concept consists of multiple levels of containment: primary (interior steel wall), secondary (drums or boxes, drum liners, polyethylene bags) and tertiary (exterior steel wall). A truss system provides rigidity and support for the inner and outer containment walls. The polyurethane foam provides impact and thermal protection to the contents. The inner containment is not directly attached to the outer containment but is supported by the polyurethane foam. The outer containment is designed with fusible plugs to allow relief of internal pressure generated from pyrolysis gases formed when the foam decomposes during a fire associated accident condition. A schematic drawing of the TRUPACT is presented in Figure 1.

The foam selected for this study is commercially available and was obtained from General Plastics Manufacturing Company of Tacoma, Washington. Two foam densities (64 and 96 kg/m^3) were representative of the strength of foam (0.689 and 1.379 MPa respectively) which could be used in the packaging. The foam is closed cell in nature and may be foamed in place.

*Work sponsored by the U. S. Department of Energy under Contract DE-AC04-76DP00789.

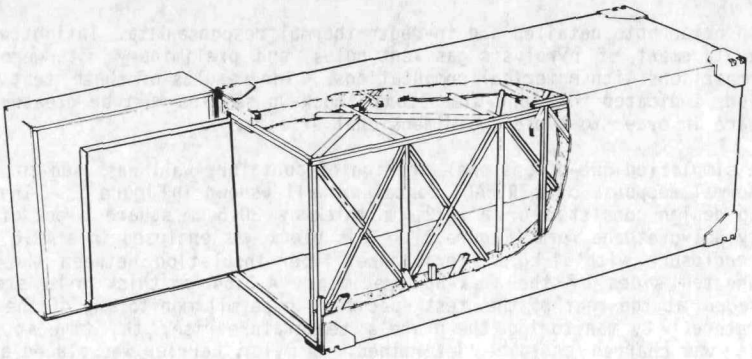


Figure 1. Isometric of TRUPACT Container

EXPERIMENTAL MODEL

The experimental investigation of the thermal response of a prototype TRU waste container, when subjected to a fire environment, is a formidable task. In such an investigation many competing complexities exist, such as: three-dimensionality, boundary condition determination, foam decomposition and shrinkage, transpiration, and combustion. However, a well defined one-dimensional experimental investigation of a section of the container could be used to generate data to produce a thermal model. When used in complex multidimensional numerical codes, such a thermal model could be used to predict the response of prototype containers. Effects of nonhomogeneity and anisotropy can also be included in the model. Nonhomogeneity can be examined one-dimensionally by evaluating the repeatability of the property data from samples of different sizes. Anisotropy can be addressed by evaluating the repeatability of property data when the samples are tested at various directions (e.g., orthogonal directions). With the above rationale, the present study is a one-dimensional investigation of a section of the TRU waste container when subjected to an almost step change in external surface temperature. Further, it is assumed that the foam is undamaged (i.e., uncrushed) and exists in a nonoxidizing environment (i.e., secondary container undamaged). Effects of nonhomogenity, anisotropy and oxidizing environments will be considered in future studies.

CHARACTERIZATION AND MOCK-UP TESTS

Initially a series of oven tests were performed to determine the response of virgin foams to low heating rates. The preliminary results from these tests, when compared with higher heating rate tests, indicated that foam decomposition is highly heat rate dependent. At low quasi-steady heating rates, the foam changes to a [illegible]ing black viscous liquid which swells on heating and contracts upon cooling. [illegible]ry high heating rates, pyrolysis gases are formed as the foam decomposes and a 20% by weight residual char remains. The structure of the char produced in the higher heating rate tests varies from a fine cellular structure at the interior surface to large voids in the exterior. This porous char acts as a thermal radiation shield which reduces significantly the radiation transport and thermal penetration rates and hence reduces the heat load to the container contents.

Additional characterization tests, involving one-dimensional heating of a 10.2 cm cube sample, served as an aid in designing a mock-up test configuration which

would provide both detailed and in-depth thermal response data. Insight was gained on the placement of pyrolysis gas vent holes, and preliminary data were provided for comparisons with numerical computations. The results of these tests and computations indicated that one-dimensional mock-up samples must be greater than 30 cm square in order to avoid two-dimensional effects.

A simplified one-dimensional section of container wall was used to determine the thermal response of a TRUPACT container wall (shown in Figure 1). The selected mock-up design consisted of a 10.2 cm thick by 30.5 cm square block of 64 kg/m^3 density polyurethane foam (Figure 2). This block was enclosed in a 40.6 cm square steel enclosure with 64 kg/m^3 ceriblanket fiber insulation between the enclosure and unheated sides of the mock-up specimen. A 0.64 cm thick mild steel plate was placed at the rear of the test specimen to permit monitoring of the ba[redacted]l temperature. By monitoring the plate's temperature rise, the time at wh[redacted]l the foam was charred could be determined. A mylar barrier was placed arou[redacted]he girth of the test specimen to prevent water from condensing on the steel back plate. Nitrogen purge lines were attached to the enclosure to provide an inert (nonoxidizing) environment. Exhaust vents at the sides and top of the enclosure were used to remove the pyrolysis gases. Thermocouple instrumentation in the foam provided in-depth and planar monitoring of the thermal penetration rate and the one-dimensionality of the thermal response.

The front surface of the mock-up box was heated by an array of thirty-six quartz-halogen lamps. Six specular aluminum lamp holders, each containing six lamps, were used. When assembled, this array provided a 40.6 cm x 53.3 cm heat source with an average power output of 20.6 W/cm^2. Figures 3a and 3b show the assembled array of quartz lamps, and Figure 4 shows the experimental configuration. The holders are individually water cooled; additional ports exist in the holders for gas cooling the quartz windows, lamp end seals, and lamp envelopes with nitrogen. The lamp array could be operated manually or automatically by a feedback control loop. A thermocouple located at the center of the back face of the mock-up box was used in the feedback loop to automatically control the power output from the lamps.

An 80 channel Vidar digital data acquisition system was used for recording thermocouple test results. This system produces a computer-compatible magnetic tape for subsequent computerized data reduction. In addition, a Honeywell 1858 Visicorder was used for monitoring the temperature during the experiment.

At time zero the front surface of the mock-up box containing the 64.0 kg/m^3 density foam was subjected to an almost step change in temperature (~ 1100 K). After 2260 s the back plate temperature had increased to 450 K and the pyrolysis gas venting had stopped for over 1000 s. It was concluded that charring was complete and that sufficient backplate data had been collected to facilitate analysis. The recorded temperature responses of the in-depth thermocouples are shown in Figure 5. The locations of these thermocouples are shown in Figure 6.

ANALYTICAL MODELING

The Aerotherm Charring Material Response and Ablation Program, Version 3, (CMA3) was selected as the basic modeling tool. CMA3 is a coded procedure for calculating the in-depth thermal response of a charring and ablating material. It requires as input: geometry, initial temperature, heating rate, and initial density. Additional temperature-dependent thermal properties such as density,

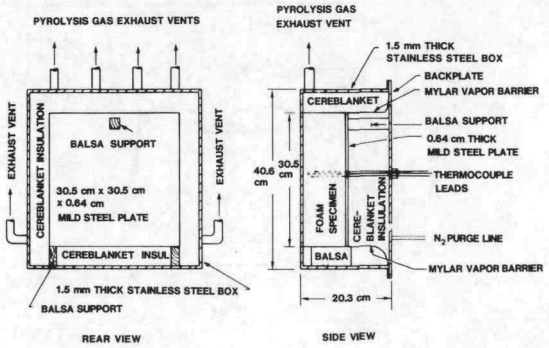
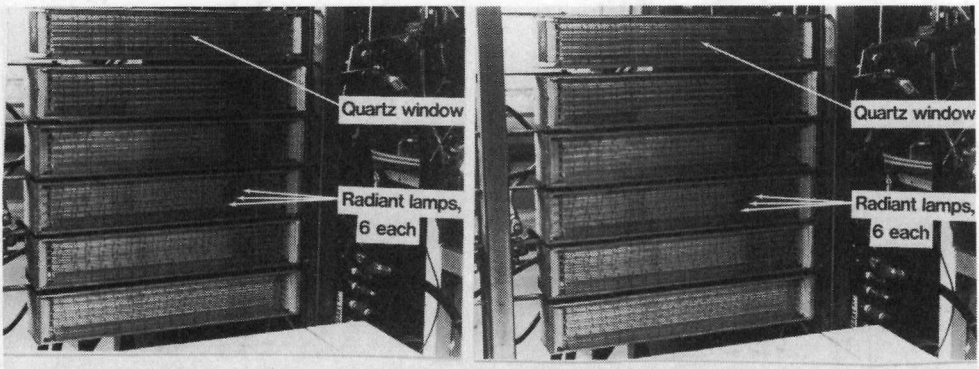


Figure 2. Cutaway views of the one-dimensional mock-up test configuration



a. (front view)

b. (rear view)

Figure 3. 50 kW quartz-halogen lamps

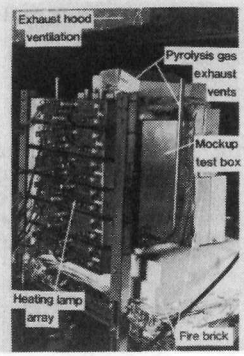


Figure 4. Experimental setup with test sample inside the mock-up test box

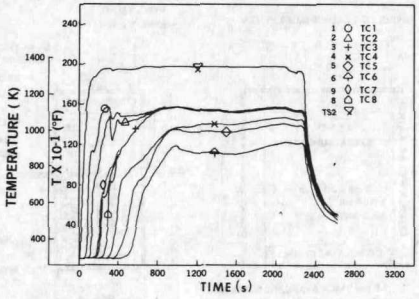
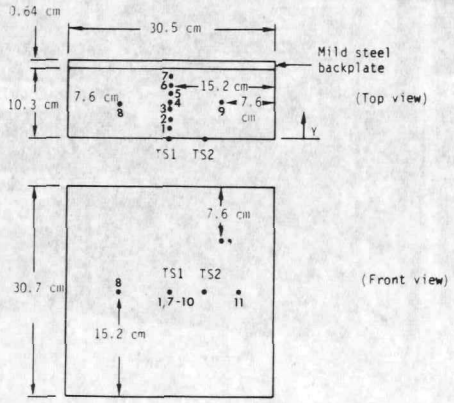


Figure 5. Temporal response of thermocouples located in-depth in foam sample (see Figure 6 for location; integers identify different thermocouples)



Thermocouple Number	Depth, y (cm)
TS1	0.00
TS2	0.00
1	1.09
2	2.13
3	3.53
4	5.69
5	6.50
6	7.44
7	8.69
8	4.38
9	3.68
10	10.92
11	10.92

Figure 6. Location and dimensional specifications of thermocouples and mock-up test sample

specific heat, thermal conductivity, heat of reaction, and values for the Arrhenius parameters are also needed. The specific heat, heat of reaction, density, and values for the Arrhenius parameters were obtained experimentally using differential scanning calorimetry (DSC) and thermogravimetric analysis (TGA) techniques. Estimates for the effective thermal conductivity of the polyurethane foams were obtained separately for the charred and uncharred foams. These estimates were made using temporal temperature response data obtained from the mock-up test.

The numerical computations were used to: (1) aid in establishing (using the experimental results) values of the effective thermal conductivity for charred and uncharred polyurethane foams; (2) compare the numerical predictions, using the thermal model for the foam, with the experimental results; and (3) make predictions of the one-dimensional thermal response of the TRU waste container when subjected to a step change in temperature of 1073 K for at least 1800 s (minimum regulatory requirement for the TRU waste container).

Because of the uncertainty of the flow direction of the pyrolysis gases in the experiments, transpiration cooling effects were neglected. Therefore, upon decomposition of the foam it was assumed that the pyrolysis gases found their way [redacted] fissure and escaped before interacting thermally with the char. Based on [redacted]-test experimental observations, a radiation gap was included in the model at [redacted] interface between the back face of the foam sample and the backplate.

Comparative numerical computations were made for the experiments using the dimensional specifications shown in Figure 6 and the front surface temperature as recorded by the surface thermocouple, TS2. As shown in Figure 7, the agreement between the numerical computations and experiment is best before foam decomposition and near the end of the experiment. It is believed that the poor agreement during decomposition was due to some extent to the neglect of transpiration effects and the uncertainty in the effective thermal conductivity, which is a strong function of temperature and the coupled chemistry of decomposition. The agreement between computations and the experimental data was enhanced when the foam decomposition was assumed to occur in a three-component reaction, and after (updated work) a better data fit was obtained for the TGA data.

Additional computations were made for 64 kg/m^3 foam to predict the effect of foam thickness on the one-dimensional thermal response of (1) the primary containment wall, and (2) a portion of TRU waste. A schematic diagram of the container wall and contents is presented in Figure 8. It should be noted that the air gap thickness indicated in Figure 8 is an effective thickness which would result if all the waste could occupy a volume with no air gaps. In addition, the term paper refers to materials such as paper, rags, glass, etc. The 64 kg/m^3 foam thicknesses evaluated were 0, 10, 20, and 35 cm. Predictions were made for the cases where the exterior wall was exposed to 1111 K (2000 R) and 1333 K (2400 R) heat sources (Figure 9). Both temperatures were in excess of 1073 K, which is specified in the regulatory criteria.

For the case where the external wall is instantaneously exposed to 1111 K and the foam thickness is 10.2 cm, the temperature rise of the interior wall will be less than 20 K after 1800 s and less than 352 K after 3600 s. When the foam thickness is increased to a value of 20.3 cm, there is no change in temperature of the interior wall after 3600 s. Next, the case is considered where the exterior surface is instantaneously brought to a temperature of 1333 K and the foam thickness is 20.3 cm. For this case, the temperature rise of the interior wall is 0 K after 1800 s and 23 K after 3600 s.

[redacted] Figure 10 shows the mean thermal response of the water which makes up a portion [redacted] the waste. Again, the foam thickness is used as a parameter. In cases where the foam thickness is 10.2 cm, the temperature of the water remains below the boiling point for a time of 3600 s. For cases where foam thickness is considerably less than 10.2 cm, perhaps due to impact damage or puncture damage or to the presence of partial voids, significant thermal penetration into the waste is possible. Work is underway to model the radiation transport through the waste gases for the latter case.

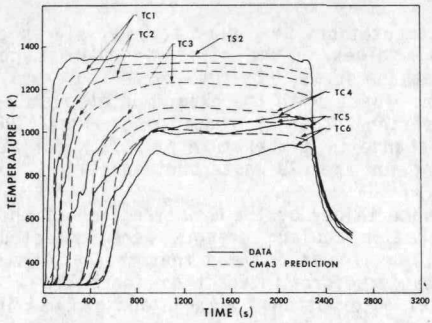


Figure 7. Comparison of one-dimensional numerical predictions with the experimental temperature response in the mock-up test sample. (Thermocouple location defined in Figure 6.)

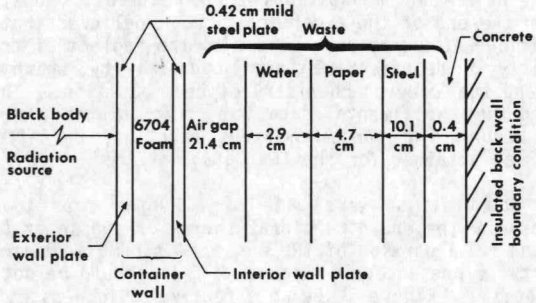


Figure 8. One-dimensional model of the TRU container wall and contents.

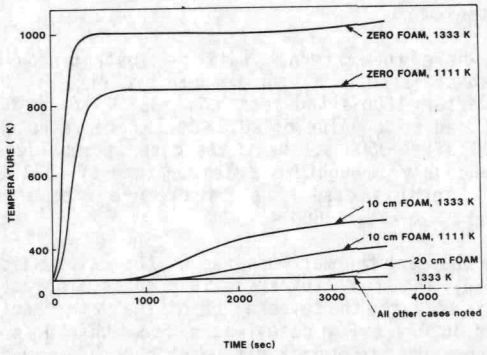


Figure 9. One-dimensional predictions of interior wall temperature response when the exterior wall is subjected to a step change in temperature (1111 or 1333 K). These predictions do not include modifications associated with the updated work mentioned previously.

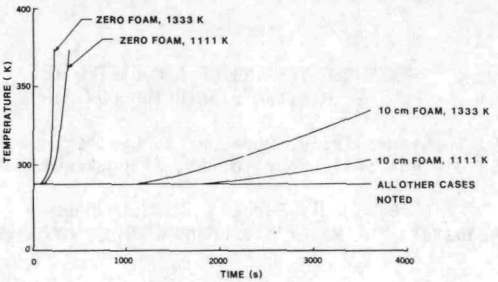


Figure 10. One-dimensional predictions of the mean temperature response of the water which is contained inside the TRU container when the exterior wall is subjected to a step change in temperature (1111 or 1333 K). Also, see Figure 8 for wall configuration.

CONCLUSIONS

Procedures have been established for modeling the thermal response of TRU container walls exposed to a fire environment. The effort included simulation testing and thermal modeling of the wall material. In this study, both testing and modeling were directed at determining a one-dimensional thermal model for undamaged polyurethane foam. The foam was assumed to exist in a nonoxidizing environment and was exposed to an almost step change in surface temperature.

Although preliminary, the results of this study indicate that if the TRU waste container wall includes a polyurethane foam (64 kg/m^3 density) of thickness greater than 20 cm and the wall is otherwise undamaged, there will be no change in the waste content temperature where the container is subjected to a surface temperature as high as 1333 K for times less than 3600 s. Further improvements are needed in the thermal model to include transpiration, better estimates of the temperature-dependent thermal conductivity, effects of damaged wall structure and radiation absorption effects for the charred foam.

REFERENCES

1. K. E. Suchsland, K. C. Kwong, and E. F. Fretter, "Thermal Characterization of TRU Waste Shipping Container Wall," Acurex Final Report FR-79-21/AS, Aerotherm Corporation, Mountain View, CA, to be published.

2. R. G. Eakes, G. H. Lamoreaux, L. E. Romesberg, S. H. Sutherland, and T. A. Duffey, "TRU Waste Transportation Package Development," Sixth International Symposium on Packaging and Transportation of Radioactive Materials, Berlin, Federal Republic of Germany, November 10-14, 1980.

3. G. H. Lamoreaux, L. E. Romesberg, and S. H. Sutherland, "Contact-Handled Transuranic Transportation System Structural Analysis," Sixth International Symposium on Packaging and Transportation of Radioactive Materials, Berlin, Federal Republic of Germany, November 10-14, 1980.

EXTENDED TESTING OF A MODIFIED 18B PLUTONIUM NITRATE SHIPPING CONTAINER*

H. R. Yoshimura, R. B. Pope, W. B. Leisher, B. J. Joseph
Sandia National Laboratories, Albuquerque, NM, USA

B. Schulz-Forberg, H. W. Hubner
Bundesanstalt fur Materialprufung (BAM), Berlin (West), FRG

INTRODUCTION

It has been found in risk assessment studies performed by Sandia National Laboratories and Pacific Northwest Laboratories that some accident environments may be encountered during air transport that exceed those specified in the current regulations. NUREG 0360, "Qualification Criteria to Certify a Package for Air Transport of Plutonium," a criteria promulgated by the US Nuclear Regulatory Commission, required package exposure to environments far beyond that specified in IAEA Safety Series No. 6. In an effort to further understand the effects on packagings of environments in excess of those specified in the regulations, additional tests in the form of extended fire testing, extended drop testing followed by fire, dynamic crush and extended punch were performed on an 18B plutonium nitrate shipping container by Bundesanstalt fur Materialprufung (BAM). These results were reported in the Fifth International Symposium on Packaging and Transportation of Radioactive Materials[1]. Since then, the 18B packaging has been modified to improve the crush resistance. This paper reports on the extended tests involving the modified 18B packaging.

CONTAINER DESCRIPTION

In the Federal Republic of Germany, air shipments of plutonium nitrate solutions are transported in 18B packagings designed to the requirements of IAEA Safety Series No. 6. The 18B packaging is basically a cylindrical foam overpack which provides impact and thermal protection to its contents (Figure 1). The overpack is fabricated with two concentric steel shells with the intervening annulus filled with 80-kg/m^3 phenolic foam. Modifications were made to the test unit in the form of added reinforcing rings, increased wall thickness, and changed shell material from mild steel to stainless steel to provide greater crush resistance. The 600-mm diameter, 1700-mm long outer shell of the test unit is constructed of 4-mm stainless steel. The inner shell is also made of 4-mm stainless steel and consists of two cylindrical chambers. The longer is 900 mm by 145 mm in diameter; the shorter, 230 mm by 170 mm in diameter. Additional thermal protection is provided by a 10-mm layer of asbestos mat around the inner shell. The [illegible] of the overpack is foam filled and contains a conical, perforated, stainless steel load spreader. Two closure heads are provided on the overpack. An inner stainless-steel closure head with asbestos mat is bolted to the inner shell and retains the titanium vessel used to contain the plutonium nitrate solution. An outer stainless-steel, foam-filled closure head attached to the outer shell by means of bolts provides impact and thermal protection. A cylindrical, titanium containment

*Work sponsored by U.S. Department of Energy under Contract DE-AC04-76DP00789.

vessel, which may contain a maximum allowable quantity of 9 litres of plutonium-nitrate solution (350 g Pu/litre) with a source heat of 30 W, is carried within the protective overpack. The package has a total mass of 206.4 kg.

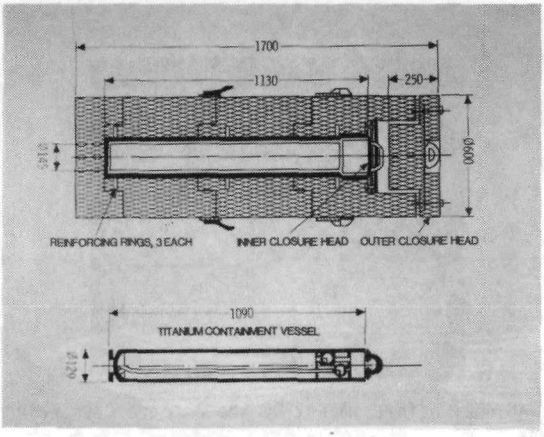


Figure 1. Cross section of an 18B Plutonium Nitrate Shipping Container

MODIFIED 18B PACKAGING TESTS

Within the IAEA Regulations, structural loading of the packaging is represented primarily by drop testing. It has been postulated that for lightweight packagings, the 9-m drop does not appear to adequately simulate an extra-severe mechanical environment (impact and crush). BAM proposed a dynamic crush test in which the specimen is placed on an unyielding target and impacted by a rigid 2-tonne mass (1-m^2 impact surface) dropped from a height of 9 m. The dynamic crush test proposed by BAM may be easily performed at existing drop-test facilities with only minor modifications. BAM developed this test in an attempt to establish a crush environment for small radioactive material transport packages.[2]

The criteria in NUREG 0360 specifies, as one of a series of sequential tests, an impact of the container in the most damaging orientation onto an unyielding surface at a velocity in excess of 129 m/s. This impact test requires a special facility such as the one developed by Sandia National Laboratories as a part of its test activities. The rocket pulldown facility consists of a tensioned cable (wire rope) suspended over a mountain canyon, a trolley that is used to lift the carriage and test unit from a ground-based target to the aerial cable, an unyielding target, and a rocket sled and track system. The aerial cable incorporates all of the necessary equipment to hoist and position the test unit for desired flight trajectory, altitude, and velocity.

Dynamic Crush Tests

The first dynamic crush test was performed on a prototype packaging and was previously reported.[1] Figure 2 shows an unmodified 18B container before impact by a 2 tonne impact weight dropped from a height of nine meters. To briefly review the results, the test produced very severe damage to the packaging (Figure 3).

The outer shell diameter of 600 mm was pressed down to 230 mm; the inner tube diameter decreased from 145 to 32 mm (Figure 4). Loss of the vessel's liquid contents was possible, but could not be directly assessed because a simulated payload was not used in this test.

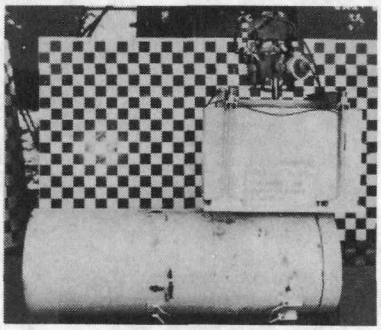


Figure 2. 18B Container before impact by the 2 tonne drop weight from a height of 9 m.

Figure 3. 18B container after a 9-m dynamic crush test.

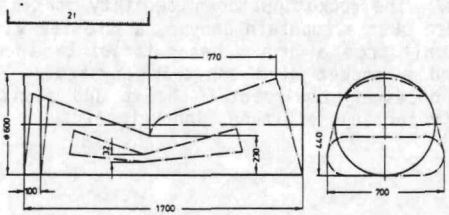


Figure 4. Deformation of the 18B container after a 9-m dynamic crush test (Reference 3).

To improve the crush resistance of the packaging, the design was modified as shown in Figure 1. The dynamic crush test of the modified packaging produced minor deformations of the inner tube which was comparable to the damage produced by a 185-m low altitude drop as reported previously,[1] and the containment vessel remained leaktight. The crush resistance of the container was substantially improved as shown in Figure 5.

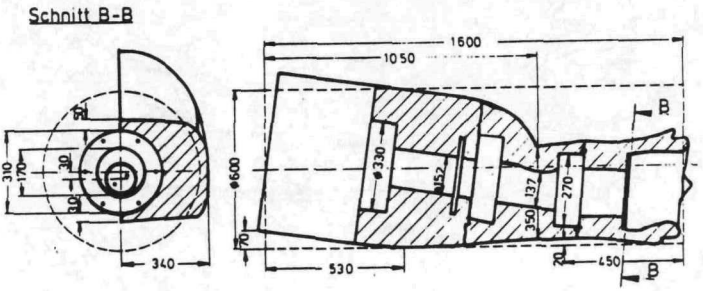


Figure 5. Deformation of the Modified 18B Container after a 9-m dynamic crush test (Reference 3).

High Speed Impact Test

A high-speed impact test of a modified packaging onto an unyielding surface was conducted by Sandia. The test was performed to compare the damage resulting from the extended drop and dynamic crush tests performed by BAM with the damage resulting from the impact environment (129 m/s) specified in NUREG 0360.

The rocket pulldown facility at Sandia National Laboratories, which has previously accommodated a number of similar impact tests, was used for the 18B package test. The general arrangement of the facility is shown in Figure 6.

The impact target is constructed of an extensively reinforced 2.4 x 10^5 kg mass of concrete, approximately 6 m in diameter and 3.5 m deep, emplaced in compacted earth. The target is faced with a 3- x 3-m slab of steel battleship armor, 7.6 to 12.7 cm thick. This facing is welded to the steel concrete-reinforcing members and is grouted to the concrete base with special grout. With respect to [redacted] kg test unit, this target is consider to be "unyielding," as defined in [redacted] Safety Series No. 37.

The test configuration and parameters were similar to those used for the NRC sponsored Plutonium Accident Resistant Container test program.[4] The test vehicle was suspended at a height of about 56 m perpendicular to the unyielding target. A rocket sled positioned on a monorail track on the ground accelerated the test vehicle along its vertical flight path by wire rope tow lines. The towline was a continuous length of wire rope looped around the test unit with the opposite ends attached to the rocket sled. Split stainless steel collars attached to both

ends of the 18B packaging were used to position the tow lines over the test unit's center of gravity and to prevent the wire rope from cutting into the thin-walled structure of the package. The wire rope was turned from the vertical direction by pulleys that straddle the target base, and then routed horizontally to the rocket sled. Figure 7 shows the modified 18B container attached to the carriage prior to testing.

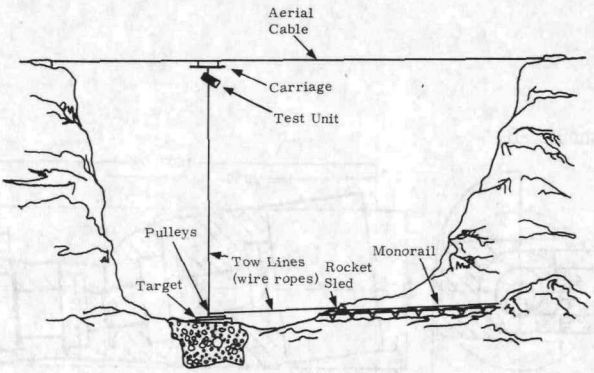


Figure 6. General arrangement of rocket pulldown facility.

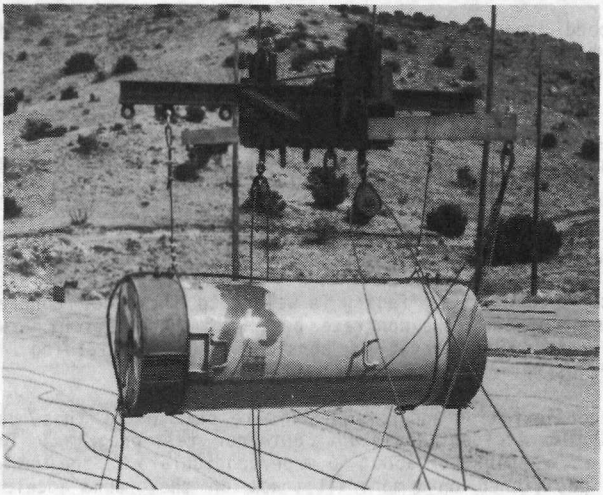


Figure 7. Modified 18B container prior to testing.

High-velocity aerial rocket (HVAR) motors, each having an impulse of 22.3 kN s, were used to tow the test unit to the desired impact velocity. The impact velocity could be varied by adjusting the package height above the target, the time of package release after rocket motor ignition, and by varying the numbers of rocket motors installed on the sled. Before the actual test, a series of calibration tests were performed to ensure that the correct test conditions were attained.

The modified 18B packaging was impacted at a velocity of 137 m/s. Substantial crushing of the foam overpack and extensive deformation of the cavity occurred, which caused release of the titanium containment vessel's liquid surrogate contents (Figure 8).

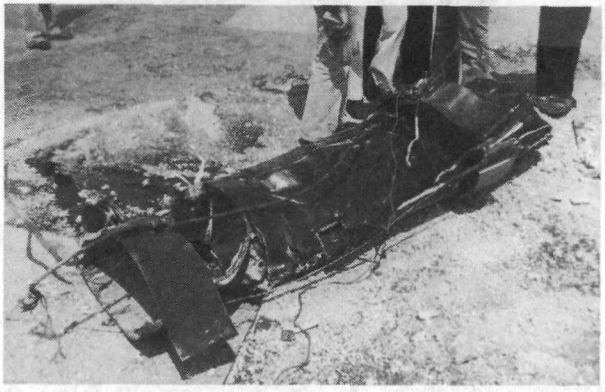


Figure 8. As landed condition of the modified 18B shipping container showing leakage of colored water contents.

Several weld failures were noted in a closer inspection of the overpack. One failure was observed in the weld between the 4-mm-thick inner stainless steel tube and 20-mm-thick reinforcing ring (Figure 9). Weld failure also occurred at the interface between the inner head flange and inner stainless steel tube. A comparison of the final shape of the deformed packaging and original packaging is shown in Figure 10.

Large deformations to the 20-mm-thick reinforcing rings and the 4-mm-thick inner tube were observed. The reinforcing rings were ineffective in protecting the container in a 137-m/s impact against an unyielding target. Nearly complete crush-up of the impact-absorbing foam material occurred, thereby negating any significant thermal protection for a subsequent fire test.[5]

SUMMARY

The container damage observed as the result of the high-speed pulldown impact test was more severe than that of either the 185-m free-fall drop of a prototype container onto a semirigid surface or the crush environment produced by a 9-m drop of a 2-tonne block onto a modified container resting on an unyielding surface. In comparison to the extended tests, the 9-m regulatory drop test onto an unyielding surface of the prototype packaging in its most damaging orientation produced the

Figure 9. Fractured weld between the 4-mm-thick inner stainless steel tube and 20-mm-thick reinforcing ring.

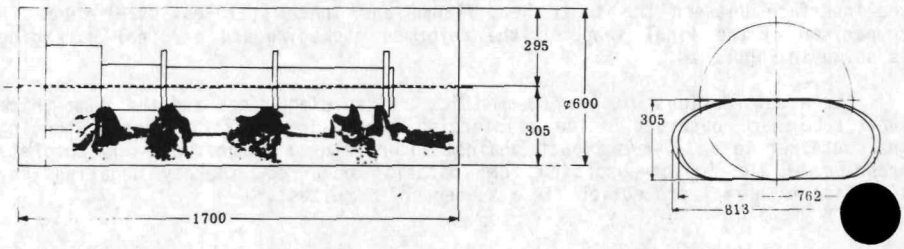


Figure 10. Deformed 18B packaging after a 137-m/s impact test (includes spring-back of outer shell).

least amount of damage. Very little deformation in the overpack was observed, and there was no influence on the fire resistivity and leaktightness of the containment vessel.[1] Table I shows a comparison of the impact test methods.

Table I. *Comparison of Impact Test Methods*

Method	Target	Total Kinetic Energy Available (kJ)	Deformation Observed		Pu Vessel
			Overpack	Inner Tube	
[redacted]rop (Regulatory)	Unyielding	18.2	Minor	No damage	Intact
9-m Dynamic Crush	Unyielding	176.4	250 mm	15 mm	Intact
185-m Drop Test	Semi-rigid	374.2	380 mm	50 mm	Intact
129-m/s Impact Test	Unyielding	1717.4	~100 %	105 mm	Leaked

The packaging's deformation behavior depends not only on impact orientation and velocity, but also on the impact surface; i.e., unyielding target, hard soil, soft soil, or hard rock.

Based on the results of the joint FRG/US test program, it is apparent that the 18B packaging, designed to withstand the environments specified in IAEA Safety Series No. 6., can withstand extended environments including longer duration fires and higher velocity impacts on yielding targets.[1] When modified with ring stiffeners, the packaging withstood a dynamic crush test, but did not survive the high speed impact onto an unyielding surface as specified in NUREG 0360.

REFERENCES

1. Heinz W. Hubner, "Extended Testing of Plutonium Shipping Containers," Fifth International Symposium on Packaging and Transportation of Radioactive Materials, Las Vegas, NV, May 1978, pp 235-250.

2. B. Schulz-Forberg and R. J. A. Neider, "Improvement of the IAEA-Safety System, What Is Necessary?", Fifth International Symposium on Packaging and Transportation of Radioactive Materials, Las Vegas, NV, May 1978, pp 825-830.

3. B. Schulz-Forberg and H. W. Hubner, "Klassifizierung Und Sicherheitsreserven von Transportbehaltern fur Radioacktive Stoffe," (in German), Bundesanstalt fur Materialprufung, Berlin, Federal Republic of Germany, March 1979.

4. J. A. Andersen, "Plutonium Accident Resistant Container Project," Fifth International Symposium on Packaging and Transportation of Radioactive Materials, [redacted]s Vegas, NV, May 1978, pp 251-261.

5. H. R. Yoshimura, R. B. Pope, W. B. Leisher, and B. J. Joseph, "High-Speed Impact Test of an Air Transportable Plutonium Nitrate Shipping Container," SAND79-1287, Sandia National Laboratories, Albuquerque, NM, April 1980.

FLEET SERVICING FACILITIES FOR TESTING AND MAINTAINING RAIL AND TRUCK SPENT FUEL TRANSPORT SYSTEMS*

C. D. Watson, B. J. Hudson,** M. K. Preston
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

D. A. Keith†
Union Carbide Corporation, Y-12
Oak Ridge, Tennessee 37830

P. N. McCreery, W. Knox, E. M. Easterling,
A. S. Lamprey, and G. Wiedemann
Allied-General Nuclear Services
Barnwell, South Carolina 29812

This paper presents design concepts and results of studies of Fleet Servicing Facilities (FSF). The FSFs in these studies are intended to address potential operational and maintenance problems associated with spent fuel transport. It is expected that transport systems designed for the shipment of other waste types (e.g., high-, intermediate-, and low-level wastes, etc.) could be handled in a similar facility.

Specifically, the FSF is intended to be used for routine servicing, preventive maintenance, and for performing requalification license compliance tests and inspections, minor repairs, and decontamination of both the transportation casks and their associated rail cars or tractor-trailers.[1] None of the waste handling plants in the United States presently receiving radioactive wastes have an onsite FSF, nor is there an existing "third party" facility providing all of these services. This situation has caused the General Accounting Office (GAO) to express concern regarding the quality of waste transport system maintenance once the transport system is placed into service.[2] Thus a need is indicated for FSFs or their equivalent at various radioactive materials receiving sites.

Normal functions performed at sites currently receiving radioactive materials in the United States are principally the following: monitoring for contamination, off-loading of the casks and ancillary equipment, unloading and storage of the

*Research sponsored by the Office of Nuclear Waste Management, U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

**System Development Corporation, Oak Ridge, Tennessee (previously of the Chemical Technology Division, Oak Ridge National Laboratory).

†Oak Ridge Y-12 Plant, operated by Union Carbide Corporation Nuclear Division for the U.S. Department of Energy.

cask contents, seal inspection and/or replacement, decontamination and on-loading of the empty cask. A Fleet Servicing Facility would provide all of these functions but would also include these additional capabilities:

1. The facility would maintain an inventory of spare parts to be used as required to minimize cask idle time.

2. Cask/vehicle inspections would be performed prior to the departure of each empty cask. Repairs required by either the cask or vehicle would be performed. All tests and checks required by regulations or warranties would also be performed.

3. Large components, such as special yokes or energy absorbers, that might be used at several sites also would be cleaned for shipment.

4. The facility would be able to:

   - fabricate and test special tools and fittings;
   - provide for the storage of bulky components such as impact structures, lifting yokes, spare parts, etc;
   - decontaminate parts completely;
   - provide a shop to supply all mechanical/maintenance services required by the casks and the transport system.

In the evaluation of the facility requirements for these functions and needs, this paper compares the respective capital costs and operating characteristics of the following three concepts of a spent fuel cask transportation FSF:

1. Integrated FSF - a functional operating area included in the original design and construction of a parent facility, such as an away-from-reactor (AFR) storage plant, reprocessing plant, waste repository, etc., which is totally supplied by the parent's services, manpower resources, and other capabilities. For the purposes of this paper an artist's rendering is presented in Fig. 1, and the sequential processing of a cask (truck and rail) is shown in Fig. 2.

2. Colocated FSF - a unit built in the proximity of an existing parent unit which shares services, manpower resources, and capabilities to the extent that the parent can provide. In some cases, the Colocated FSF may be attached directly to the selected existing parent facility (see Figs. 3 and 4).

3. Independent FSF - a totally self-supporting unit that does not depend on the proximity of a parent reprocessing plant, AFR, etc., for any type of support.

A previous study[3] examined the characteristics of an Independent FSF, and this concept became the basis for comparison with the Integrated and Colocated concepts. The facility was designed to receive 2000 MTU (as heavy metal) of spent fuel per year (75% of which was delivered by rail, and 25% of which was delivered by truck). A design concept and cost study of each facility was completed and provided information on the following items:

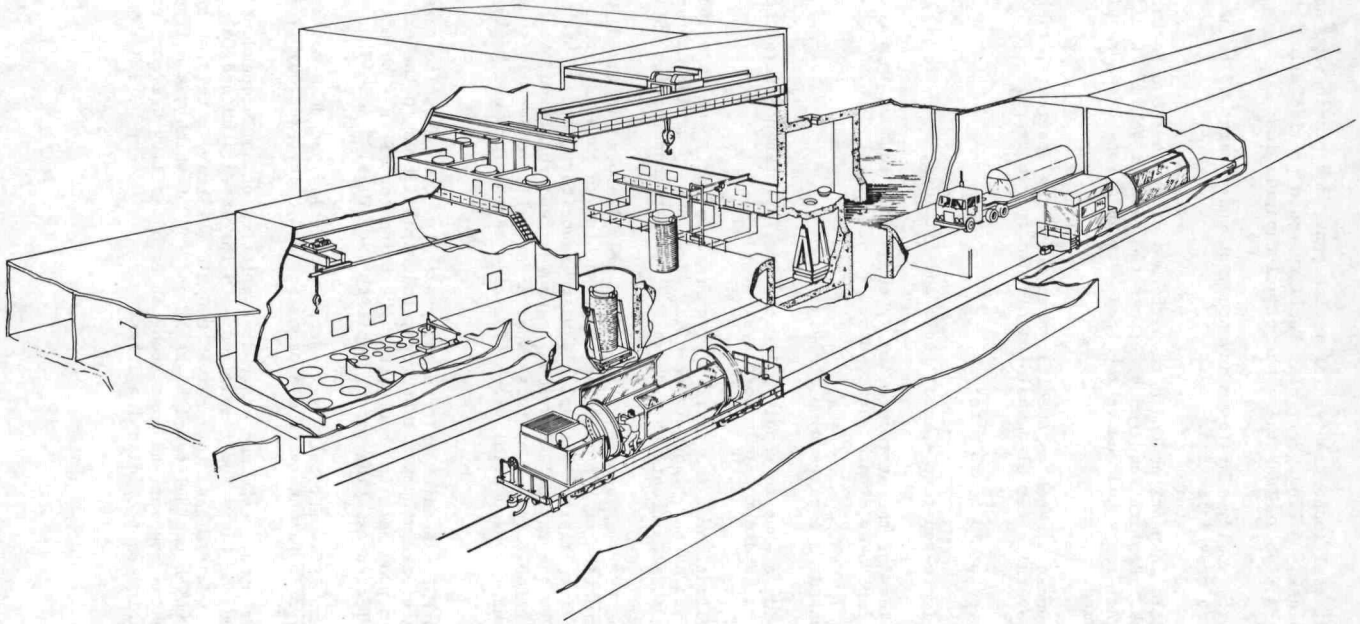


Fig. 1. Artists rendering — Integrated Fleet Servicing Facility.

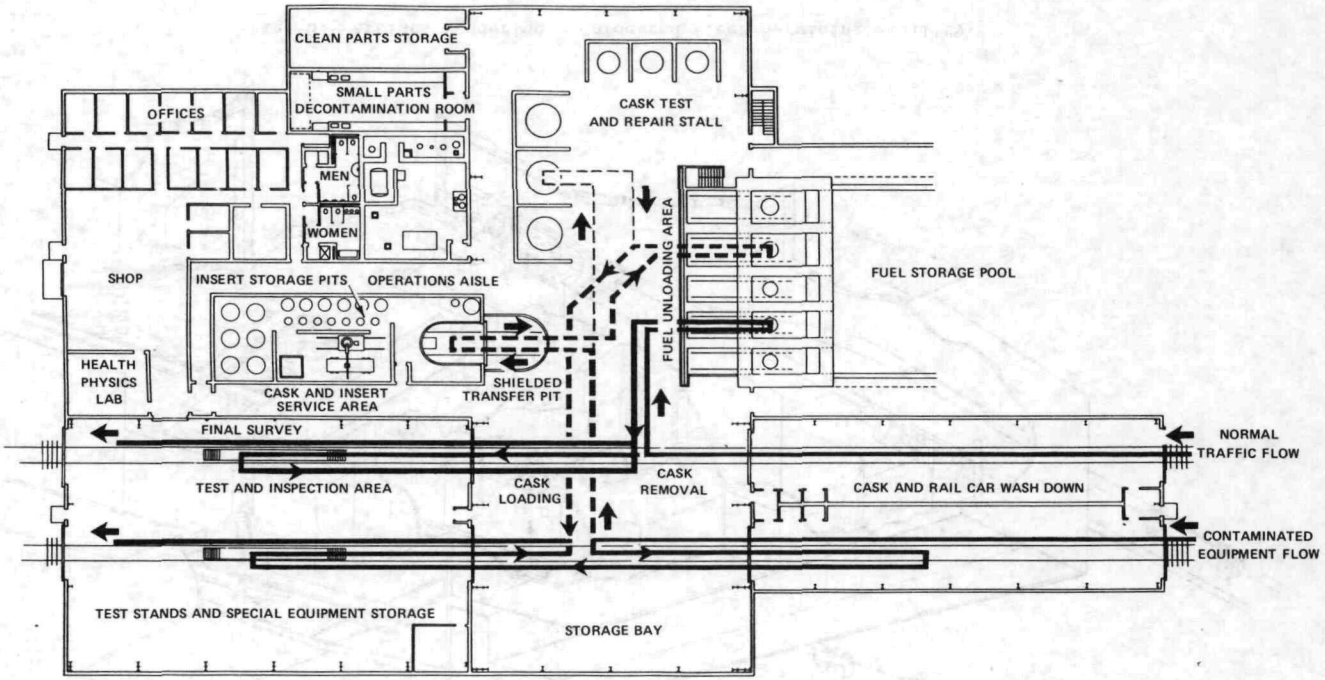


Fig. 2. Normal traffic plan — Integrated Fleet Servicing Facility.

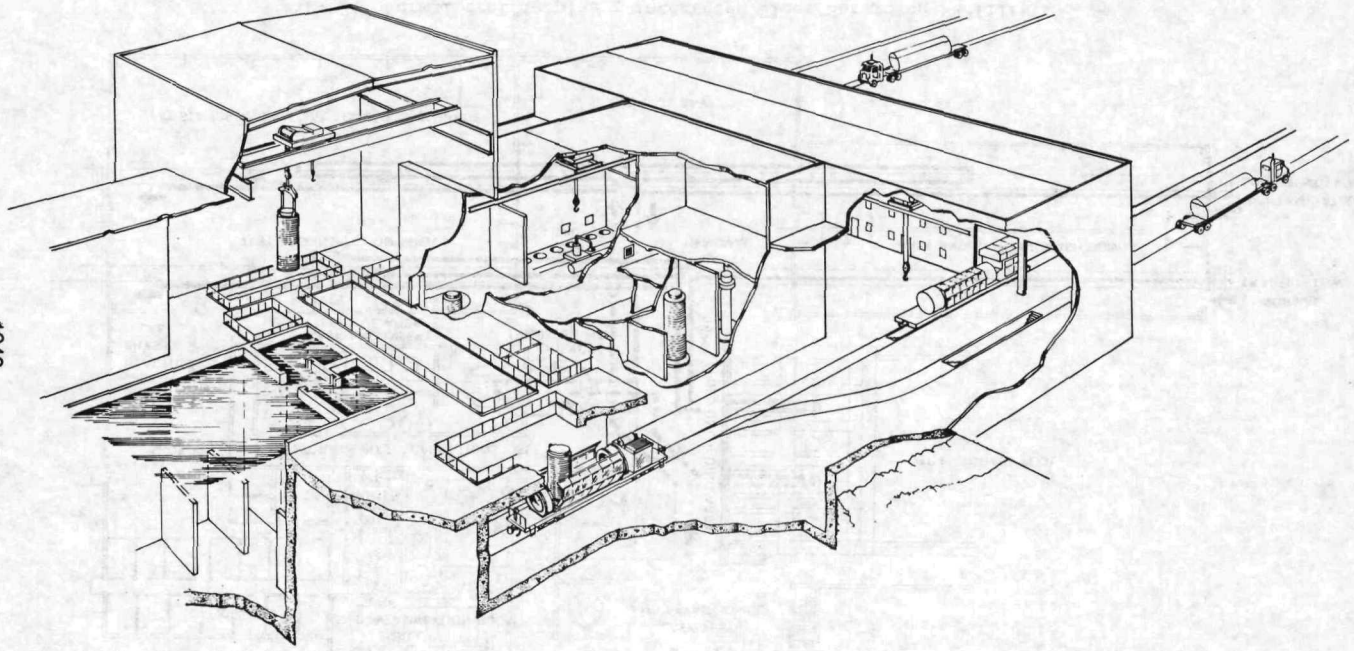


Fig. 3. Artists rendering — Colocated Fleet Servicing Facility.

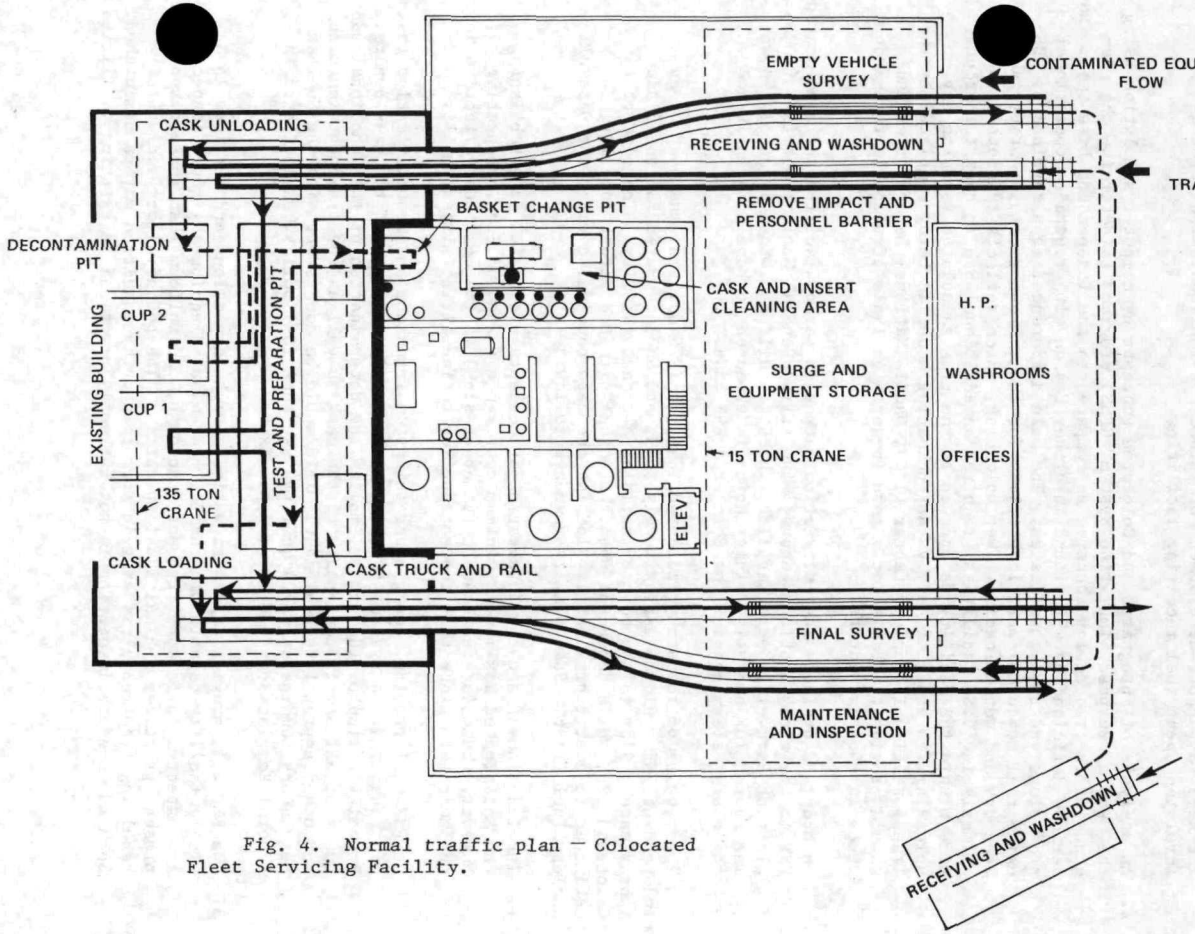


Fig. 4. Normal traffic plan — Colocated Fleet Servicing Facility.

1. The relative feasibility and capital costs of the three basic design concepts; the basis for selecting a concept for further on-going detailed design work; the basis for selecting a parent facility concept upon which a more-detailed, realistic construction and operational cost analysis could be produced; and the need to support development of future DOE implementation strategy involving spent fuel receiving facilities.

2. An analysis of Integrated and Colocated concepts by computer simulation using the General Purpose Simulation System (GPSS) program[4] to determine the adequacy of the two design concepts as compared to the Independent base case. No serious deficiencies were found in either design with regard to operational provisions and throughput rates. The main difference, as determined by GPSS using real measured operational times for the various receiving steps was between dry and wet unloading. Wet unloading, which reflects the method used at all existing U.S. facilities, required an additional 8 h in cask turnaround time within the facility as opposed to the time calculated assuming dry unloading. A study of wet- vs dry-unloading modes is being completed by the Transportation Technology Center (TTC), Sandia National Laboratories and should provide the basis for a more in-depth analysis of probable turn-around times in the near future.

   In order to evaluate the interaction between a parent facility and a colocated FSF in the study mentioned above,[1] the Allied-General Nuclear Services (AGNS) plant at Barnwell, South Carolina, which is built and whose layout is known, was chosen for analysis. This choice is expected to have some influence on the costs determined for the Colocated FSF case.

The results of the study indicated that, in 1979 dollars, the cost of an Independent FSF would be approximately $26M, and an Integrated FSF would add approximately $13M to the cost of building a parent facility. The cost of a Colocated FSF, while dependent upon the actual site and parent facility, is estimated to cost approximately $16M for the case considered here, the Barnwell Nuclear Fuel Plant. Some summary comments follow:

1. The Independent and the Integrated FSF concepts are useful in establishing the anticipated maximum-to-minimum cost range ($26M and $13M, respectively); however, the Colocated concept (at about $16M) is the most cost-effective in the near term since this concept applies to facilities already built.

2. An FSF will provide, among other things, maintenance and requalification tests of transportation systems by an organization independent of the cask owners. Requalification of the system implies the determination of the functional adequacy of all significant parts of the cask as well as the transport vehicles and other appropriate equipment. This would be consistent with the Nuclear Regulatory Commission (NRC) quality assurance (QA) policy, as indicated in Federal Regulations 10 CFR 50, Appendix B.

3. The FSF will provide transport vehicles with inspections and services in a single facility capable of handling contaminated equipment. Such inspections are expected to be more frequent, thorough, and uniform than those given under present practices by individual operators. The potential for accidents resulting from vehicles being in poor roadworthy condition may be diminished. Such an in-depth inspection is not currently practical in existing facilities.

4. As fleet sizes grow and components are increasingly interchanged, a central control system at each FSF would provide compliance assurance more economically than can a number of small, decentralized, specific-purpose facilities.

5. The FSF concept provides a positive response to the criticisms of GAO regarding cask maintenance, and it is recommended that studies continue on the FSF with greater emphasis placed on the more feasible concepts and near-term needs.

6. An in-depth study and analysis of an FSF should provide similar information for transport systems and waste forms other than spent fuel.

It is important to note that since the colocated concept appeared to be the st cost-effective, and since the type and location of the parent facility could ve a significant influence on the cost of the facility, it would be desirable o study the concept using parent facilities other than those at the AGNS plant; for example, the General Electric plant facilities at Morris, Illinois. Further studies might also include an investigation of the effect of interaction between spent fuel and other waste forms at an FSF.

1. C. D. Watson et al., Fleet Servicing Facilities for Servicing, Maintaining, and Testing Rail and Truck Radioactive Waste Transport Systems: Functional Requirements, Technical Design Concepts and Options, and Cost Estimates and Comparisons, ORNL/Sub-79/13866/1 (May 1980).

2. General Accounting Office (GAO) report, Federal Actions Are Needed To Improve Safety and Security of Nuclear Materials Transportation, EMD-79-18 (May 1979).

3. P. N. McCreery et al., The Conceptual Design of a Spent Fuel Cask Fleet Servicing Facility, AGNS-1040-1.5-48 (September 1978).

4. General Purpose Simulation System/360, Introductory Users Manual, GH20-0304-4, International Business Machines, White Plains, New York 10604 (November 1969).

SESSION XIX
EXTENDED TESTING I

RELATIVE RESPONSE OF TYPE B PACKAGINGS TO REGULATORY AND OTHER IMPACT TEST ENVIRONMENTS*

J. D. McClure, H. R. Yoshimura, R. B. Pope and R. M. Jefferson
Sandia National Laboratories, Albuquerque, NM, USA

R. D. Seagren, L. B. Shappert
Oak Ridge National Laboratory, Oak Ridge, TN, USA

INTRODUCTION

The International Atomic Energy Agency (IAEA) regulations (Safety Series No. 6) specify that a Type B packaging must be subjected to tests which demonstrate the ability of the packaging to withstand accident conditions in transport. One of these tests consists of the specimen (packaging) falling 9 m onto a flat, horizontal, "unyielding" surface so as to suffer the maximum damage. It has been reasoned that the 9 m drop is insufficient since it provides an impact velocity of only 13.4 m/s, whereas truck and rail vehicles often travel significantly faster (at speeds of 30 m/s or greater), and aircraft travel is faster yet. It is also often stated that very complex impact orientations and targets should be used to better replicate specific accident scenarios. In reasoning that the velocity, packaging orientation and target for the impact test are inadequate, one must not ignore the actual conditions which exist in most transportation accidents and the philosophy which was used in establishing the above specified test condition.

Fairbairn and George[1] contend that "much of the energy of impact is absorbed by damage to the rail vehicle or aircraft, and as a result, it was considered most unlikely that a package being carried will be subject to an impact significantly exceeding that in the 30 ft drop onto the very hard target." Also, very few "unyielding" targets exist, and the likelihood of an impact actually occurring where the packaging will be oriented for maximum damage is low.

Most significantly, however, the specified test was not intended to represent a specific accident scenario. Rather, the test was developed to provide a level of damage to packagings which could be expected in most high-velocity accident situations.[1] McClure[2] has shown by statistically analyzing accident impact situations that the required test protects against nearly 100 percent of all truck and rail accidents when velocity change is used as the criteria for judging accident severity.

Generally, when packagings are designed and subsequently qualified (licensed), there are development tests followed by a series of formal licensing tests. The licensing tests are limited to those tests required by regulations and these tests are followed by an assessment to demonstrate that the packagings meet specified acceptance criteria (shielding and allowed leakage rates).

*Work sponsored by the US Department of Energy under Contract DE-AC04-76DP00789.

Packaging tests at impact velocities which exceed the regulatory test levels and for realistic target conditions such as soil, hard rock, etc., provide useful data for the assessment of the adequacy of the regulatory requirements. In addition, such extra-regulatory testing provides information related to the determination of margins of safety by the identification of the failure limits of specific packaging designs. Such information is also used to establish release fractions for detailed transportation risk analysis studies.

During the past five years, several analysis and test programs have been performed to determine packaging response to various environmental impacts. In this paper, the results of these programs will be consolidated to demonstrate that the impact test specified in the regulations does indeed represent a very severe environment which significantly tests the structural and containment integrity of packagings.

The effects of target hardness, impact velocity and intervening transportation vehicle structure will be characterized by studying the extent of packaging damage for different impact conditions. While these data have been previously published, they have never been consolidated for the purpose of demonstrating the validity of the required test. The data reviewed here involve analyses and tests where packagings were exposed to 9 m drop tests and to drop tests and other high velocity impacts different from those required by current regulatory tests. The test data involve essentially four variables: radioactive material packaging design, including size (or mass); the impact velocity of the packaging; the orientation of packaging and impact target and the impact target hardness.

The 9 m drop test (impact) environment will be used as the basis of comparison for formulating the relative response arguments presented in this paper. It may be possible to formulate additional or similar relative response arguments for other accident environment categories such as thermal (fire), crush, puncture or immersion. Such assessments are left to other studies.

IMPACT-VELOCITY/TARGET-HARDNESS RELATIONS

At least one analytical model has been developed[3] to define the relationship between impact velocity and target hardness. The model involves the impact of a rigid sphere onto an elastic half-plane and uses equivalent accelerations to define the impact velocity relationships for yielding and unyielding targets. Steel was used to represent an unyielding target. Other materials such as rock and soil were used to represent realistic impact targets. An impact velocity ratio was calculated, V_y/V_s, where V_y represents an impact velocity onto a yielding target which produces an acceleration equivalent to that from an impact onto an unyielding target at velocity V_s. Equation 1 gives the relationship for the equivalent damage (i.e., acceleration) velocity ratio V_y/V_s. As calculated, V_y/V_s is 2.2 for hard rock, 2.5 for soft rock, 3.4 for hard soil, 4.5 for water and 7.1 for soft soil. Thus, to obtain damage in a packaging equivalent to that resulting from a 13.4 m/s impact onto an unyielding target, the model predicts that velocities must be approximately 29.5 m/s for hard rock, to as high as 95 m/s for soft soil.

$$\frac{V_y}{V_s} = \left[\frac{1 - \nu_y^2}{1 - \nu_s^2}\right] \left[\frac{E_s}{E_y}\right]^{1/3} \qquad (1)$$

ν_y, ν_s Poisson's ratio (yielding and unyielding targets)
E_y, E_s Modulus of elasticity (yielding and unyielding targets)

This relationship between impact velocity and target hardness represents a damage equivalence for various impact target conditions where the steel target represents an unyielding target, concrete represents a target of intermediate hardness and soil represents a relatively soft target. These target hardness categorizations represent only a crude grouping of target conditions.

Bonzon and Schamaun[4,5] performed an analytical and an experimental study on LLD-1 and 6M packagings. The LLD-1 consists of cylindrical outer and inner steel packagings supported within a steel cage structure having a total mass [redacted]minally 59 kg. The 6M consists of a small cylindrical steel drum with an [redacted]nal steel vessel having a mass of approximately 23 kg. Eleven impact tests were performed for each packaging on either a steel-faced concrete (unyielding) target, a concrete target or a soil target.

The LLD-1 packaging demonstrated a sensitivity to impact angle. The side impact was the most severe in terms of gross structural failure. From the LLD-1 test data, a velocity/target-hardness equivalency was established based upon equivalent levels of damage. For the steel (unyielding) target, the velocity was 90 m/s; for concrete, 125 m/s and for soil, 205 m/s. This provides relative velocity ratios of V_y/V_s = 1.4 for concrete and 2.3 for soil.

The 6M inner container showed less sensitivity to impact angle. The center of mass above the closure edge resulted in the most severe failures. Here, impact velocity/target-hardness equivalence values were 95 m/s for steel, 125 m/s for concrete and 230 m/s for soil. The resulting velocity ratios are V_y/V_s = 1.3 for concrete and 2.4 for soil.

A second series of impact tests are described in work performed by Waddoups[6] and Shappert.[7] These tests involved impact of BE-83 casks, one dropped 610 m from a helicopter onto undisturbed, hardpan desert soil and the second dropped 9 m onto an unyielding target. The BE-83 cask had a mass of 3050 kg.

When dropped from 610 m, the BE-83 cask had an impact velocity of 110 m/s, impacted near side-on with closure end lower and penetrated the hard soil target. There was no observable damage to the packaging following the 610 m drop test.

The 9 m drop test of the BE-83 cask was performed in an end-on configuration. Some cask deformation and weld failures were noted. Thus, the 9 m drop test onto an unyielding target provided greater mechanical damage to the BE-83 cask than did the 610 m drop test onto a hard soil target. From these test results a relative velocity ratio of V_y/V_s > 110/13.4 = 8.2 for the soil impact can be inferred.

A third series of tests involved the full scale crash testing of a rail guided truck-trailer-cask system into a 627 tonne concrete target which was back-[redacted] with 1545 tonne of soil,[8] and the dropping of a cask from 9 m onto an u[redacted]ding target. Both tests involved end-on impacts of a cask approximately 4 m in length and 1 m in diameter, which weighed 20.4 tonne. The cask, of steel/lead construction, had been retired from service because of design features which were no longer acceptable such as exposed external piping.

In the crash tests, the system was traveling at 27.3 m/s at impact in one test, and at 37.6 m/s at impact in a second test. The truck and trailer interposed between the cask and the target acted as a moderator for the target hardness. For this impact environment, the concrete target with intervening truck-trailer

structure can be thought of as a relatively soft target. No significant structural damage to the cask was found. A peak deceleration of approximately 18 g was measured in the 27.3 m/s impact and of approximately 70 g in the 37.6 m/s impact.

For the 9 m drop test, no damage to the cask could be found. A peak deceleration of approximately 115 g was measured.

Thus, in terms of deceleration, the 9 m drop test onto an unyielding target was more severe than the 37.6 m/s crash test. In terms of structural damage to the cask, the two tests produced comparable results. However, a relative velocity ratio for a relatively soft target of $V_y/V_s > 37.6/13.4 = 2.8$ can be derived from the deceleration data. The range in V_y/V_s is partially a result of different packaging designs, impact orientations and a variation in the physical properties of the impact targets.

The relative velocity ratios developed for different conditions using both analysis and test results, are compared in Table I. Generally, for moderate hardness targets an approximate doubling of velocity is required, and for relatively soft targets, at least a tripling of velocity is required to attain equivalent packaging damage.

TABLE I

Impact Velocity Ratio - Damage or Deceleration Equivalence

Target Description	Type of Target	Basis of Comparison	V_y/V_s*	Source of Data
Unyielding	Concrete-Backed Steel	Deceleration or Damage	1.0	Definition
Moderate Hardness	Hard Rock	Deceleration	2.2	Analysis[3]
	Soft Rock	Deceleration	2.5	Analysis[3]
	Concrete	Damage	1.3 to 1.4	Test[4,5]
Relatively Soft	Hard soil	Deceleration	3.4	Analysis[3]
	Water	Deceleration	4.5	Analysis[3]
	Soft Soil	Deceleration	7.1	Analysis[3]
	Desert Soil	Damage	2.3 to 2.4	Test[4,5]
	Desert Soil	Damage	> 8.2	Test[6,7]
	Concrete/Truck-Trailer	Deceleration	> 2.8	Test[8]

*V_y is velocity of impact for yielding target and V_s is velocity of impact for unyielding target providing equivalent packaging damage or deceleration.

The protection provided by existing packaging designs is demonstrated by the cumulative probability curve[9] of impact velocity shown in Figure 1, which can be used to establish a figure of merit on "protection" provided by the regulatory test. Target hardness must be included to provide an adequate description of the true accident environment, which is accomplished in Figure 1 by identifying different velocities for the different targets. Very high levels of protection are provided by existing regulatory testing (greater than 99.5 percent) when

target hardness is not considered. For the softer, more realistic targets, the damage equivalent velocity ratios developed in this paper indicate the levels of protection provided by the regulation are even higher, approaching 100 percent. These damage equivalent points are shown on the protection curves of Figure 1.

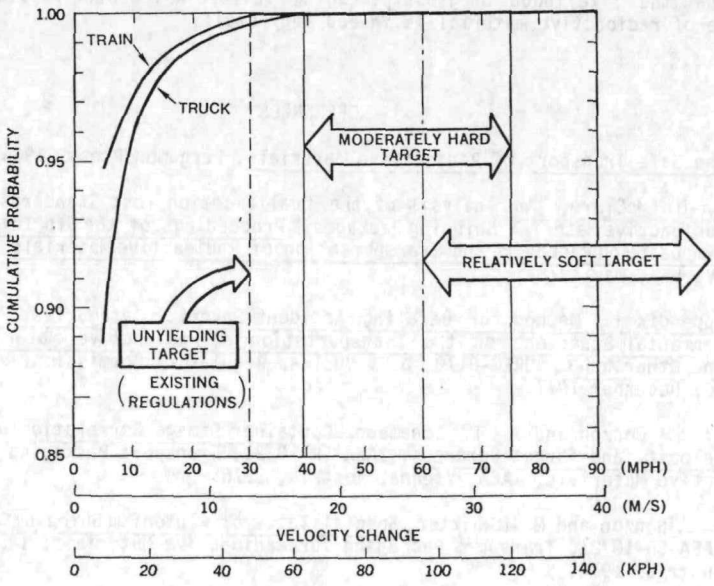


Figure 1. Damage Equivalence-Protection Diagram

CONCLUDING REMARKS

The information presented above has emphasized the magnitude of the environmental parameters which might present a mechanical insult to a radioactive material packaging design. The equivalence of these environmental tests with other than non-yielding targets has been discussed.

The information presented in this paper indicates that the current IAEA regulatory drop (impact) test provides a damage equivalence to severe impact-like transport accidents. The analysis in this paper makes a comparison between the regulatory impact test velocity and the velocity that can actually be sustained by a package in transport. Such comparisons are often incomplete because they ignore the energy-absorbing capabilities of the vehicle structure or the hardness of the target. Therefore, it must be emphasized that the complete description of an impact accident situation couples the impact velocity with the target hardness.

It must be re-emphasized that a candidate packaging design must first be subjected to the "accident conditions of transport" and then the packaging must meet strict acceptance criteria for shielding and containment of radioactive contents. Essentially two types of comparisons must be made. The resultant damage

from a 9 m drop onto an unyielding target must be placed in perspective with the damage sustained in impacts onto yielding targets at various orientations. Such a comparison must be combined with stringent post-test acceptance criteria. The combination of severe environmental testing, a consideration of target hardness conditions and stringent post-test acceptance criteria lead to the conclusion that the likelihood of gross packaging failure which will provide a large release of radioactive material is indeed very small.

REFERENCES

1. The Safe Transport of Radioactive Materials, Pergamon Press, 1966.

2. J. D. McClure, "An Analysis of the Qualification Test Standards for Small Radioactive Material Shipping Packages," Proceedings of the 5th International Symposium on Packaging and Transportation of Radioactive Materials, Las Vegas, NV, USA, May 1978.

3. Appendix H, Method for Derating Accident Severity Categories, Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170, U.S. Nuclear Regulatory Commission, Washington, DC, December 1977.

4. L. L. Bonzon and J. T. Schamaun, Container Damage Correlation with Impact Velocity and Target Hardness, IAEA-SR-10/21, Transport Packaging for Radioactive Materials, IAEA, Vienna, Austria, 1976.

5. L. L. Bonzon and M. McWhirter, Special Tests of Plutonium Shipping Containers, IAEA-SR-10/21, Transport Packaging for Radioactive Materials, IAEA, Vienna, Austria, 1976.

6. I. G. Waddoups, Air Drop Test of Shielded Radioactive Material Containers, SAND75-0276, Sandia National Laboratories, Albuquerque, NM, USA, September 1975.

7. L. B. Shappert, N. C. Bradley, J. H. Evans, M. C. Jurgenson, The Obsolete Cask Test Program: Test No. 2, Vol. 16, ORNL-TM-1312, Oak Ridge National Laboratory, Oak Ridge, TN, April 1975.

8. Michael Huerta, Analysis, Scale Modeling, and Full Scale Tests of a Truck-Spent-Nuclear-Fuel Shipping System in High Velocity Impacts Against a Rigid Barrier, SAND77-0270, Sandia National Laboratories, Albuquerque, NM, USA, April 1978.

9. A. W. Dennis, "Predicted Occurrence Rate of Severe Transportation Accidents Involving Large Casks," Proceedings of the 5th International Symposium on Packaging and Transportation of Radioactive Materials, Las Vegas, NV, USA, May 1978.

10. Para 718-721, Safety Series No. 6, Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition, IAEA, Vienna, Austria, 1973.

DISCUSSION

Question by R. F. Barker, Austria: Do you or can you show the effect of weight of the package in terms of equivalent damage for the 9 m drop test on unyielding targets? Would the factors for target yield vary with weight of package?

Answer: I believe that package mass can be included as a parameter and we shall attempt to make such a display. I am not certain that package mass would have a significant effect, I would have to think about that.

Question by G. Schmidt, FR Germany: In the first part of your paper you mentioned four essential variables; the first of these being the packaging design. Could you please give some figures concerning the probably most important design feature of packagings, namely the impact limiters usually attached. I.e., how do such impact limiters influence the effects and relations shown in your paper, or is this influence already included?

Answer: The impact limiters (or other such mitigating devices) could have an effect on the results - especially on relatively hard targets.

CLASSIFICATION AND SAFETY MARGINS OF RAM PACKAGINGS - RESULTS OF A RESEARCH PROGRAMME

B. Schulz-Forberg, H. W. Hübner
Bundesanstalt für Materialprüfung (BAM)
Unter den Eichen 87, D-1000 Berlin 45

Sponsored by the German Federal Ministry of the Interior, the "Bundesanstalt für Materialprüfung (BAM)" has carried out a research programme to determine the safety margins of RAM packagings used in the Federal Republic of Germany, to investigate the failure mechanisms and, based on these data, to establish a classification system for different groups of RAM packagings [1] .

In general, packagings for the safe transport of radioactive materials which are designed and constructed according to the IAEA Regulations, are considered to be safe enough under national and international law. Besides this, there is the question of safety margins and failure mechanisms, i.e. what kind of stresses beyond the legally required limits can be sustained, when and how do they fail and how the packagings can be classified accordingly.

To answer these questions it was necessary to define test parameters oriented to real accident situations, which on the one hand represent an extension of already known test conditions and had to be specified on the other especially for so-called light-weight packagings (for α-emitters or fresh fuel elements with plutonium compounds).

TEST FACILITIES

To carry out extended and/or modified testings, i.e. extension of the drop height from 9 m to 200 m, fire duration from 30 min. to 75/90 min., the development of new test facilities and programmes as well as the modification of those already in existance proved to be necessary.

To carry out operative tests, an open fire test facility was established to determine

- the behaviour of different insulation materials (i.e. gassing) under long term fire stress, and
- safe and proper handling of possible bursting of liquid-filled packagings with regard to test personnel and equipment.

The fire test facility consists of a 15 m^2 fire trough, which can be divided into areas of different sizes, an emergency flap operated by a remote-controlled cable system and an extinguishing bunker (15 m^3) filled with CO_2 gas during testing into which burning liquid can pour. All-round wind shields guarantee the required full

engulfment while by continuous refilling of fuel oil and water (to cool the fire trough) all required fire durations can be obtained.

Comparative heat calculations can be carried out by means of a one-dimensional heat programme. Though there are already more sophisticated programmes, results are still satisfactory. The heat conductivity figures for new insulation materials, necessary for the heat calculations, were determined with a newly developed heat conductivity measuring device.

BAM drop test facilities were used to perform extended drop tests especially for the crush tests. The 200 m drop tests were all carried [redacted] using helicopters.

To determine the safety margins with regard to extended mechanical stresses, the packagings were dropped from a height of 200 m. For light-weight packagings, the 200 m represent a drop height by which these packagings reach their terminal sinking velocity; for heavy packagings, i.e. spent fuel element casks, drops from the highest bridges could be simulated as well as highest collision speeds.

We consider light-weight packagings, i.e. packagings with a high volume/mass ratio, to be exposed to significantly lower mechanical stresses due to the 9 m drop test than casks with their heavy shieldings. Therefore we developed the parameters of a so-called dynamic crush test, i.e. the dropping of a crush weight of 2 tonnes with a base of 1 m^2 from 9 m. If such tests should be mandatory in the future, existing drop test facilities could be used.

The main intention of the extended fire test was to estimate the thermal stresses up to failure of the packagings. Therefore the light-weight packagings in particular were exposed to simulated accident fires up to a 90 min. duration where mean flame temperatures lie beyond 800°C.

With the help of the temperature dependent heat conductivity figures, the failure limits were calculated. These limits lie beyond a thermal stress of 150 min. for packagings with a phenolic foam insulation. Besides this, the failure limits were calculated for a number of packagings circulating within the Federal Republic of Germany with regard to different failure mechanisms such as melting of lead and therefore loss of shielding or loss of heat insulation due to previous mechanical damage.

TEST RESULTS

Crush testing proved that dynamic crush impacts as defined by [redacted]2 t weight with a 1 m^2 base from 9 m) cause more severe damage t[redacted]ght weight packagings than the required 9 m drop test on an unyielding target, also compared with packagings of the same size but with heavy shieldings. All light-weight packagings investigated failed or showed such damage that no protection was left when exposed to a fire following an accident. However, it was later shown that with simple constructive improvement sufficient safety levels could be reached even against such stresses.

The 200 m air drop tests on light-weight packagings on hard soil, as well as calculations proved that some designs can withstand extended crash forces up to impact velocities equal to the terminal velocity without loss of integrity. The design of a T-shaped lid construction versus the normally used clamping ring closure and phenolic foam as thermal insulation versus pourable vermiculite is advantageous for the shipment of liquid α-emitting radioactive material.

A total of seven 200 m drop tests using a 2000 kg cask for irradiated graphite ball elements and a 4000 kg 1:2 model of a spent fuel element cask proved that these types of packagings represent extremely resistant constructions. Even a drop on to a 25 cm reinforced concrete target caused no loss of integrity though release of gaseous fission material can be expected.

For soft soil or mud, total burial of heavy casks has to be taken into consideration. The resulting heat accumulation due to the interrupted heat dissipation by natural convection calls for deeper investigation. First conservative calculations resulted in the recovering of such buried lead-shielded casks after 12 hours.

The packagings to be investigated against extended thermal loads showed sufficiently good safety margins due to their insulating materials. Foamed insulations such as phenolic foam were preferable to pourable insulations such as vermiculite. While heavy casks normally have a large thermal capacity, the known theoretical procedures for calculation can be seen as sufficient simulation of actual accident fire loads. Knowing the respective material data, wall construction and thermal conductivities, we can determine the amount of damage to a packaging.

CLASSIFICATION OF ESTIMATED SAFETY MARGINS

Based on collected data on type B packagings available in the Federal Republic of Germany - a total of 134 different types of packagings - we performed deeper investigations on typical packagings as representatives of their packaging groups with regard to their safety margins and failure mechanisms. The division into groups - without calling for completeness - was carried out as follows:

- radiography devices with low volume and high specific weight for generally encapsulated material;
- packagings for large sources of non-fissile material and fuel elements from test reactors, characterised by their own components for shielding or protection against accident stresses;
- packagings for α-emitters or fresh fuel elements with plutonium components with high volume and low specific weight;
- casks for spent fuel elements or high radioactive waste.

So we use the design and construction of the packagings as the first essential classification criterion.

After evaluation of failure mechanisms and safety margins, a first classification was made by a combination of failure/damage and possible accident scenarios. But because only representatives of

the different packaging groups had been investigated, only a rough matrix could be achieved. Therefore, we do need futher investigations to make more than a mere statement indicating tendency.

Failure/damage was divided into four steps, but we do not have detailed descriptions of the different steps. Furthermore, these steps should be characterised by concrete examples and not conclusively defined. This means, that other possible types of damage of the same order of magnitude are included in the respective steps.

Step 1 means, that a packaging (by stresses greater than those required by IAEA tests) is damaged so that there is leakage. Here leakage stands for mass flow or locally increased radiation dose. The magnitude of leakage shall be 1000 times the IAEA A_2 figure, i.e. $10^{-3} A_2$/week or the loss of shielding of 10 % of the surface.

Step 2 means additional damage so that extended leakage of up to 10^6 times the admissible A_2 figure or the total loss of shielding.

Step 3 means damage up to the total loss of contents.

This scheme will be completed by the Step 0, which means that there is only damage tolerated by the IAEA Regulations.

If one divides the types of accidents, which lead to the damage indicated in steps 1 to 3, into three steps accordingly, we have to incorporate the crush test, as a mechanical impact which is greater than IAEA tests require into Step 1 of these types of accidents. Step 2 covers an additional thermal load (1000 °C/h instead of 800 °C/0.5 h). Step 3 includes a drop load greater than in IAEA requirements with very high impact velocities (crash test) and an extended thermal load as well. It is assumed that the energy of the crash test is greater than that of the crush test.

If we assign so-called damage numbers (i.e. valencies) 1, 3 and 9 to the damage steps 1 to 3 and assign so-called accident numbers 1, 2 and 4 to the accident steps 1 to 3, we can define a so-called endangering factor by forming the quotient of damage number and accident number. This may be highlighted by an example. If a packaging is damaged during accident step 1 so that damage of step 3 equal to damage number 9 occurs, we will obtain an endangering factor of 9 over 1 equal to 9, etc.

We then carried out these evaluations with about 70 different types of packagings as far as possible. In doing so we could classify 56 packagings, i.e. 80 %, as packagings with good to satisfac- safety margins. 14 packagings, i.e. 20 %, showed only small ty margins. But even small safety margins mean that there are reserves against IAEA requirements.

Our classification only takes packaging and accident parameters into consideration and leaves open an evaluation with regard to the material to be shipped. The endangering factor then surely has to be combined by a number which incorporates material properties. Amount of activity and physical condition are figures which contribute to this material number.

Regarding our above mentioned packaging groups, we found the following distribution of safety margins from our investigations:

- within the group of radiography devices or equivalent, there are only packagings with high safety margins;
- because of the small number of packagings for large sources and casks for spent fuel and high radioactive waste, these are put together and mostly show high to satisfactory safety margins;
- regarding the group of packagings for plutonium compounds or equivalent, we have to establish that the percentage of packagings with satisfactory to small safety margins is significantly high. These packagings are valid type B packagings, but in our view, a very realistic accident scenario of crush stresses leads for the generally light-weight packagings to often a high amount of damage (always with regard to the drop of 2000 kg from 9 m).

Completion of the existing regulations seems to be recommendable. We have already submitted appropriate proposals.

REFERENCE

[1] B. Schulz-Forberg, H. W. Hübner
Packagings for the Transport of Radioactive Materials - Classification and Margins of Safety
Bundesanstalt für Materialprüfung (BAM), Berlin, August 1980
(available in English and German)

DISCUSSION

Question by C. B. G. Taylor, United Kingdom: You referred to an improved lid for drums, being better than the usual flat lid with retaining band. Would you describe this lid please?

Answer: This is a so-called T-shaped lid, which does not merely serve as a flat disk but acts like a cork in a bottle, so that in the case of crushing of the lid region, the lid will not pop off but stay like a cork in the container.

ANALYSIS OF HYPOTHETICAL CRASH OF A NUCLEAR FUEL TRANSPORT VEHICLE

Sabro Kikuchi and Minoru Kubo
Power Reactor and Nuclear Fuel Development Corporation

Introduction

The transport system was composed of a truck,a shock-absorbing damper (including a tie-down device),a cask and various auxillary equipment. Assuming the impact energy absorption rate by the transport vehicle at the moment of its impact against a fixed barrier,the motion,deformation and damage at various sections constituting the transport system were simulated by computer code.

The overview of the transport vehicle is shown in Fig.1. Namely, the transport system was substituted by a two-dimensional model,and the analyses were conducted on the following points employing the kinetic analytical code by the two-dimensional finite element method. This kinetic code is used non-lenear spring model using Newmark method to simulate the transport systems.The calculation time is about 1 minute.

The objects of the analysis were:

(1) The integrity of the securing device of the transport cask to the damper. The securing device is shown in Fig.2.

(2) The integrity of the shock-absorbing damper. The shock-absorbing damper is shown in Fig.3.

(3) The integrity of the tie-down device of the damper to the truck's platform. This tie-down device is shown in Fig.4.

(4) Response of each section during impact time.
(load-time relation,and cask velocity-time relation)

(5) The mode of the cask's motion.

(6) The integrity of the cask in comparison with the conditions of the 9-meter free drop test.(the impact response to the fuel assembly in the cask was calculated.)

(7) The ability to predict the integrity of the fresh fuel assembly.

Analytical model

(1) The following describes preparation of the model for calculation of the impact response at the time when the transport system as shown in Fig.3 might crash against a rigid barrier at a certain fixed velocity.

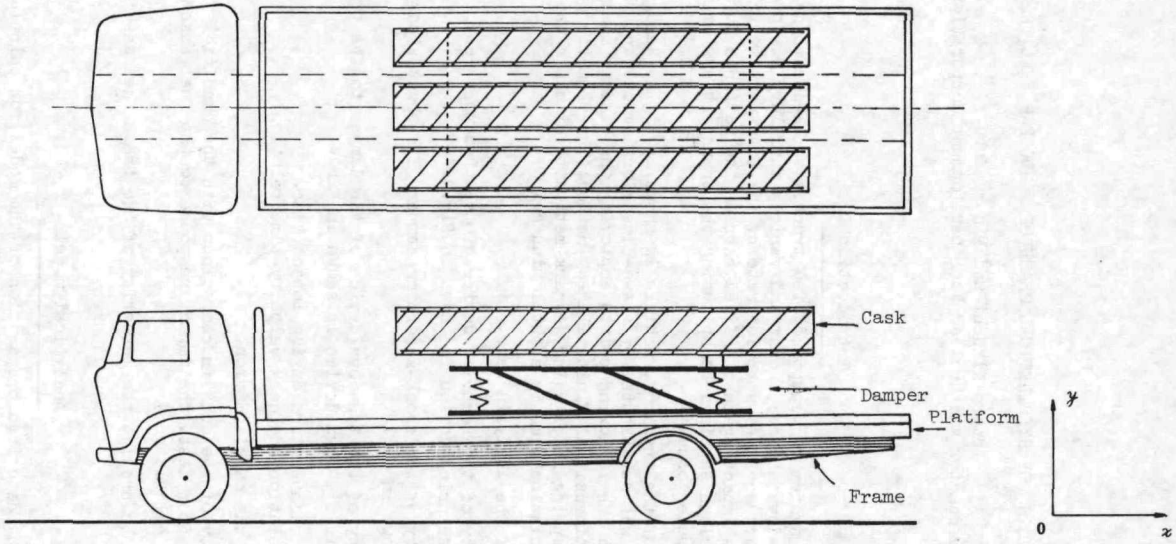


Fig.1 The overview of the transport vehicle

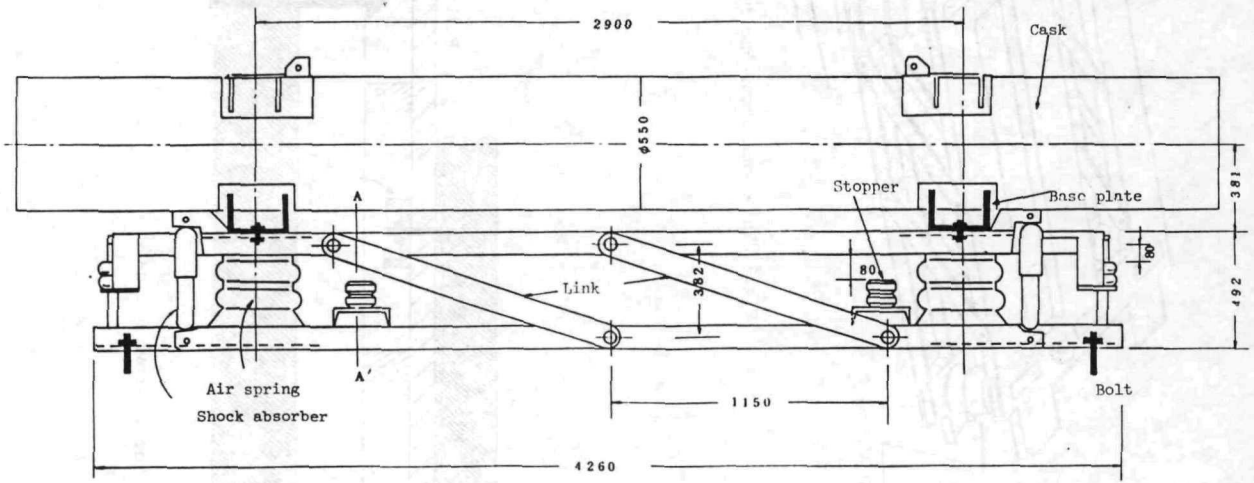


Fig.2 The securing device

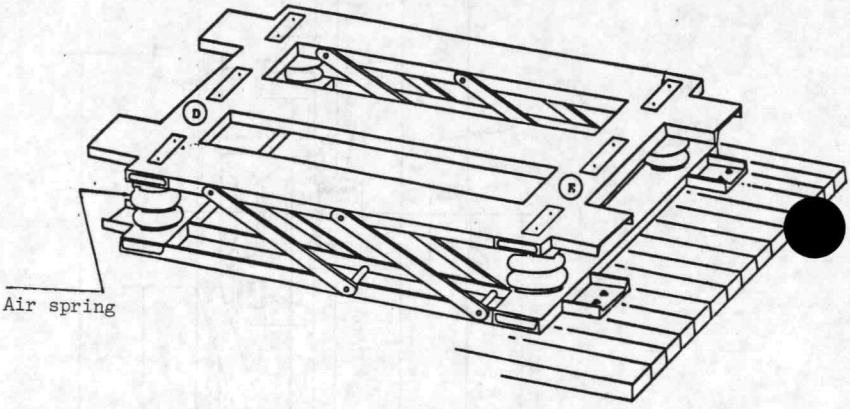


Fig.3 The shock absorbing dumper

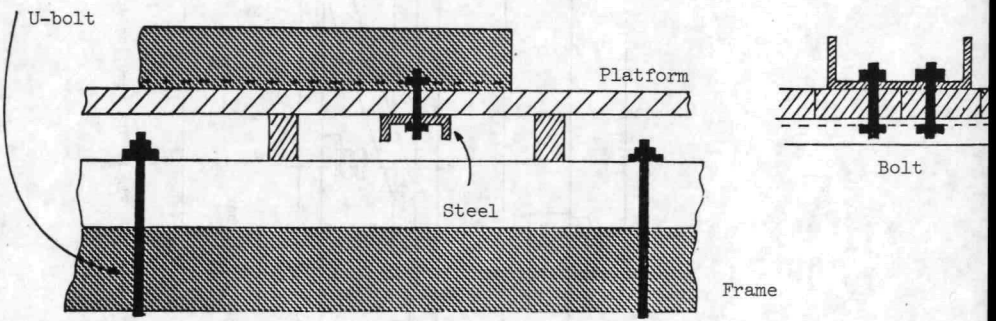


Fig 4 The tie-down device

Specifications of major constituent elements

a) Transport cask
Outer wall : Diameter 550 mm, length 5000 mm
Gross weight : 1670 Kg
Contents : Fresh fuel assembly (PuO2-UO2)
The cask is secured to the damper by 4 bolts of 16 mm in diameter via a base plate.

b) Shock-absorbing damper
Weight : 2400 Kg (made of mild steel(SS41))
Measurements : Acceleration etc.
Damper is secured to the truck's platform by 4 U-bolts of 16 mm in diameter

c) Truck

Type	:	Large-size truck
Max.loading capacity	:	About 7 metric tons
Vehicle's weight	:	About 7 metric tons
Length of pla-form	:	About 8 meters

(2) Two-dimensional finite element model
A two-dimensional finite element model as shown in Fig.6 was considered for the transport system shown in Fig.5. It is almost impossible with the presently available analytical technique to handle the crash destruction of a cabin by use of amodel. Therefore, we considered the renear part of the system only, that is, excluding the cabin, and thus we calculated the response at the time when an impact force might be applied to the point "G" of the joint section of the cabin. The input of the impact force is in the form of square wave. P, Q, R and S points are considered as springs.

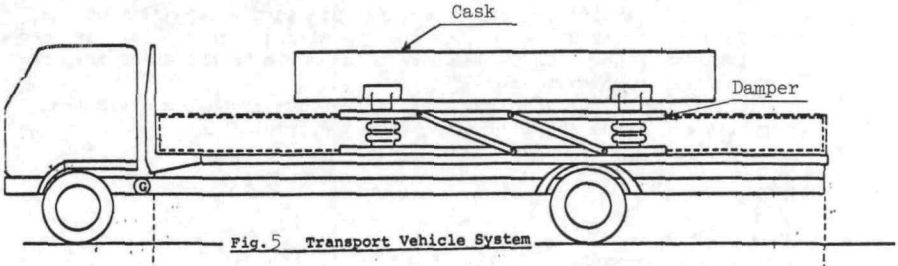


Fig.5 Transport Vehicle System

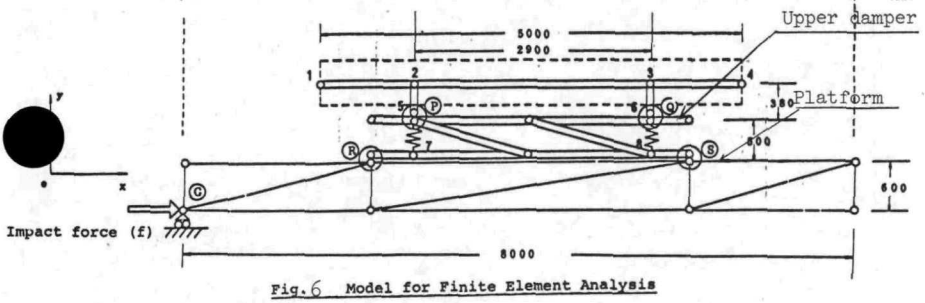


Fig.6 Model for Finite Element Analysis

Results of analysis

From the calculation results,there appears the possibility of the truck hitting a barrier head-on without appreciably slowling the considered rear part of the system.

Behavior of the truck-cask impact accident with 50 Km/h against a concrete fixed barrier by the simulation code are ,for example, shown in Fig.7,8 and 9.

Following results were confirmed:

(1) The integrity of the securing device of the transport cask to the damper was analysed.

(2) The integrity of the shock-absorbing damper was analysed.

(3) The integrity of tie-down device of the damper to the truck's platform was analysed.

(4) Response of each section during the impact time could be predicted.

(5) The mode of the cask's motion could be simulated.

(6) The integrity of the cask in comparison with the conditions of 9-meter free drop test was analysed.

(7) The calculations were worked on the shock stress on the fuel pellets cladding tubes of the fuel assemblies at a collision velocity of 50 Km/h-100 Km/h varying the shock force continuing time.
And the integrity of the fresh fuel assembly was analysed to be safe.

We could analyse the behavior of the transport systems in the impact accidents by using the simulation code.

Even with various uncertain factors such as the mechanism of the cabin's crash(impact force),the analytical code empoyed for the present analysis work has made it possible to approximate the shock response of the transport systems.

It is,however,necessary to improve the code through and via the comparison of the obtained data with experimental data.

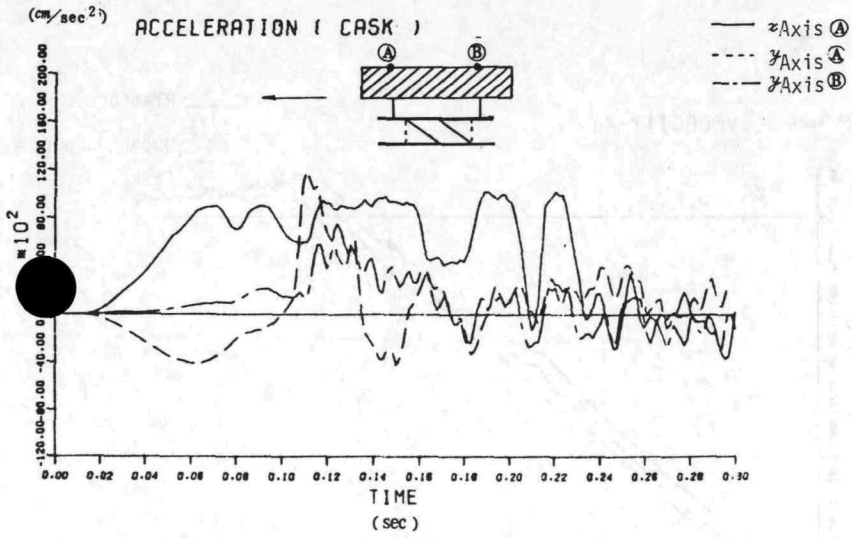


Fig.7 Acceleration of the cask

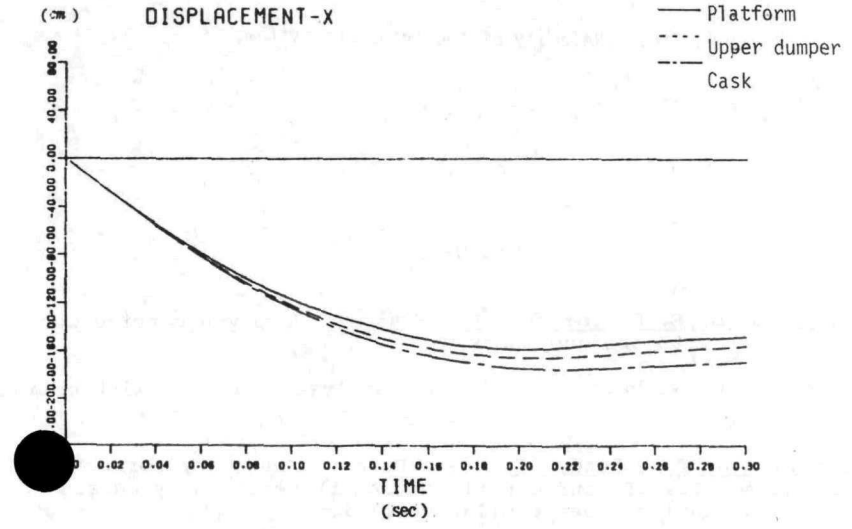


Fig.8 Displacement of the transport systems

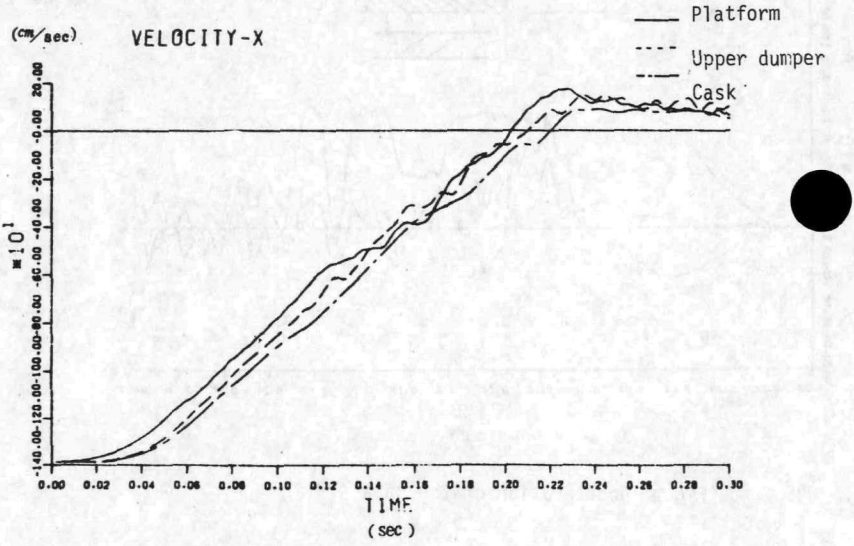


Fig. 9 Velocity of the transport systems

DISCUSSION

Question by E. G. Tomachevsky, France: Have you carried out previous tests with oblique shock?

Answer: No we have not. We only analysed the test with oblique shock.

Question by G. Cohendy, France: Have you made any correct between the results of your experiment and those given by Sandia National Laboratories under similar conditions?

Answer: We discussed the simulation model, impact time, acceleration data, etc. with Sandia National Laboratories under similar conditions.

ANALYSIS AND MODEL TESTING OF SUPER TIGER® TYPE B PACKAGING IN ACCIDENT ENVIRONMENTS*

H. R. Yoshimura, L. E. Romesberg, R. A. May, B. J. Joseph
Sandia National Laboratories, Albuquerque, NM, USA

INTRODUCTION

Sandia National Laboratories is attempting to predict the accident response of a Type B package containing drums of contact-handled transuranic waste (CH-TRU) as part of a program to evaluate the adequacy of experimental and analytical methods for assessing the safety of waste transport systems in accident environments. Impact tests using scale models[1,2] of the transportation systems are being conducted in conjunction with computer analyses.

A US NRC certificated Type B Package known as the Super Tiger®,† a crushable foam and steel overpack container for carrying Type A containers by either highway or rail transportation, was selected for this study. Super Tiger has the overall dimensions of a standard 2.4 m x 2.4 m x 6.1 m bimodal cargo container. The interior cavity of the overpack, which is 1.9 m x 1.9 m x 4.4 m, carries two tiers of 208-litre (55-gal) drums in a 3 x 7 array, totaling 42 drums. Construction of the Super Tiger consists of two steel shells separated by rigid polyurethane foam which provides both impact and thermal protection (Figure 1). The highway configuration, consisting of a tractor, flat-bed trailer and overpack, was used for both the scale model tests and the numerical analyses.

Prior to scale model accident testing, full scale and subscale impact testing of the contents and subscale regulatory drop testing of the packaging were performed to obtain data to assess the adequacy of the subscale models. The accident scenarios selected for the waste packaging tests were similar to those of a spent fuel cask test program conducted previously by Sandia.[3] The first was a 97 km/h (60 mph) head-on impact of the tractor-trailer-overpack system into a rigid barrier. The second was a 97 km/h (60 mph) locomotive grade crossing impact. Both accident scenarios represent very severe impact environments.

CONTENTS RESPONSE EVALUATION

Prior to performing the scaled transportation accident tests, full scale and subscale models of various subsystems (i.e., 208-litre drums and polyurethane foam) were statically and dynamically tested. Additional information on the drum tests is presented in a companion PATRAM paper.[4]

Crush tests of 208-litre 17 C drums (Type A packagings) loaded laterally in a static testing machine were performed to obtain load-deflection data for use in computer codes. Drum failure, defined by loss of head closure seal, occurred at approximately 10% crush (6.5 cm deflection) of the head. Loss of closure was defined as the point where the drum lid buckled and separated from the clamp ring and drum body.

*Work sponsored by U.S. Department of Energy under contract DE-AC04-76DP00789.
†Trademark, Protective Packaging, Incorporated

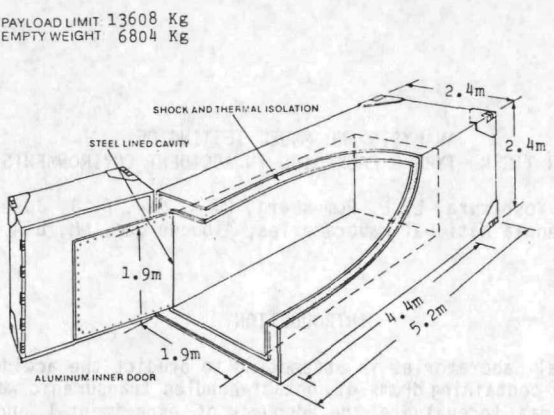


Figure 1. Schematic diagram of Super Tiger packaging

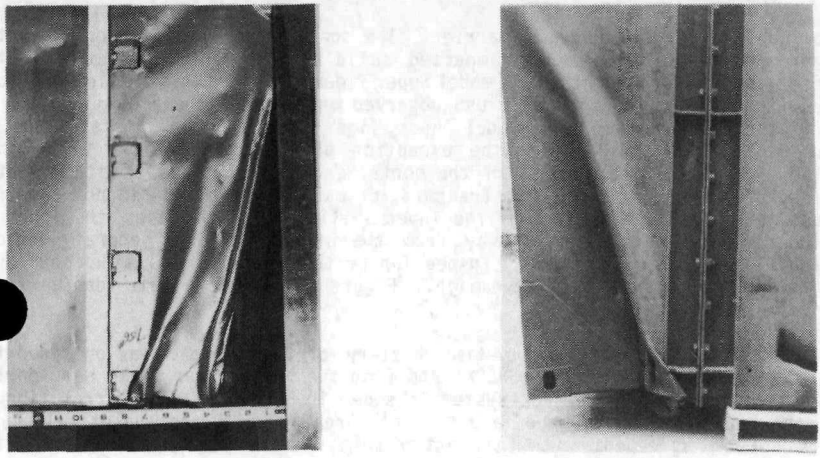
SCALE MODEL REGULATORY DROP TEST

After many of the subsystem tests had been performed, a one-quarter scale Super Tiger model, loaded with scale models of drums weighing 318 kg, was dropped 9 m onto an unyielding target in a center-of-gravity-over-corner orientation. Comparison of the crush sustained by the scale model to the crush of the prototype, reported in Reference 3, shows good correlation. Post-test measurements indicate the corner of the scale model crushed inward approximately 9 cm (36 cm full scale) while the prototype was reported to have crushed "less than 14 inches" (35.6 cm).[5] Visually, the damage appeared similar (Figure 2) and the inner and outer doors remained securely attached in both tests. Based on this test, it was estimated that approximately two-thirds of the drums in a full scale Super Tiger would be crushed sufficiently to open the head closure. These tests have provided realistic data for describing subsystems in analytical models and also provided verification of the adequacy of the scale model designs used in the subscale transportation system tests.

SCALE MODEL TRANSPORTATION SYSTEM ANALYSIS AND TESTS

Computer analyses have concentrated on the head-on impact scenario for the tractor-trailer-overpack system. A one-dimensional, lumped parameter model (SHOCK)[6] was used to predict overall system response (Figure 3a). A two-dimensional finite element model (HONDO)[7] was used to predict overpack and contents response (Figure 3b). The lumped parameter model indicated that the Super Tiger broke free from the trailer early in the crash and impacted the target at approximately 80 km/h (50 mph). The finite element model, using input from the lumped parameter model, showed a large amount of deformation in the Super Tiger and little deformation of the drums. These calculated deformations were not large enough to rupture the overpack system.

Scale model tests of the transportation system head-on impact have been conducted using both one-eighth and one-quarter scale models of the transportation system. The results of both tests were similar. After the initial impact the kingpin connection between the tractor and trailer broke, causing the front of the trailer to crush the



1/4 scale model

full scale

Figure 2. Post test view of a scale model and full scale Super Tiger packaging after a 9-m center of gravity over corner drop test onto an unyielding surface

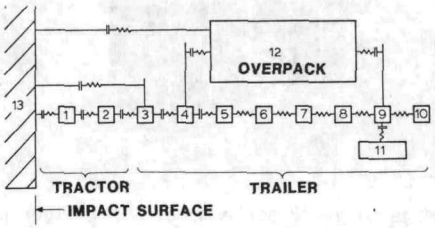


Figure 3a. Lumped parameter model of a truck mounted Super Tiger in a head-on impact into a barrier.

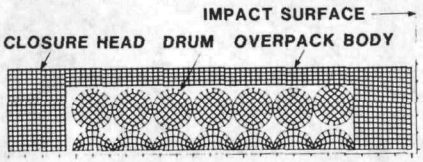


Figure 3b. Finite element model of the Super Tiger and drum contents for an head-on impact into a barrier.

tractor cab against the concrete barrier. The scale model of the Super Tiger then broke free from the trailer and impacted solid structure. Post-test inspection indicated that the one-eighth scale model Super Tiger exhibited very little structural damage with only 1 cm (0.4 in) of crush observed on the impact end (Figure 4). The crush of the one-quarter scale model Super Tiger was similar on a scaled basis to the one-eighth scale model with the exception of a large deformed area in the lower portion of the outer shell of the container caused by impact with the steel block representing the engine and transmission mass. The block did not penetrate or cause any localized failure of the inner shell. In both cases the inner and outer doors, which were oriented away from the impact, remained securely attached and visually appeared undamaged. Inspection of the model drums inside both overpacks revealed a large amount of damage. Figure 5 shows the drum damage from a one-eighth scale model test.

A comparison of the velocity-time history of the overpack as obtained from the lumped parameter analysis (SHOCK) and from the one-quarter scale head-on test for the tractor-trailer-overpack system is shown in Figure 6. The correlation between the test results is good, and the only area in which the analytical prediction is not in agreement with the test results is the overpack response after impact with solid structure.

Figure 4. Post test damage to a 1/8 scale model Super Tiger packaging after a 97 km/h head-on impact test

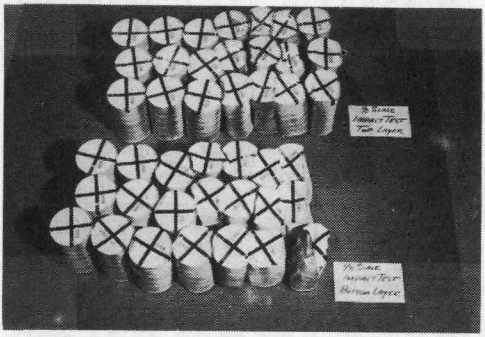


Figure 5. 1/8 scale model drums after a 97 km/h impact of a Super Tiger packaging into a concrete barrier

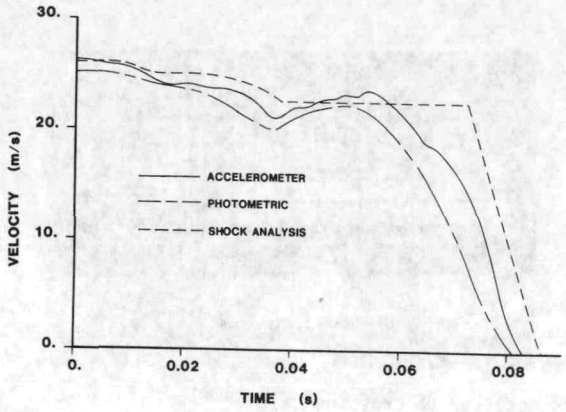


Figure 6. Comparison of velocity-time histories for the one-quarter scale model truck mounted Super Tiger in a 97 km/h head-on impact

The analytical model predicts abrupt deceleration upon impact, while the scale model shows a more gradual velocity change. This can be attributed to the debris crushing in front of, and gradually slowing, the overpack. The analytical model can be modified to include this effect by slightly softening coupling 12-13 shown in Figure 3a.

While no attempt has been made to analytically simulate grade crossing accidents because of the lack of a suitable three-dimensional analysis technique, two scale model tests have been performed. In both the one-eighth and one-quarter scale tests, a scale model locomotive traveling at 97 km/h (60 mph) impacted the center of the scale model Super Tiger with the transportation system placed perpendicular to the direction of travel of the locomotive (Figure 7). In both cases the Super Tiger and trailer were thrown to the side of the rail after impact. The outer cover of the one-eighth scale model Super Tiger detached during the subsequent tumbling impacts (Figure 8). This detachment occurred when the main body of the overpack separated from the trailer. The inner cover buckled but remained attached to the container. In the one-quarter scale test the Super Tiger remained attached to the trailer and both the inner and outer covers partially separated during the initial impact. Although the model drums were visible, all remained inside the overpack (Figure 9). In both cases there was deformation of both the inner and outer shells of the Super Tiger at the point of impact.

RELEASE ESTIMATES

In the Environmental Impact Statement for the Waste Isolation Pilot Plant,[8] for a hypothetical accident involving a Type B packaging carrying contact-handled transuranic waste, it was estimated that approximately 5% of the contents would be released to the environment.

From a comparison of contents response in each of the scale model Super Tiger tests, it can be inferred that approximately two-thirds of the drums in a full scale Super Tiger would have been crushed sufficiently to open the head closures. For the scale model tests performed, the greatest amount lost from the simulated drums

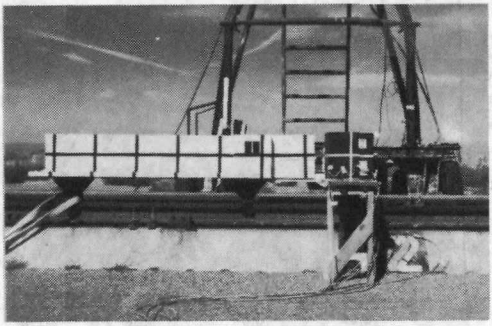


Figure 7. Scale model grade crossing test

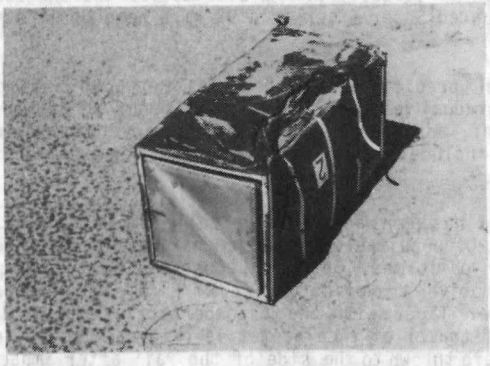


Figure 8. Post test view of a 1/8 scale model Super Tiger after a 97 km/h grade crossing impact test.

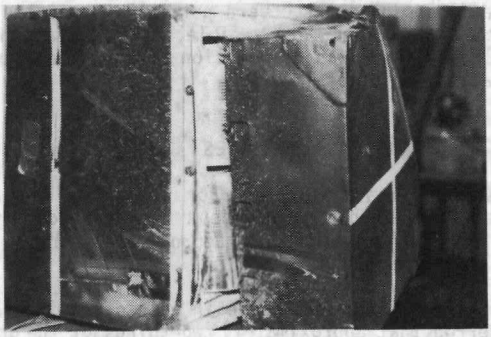


Figure 9. Post test view of a 1/4 scale model Super Tiger after a 97 km/h grade crossing impact test.

was measured to be less than 5% by weight of the total contents. This was accomplished without including in the models the additional containment protection offered by the polyethylene bag and liner inside the drum. In the four scale model tests of the Super Tiger, where the inner cover and seal remained attached, there was no evidence that any material was released to the environment. In the quarter-scale grade-crossing test, where both inner and outer doors partially separated, it was estimated that less than 0.5% of the surrogate waste by weight was released to the environment. The amount released in a full scale grade crossing test should be less because of the additional layers of containment in the drums, i.e., polyethylene bags and plastic liners. If the results of testing the scale model Super Tiger in severe accident environments were extrapolated to similar full scale tests, it is estimated that the release fraction of material in both scenarios studied would be expected to be less than the 5% estimated in the WIPP EIS.[8]

SUMMARY

In summary, based on previous scale model test results with more rigid systems and the subsystem tests on drums, it is believed that the scaled models realistically replicate full scale system behavior. Future work will be performed to obtain improved stiffness data on the Type A containers. These data will be incorporated into the finite element model, and improved correlation with the test results is expected. Review of the scale model transport system test results indicated that the method of attachment of the Super Tiger to the trailer was the primary cause for detachment of the outer door during the one-eighth scale grade-crossing test. Although the container seal on the scale model of Super Tiger was not adequately modeled to provide a leak-tight seal, loss of the existing seal in a full scale test can be inferred from the results of the one-quarter scale model grade-crossing test. In each test, approximately two-thirds of the model drums were estimated to have deformed sufficiently to predict loss of drum head closure seal, with several partially losing their contents within the overpack. In no case were drums ejected from the overpack, nor was there evidence of material loss in excess of the amount assumed in the WIPP EIS from any of the Super Tiger models tested.

REFERENCES

1. Yoshimura, H. R., and Huerta, M., "Full-Scale Tests of Spent-Nuclear-Fuel Shipping System," International Atomic Energy Agency Paper No. SR-10/17, August 1976.

2. Murphy, G., Similitude in Engineering, Ronald Press Company, NY (1950).

3. Yoshimura, H. R., "Full Scale Simulation of Accidents on Spent-Nuclear-Fuel Shipping Systems," Proceeding of the Fifth Internation Symposium on Packaging and Transportation of Radioactive Materials, May 1978, pp 463-476.

4. [redacted]y, R. A., Romesberg, L. E. and Yoshimura, H. R., Sandia National Laboratories, and Baker, W. E., and Hokanson, J. C., Southwest Research Institute, "Analytical and Empirical Evaluation of Low Level Waste Drum Response to Accident Environments," Sixth International Symposium, Packaging and Transportation of Radioactive Materials. November 10-14, 1980, Berlin, Federal Republic of Germany.

5. Hansen, L. J., and Haelsig, R. T., "Engineering Evaluation of the Super Tiger Overpack Designed for the Shipment of Large Quantities of Hazardous Materials," Mechanics Research Inc., Tacoma, Washington, May 4, 1970.

6. Gabrielson, V. K., and Reese, R. T., "SHOCK Code Users Manual, A Computer Code to Solve the Dynamic Response of Lumped Mass Systems," Sandia Laboratories, Livermore, California, Report No. SCL-DR-69-68, November 1969.

7. Key. S. W., "HONDO, a Finite Element Computer Program for the Large Deformation Dynamic Response of Axisymmetric Solids," Sandia Laboratories, Albuquerque, New Mexico, Report No. SLA-74-0039, April 1974.

8. Environmental Impact Statement Waste Isolation Pilot Plant, DOE/EB-0026-D, US Department of Energy, Washington, DC, December 21, 1979.

DISCUSSION

Question by C. B. G. Taylor, United Kingdom: Do you have in mind any improved type of lid closure in these drums? If so, please describe it.

Answer: Yes, Sandia Laboratories has evaluated methods for improving clamp ring closures and has developed an improved closure design. I believe this was reported by Bonzon and Otts in an IAEA seminar in 1976. To my knowledge, the use of the improved closure design has not been implemented in any quantity.

THERMAL ANALYSIS OF HNPF SPENT FUEL SHIPPING CONTAINER IN TORCH ENVIRONMENTS

Philip E. Eggers and Herman L. Crawford
Ridihalgh, Eggers and Associates, Inc., Columbus, OH, USA

A. A. Trujillo, R. B. Pope, and H. R. Yoshimura
Sandia National Laboratories, Albuquerque, NM, USA

SUMMARY

A thermal analysis of a modified HNPF spent fuel shipping container (cask) has been performed to predict cask temperatrues in simulated torch fire environments. In addition, this analysis also provided a basis for the modifications leading to the design of external cooling fins and a pressure relief valve system. An analysis of the torch fire literature was conducted in order to develop an empirically based thermal model for the torch environment. The scope of this study was analytical in nature and involved the use of a computer-based heat transfer analysis method. The product of this study is the prediction of cask temperatures and internal pressures under normal and torch fire accident conditions. The results will be compared with test results which will be obtained in early 1981.

INTRODUCTION

As a result of its concern with torch-fire accidents, the Federal Railroad Administration of the United States Department of Transportation (DOT) has obtained experimental data on torch fires. In these accidents, a hydrocarbon fuel torch emanating from a damaged tank car may impinge on an adjacent vessel. The torch-type fire involves high heat transfer coefficients over a localized area. A torch fire exposure test has been incorporated into regulations for hydrocarbon fuel tank car design. Although torch fire tests on spent fuel shipping casks are not required by U.S. regulations, the DOT was sufficiently interested in the effects of such tests that they contracted with the Transportation Technology Center at Sandia National Laboratories (SNL) to evaluate the effects of torch fires on spent fuel shipping casks. In preparation for testing, Ridihalgh, Eggers and Associates, under contract to SNL, performed a thermal analysis of a shipping cask in simulated torch fire environments and in conjunction with SNL designed modification to the test cask. The DOT specifications[1] for the torch were used in thermal analyses of the cask, and they will also be used as guidelines during the experimental phase of the program, which is presently scheduled for early 1981.

Because of the expense and time involved in designing and fabricating a special cask for experimentation, existing casks were surveyed to determine if any were available for testing. Based on availability, as cask of older design was selected. The Hallam Nuclear Power Facility (HNPF) cask, built in 1967, was made available for testing by the Department of Energy. Contemporary casks use ethylene glycol-water solution as a neutron shield and utilize energy-absorbing devices, attached to each and, to provide protection against impacts. The HNPF cask did not have a neutron shield or impact limiters. Therefore, in order to have the modified cask resemble contemporary units, a neutron shield and impact limiters were required.

APPROACH

The first task in this study was the definition of modifications which would be made on the HNPF cask to make it more representative of modern cask designs. Cask modifications included the design of impact limiters, an external jacket, cooling surfaces, pressure relief system, and specification of the inner cavity coolant.

This analytical phase of the program included the following tasks: (1) define cask modifications required, (2) estimate the thermal response of the cask during exposure to thermal environments defined for normal regulatory, accident transport conditions, (3) develop a three-dimensional thermal model of the modified HNPF cask with a neutron shield and impact limiters, (4) review literature on heat transfer characteristics of torches, (5) estimate the thermal response of the cask to a torch fire on the neutron shield with the neutron shield intact and with the neutron shield lost, and (6) evaluate the thermal response of the cask to a torch fire on the top, i.e., closure end, of the cask with the impact limiter in place. The findings of this phase will be presented in detail in the final report of this study.[2]

The experimental phase of the program will involve the following tasks: (1) torch fire testing, and (2) analysis of test results and comparison with analytical results. The results of the experimental phase of the program will be reported following the tests.

CASK MODIFICATION

The cask neutron shield was designed under the assumptions that the cask would carry four BWR spent fuel assemblies, each with a decay heat rate of 3.36 kW, which corresponds to a 120-day out-of-core cooling time and a burnup of 5600 megawatt days. The designs for the external cooling surface and the pressure relief system were based upon the thermal environments defined for regulatory normal and accident transport conditions.

Impact limiters were designed to the 9-m free drop onto an unyielding surface criteria as specified in IAEA Safety Series No. 6, Paragraph 708. The impact limiters for the modified HNPF cask use redwood as the energy absorbing material. The cask, when built, was 0.95 m in diameter, 6 m long, and had a mass of 32.6 tonne. The modified cask, excluding impact limiters, is nominally 1.45 meters in diameter and 6 meters long and has a mass of 38 tonne. Figure 1 presents the major physical features of the modified HNPF cask.

Heat Transfer Characteristics of Torches

A review of the literature indicated Reference 3 is the most relevant reference concerning the heat transfer characteristics of torches. The experimental study described in Reference 3 has become the basis for developing regulations on the construction of railroad tank car carrying hazardous materials.

A flat plate thermal model was developed to derive the heat transfer coefficient to be used with the three-dimensional model of HNPF cask. Assuming quarter symmetry, a 122 cm X 122 cm X 1.6 cm thick carbon steel plate was modeled according to the test details described in Reference 1 and analyzed using the TRUMP heat transfer computer program.[4]

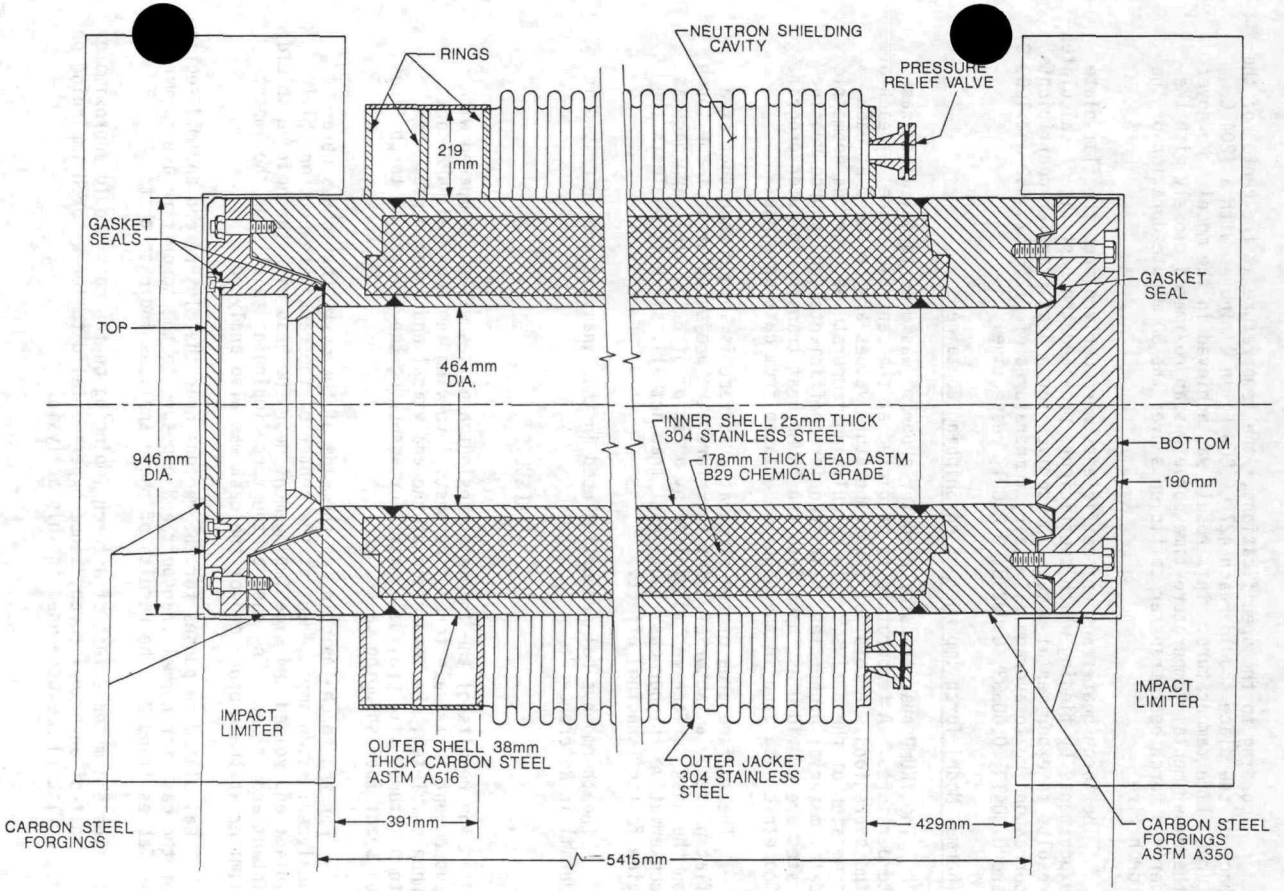


FIGURE 1. MODIFIED HNPF CASK USED IN TORCH ANALYSIS SHOWING IMPACT LIMITERS AND NEUTRON SHIELD

According to the specifications,[1] the temperature at the center of the back of the plate should reach 427 C (800 F) in 4 minutes with a 1204 C torch flame temperature. This result was achieved in the model. Moreover, the experimental temperature-time curve was approximated closely with the derived torch heat transfer cofficients over the 30 minute duration of the torch fire.

The torch heat transfer coefficients were derived from the flat plate experiments by finding which coefficients provided the best fit of calculated results to experimental data. The derived torch heat transfer coefficients corresponding to average film (air) temperatures of 612, 752, 852 and 1000 C are 0.00413, 0.00333, 0.00238, 0.00001, respectively.

Thermal Model Torch Impingement on Shipping Containers

The TRUMP heat transfer computer program[4] was used to calculate cask temperatures. A temperature is predicted at the centroid (node) of each of the subregions. A total of 549 temperature values are produced at each time step of the analysis. In calculating temperatures, various geometric data, material properties, heat transfer coefficients, and source heating rates are provided as input into the TRUMP heat transfer computer program. Geometric data used in the paper were taken from cask drawings.

The surface area of the corrugated surface is larger than that of a flat plate by a factor of 2.14; therefore, the projected area of the torch on the cask surface was multiplied by a factor of 2.14. This approach is conservative (higher cask heat input than actual), since it neglects the view factor (shading) effect.

The thermophysical properties used in this analysis are described in detail in Reference 2.

RESULTS

The results of the thermal transient analyses will be compared with subsequent full-scale torch fire tests involving the modified HNPF cask. When completed, the comparison of the analytical and experimental cask temperatures will provide a basis for assessing the effects of torch fires on spent fuel shipping casks.

For the torch impinging on the side of the cask, four cases were analyzed: side impingement at the center of the cask with neutron shield intact and voided and side impingement near the head with the neutron shield intact and voided. The case of the torch impinging axially on the impact limiter at the closure end of the cask was also analyzed.

Calculated responses for the regions near the closure end and mid-section of the cask for normal transport and exposure to the torch fire are summarized in Tables 1 and 2. The results of these analyses indicate that:

1. The estimated effect of a torch impinging on the cask should approximate that of an actual torch, since experimental data for a torch impinging on a flat plate were used in this analysis.

TABLE 1. SUMMARY OF CALCULATED THERMAL RESPONSE NEAR THE CLOSURE END OF THE MODIFIED HNPF SHIPPING CONTAINER FOR 13.45 KW DECAY HEAT AND 2.21 KW SOLAR HEAT LOAD

THERMAL EXPOSURE CONDITION	MAXIMUM TEMPERATURE, C			
	Outer Jacket 1[a]	Outer Shell 2	Inner Shell 3	Gasket Seal 4
Normal Transport	82	86	95	83
[redacted]-Hour Engulfing Fire[c]	704	315	213[b]	139
Half-Hour Torch Impingement on:				
• Side near closure end, neutron shield intact	237	178	153	112
• Side near closure end, neutron shield voided	740	269	169	100
• Impact limiter	Negligible effect on cask temperatures			

(a) Calculated temperature refers to this region of shipping container.
(b) Maximum temperatrue occurs 0.5 hour after end of fire.
(c) 10CFR Part 71 Engulfing Fire, 800 C, neutron shield lost at start of fire.

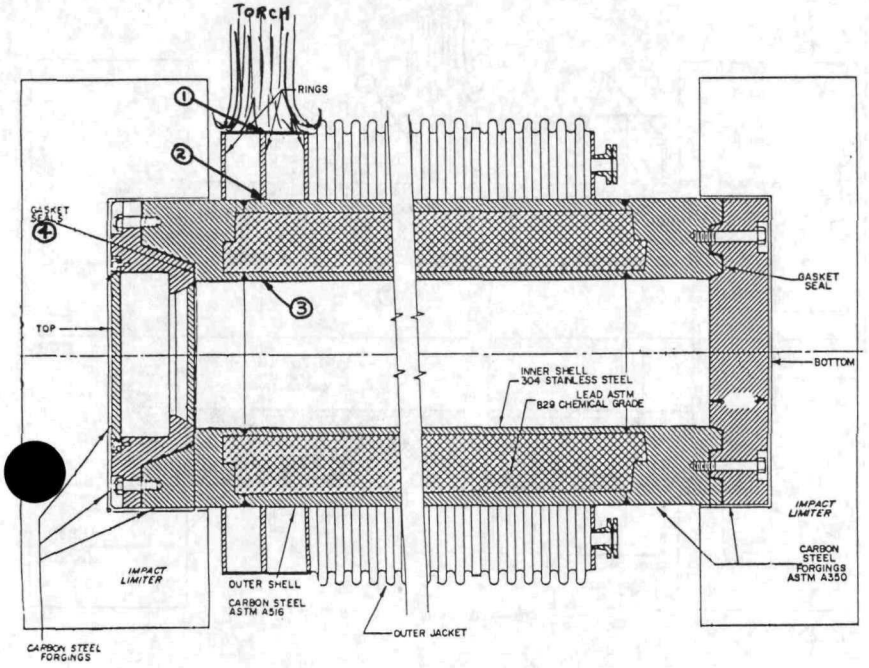
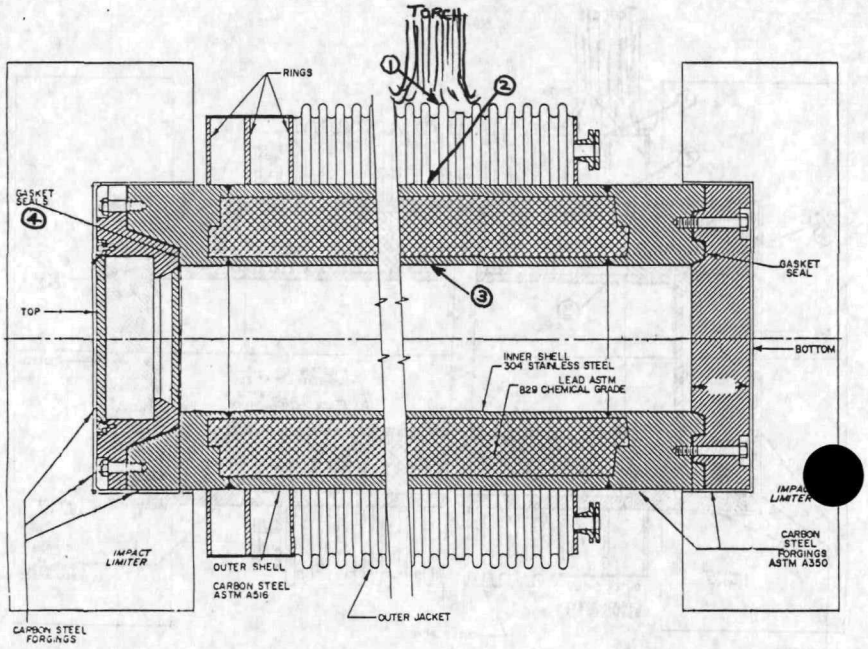


TABLE 2. SUMMARY OF CALCULATED THERMAL RESPONSE AT MID-SECTION OF THE MODIFIED HNPF SHIPPING CONTAINER FOR 13.45 KW DECAY HEAT AND 2.21 KW SOLAR HEAT LOAD

THERMAL EXPOSURE CONDITION	MAXIMUM TEMPERATURE, C			
	Outer Jacket 1[a]	Outer Shell 2	Inner Shell 3	Gasket Seal 4
Normal Transport	100	106	122	83
Half-Hour Engulfing Fire[c]	710	331	249[b]	139
Half-Hour Torch Impingement on:				
• Side near mid-section, neutron shield intact	175	153	149	90
• Side near mid-section, neutron shield voided	738	241	179	91

(a) Calculated temperature refers to this region of shipping container.
(b) Maximum temperature occurs 0.5 hour after end of fire.
(c) 10CFR Part 71 Engulfing Fire, 800 C, neutron shield lost at start of fire.



2. The 1204 C, 0.5-hour torch fire specified for U.S. railroad tank car design when applied to the modified HNPF shipping cask will not cause release of the neutron shield due to overpressure. Also, these results indicate that the maximum cask seal temperatrue will not exceed 112 C, which is below the temperature at which many elastomeric seals fail, and therefore, the 0.5-hour torch fire should not result in the release of any contents from within the inner cavity.

3. The quantity of heat absorbed by the cask during a side impingement of the torch fire with neutron shield intact is greater by a factor of 9 than if the neutron shield is lost prior to torch exposure. However, the maximum lead temperature on the outer periphery of the gamma shield is greater if the neutron shield is lost because heat is radiated directly across the empty cavity. With neutron shield intact, the liquid absorbs the heat from the fire and redistributes it around the circumference of the cavity by natural convection; therefore, the lead outer surface temperature is less than that with the neutron shield lost.

4. For torch impingement, at the mid-section of the cask, higher internal cask temperatures will occur with the neutron shield intact because of the greater heat transfer qualities of circulating liquid compared with those of an empty air space. However, because heat is absorbed in the neutron shield and is conducted inward after the torch fire exposure, the higher internal temperatures occur about 1 hour after the end of the torch fire.

5. The specified torch fire will not penetrate through an impact limiter. As a result, a negligible increase of cask temperatures is predicted with the torch impinging on the top end of the cask.

6. No torch impingement condition was found which would cause the cask to exceed temperatures which can be expected in the regulatory engulfing fire. Thus, the torch environment is less severe than the engulfing fire for which casks are designed.

REFERENCES

1. "Specifications for Tank Cars," 49 CFR Part 179, Federal Register, Vol. 42, No. 179, Sept. 15, 1977, pp. 46306-46315.

2. Crawford, H.L., and Eggers, P. E., Thermal Analysis of HNPF Spent Fuel Shipping Container in Torch Environments, Final Report prepared by Ridihalgh, Eggers and Associates for Sandia National Laboratories under Contract No. 13-6483, TTC-0037, to be published.

3. Anderson, C., et. al., "Comparison of Various Thermal Systems for the Protection of Rail Tank Cars Tested at the FRA/BRL Torching-Facility," Interim Memorandum Report No. 459, USA Ballistic Research Laboratories, Aberdeen Proving Ground, Maryland, Dec., 1975.

4. Edwards, A. L., TRUMP: A Computer Program for Transient and Steady State Temperature Distributions in Multidimensional Systems, UCRL-14754, Livermore, California, September 1, 1972.

Work sponsored by U.S. Department of Energy under contract DE-ACOA-96DP00789 through Sandia National Laboratories, Transportation Technology Center.

SESSION XX
STRUCTURAL AND THERMAL EVALUATION II

CANDU IRRADIATED FUEL TRANSPORTATION SYSTEM - DYNAMICS ANALYSIS

T. Loewen, J.W. Forest, M.A. Elbestawi
Ontario Hydro Research Division

ABSTRACT

A study to assess the structural integrity of irradiated fuel bundles during rail and road transport is presented. The assessment is based on impact and fatigue tests on irradiated fuel bundles, shaker tests of a loaded multi-tube transportation module and analyses of vehicle-flask response to various transportation environments. The results indicate that the irradiated fuel bundles will not be damaged while being transported with standard tractor-semitrailers or railcars equipped with hydraulic draft gear of suitable travel.

INTRODUCTION

The irradiated fuel bundles from Ontario Hydro's nuclear power generating stations are stored in water-filled bays. In the bays the fuel bundles (approximately 50 cm long, 10 cm in diameter and weighing 24 kg) will be held in modules consisting of 48 horizontal tubes supported by two tubesheets. Each module tube holds 2 bundles. It is planned to transport the modules with the fuel bundles from the stations to a central fuel management site in the future.

Concepts and requirements for road and rail transportation systems are being developed by Ontario Hydro(1). One of the transportation system requirements is that physical deterioration such as bundle disassembly, distortion or pencil breakage should not occur during normal transportation due to the shock and vibration environment to facilitate fuel handling by automated equipment after transportation.

The objectives of the study presented in this paper were to:

1. Determine whether the fuel bundle shock and fatigue strengths exceeded the transient and steady-state excitations the fuel bundles would be subjected to during transportation by rail or road.

2. Provide experimental data on the dynamic characteristics of the fuel bundles and module to assist in transportation system design and assessment.

3. Develop computer programs for calculating the response of vehicle-flask-tiedown systems to transportation environments to assist in transportation equipment design development.

FUEL BUNDLE TESTS

Impact and fatigue tests were performed on a number of fuel bundles in AECL's Whiteshell Nuclear Research Establishment hot cells(2). The bundle

elements consist of tubes containing sintered pellets of natural uranium oxide. The tubes are sealed with welded on end caps which are welded to the bundle end plate (Fig. 1). The bundle structural materials are zirconium alloys.

The test fixtures holding the fuel bundles were designed to simulate the support provided by the module in the proposed transportation system. This enabled the tests to identify the fuel bundle failure modes and the module tube excitation levels above which failure of the bundles in the module tube would occur.

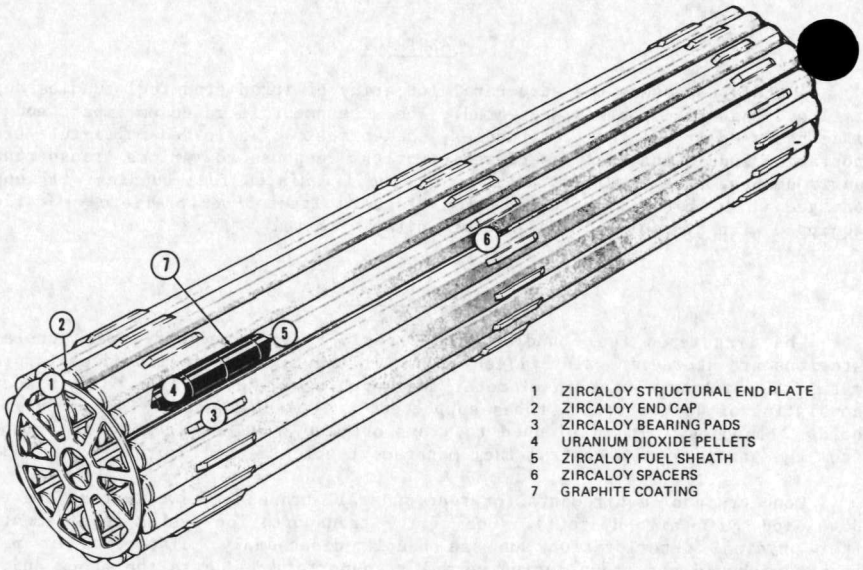


FIGURE 1

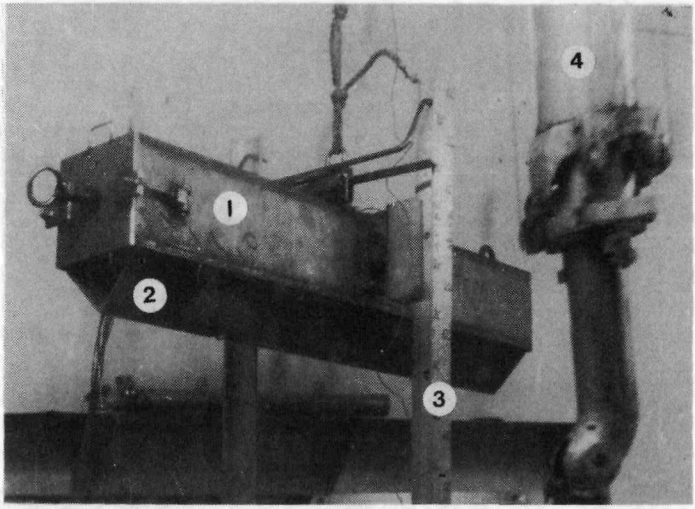
FUEL BUNDLE FOR PICKERING REACTOR

Impact Tests

Fuel bundle impact tests were performed in both the lateral and axial directions. The axial tests were conducted on one fuel bundle in a tube having the same diameter and wall thickness as the module tubes. The tube was suported by two tubesheets enclosed in a rigid casing. For the lateral impact tests two bundles were contained in the test fixture to closely simulate the two bundle per tube arrangement in the modules so that the lateral tube response and fuel b[redacted]dle interaction would be included (Fig. 2). In this case the tubesheet suppor[redacted]e in the same location as in the module. The fixtures also incorporated a [redacted]ce for axially preloading the bundles, as this provision is being considered for the transportation system.

The shock loading was obtained by dropping the fixture, guided by rails, into a bed of sand. The rise time of the fixture response to the shock was shaped, by the addition of suitable penetrators on the fixture, to be in the 5 to 10 ms range which is typical for railcar end-impacts.

Two 28 element (Pickering) and two 37 element (Bruce) bundles were tested in the axial mode and one of each type were tested in the lateral mode. The tests were started at a low drop height which was increased in increments until the fuel bundle failed or the maximum drop height allowed (92 cm) by the hot cell test facility was achieved. After every test the bundle was removed from the fixture and inspected for structural defects. None of the bundles failed. The maximum peak decelerations achieved were: Pickering bundle - 165.8 g axial and 55.0 g lateral; Bruce bundle 175.0 g axial and 78.0 g lateral.



(1) Test fixture containing 2 fuel bundles in hot cell
(2) Penetrator
(3) Guide rails
(4) Manipulator Arm

FIGURE 2

TEST FIXTURE - LATERAL IMPACT TESTS

Fatigue Tests

The fatigue tests were performed with the same test fixture as the lateral impact tests, containing two bundles. A continuous sinusoidal excitation was provided by an electronically controlled hydraulic shaker. An input excitation frequency of 21.25 Hz was chosen to avoid bundle, fixture and shaker resonances. The input vibration level was monitored with an accelerometer at a central location at the top of the fixture. Fuel bundle failures were detected primarily by periodic bundle inspection, but on-line waveform and spectrum analyses, as well as sine-sweep resonance searches were also utilized.

Since the maximum time in transit would likely be less than 24 hours, the tests were conducted to determine the maximum input level, applied for 24 hours, for which the bundles would not fail. On this basis the maximum allowable peak acceleration levels were found to be 0.5 g and 0.75 g for the five Bruce and five Pickering bundles tested respectively. Both types of bundles were also found to be able to withstand excitation levels of 2-3 g for periods upto 1 hour.

Several types of fuel bundle failure occurred. The most common types were by fracture of the end plate near the end cap of the fuel elements, and by an end cap pulling out of the end plate. There was also one incidence of a circumferential crack in the fuel element sheath. The failed fuel bundle components were sectioned, metallographic examinations were performed and the end plate hydrogen/deuterium concentration was analyzed. The hydrogen/deuterium concentration in the irradiated bundles ranged from 25-112 ug/g as compared to the normal as-received concentration of 10-15 ug/g.

MODULE TESTS

Shaker table tests of a fully loaded module (Fig. 3) were performed to determine the module-bundle dynamic characteristics. Three of the module tubes on which accelerometers and strain gauges were mounted contained unirradiated fuel bundles while the remaining 45 tubes held steel billets of similar external dimensions and mass as the bundles. The module was rigidly fixed to the shaker table and an axial preload was applied to the bundles. Sweep tests covering the frequency range of 5 to 200 Hz with a sinusoidal input at a level of 0.2 to 0.3 g peak were performed for three module axes independently. Random and half-sine-pulse impact tests were also performed in the vertical axis.

Analysis of the tube response as measured by the accelerometers and strain gauges indicated that the major response of the module tubes occurred in the range of 25 to 30 Hz. The increased response in this frequency range was determined to be due to inter element bundle action and bundle rocking in the tube. However, the amplified tube response was less than two times the shaker table input. At other frequencies less than 200 Hz the tube amplification was negligible.

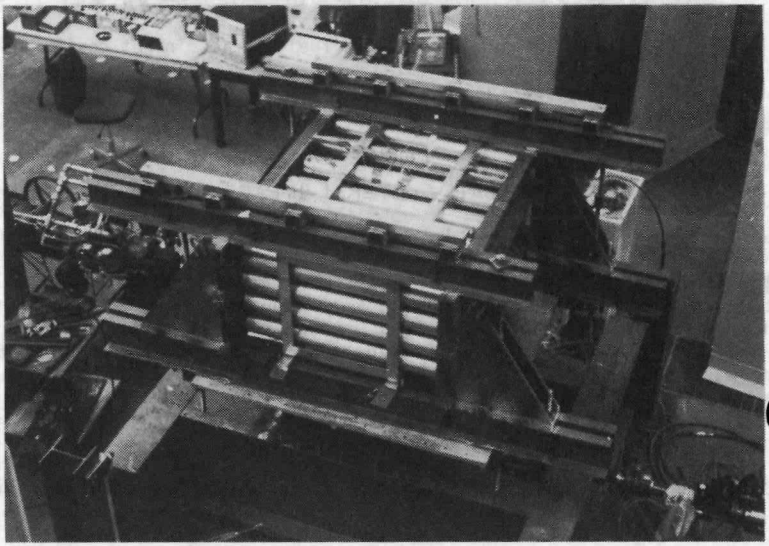


FIGURE 3
LOADED TRANSPORTATION MODULE ON SHAKER TABLE

RAILCAR END IMPACT RESPONSE

A standard 90 tonnes freight car carrying a 62 tonnes flask was modeled, using two-dimensional beam elements. A computer program was developed to compute the system natural frequencies, mode shapes and transient response to end impact with different combinations of draft gear; friction-friction, hydraulic-hydraulic, and friction-hydraulic(3).

The most important results were obtained from analysis of the horizontal and vertical acceleration of the vehicle and flask at various locations for a range of end impact speeds up to 16 km/h (Fig. 4). The analysis indicated that with a standard draft gear (89 mm travel) the impact energy absorption capability of the gear was exceeded for impact speeds in excess of 6.4 km/h. This resulted, for example, in an increase in the horizontal deceleration of the railcar at the point of impact from 21 g at 6.4 km/h to 475 g at 16.0 km/h. With a draft gear having a 254 mm travel the deceleration at the point of impact is reduced to 53 g at an impact speed of 16 km/h.

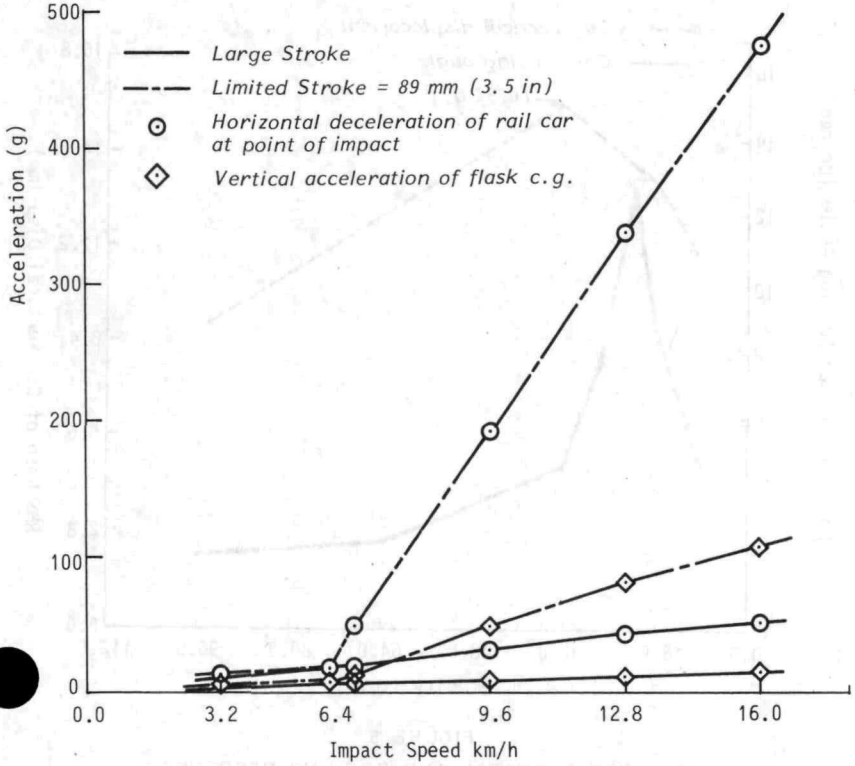


FIGURE 4

PEAK ACCELERATION DURING IMPACT
FOR FRICTION-HYDRAULIC DRAFT GEAR COMBINATION

RAILCAR STEADY-STATE RESPONSE

The railcar and flask were modelled as rigid bodies mounted on two trucks. Each truck consists of two wheel sets, two side frames with springs and friction dampers. The car body sits on two bolsters (one on each truck) at the centre plate location. In this manner, a swivel movement can take place between the bolster and the car body. A nine degrees of freedom nonlinear model was used to adequately include all the important characteristics of the railcar. The railcar excitations due to track segmentation and random irregularities such as surface roughness and rail installation are represented by a combination of periodic and random processes(4).

The analytical results indicate that for speeds up to 100 km/h, the ma[illegible] railcar rocking response of 12.8° (rms) and vertical acceleration of 0.32 g [illegible]-curred at 19 km/h and 32 km/h respectively (Fig. 5).

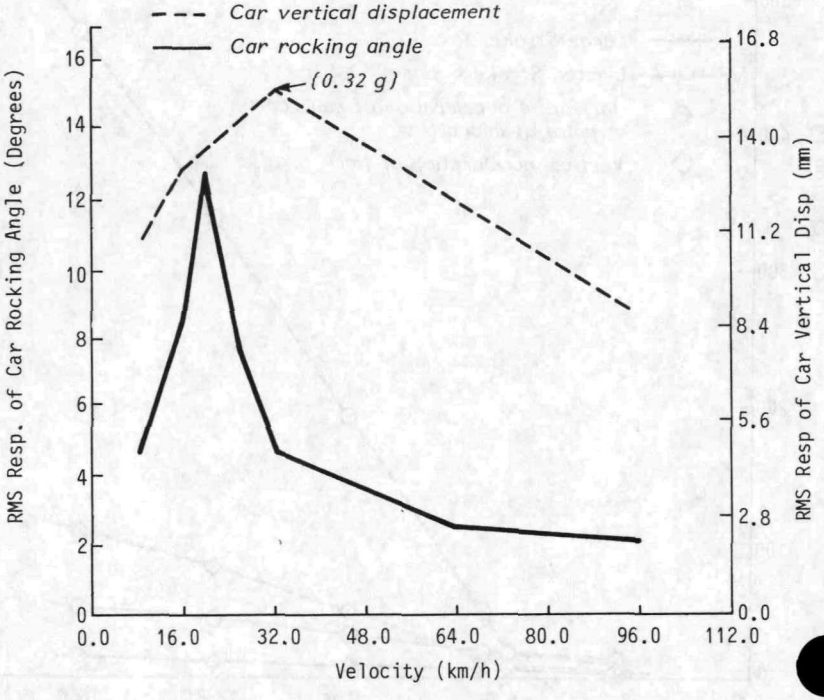


FIGURE 5

RAILCAR VERTICAL AND ROCKING RESPONSE VERSUS RAILCAR SPEED

TRACTOR-SEMITRAILER STEADY-STATE RESPONSE

The response to road excitations of a standard tractor-semitrailer carrying a 27 tonnes flask was analyzed utilizing a two-dimensional model considering the longitudinal and vertical planes(5). The cab, engine and tractor chassis is considered as one rigid body as is the semitrailer with the flask-tiedown system. The vehicle suspension is represented by linear springs and linear viscous dampers. Each of the tires on the tractor and semitrailer is assumed to be a linear spring and damper having point contact with the road surface. The model allows six degrees of freedom: pitching and bouncing motions of the tractor centre of gravity, pitching motion of the semitrailer centre of gravity and vertical motion of each wheel and axle assembly (Fig. 6).

The excitation of the transport system due to road irregularities was represented as a stationary Gaussian random signal. The vehicle and flask response to a range of road surface roughnesses from good to very poor was obtained in terms of the power spectral density. Also, the root mean square acceleration of the flask for the range of road classes was computed. Generally, the vertical acceleration of the flask increases with vehicle speed and road roughness. A system resonance resulting in an amplified response is indicated at 40 km/h. The maximum flask acceleration predicted at speeds up to 100 km/h is 0.2 g rms on a very poor road (Fig. 7).

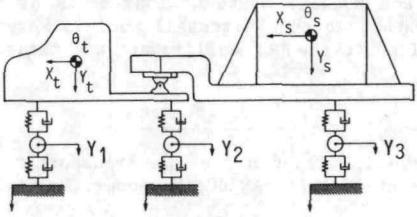


FIGURE 6
ARTICULATED VEHICLE MODEL

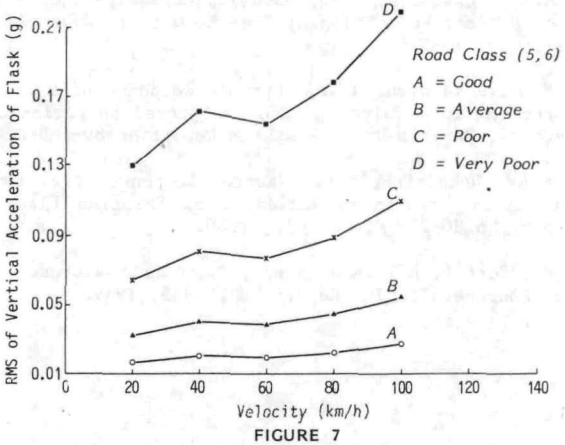


FIGURE 7
VERTICAL ACCELERATION OF THE FLASK VEHICLE SPEED

CONCLUSIONS

1. The maximum steady-state flask response during rail or road transport, as determined by computer analyses has been found to be less than the experimentally determined fuel bundle fatigue strength. Thus most fuel bundles are not expected to fail during transportation under the conditions considered.

2. Railcars should be equipped with hydraulic draft gear having sufficient travel to prevent fuel bundle failures and excessive loads on the railcar, flask and tie down system due to end-impacts. The analytical results indicate that for end-impact speeds up to 16 km/h, the draft gear should have a travel of at least 254 mm.

3. It is expected that the experimental fuel bundle strength data, determination of module dynamic characteristics and analyses techniques developed will assist in development of a reliable and efficient irradiated fuel transportation system.

ACKNOWLEDGEMENT

Objective setting and funding for this program were provided by the Nuclear Materials Management and Nuclear Systems Departments of Ontario Hydro. Atomic Energy of Canada Ltd staff at the Whiteshell Nuclear Research Establishment made significant contributions to the hot cell impact and fatigue tests.

REFERENCES

(1) Kortright, R., et al, "Program for the Transportation of Irradiated Candu Fuel", to be presented at PATRAM'80, November 10-14, 1980.

(2) Forest, J.W. Forest, "Impact and Fatigue Strength of Irradiated CANDU Fuel Bundles", Ontario Hydro Research Report No 80-321-K, August 14, 1980.

(3) Elbestawi, M.A., Dokainish, M.A., "End Impact Analysis of a Railcar Carrying an Irradiated Nuclear Fuel Shipping Flask", Ontario Hydro Research Report No 80-53-K, March 19, 1980.

(4) Elbestawi, M.A., Dokainish, M.A., "Dynamic Response of a Railcar Carrying an Irradiated Nuclear Fuel Shipping Flask Subjected to Periodic and Stochastic Rail Excitation", Ontario Hydro Research Report No 80-190-K, May 30, 1980.

(5) Elbestawi, M.A., Dokainish, M.A., "Random Response of a Tractor-Semitrailer System Carrying an Irradiated Nuclear Fuel Shipping Flask, Ontario Hydro Research Report No 80-209-K, June 12, 1980.

(6) LaBarre, R.P., Forbes, R.T. and Andrew, S., "The Measurement and Analys Road Surface Roughness", MIRA Report No 197915, 1969.

DISCUSSION

Question by N. A. Eisenberg, USA: I have a two part question: 1. Were the vibration input spectra used in the computer modeling obtained from measured data or theoretical models? 2. You indicated that fatigue measurements were performed at non-resonant frequencies; however, the frequency content of shock and vibration experienced in transport are likely to be rather broad band. Would you expect fatigue failure threshold to vary substantially with frequency, particularly at or near resonant frequencies?

Answer: 1. The input spectra were based on theoretical models. The theoretical models were compared with measured data and found to be in reasonably good agreement. 2. A shaker input excitation frequency at other than a resonant frequency was chosen for the fatigue tests so that a constant input amplitude level could be maintained by the shaker. The actual input to the fuel bundle was broadband due to fuel bundle/tube interaction. Sweep tests were also performed to obtain the input amplification at resonances. The amplification at resonant frequencies is considered in assessing the fuel bundle environment.

Question by G. Malkoske, Canada: What difference in 'g' levels for the fuel bundels have you found between those experienced in transportation and those determined by testing?

Answer: During transportation of fuel bundles by road the input 'g' levels measured have been generally less than 0.2 g with some transients up to 1 g. The fuel bundle fatigue failure threshold levels determined by test exceed the levels experienced during transportation. The test values are given in the paper. Also, the peak values during transportation occur for only brief periods whereas during testing the bundles were subjected to the threshold input levels continuously for 24 hours and to higher levels for durations of approximately one hour.

MEASUREMENT OF VIBRATIONAL EFFECTS ON AN ASSEMBLY OF PWR FUEL ELEMENTS DURING HANDLING AND TRANSPORT BY ROAD AND RAIL

J. P. Prulhière
Commissariat à l'Energie Atomique,
Centre d'Etudes Scientifiques et Techniques d'Aquitaine,
Le Barp, 33830 Belin Beliet, France

F. Israel
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
B.P. no. 6, 92260 Fontenay-aux-Roses, France

Note: This is a summarised version of the full proceedings text. For more details please contact Mr. J. P. Prulhière at the above address.

1. INTRODUCTION

It is necessary to know the vibrational environment affecting a fuel element during the usual stages of its lifetime - transport, handling - for calculating the failure probability of the fuel element cans during back-stages between nuclear power stations and reprocessing stations, and carrying out simulation tests on fuel element subassemblies in the laboratory from actual data.

This paper presents the results obtained by the "Centre d'Etudes Scientifiques et Techniques d'Aquitaine - that is a CEA establishment in France - during a test series in the course of the first quarter of 1980.

2. TEST PROGRAMME

This test concerns a simulation of the transport of irradiated fuel elements of the PWR type between a nuclear power station located in South-East France and the reprocessing station at La Hague.

Three stages were studied:

- a lorry stage describing transport between, on the one hand the nuclear reactors and the nearest goods station, and on the other hand Cherbourg station and La Hague centre;
- a train stage;
- a handling stage.

The starting point was located at the Bugey nuclear power station and the point of arrival at the Tricastin nuclear power station.

Between departure from the nuclear power station and arrival at the reprocessing station many handling processes are necessary. They consist of rolling, lifting, shifting and setting down the cask on a suitable means of conveyance, that is to say a lorry (in the power station) and specialised lorries and trucks.

The shipping procedure can be divided as follows:

- transport of the cask to the loading point;
- loading on to the lorry and shifting - mean duration: 3 hours;
- transport from La Hague to the railway station - mean duration: 2 hours;
- loading and setting down the cask on the truck - mean duration: 3 hours;
- joining the truck to the instrumented railway coach - mean duration: 3 hours including checking of the measuring instruments;
- railway transport from La Hague to Tricastin - duration ranging from 4 to 6 days;
- unloading on to the truck and shifting;
- setting down the cask on the ground.

Planning of the tests took 3 months including one month for setting up instruments.

Railway transport is common and it must be pointed out that 1000 km transport takes from 6 to 8 days, which involves long stays in marshalling yards.

The specimen to be tested was a PWR assembly mock-up installed in a TN 12 transport cask of 100 metric tonsbelonging to the Transnucléaire company. It was not fitted out with support baskets for twelve assemblies.

3. MEASURING INSTRUMENTS SET UP

The measuring instruments set up, including 34 simultaneous measuring channels, were chiefly composed of accelerometers.

The prospective assembly point during a transport stage was simulated by an extremely rigid structure. As the mass of the basket could not be reproduced, the bond between the basket and the cask shell was made rigid whereas there is in fact a large clearance.

The measuring channels located outside the cask were placed either on the trunnions or on the truck floor - rigidly bound to the truck floor.

Because it is difficult to have access to the inside of the cask after closing the lid (it is necessary to foresee shifting, therefore heavy specific equipment and in particular a 120 t crane) all the internal instruments of the specimen were doubled.

Grids were chosen (eight three-axes accelerometers). The micro deformations on a few rods were measured, in order to compare them with the existing data from structure computer codes.

A railway coach laboratory belonging to the SNCF (French national railway company) was coupled on to the truck in order to carry measuring and computing electronic hardware and accommodate employees during the tests.

Experimental data were recorded:

- on analogical magnetic tape (34 simultaneous channels);
- digitally by means of a specially built computer (maximum capacity: 16 simultaneous channels).

In addition to measuring the values of the parameters referring to the truck floor and the inside of the cask, the radial acceleration of a train wheel (that is a rotating vector) was recorded by means of a telemetering process.

For handling and road transport stages, more compact measuring electronic hardware operating by battery was used.

4. PROCESSING (with examples)

The large amount of information recorded during the tests required data condensation techniques in common use in signal processing technology. The most common ones will be cited as an example (calculation of effective values, power spectral densities, shock spectra, etc.):

- Signal as a function of time

The easiest processing consists of plotting signal against time.

Accelerations were measured on the first grid as well as micro-deformations collected on two rods during the handling stage.

Low acceleration levels (peak values 0.3 m/s^2) and the appearance of elastic bending waves (30 Hz) on rods along the vertical axis can be observed.

- Effective value as a function of time

Unfortunately, this first method does not condense data. However, it can be analogically achieved easily enough by computing the effective values of accelerations in a given frequency band and chosen on a sampling basis.

(The effective value of a signal is the integral of the rectified signal. In the case where the mean value of the signal is zero, it is equal to the standard deviation of the measured magnitude distribution).

A thirty minute recording covers the accelerations measured on the first grid during the rail stage where the train speed ranging from 0 to 80 km/h is indicated on the same sheet.

We noted that when the train stopped and when it went over switch points, measured effective levels reach 0.25 g. There is no correl-

ation between velocity and measured levels, but we noted that the highest levels resulted from switch points.

The method makes it possible to choose the zones whose γ levels are the highest, and contract data to a large extent (effective value calculation is made every 0.3 s whereas peak values can result from frequencies nearing 150 Hz).

- Power spectral density

It is important to know frequencies from measured signals for checking whether the frequencies received are those that make the structure vibrate. Power spectral density (PSD) is currently used for that purpose. PSD is the average of the Fourier transformation modulus from a time signal. It can be analogically obtained by means of a band pass filter whose central frequency has been fluctuated, a rectifier and an integrater; but small special-purpose computers have superseded such a device.

It is then possible to combine the above mentioned time and frequency processes.

- Another frequency processing type (SANDIA)

Another method developed by Sandia Laboratories consists of examining the signals after processing in band pass filters, computing their histogram distribution and statistical parameters (average, standard deviation) and representing the maximum level consistent with a chosen confidence level (for instance 95.6 % for 2σ).

An outline of processes meant for plotting peak values (with a confidence level) is as follows:

- acceleration signal to be processed;
- output signal from the band pass filter;
- histogram of the peaks;
- statistical parameter computation - average, standard deviation asymmetry, etc.;
- plot of possible levels (with a given confidence level);
- asymmetrics;
- oblateness

- Shock spectra

In the case of short phenomena (as regards the response of the structure) the most widely known process consists of calculating a shock spectrum defined by the maximum displacement of a linear mechanical resonator. Each resonator is chosen in order to obtain the same damping. When the resonance frequency varies, a curve called a "shock spectrum" is obtained.

5. RESULTS

As most processes concern the train stage, the results from this stage will be emphasised. They are compared with those provided by Sandia in 1971.

- In-situ measurements required

The environment measured on a fuel element assembly is necessarily different from that existing on the truck floor. It depends on the transfer functions of the structures linking floor and assembly.

In order to account for this fact, the acceleration measur[illegible] along the axis at different places in the structure (train whee[illegible] truck floor, cask trunnion, assembly) is represented on the sam[illegible] plain.

Signals with very different magnitudes and shapes are always noted, but only some correlated frequencies.

Fig. 1 shows accelerations measured during rail transport and measurement point influence. (Note: time signals AC 4, AC 3 and AGE are drawn on the same scale).

Investigating vibrations measured on the assembly shows the environment to be hardly ever homogeneous over all the grids.

When examining signals measured on the truck floor (between two points) and trunnions, the results are even more convincing. The same conclusion is reached after studying the main frequencies noted. They are different, according to their place and the measuring axis.

- Comparison with Sandia tests

CESTA data were compared from one axis to another with those provided by Sandia.

Data are given in Fig. 2.

We find that the environments calculated by Sandia Laboratories are more important than those which we measured, except for one or two frequency bands.

In the first place the comparison involves similar directions but different measurement points (the floor as regards Sandia, the assembly as regards CESTA), for Sandia Laboratories did not carry out experiments on the cask with three-axes transducers.

This may partly allow for the difference.

It is likely that the levels measured are influenced to a [illegible]ge extent by the condition of the railway lines.

Moreover we think that the differences between the two series of results are amplified more by the statistical processing of signals by Sandia Laboratories, because they consider the distribution of peak resulting in a given confidence level to be symmetric, whereas it is not in every case. This hypothesis which is safety-orientated re-

sults in great increases in the levels measured and accounts for the differences between the results to a large extent.

We have discarded such a method and prefer techniques which are not based on such hypotheses.

Peak to peak levels met on the assembly during rail/road stages and handling, and frequencies occurring most often have been collected into data tables. Results are shown in Fig. 3.

6. CONCLUSION

The measurement of vibrational parameters (acceleration and microdeformations) was carried out on a fuel element assembly during normal environmental stages (rail/road transport and handling).

Its utilisation showed:

- acceleration levels lower than those measured by Sandia Laboratories during a similar test;
- difficulties in comparing two tests, a priori similar, on account of the differences between measurement locations;
- discrepancies between results proper to the American document, which probably resulted in overrating levels actually measured;
- the necessity for carrying out measurements in situ;
- resonances in several subassemblies (cask supports, grids, bars, etc.);
- very low levels during handling stages.

Nevertheless, the characteristics of the chosen environment do not encompass all possible normal events. As a matter of fact:

- many precautions were taken during handling and transport by road, shocks being avoided as far as possible. That was mainly due to the new type of cask and the technical help lent during the test (presence of the constructor for instance), which would not occur when the cask is in service;
- some stages of transport which did not take place (but which took place in the US and from which high levels of acceleration were derived) are based only on regulations. Especially hump-switching in a marshalling yard and end-on collisions.

Therefore the levels measured must be considered as basic values.

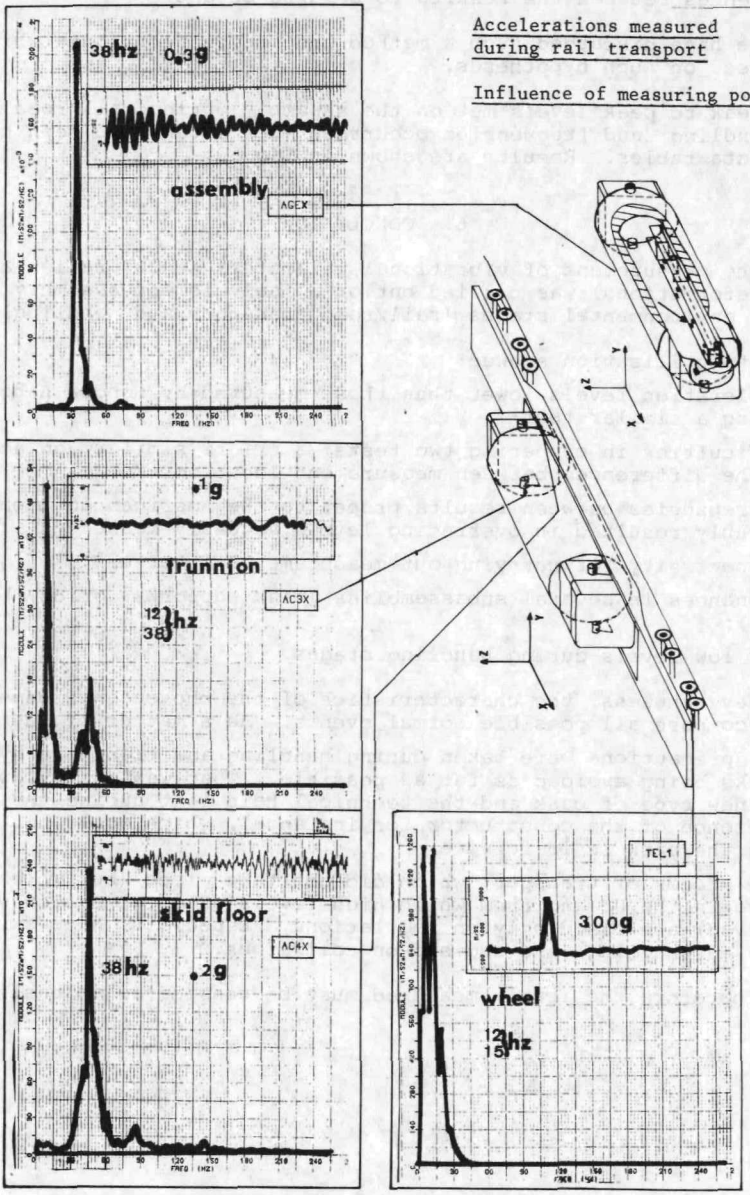
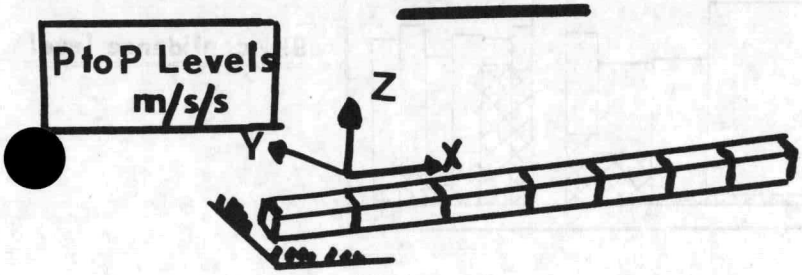

Accelerations measured during rail transport
Influence of measuring point
38hz 0,3g
assembly
AGEX
•1g
trunnion
AC3X
12
38 hz
skid floor
AC4X
38hz .2g
TEL1
300g
wheel
12
15 hz

Figure 1

RESULTS

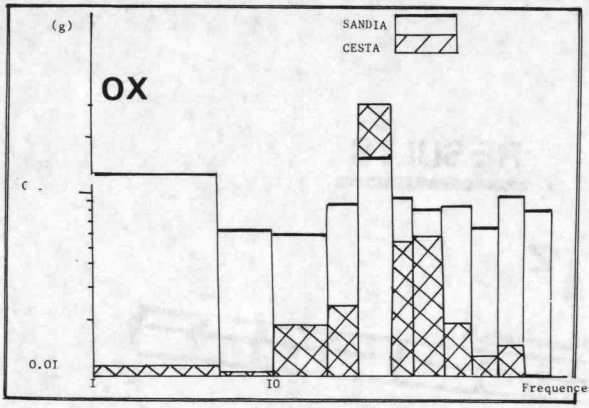


	RAIL	ROAD	HANDLING
X	6	1.5	2
Y	4	1 to 5	2
Z	4.5	.6 to 4	2

FREQUENCY

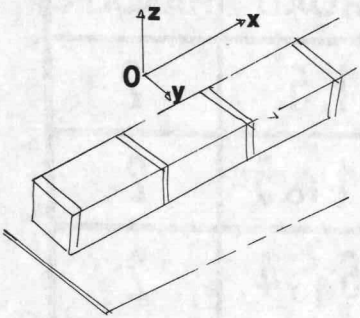
X	35	35
Y	3 10 35	"
Z	4 12 24 36	4 10

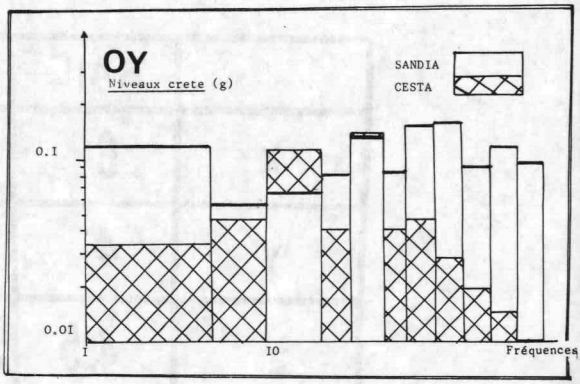
Figure 2



RAIL

99% confidence level





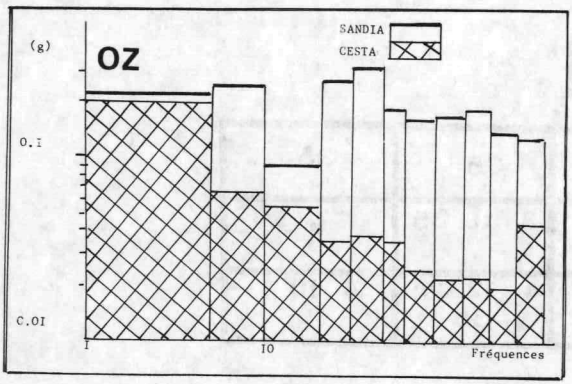


Figure 3

DISCUSSION

Question by A. Müller, FR Germany: What is the conclusion of your investigations on spent fuel elements in real transport? Do you have experience in fuel element behaviour in real transport situations, especially with respect to fuel element failure during transport?

Answer: We wanted to exactly understand the vibrational environment on the PWR fuel element assembly and not what happens on the truck floor or the cask wall such as has been done up until now (e.g. S[redacted]A 1971). From the results provided, we now want to calculate d[redacted]e to the grid spacers and the fuel rods with the help of a damage calculation programme using the MINER cumulative hypothesis. We will check this theory with fatigue tests (using the real measured environment) up to rupture of the fuel rods and the gauges. The results will be know in 1981.

Question by H.-J. Engelmann, FR Germany: Did you carry out your simulated tests under normal transport conditions with heat?

Answer: No, our only aim was to simulate normal transport under vibratory conditions in order to obtain test fatigue specifications on the irradiated and non-irradiated grid spacers and fuel rods. We intend to introduce the thermal aspect in obtaining the test specification by modifying the transfer functions by a temperature function.

Question by B. Balmer, Switzerland: 1. How do the measured acceleration spectra for rail transport differ from those for road transport? 2. How does the speed of travel influence the accleration spectrum?

Answer: 1. There are important differences between the rail and road environment, both between the levels and between the frequences observed. There are of course frequencies inherent in the assembly (e.g. 36 Hz) but the other frequencies characterise the mode of transport or the way in which the container is secured. 2. There is a small spectrum modification as a function of the travel speed. But for certain speeds it is preferential to excite certain modes.

DYNAMIC ANALYSIS TO ESTABLISH NORMAL SHOCK AND VIBRATION OF RADIOACTIVE MATERIAL SHIPPING PACKAGES

S. R. Fields
Hanford Engineering Development Laboratory
Westinghouse Hanford Company
Richland, Washington

Packaged radioactive material is transported within the United States by both truck and rail. Regulations governing the shipment of such material have been issued by both the U. S. Department of Transportation (US-DOT) and the U. S. Nuclear Regulatory Commission (US-NRC). The US-NRC is primarily concerned with maintaining the structural integrity of the radioactive material shipping packages. A computer model, CARDS (Cask-Railcar Dynamic Simulator) was developed by the Hanford Engineering Development Laboratory for the US-NRC to provide input data for a broad range of radioactive material package-tiedown structural assessments.

CARDS simulates the dynamic behavior of shipping packages and their transporters during normal transport conditions. The model will be used to identify parameters which significantly affect the normal shock and vibration environments which, in turn, provide the basis for determining the forces transmitted to the packages. The determination of these forces is necessary for the package-tiedown structural assessments. The objective is to determine the extent to which the shocks and vibrations experienced by the shipping packages during normal transport are influenced by, or are sensitive to, various structural parameters of the transport system (i.e., package, package supports, vehicle characteristics, etc.).

CARDS is a complex two-dimensional, multi-degree-of-freedom model which determines the horizontal (longitudinal), vertical, and rotational motion of both the package and its transporter following impact with an anvil train during coupling operations. The model also determines the longitudinal motion of each of four (or more) ballast-filled hopper cars in the anvil train. CARDS calculates the coupler force on the cask-railcar due to the impact, considering draft gear characteristics and shifting of the cargoes on the cask-rail car and on the anvil cars. The formulation of CARDS provides flexibility for the simulation of a broad range of package-car, and package-car-anvil train configurations.

Validation of the model has been partially established by the simulation of Test #3 of 18 instrumented impact tests conducted at the Savannah River Laboratories (SRL) in July and August of 1978. In these tests, several hammer car configurations (spent fuel shipping cask-flat rail car configurations) were coupled, at various impact speeds, to a stationary anvil train consisting of four ballast-loaded hopper cars. In the test simulated, a 40-ton Hallam cask was mounted on a 70-ton Seaboard Coastline flat bulkhead railcar with standard couplers. Horizontal movement of the cask on the railcar was opposed

by a rigid stop at the struck end, and vertical movement was opposed by six vertical bolts. This cask-rail car configuration was coupled to the anvil train at an impact speed of 10.5 miles per hour.

A measure of model validation was established by the application of two separate techniques. The simplest and most straightforward technique was visual comparison of experimentally measured response variables with the corresponding values calculated using the model. The second technique used is a statistical technique based on Theil's inequality coefficients. A Theil's inequality coefficient for each response variable was computed from comparisons of its time-varying predicted and measured values. These individual coefficients are combined into a single figure of merit (Theil's multiple inequality coefficient) which varies from 0 (best agreement) to 1 (poorest agreement).

An assessment of how well the CARDS model simulated the behavior of the cask-rail car system for the conditions of Test 3 of the SRL experiments was made by comparing, both visually and quantitatively, the calculated and experimental values of coupler force, the horizontal force of interaction between the cask and rail car, the horizontal acceleration of the rail car, the horizontal acceleration of the cask, the vertical acceleration of the cask at the far end, and the vertical acceleration of the cask at the struck end. The coupler force was the force of excitation causing the system to vibrate.

At first, there was good agreement only between the calculated and experimental values of the horizontal forces and accelerations. The vertical accelerations of the cask at the far end did not show acceptable agreement when compared both visually and quantitatively. A review of high speed films of the tests and of system structural features revealed the cause of this lack of agreement. The high speed films showed that water was ejected from the collar around the cask at the far end at impact (rain water had collected under the collar during a rain storm the night before the test). It was also recalled that a rubber gasket or shim was used under the collar. This suggested that the rubber, or gap, or both, could cause both an increase in the magnitude and frequency of the acceleration readings at the far end, precisely the characteristics needed in the calculated results to achieve agreement. Double integration of the measured accelerations gave displacements which confirmed this conclusion. Therefore, a non-linear stiffness coefficient was devised for a rear tiedown consisting of a series combination of an initial gap between the cask and its collar, a rubber shim, and then the intended tiedown structure. A corresponding damping coefficient was also devised. When these new coefficients were used in the model, a dramatic improvement in the agreement was realized (especially in the visual comparison).

To assess the performance of CARDS, two cases were evaluated. In the first case, the experimentally measured coupler force was used and, in the second case, the coupler force used was that calculated by the model. Due to space and time limitations, only selected results from the first case will be presented. Visual comparisons of selected response variables are presented in Figures 1 through 4. Figure 1 shows the longitudinal force of interaction between the cask and rail car. Figure 2 presents the horizontal acceleration of the cask.

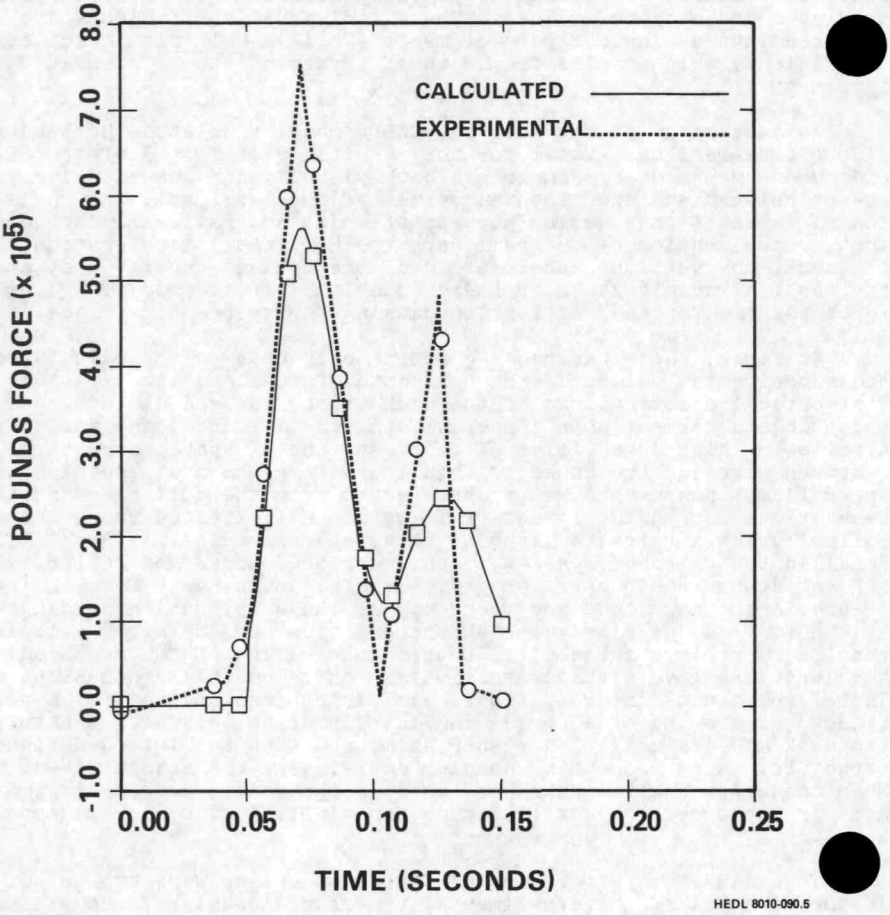


FIGURE 1. Longitudinal Force of Interaction Between Cask and Rail Car

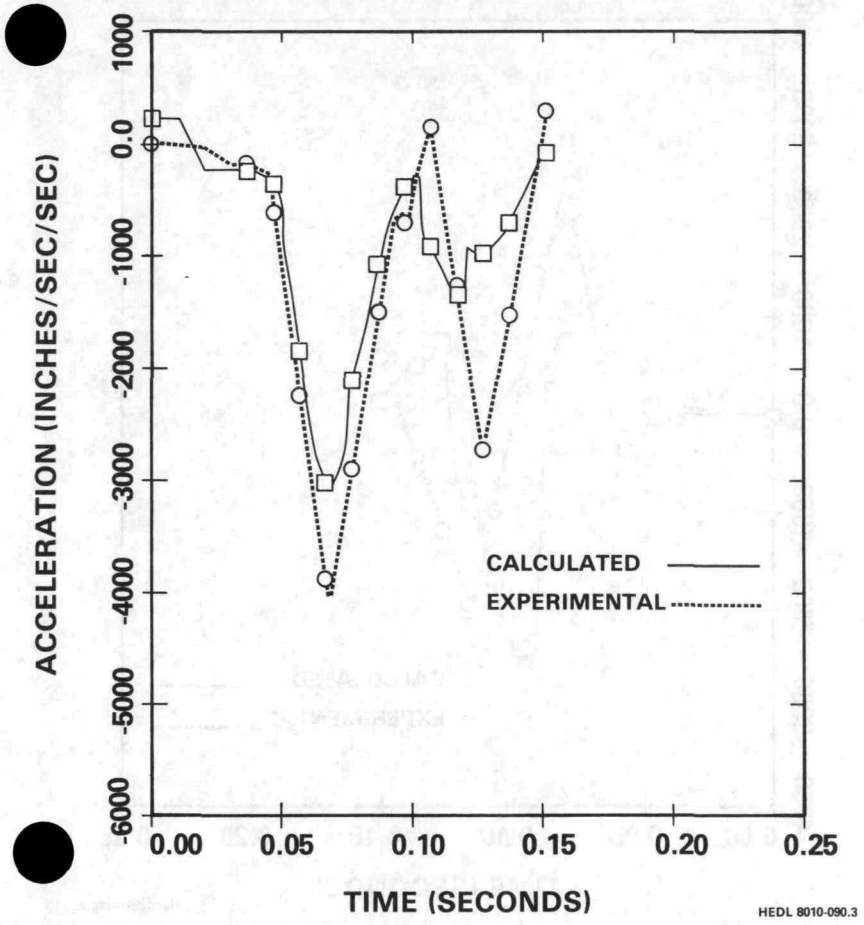


FIGURE 2. Horizontal Acceleration of Cask

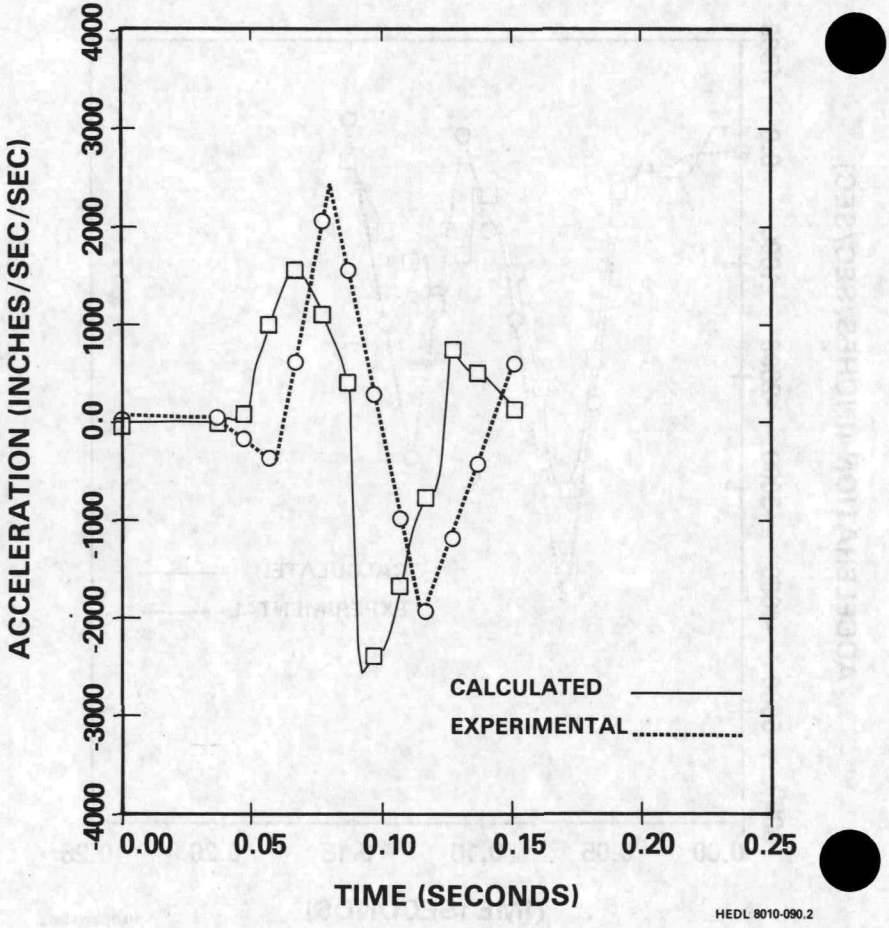


FIGURE 3. Vertical Acceleration of Cask at Far End

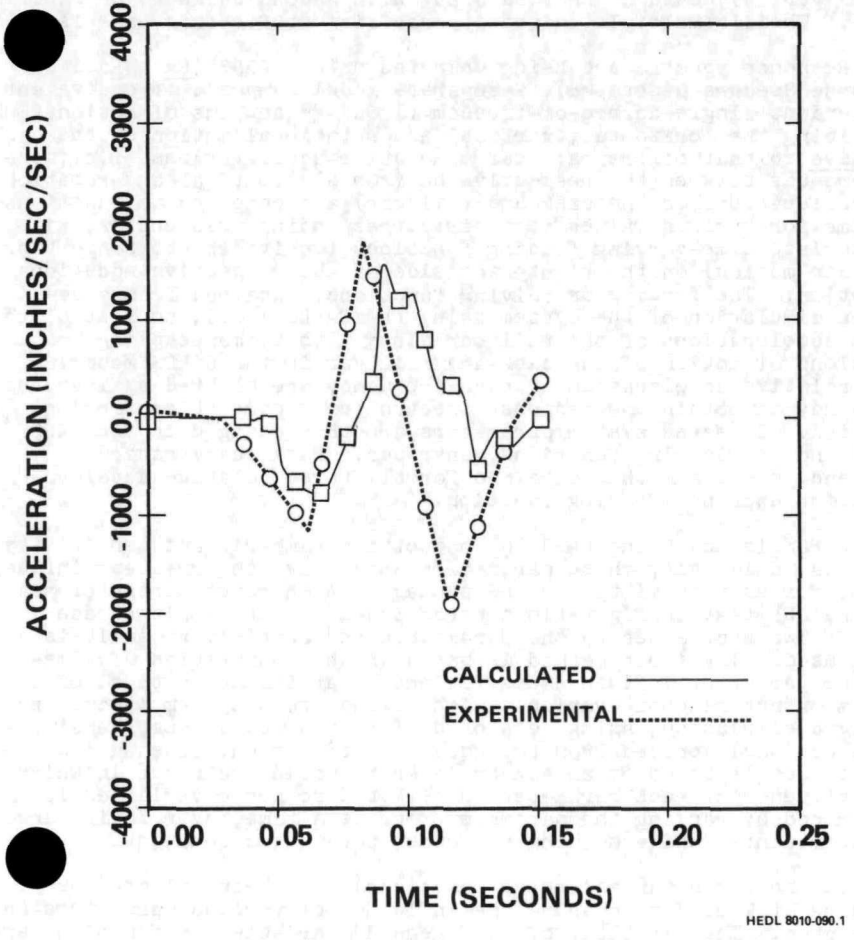


FIGURE 4. Vertical Acceleration of Cask at Struck End

Figures 3 and 4 show the vertical accelerations of the cask at the far and struck ends, respectively. Quantitative comparisons of each pair of response variables were made by computing Theil's inequality coefficients. Only the vertical accelerations of the cask produced inequality coefficients above 0.5. However, Figures 3 and 4 show that good visual agreement exists between the vertical accelerations. Both the magnitude and frequency of these plots are in good agreement. It appears, however, that better quantitative agreement could be obtained if the calculated vertical acceleration at the far end (Figure 3) could be made to shift approximately 0.025 second forward on the time scale, and if the calculated vertical acceleration at the struck end (Figure 4) could be shifted about 0.02 second backward on the time scale. Theil's multiple inequality coefficient for this case is .059.

Response spectra are being computed using CARRS (Cask Rail Car Response Spectra Generator), a separate model composed of equivalent independent single-degree-of-freedom (1 DOF) equations of motion (EOM) describing the horizontal, vertical and rotational motion of the cask relative to that of the rail car. Relative accelerations (i.e., the differences between the respective horizontal, vertical and rotational accelerations of the cask and rail car) are computed as functions of time for various values of their corresponding frequencies, with appropriate time-varying forcing functions (equivalent to support or rail car motion) on the right-hand side of the respective equations of motion. The forcing or driving functions, obtained from a detailed simulation of the system using the CARDS model, consist of the three accelerations of the rail car along with terms coupling the equations of motion of the cask and rail car in the CARDS model. Peak relative acceleration at each frequency are plotted against the frequency to obtain the response spectra for a particular forcing function. Selected system parameters are then changed in the CARDS model and new forcing functions generated. Peak acceleration vs frequency plots are thus obtained for all three relative accelerations for each new forcing function.

CARDS is now being used to conduct a parametric and sensitivity analysis to identify those parameters which have the greatest influence on the forces transmitted to the package. A thorough parameter search for the SRL test configurations established the appropriate base cases. Two approaches to the parametric and sensitivity analysis are being used. The first method is based on the computation of time-varying parameter influence coefficients (partial derivatives of a system output response variable with respect to a system parameter) during a simulation, using sets of differential equations (sensitivity equations) derived from the equations of motion. The second method used is based on an Algebraic Monte Carlo technique in which the influence of each parameter on selected response variables is determined by varying the parameters one at a time, over their ranges of uncertainty, while holding all other parameters constant.

Two response variables (or functions) have been selected as "Figures of Merit" to measure the influence of various parameters on the system. The first set of "influenced" variables or functions are the forcing functions driving the 1 DOF EOMs of the CARRS model. These forcing functions indirectly transmit parameter influence to the response spectra. System sensitivity is therefore indicated by changes in the response spectra. The second "Figure of Merit" selected is Theil's multiple inequality coefficient.

The CARDS model consists of twelve equations of motion, one derived for each degree of freedom (generalized coordinate), and supplementary auxiliary equations. The equations of motion were derived from an energy balance (expressed in generalized coordinates) on the system. The entire model definition was written in the Advanced Continuous Simulation Language (ACSL) which was developed for the purpose of modeling systems described by time-dependent, non-linear differential equations and/or transfer functions. Of the four numerical integration algorithms available in ACSL, the Runge-Kutta fourth-order algorithm was chosen for the CARDS model.

DISCUSSION

Question by J.-P. Prulhière, France: The model you have presented is a model with discrete elements. How did you choose the value of the parameters (e.g. stiffness)?

Answer by N. A. Eisenberg, USA: Although I am not the author of this paper, I shall attempt to adequately and accurately respond to your question.

Since this is a lumped model, rather than a distributed system model, it is necessary to estimate parameter values for the system components assumed in the model. Some parameter values, e.g. the masses of the cask, railcar, and wheel assemblies, arise rather transparently from direct measurement or design data. Other parameter values, e.g. the compliance (spring) coefficients and frictional (damping) coefficients for tiedowns and other connecting structural members, are obtained by conventional engineering calculations involving the dimensions and material properties of the components. For certain nonlinear components, such as the draft gear, experimentally measured values of parameters are used. Furthermore, it is my understanding that the author adjusted some parameter values, so that the computer simulation results matched optimally with certain experimental tests used to "calibrate" or benchmark the simulation. However, once parameters were fixed for these few cases, no further adjustments were made.

Question by T. Loewen, Canada: Which type of draft did you consider and how did you model the draft gear?

Answer by N. A. Eisenberg, USA: Although I am not the author of this paper, I shall attempt to adequately and accurately respond to your question.

Both conventional and long-travel cushioned draft gear were considered. The simulation language (ASCL) permits characterisation of the inherent nonlinear behaviour of the draft gear (e.g. the damping force is proportional to the coupling force, rather than the velocity of travel, as in a linear component). Furthermore, bottoming-out of the draft gear at the end of its travel, with the concommitant increase in stiffness and stress, is also modelled.

DISCUSSION

Question by D. Haon, France: This question is relevant to the first three papers of this session.

Is it possible to say whether the tests carried out showed partial destruction of the fuel elements transported, which, according to Mr. Walker in paper 029 should be taken into consideration in criticality studies for transport containers?

Answer by G. Chevalier, France: In France we have elaborated a shock absorber for an assembly (15 x 15). This shock absorber constitutes a deceleration identical to that seen in a container with 12 assemblies after a drop of 9 m. We will have the results on deformation and ruptures of the assembly in a year's time.

Generally speaking, there are few studies on this type of behaviour of assemblies, in particular if they are irradiated. These are studies which could justify whether criticality studies on assemblies after an accident are necessary.

RECENT DEVELOPMENTS IN PUNCTURE ANALYSIS TECHNIQUES OF NUCLEAR SHIPPING CONTAINER COMPONENTS*

by

R. C. Shieh

9937 Dellcastle Road
Gaithersburg, Maryland, USA 20760

SUMMARY

This paper identifies a number of previously used, deficient design analysis/safety review bases and methods of analysis for nuclear shipping cask designs under the regulatory puncture loading. It also reviews previous experimental/empirical studies in the subject areas and points out some shortcomings in these studies. Recent developments in the subject technological area are discussed and major research results presented. Some potential areas of future research are also identified.

1. INTRODUCTION

Title 10, Part 71 of the Code of Federal Regulations sets forth the requirements governing the shipment and performance of shipping casks under accident conditions. Because of the requirement of subjecting the cask to the accident fire test following the puncture test, it is a generally accepted fact that the outer steel shell layer of a lead shielding cask not be punctured; otherwise, loss of the lead shielding in the puncture region will result under the subsequent fire condition.

Despite the necessity of meeting the regulatory requirement, the puncture-safe design technology of shipping casks has not been fully developed yet. In the past, inadequate general design bases, analysis methods and/or empirical data/equations have been often used in cask puncture-safe designs, and the designs routinely accepted.

Earlier systematic studies (Refs. 1-3) to develop puncture-safe design analysis techniques had employed the experimental/empirical approaches. More recently, the validity of these experimental/empirical data were examined and questioned, and appropriate empirical equations for lead-backed cylindrical steel shells have been developed in Ref. 4.

In Ref. 5, Lawrence Livermore Laboratory, under the sponsorship of U.S.-NRC, performed both experimental and analytical studies of the axisymmetrical puncture problems of laminated, circular stainless steel plate assemblies. A finite element type computerized study has been developed, but both computational efficiency and accuracy need to be improved before it become a practical and useful cask licensing or design tool for circular ctural components of lead-shielded steel casks. The experimental study yielded some useful data or confirmatory information for puncture force/energy responses of lead - and uranium - backed circular steel plate assemblies.

The analytical techniques/design curves applicable to the design of circular steel plates subjected to an axisymmetric, quasi-static or low velocity impact puncture loading have been developed in Ref. 6. A continuum

* This work was partially completed at the U.S. Nuclear Regulatory Commission where the author was a structural engineer(1975-80).

approach was used in the study.

The main purposes of the present paper are to critically review the previously developed analytical techniques/design curves, point out some shortcomings in these studies, identify inadequate general design bases and analytical methods used in cask design/licensing review, summarize the recent developments in puncture analysis techniques, and identify the areas in which additional research efforts are needed.

The problem areas in puncture-free design analysis of cask structural components are presented in Section 2. The critiques of the previous studies on puncture analysis techniques are given in Section 3 while recent developments in the techniques are given in Section 4. Section 5 identifies the areas in which additional puncture research efforts are needed.

2. SOME DEFICIENCIES/PROBLEM AREAS IN CASK PUNCTURE DESIGN ANALYSIS

During the past five years in reviewing structural designs of shipping casks, the author has found that almost the entire cask puncture design analysis/safety review bases (including a previously used 1414 kips-punch-force-puncture design criterion), and methods of analysis were in a technically unsound or questionable footing. Some examples are:

(a) The commonly used design/analysis/evaluation bases/practices derived from the assumption of punch buckling failure at the onset of punch yielding, including the previously stated position of 50 ksi-maximum-punch-stress puncture design criterion, has been found invalid or of questionable validity and could lead to grossly unconservative analysis/design results for thicker container wall cases (e.g. more than 1-inch thick lead-backed stainless steel shell; for 2-inch thick lead-backed stainless steel, the maximum punch stress at incipient puncture is about 100 ksi at room shell temperature. See Refs. 5 and 8).

(b) The widely used design curves contained in the ORNL Cask Designer's Guide (also Nelms'equations) for puncture analysis/evaluation of lead-backed, steel cask shell design may give unconservative results partly due to use of solid lead core interior or unrealistically (almost infinitely) thick lead backing in generating test data that were used in developing these curves/equations; experimental evidence shows that these curves/equations may overpredict the puncture energy absorbing capacity of a realistic cask steel shell by as much as 50%!

(c) Neglect of important elevated temperature effects (e.g., in NLI-10/24 cask design), that would or could reduce puncture resistance capacity of a lead-backed cask steel plate or shell by as much as 50% at 400°F temperature, and low temperature effects, which may cause carbon steel casks to fracture in a non-ductile manner.

(d) Questionable validity of misapplication of the existing empirical puncture design equations beyond their ranges of applicability, e.g., generally unconservative use of Nelms' puncture equations in laminated or multiconcentric steel shell puncture analysis in which two shells are separated by air or water.

(e) General unconservative puncture analysis and design practices by neglecting the portion of applied puncture energy contributed by punch displacement after cask/punch bar contact--a practice which could lead to significantly unconservative results, particularly if the cask is of a concentric shell type with the outer and inner shells separated by air or water and the former is punctured through.

(f) Shortcomings (substandards) in puncture scale model testing techniques/procedures/practices such as neglect of several important scaling distortion effects.

REVIEW/CRITIQUE OF PREVIOUS STUDIES

(a) ORNL Studies (Refs. 1 and 2)

Earlier systematic experimental studies of the incipient puncture behavior of lead-backed carbon and stainless steel plate and shell assemblies were made at Oak Ridge National Laboratories (ORNL) (Refs. 1 and 2). In these studies, puncture test data were generated by simplified scale model or prototype drop tests in which the lead backing was either an infinitely thick layer compared with the steel shell layer or solid interior core of the circular cylindrical steel shell. These studies have produced the well-known Nelms' puncture equation which have been widely used in cask shell design by cask designers and safety review by NRC reviewers. Correlation between the equation-predicted results and obsolete cylindrical cask test results (Ref. 7) show that the Nelms' empirical puncture equation over-predicted the test results by about 40 to 50% (See Fig. 1), thus, use of this equation in cylindrical cask design may result in nonconservative design. This is hardly surprising because the scale models and prototype structures used in the tests deviate too much from realistic cask structures (e.g., lead backing thickness). Indeed, recent confirmatory study by LLL (Lawrence Livermore Laboratory) (Ref. 5) shows that the thickness of lead layer in a lead-backed steel plate has significant effects on incipient puncture behavior of steel plates. Besides this, Nelms' equation and the design curves contained in the ORNL Cask Designer's Guide may give nonconservative design results because they did not take into account test data scatter.

(b) Sakamato, et al, Study (Ref. 3)

Later, Sakamoto, et al. conducted a series of scale model tests aiming at studying the effects of cask shell diameter and rounded punch edge radius on incipient puncture energy behavior. The model casks used in the tests were scaled-down versions of a realistic 80 ton prototype cylindrical cask of steel/lead composite shell construction. Static, impact machine and drop puncture tests were performed. An empirical puncture energy equation which, in general, predicts stronger cask puncture resistance than that of Nelms' has been developed. The applicability of most of these test and thus, the empirical equation in cask puncture-safe design, needs additional confirmation because machine impact test velocities were not shown to be in the vicinity of regulatory puncture drop velocity, and most of the drop test heights deviated too much from the regulatory drop height of 40 inches. (According to the scaling laws, drop height in the scale model tests should be 40 inches).

4. RECENT DEVELOPMENT

(a) Puncture Studies by the Writer (Refs. 4 and 6)

(i) In Ref. 4, the writer performed an in-depth review of the existing puncture data and accomplished the following:

- Developed an empirical incipient shell puncture energy equation of lead-backed, steel cask shells. This was done on the basis of the ORNL obsolete (realistic) cask puncture test data (Ref. 7) and those scale cask model drop puncture test data of Sakamoto, et al (Ref. 3) with drop heights in the vicinity of 40-inches. The effect of regulatory punch edge radius of 1/4-inch on incipient puncture energy has been incorporated in the equation. In order to account for uncertainties (such as test data scatter, possible inaccuracy in test data due to deviation of the test drop heights from the regulatory drop height of 40 inches, etc.) it was recommended that use of the empirical incipient puncture equation in cask shell puncture-free design should be made by reducing the incipient puncture energy by at least 20%. In this way (with 20% safety margin), the following empirical puncture design equation was obtained in Ref. 4 for lead-shielded, steel cask shells:

$$W < 14.8[1+ 6/D + (41/D)^2]t^{1.7}\sigma_\mu/(h+ \delta_c) \qquad (1)$$

$$(0.5" \leq t \leq 2"; \ 45" \leq D \leq 90", \text{ lead thickness} \geq 4 \text{ inches})$$

where W is the total cask weight, h the cask drop height (= 40 inches), δ_c the cask center of gravity displacement after cask/punch bar contact, D the cask shell outer diameter, t the cask outher steel shell thickness, and σ_μ the ultimate strength of the outer steel shell; all units are in inches, kips or their combination. For a single composite shell type cask, $\delta_c \simeq 4\ t^{0.87}$ (inches) can be used in Eq. (1).

- Identified several scaling distortion effects and formulated a technically sound remedial measure for treating these effects in use of scale model testing techniques.

- Introduced/applied similitude laws and standardized data analysis/parametric study procedure in empirical puncture equation development via scale model testing techniques.

- Identified significant discrepancies between the existing empirical puncture equations and shell puncture test data of realistic, cylindrical, lead-backed, steel cask shells and their main causes, and developed a technically sound empirical puncture design equation as mentioned above.

(ii) In Ref. 6, the axisymmetric problem of a circular metal plate subjected to a punch load at its center has been studied. An efficient analytical predictive technique has been developed, and puncture-safe design curves for stainless steel plates at room and elevated (400°F) temperature conditions were presented (see Fig. 2). The computational time involved in computing the incipient puncture responses (energy, displacement, etc.) for a given case is not more than a few seconds as compared with the LLL finite

element NIKE-2D model of 10 to 50 minutes (see item (b) below).

Recently, the above study has been extended to the carbon steel case. Figure 2 shows the nondimentional incipient puncture energy $\overline{V}_c = V_c/\sigma_\mu t^3$ and punch displacement $U_c = \delta_c/t$ as functions of d/t for various values of d/D - values for both stainless and carbon steel plates. Here, V_c is incipient puncture energy, d the punch diameter, and σ_μ, D, t and δ_c are defined as in Eq. (1). Use of these curves in puncture-safe design of a circular flat steel cask plate should be made only after reducing the puncture energy absorbing capacity by 30% to allow a margin of safety for inaccuracy, uncertainty etc.

Figure 3 shows the excellent correlation between the analytically predicted and experimentally obtained punch force responses as a function of punch displacement.

(b) LLL - Study

In Ref. 5, Lawrence Livermore Laboratory (LLL) performed both analytical and experimental puncture studies of lead-backed and uranium backed circular flat stainless steel plate assemblies. A finite element computer code, NIKE-2D, has been developed. The computational time in solving a typical static or dynamic puncture problem of a lead-backed, stainless steel plate using the NIKE-2D model has been estimated as 10 to 50 minutes of CDC-7600 cpu time. The NIKE-2D model also does not possess capability of modeling strain rate sensitive material behavior on inelastic deformation. Therefore, additional efforts are required to improve the computational efficiency and dynamic modeling capability of rate sensitive materials (such as steel and lead) before the NIKE-2D model can become a useful tool in predicting accurately the puncture behavior or in serving as a practical analytical tool in licensing review or design of shipping cask circular plate components.

In the experimental study area, the study also yielded the following data and confirming information:

- Verified that the plate puncture fracture criterion for lead-backed stainless steel plates is of shear failure type in the range of parameters studied.

- Experimentally generated puncture data for lead- and uranium-backed, stainless steel plate assemblies at room and elevated temperature environments.

- Confirmed the importance of elevated temperature effects in degrading the puncture resistance capacity of lead-backed steel plates; e.g., at 200°F and 400°F temperatures the puncture energy absorbing capacity of the plate assemblies may be reduced from that of room temperature by as much as 20% and 50%, respectively.

- Developed incipient puncture force data for lead-backed stainless steel plate assemblies at room temperature and confirmed inadequacy of using the 50 ksi maximum punch stress as the cask puncture design basis if shell thickness is greater than 1-inch under the room temperature environment. From these data, the following maximum punch stress equation at the incipient plate puncture has been derived

(Ref. 8):

$$\sigma_c = 47\ t\ (ksi) \qquad (2)$$

- Confirmed that lead thickness has significant effects on the puncture energy absorbing capacity of lead-backed cask plate and found the puncture energy absorbing capacity of such plates being increased by only about 10% due to the presence of the regulatory punch edge radius in the tested punch bars as opposed to square-edged punch.

- Found that the puncture resistance capacity of a uranium-backed stainless steel plate assembly is approximately three to five tim[redacted] that of the corresponding lead-backed stainless steel plate case

- Showed that for a given lead layer thickness and temperature, the incipient puncture energy is not only a function of punch diameter (d), plate thickness (t), but also a strong function of plate diameter (D).

However, it must be mentioned that numerous comments and conclusions made or drawn in Ref. 5 cannot be supported by existing data or need to be backed-up by additional studies/data (see Ref. 8). Some examples are:

- The conclusions of dynamic puncture energy absorbing capacity being less than the corresponding static one based on the information given in Table 4 of Ref. 5 cannot be supported because the dynamic energy data in this table are believed not accurate enough. The more accurate values are those enclosed in parentheses in Table 1 below (Refs. 5 and 8).

TABLE 1: Static and dynamic test failure data for lead-backed plates when backing thickness = punch diameter.

Plate thickness, in.	d/t	Punch diam, in.	Average dynamic test data		Static test data	
			Max Force, lb	Puncture energy, V_c in.-lb	Max force lb	Puncture energy, in.-lb
0.05	12	0.6	8,100	2,300 ($2{,}590<V_c<2900$)	7,000	2,300
0.105	5.71	0.6	--	$4{,}800<V_c<7{,}400$[a]	13,800	5,600
0.20	3	0.6	25,600	12,600 ($10{,}820<V_c<15{,}670$)	25,400	14,600[b]
0.117	12.82	1.5	46,500	22,900 ($25{,}300<V_c<30{,}200$)	39,000	23,300
0.540	2.78	1.5	176,500	$220{,}000<V_c<235{,}000$[a]	175,000	275,000

[a] Inconsistency or the absence of data prevents closer determination of puncture energy

[b] Average of two static tests, V_c=13.8 and 15.d kip-in.

In the last case of Table 1, the dynamic puncture energy absorbing capacity is seen to be less than that of the corresponding static case. However, in this case, the drop puncture test height was 118

inches, which are about 3 times of the regulatory drop height. In any case, that the fact that dynamic/static puncture resistance energy ratio may be less than 1 is not entirely new. In Ref. 9, whether this ratio will be greater or less than 1 for carbon steel beam puncture was shown to be a function of carbon steel contents and punch impact velocity.

- An attempt was made in Ref. 5 to correlate the newly developed, incipient static and dynamic puncture energy data for lead-backed, circular stainless steel plates with those of the previously, independently developed test results and empirical equations of lead-backed, rectangular steel plates of Refs. 1-4 (see Fig. 4). As might be expected, a good correlation was not obtained because of complete or vast differences in the configuration and boundary conditions of the tested assemblies, test methods, etc. Also, the nondimensional incipient puncture energy $\overline{V}_c$ of lead-backed, circular steel plate assemblies is a strong function of punch diameter/plate diameter ration (d/D_p), i.e, $\overline{V}_c$ decreases with increase in d/D_p - value -- the information that was not pointed out in Ref 5 or shown in the Fig. 11 of Ref. 5 (i.e., Fig. 4 of the present paper).

- Comments made on material strain rate effects on puncture behavior were mostly inaccurate or invalid. Figure 5, taken from Ref. 5, shows the importance of material strain rate effects on the punch force response as a function of punch and plate displacement.

5. POTENTIAL AREAS OF FUTURE CASK PUNCTURE RESEARCH

The puncture analysis techniques of nuclear shipping casks are still not well developed. The chart in Figure 6 shows the major areas of cask puncture research and the areas in which puncture studies and data exist. Some potential areas of additional confirmatory research to be pursued, therefore, are:

(a) Investigation of puncture resistance and fracturing behavior of those cask materials (e.g., certain types of carbon steel) that are not very ductile or which can lose ductility in a low temperature environment, and fracturing behavior of uranium backing material.

(b) Investigation of puncture resistance capacity of unbacked steel shells.

(c) Generation of additional puncture data (for independent confirmatory purposes) and investigation of effects of elevated temperature on puncture resistance capacity of lead-shielded, circular cylindrical cask shells.

(d) Investigation of effects of steel plate/shell lamination, cask drop puncture velocity (including quasi-static case), initial stress, etc., on cask plate/shell puncture resistance capacity.

6. CONCLUSION

A number of deficient or questionable technical bases and problem areas in puncture-free design analysis and safety review of nuclear shipping casks subjected to the regulatory puncture loading were identified; these included

the widely used design basis of 50 ksi-punch-stress puncture design criterion and design curves given in the ORNL Cask Designer's Guide and Nelms' empirical puncture design equation.

Review/critique of the previous puncture studies (Ref. 1-3) were made, which resulted in identification of some inadequate or questionable test/ empirical results.

Recent developments in puncture analysis techniques (Refs. 4-6) were discussed, and appropriate empirical results, design curves and information, and other research results, were summarized or presented in the form of equations (Eqs. 1 and 2) table,figures (Figs. 2-5).

The potential areas of future puncture research were also identified.

In summary, state-of-the-art in cask puncture design analysis technology is still in its development stage and additional research efforts must be expanded to build up the technology.

REFERENCES

1. Spaller, A. E., "Structural Analysis of Shipping Casks - Vol. 2 Resistance to Puncture," ORNL-TM-1312 Vol. 2, Sept. 1966.
2. Nelms, H.A., "Structural Analysis of Shipping Casks," Vol. 3, ORNL-TM-1312.
3. Sakamoto, et al, "An Experimental Study on Puncture Resistance of Spent Fuel Shipping Cask by Drop Impact Tests," 4th Int. Symp. on Packaging and Transportation of Radioactive Materials, pp 262-276, Sept. 22-27, 1974, Miami Beach, Florida.
4. R. C. Shieh, "Empirical Equations for Puncture Analysis of Lead-Shielded Spent Fuel Shipping Casks," Proceedings of the 5th International Symposium Packaging and Transportation of Radioactive Material, Las Vegas, May 7-12, 1978, pp. 341-350.
5. Larder, R. A., and Arthur, D., "Puncture of Shielded Radioactive Material Shipping Containers," NUREG/CR-0930, Lawrence Livermore Laboratory, Dec. 1978.
6. R. C. Shieh, "Analytical Puncture Study of Circular Metal Plates," to be published in ASME Journal of Pressure Vessel Technology.
7. L. B. Shappert, "A guide to the Design of Shipping Casks for the Transportation of Radioactive Materials," USAEC Report ORNL-TM-681, Oak Ridge National Laboratory, April 1965.
8. R. C. Shieh, "Cask Structural Puncture Research - Contract Technical Assistance, Research Progress Report and Recommendations," Memo to C.E. MacDonald, Jan. 16, 1980. at NRC.
9. Stock, T.A. and Wingrove, A.L., "The Energy Required for High-Speed Shearing of Steel," Journal of Mechanical Engineering Science, Vol. 13, No. 2, 1971, pp 110-115.

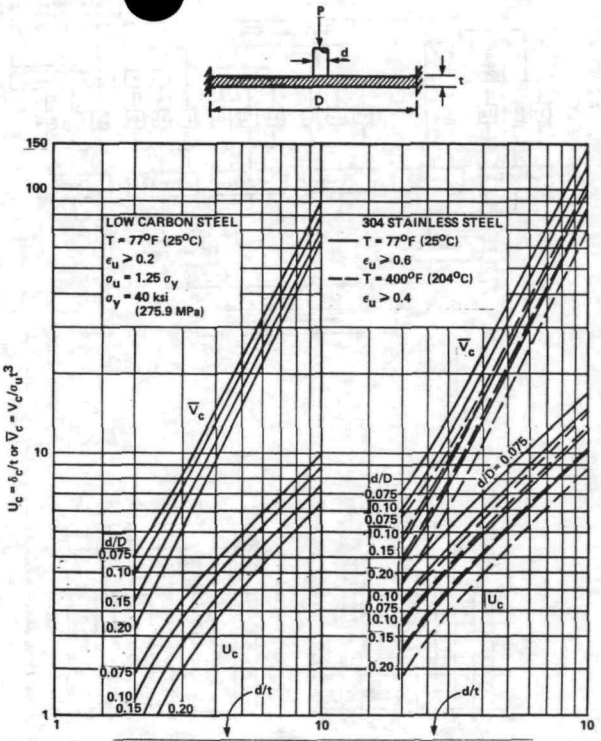


Figure 2. Nondimensional Incipient Puncture Energy $\bar{V}_c$ and Punch Displacement U_c as Functions of d/t for Various Values of d/D

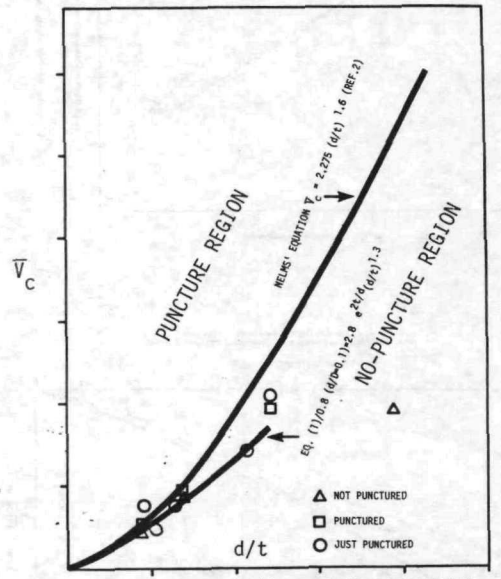


Figure 1.-Correlations of Eq. (1), Nelms' Equation and Puncture Test Data for Lead-Backed Prototype and Obsolete A-245 Cask Steel Shells

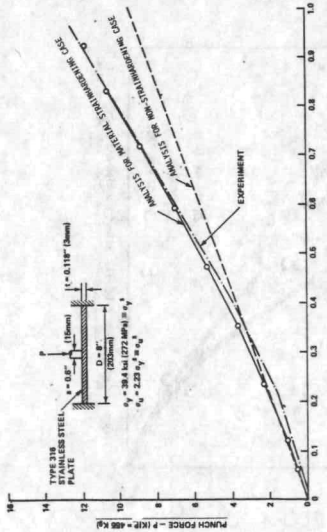


Figure 3 Analytical/Experimental Correlations of the Static Punch Force-Displacement Relationships for the Clamped Edge Plate Case.

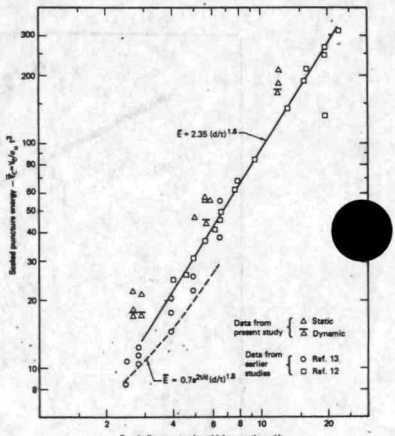


FIG. 4. Scaled puncture energy comparisons.

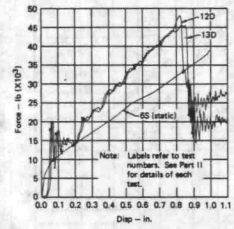


FIG. 5a Experimental force vs displacement for lead-backed 0.117-in. SS plates.

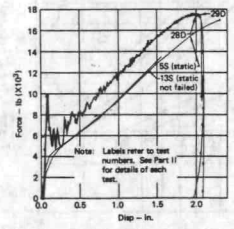


FIG. 5b Experimental force vs displacement for lead-backed 0.540-in. SS plates.

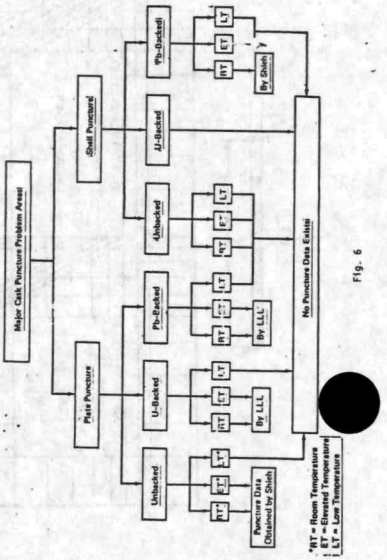


Fig. 6

ANALYTICAL AND EMPIRICAL EVALUATION OF LOW-LEVEL WASTE DRUM RESPONSE TO ACCIDENT ENVIRONMENTS*

R. A. May, L. E. Romesberg, and H. R. Yoshimura
Sandia National Laboratories, Albuquerque, NM, USA

W. E. Baker and J. C. Hokanson
Southwest Research Institute, San Antonio, TX, USA

INTRODUCTION

As part of an overall program to predict the response of low-level nuclear waste transportation systems to accident environments, Sandia National Laboratories (SNL) has been evaluating the structural response of 208 litre (55 gallon), 17C drums (Type A packagings), containing contact-handled transuranic waste (CH-TRU), to impact environments. Early analytical studies of entire transportation systems indicated that the Type A drum behavior had a strong effect on the systems' structural response. In addition, it is desirable to predict the failure of these drums as part of the overall assessment of package response in accident environments to serve as inputs to environmental impact statements. Finally, accurate definition of scale model drum response would allow less-expensive scale model tests to be performed during packaging development and assessment activities.

In order to assess the feasibility of simulating the drum response using existing computer codes, as well as to verify the adequacy of the subscale models, static and dynamic tests of full scale and subscale drums were conducted. This paper briefly describes the results of static and dynamic tests using full scale 17C drums and compares them to the results obtained from subscale tests and computer analyses of equivalent tests. More detailed information can be found in Ref. [1].

DESCRIPTION OF PROTOTYPE AND SUBSCALE DRUMS

Full scale drums actually used to package low-level waste were obtained from Rockwell International's Rocky Flats facility. The drum assembly consisted of a thin polyethylene bag (filled with waste) which was inside of a rigid polyethylene liner which in turn was contained by a 17C drum, Figure 1. The full scale drums were tested in four different configurations: filled with 90 kg of combustible waste, filled with either 225 kg or 320 kg of simulated sludge waste, or empty. The weights selected for the filled drums were representative of those shipped by Rocky Flats.

Scale model drums were simulated by using appropriately sized, commercially available, tinplated steel cans normally used for food packaging. Number 12 cans were used to simulate the 1/4 scale drums and #300 cans were used to simulate 1/8 scale drums. A comparison of the container sizes and thicknesses and the scaled prototype dimensions is shown in Table I. The diameters and heights compared

*Work sponsored by U. S. Department of Energy under Contract DE-AC04-76DP00789.

within 3 percent, but the material thicknesses were 15 percent too small for the 1/4 scale cans and 16 percent too large for the 1/8 scale cans. The scale model drums were filled with material taken from previously tested prototype drums or with locally obtained material of similar composition. After the cans were filled to appropriate scaled weights (some cans were not filled), the lids were attached using a commercially available can sealer.

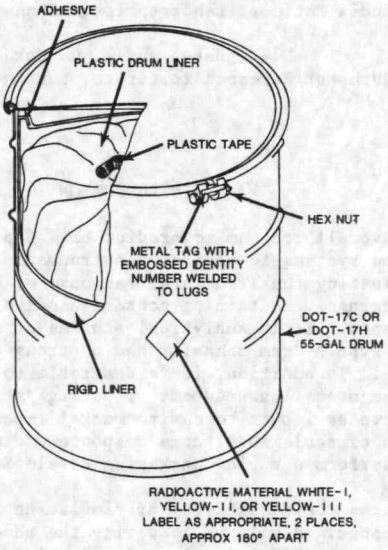


Figure 1. Typical 17C drum, Type A waste container with a rigid 2.5-mm-thick polyethylene liner

Table I
Dimensions and Comparisons of Prototype and Scale Model 17C Drums

CONTAINER	DIMENSIONS (mm) Diameter	Height	Thickness
Prototype (17-C)	606	884	1.52
#12 Can	156	222	0.32
Scaled Prototype	152	221	0.38
#300 Can	75	113	0.22
Scaled Prototype	76	110	0.19

No attempt was made to model the polyethylene bag, rigid liner or the lid clamping ring since it was assumed that accurately modeling these details would not improve test results enough to justify the expense. The effects of neglecting these details are being investigated to verify this assumption.

STATIC TESTS

The first tests consisted of quasi-static crush tests of both full scale and subscale CH-TRU drums. This testing was undertaken in order to obtain the load-deflection behavior of the prototype containers for use in the computer analyses. Also by comparing the scaled prototype results to the load-deflection [redacted] for the subscale containers, an early estimate of the adequacy of the model [redacted]iners was made. It was felt that if reasonable correlation was not found for the static tests, correlation between the same containers would not be possible in the dynamic tests, and a different model container would have to be obtained.

The full scale drums were tested in a universal testing machine which had 25-mm-thick steel plates attached to both the fixed and movable heads to assure adequate contact area during crush, Figure 2. Plots of crush force versus platen motion were generated from recording equipment attached to the testing machine. In addition, time lapse motion pictures were taken from two vantage points at the rate of one frame per second. The drums were crushed in a radial direction at a rate of 12.7 mm/min. The maximum crush of the drums was arbitrarily limited to one-half of the drum diameter, i.e., 305 mm. The scale model containers were tested in a similar manner except on a smaller capacity testing machine.

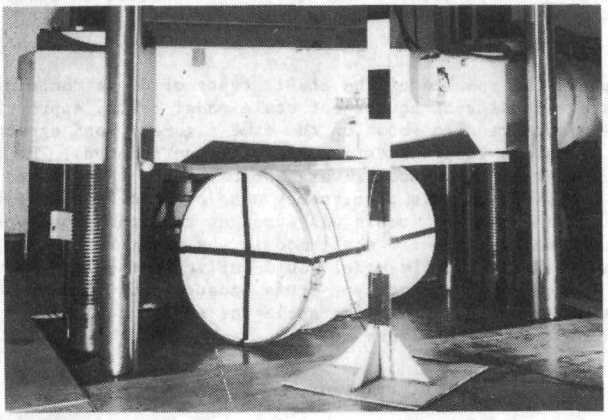


Figure 2. Static test configuration for prototype containers

Eight full scale drums were tested along with six 1/4 scale containers and ten 1/8 scale containers. The results of the prototype static crush tests show consistent load-deflection behavior, Figure 3. The initial crush phase and the occurrence of drum-lid separation can be seen to be independent of contents. The

load-deflection plots show a sharp drop in load at about 64 mm (10% crush) which corresponds to the separation of the lid from the drum body. The contents and the amount of free volume did affect the later crush stiffness and the onset and severity of lockup. The behavior of the drums containing either 225 kg or 320 kg of simulated sludge was virtually identical, while the drums containing 90 kg of combustible waste appear to be slightly stiffer.

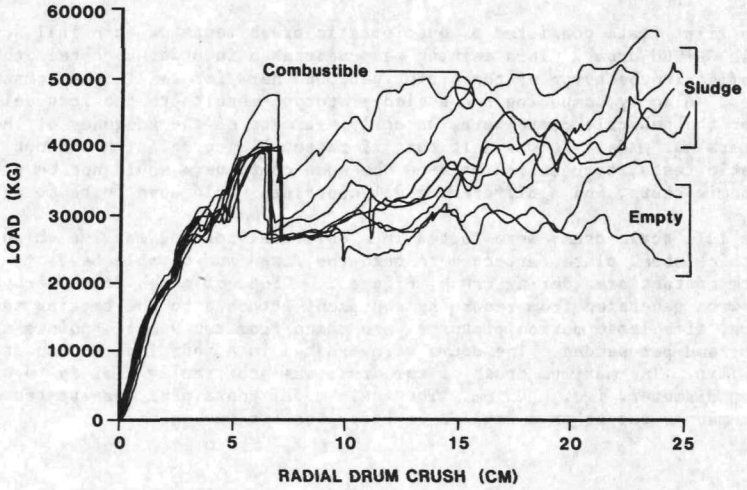


Figure 3. Comparison of the static load-deflection behavior of prototype drums - various contents

Figure 4 shows the results of the static tests of drums containing 225 kg of simulated waste. Results of equivalent scale model tests, appropriately scaled to prototype dimensions are also shown on the same figure. Good agreement between the prototype and 1/4 scale load-deflection behavior can be seen. The 1/8 scale behavior indicated that these models were approximately three times stronger than the prototype containers. This discrepancy was found to be caused by two factors: (1) the material in the scale models was stronger than the prototype material, and (2) the wall thickness in the 1/8 scale models was greater than the correctly scaled thickness. Neither scale model could duplicate the lid separation which occurred at 10% crush on the prototype drums because the prototype drums used clamping rings to secure the drum lids while the scale model container lids were rolled and crimped to the container body.

DYNAMIC TESTS

After the quasi-static prototype and subscale model tests had been completed, a series of dynamic tests was initiated. The objective of this test series was to obtain the characteristic deformed shapes and measurements for both the full scale and subscale Type A containers. Both single drum and multiple drum configurations were tested at impact velocities that might be encountered in a severe transportation accident. In some tests, a closed cell, rigid polyurethane foam, located between the container and the target, was used to simulate the impact mitigation properties of a typical transport container.

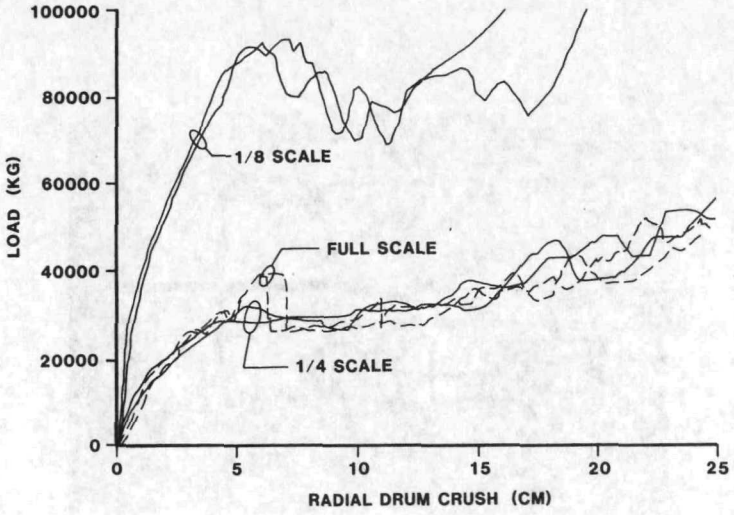


Figure 4. Comparison of the scaled load-deflection behavior for 1/8, 1/4, and full scale 225 kg containers

All dynamic tests were conducted at SNL's Coyote Canyon Drop Test Facility. The full scale drums were allowed to free-fall onto an unyielding target. It was determined that the smaller, scaled containers were subject to perturbations at release and during free-fall, therefore the scale model drums were guided into the target using a small diameter cable passing through guides attached to the cans. The target consisted of a 225 tonne concrete foundation topped by 10 cm of hardened steel plate. The multiple drum drops used a frame constructed of heavy angle stock to keep the drums in vertical alignment. No bulging or excessive deformations of the frames were observed during post-test inspections. Post-test measurements were made of the radial drum deformation, flattened impact area and, where applicable, foam deformation.

The results of the dynamic tests are consistent with the results of the static tests. Correlation was good between the prototype and 1/4 scale test results while the 1/8 scale results under-predicted the prototype deformations due to the greater stiffness. The deformed shapes of both models and prototype drums were generally in good agreement. Figure 5 shows a comparison of a prototype drum and the two subscale containers that impacted at 47 kph (29 mph). It was found that the addition of the polyurethane foam had little, if any, effect on the crush behavior o[redacted] multiple drum impacts.

Again separation of the drum lid could not be visually determined from the subscale models. However, after examining the full-scale and subscale static and dynamic test results, it was felt that lid separation in prototype tests could be predicted from scale model data. In both the static and dynamic tests, lid separation in the prototype drums occurred at approximately 10% crush. Therefore, a subscale drum which suffers a radial deformation at the lid of more than 10% infers that a full scale drum would open in the same test. This criterion can also be used to predict drum openings from the results of the computer simulations.

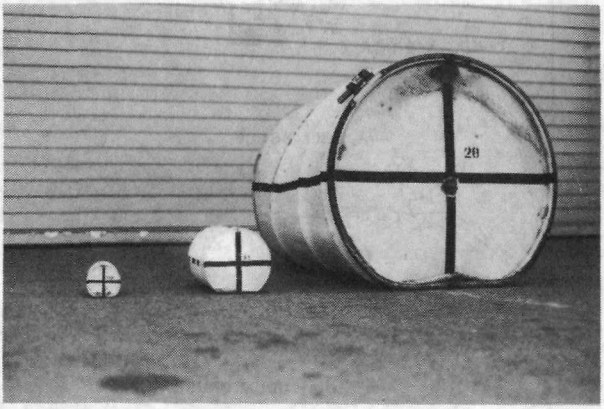


Figure 5. Post-test appearance of 1/8, 1/4, and full scale 320 kg containers following a 47 kph impact

ANALYTIC RESULTS

Two computer models were used to predict the response of an eight high array dropped from a height of either 6.4 or 9.0 meters. The first method used a lumped parameter code [2] to calculate the crush of each drum and the deformed array height. The drums were modeled in the code with a series of masses and springs which represented the weights and stiffnesses of the drums and the contents. The stiffnesses were obtained from the previously conducted static crush tests.

A finite element model using a SNL developed computer code [3] was also used. The drums were simply represented as solid cylinders of material. This representation required some average or equivalent material properties for the container system. The average material properties were determined from the previously conducted static crush tests. [1]

Both analytical models were able to predict the crush of an eight high array of drums with reasonable accuracy. Both models predicted the results for the maximum deflection of each drum, while in actuality the drums exhibit about 3.7 cm of springback. The results of both the lumped parameter model and the finite [redacted]ent model, corrected for the springback, are shown along with full scale and [redacted]ale test results for a 9.0 meter drop, Figure 6. The finite element model slightly over-predicts the crush of the top and the bottom two drums. This anomaly can be attributed to modeling the drums as a solid cylinder. The lumped parameter model gives a somewhat better prediction of drum crush, but with no indication of the deformed container shape. Both models give results which are within the expected repeatability of the tests. By using the previous criteria for lid separation, failure of the drum closure can be inferred at radial deformations greater than 6 cm (10%).

CONCLUSIONS

Based upon the results of the tests to date, it has been found that the structural response of low-level waste drums to impact environments can be generally predicted, both analytically and with subscale models. As currently represented, only the 1/4 scale models would adequately represent full scale drum deformation; however, additional work has shown that with proper heat treating, the strength of the material used in the 1/8 scale containers can be reduced to the correct value. Both analytical models give results that are expected to be within the range of behavior of the full scale drums. Failure of the drum closure can be adequately inferred from the radial deformation results of both subscale tests and computer analyses.

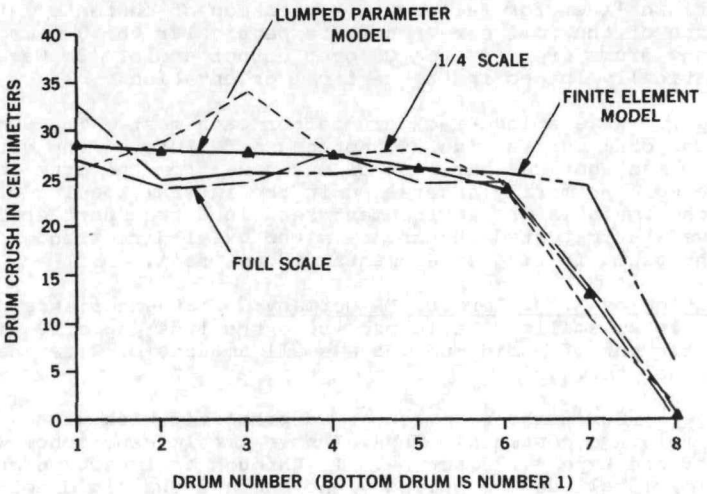


Figure 6. Comparison of full and 1/4 scale test results to analytical results; eight high stack, 9 m drop

REFERENCES

1. May, R. A., Romesberg, L. E., Lamoreaux, G. H., and Sutherland, S. H., "Analysis, Scale Modeling and Full Scale Tests of Low-Level Waste Drum Response to Accident Environments," SAND80-2517, Sandia National Laboratories, Albuquerque, NM, (to be published).

2. Gabrielson, V. K., and Reese, R. T., "SHOCK Code Users Manual, A Computer Code to Solve the Dynamic Response of Lumped-Mass Systems," Sandia National Laboratories, Livermore, California, Report No. SCL-DR-69-68, November 1969.

3. Key, S. W., "HONDO, a Finite Element Computer Program for the Large Deformation Dynamic Response of Axisymmetric Solids," Sandia National Laboratories, Albuquerque, New Mexico, Report No. SLA-79-0039, April 1974.

DISCUSSION

Question by H.-J. Engelmann, FR Germany: Have you carried out tests with the space between the filling and the lid of the drum?

Answer: The filled drums normally contain some voided space between the filling and the lid of the drum. The influence of the void space and contents on the static load-deflection curve was observed for deflections beyond that shown in the diagram. Our data show that the drums generally tended to "lock up" (have rapidly rising load in comparison to deflection) at large deformations.

Question by M. L. Bell, USA: 1. Did you examine the polyethy bag and rigid liner for failure, i.e. leakage of contents, after performance of the full size tests, in particular the 9 m drop test? 2. Were any drums dropped at a CG over corner angle? 3. Were the drums statically loaded in the vertical orientation?

Answer: We examined each drum after each test. The 9 m drop test of the drum on its side did not cause failure of the drum lid/ring attachment nor was any loss of material from the drum observed. 2. No, we were primarily interested in the lateral loading of the drum which simulates the environment seen in a transport accident. 3. Yes, we static tested the drums in the axial direction. In these tests, the drum lid/ring attachments did not fail.

Question by H.-U. Berger, FR Germany: What experience do you have with leaks smaller than those where the lids pop off? (Important for the shipment of humid mud where small amounts of water have to be taken care of).

Answer: Our tests were concerned primarily with wastes that have low moisture contents. Thus, I have little experience with leakage of the type you describe. If through an impact, gaps form in the lid, I believe the polyethylene bag and the rigid polyethylene liner would prevent or slow any leakage of free water that might be contained within the sludge.

SESSION XXI
EXTENDED TESTING II

AIR ACCIDENTS : SIMULATION FACILITIES

B. BAZELAIRE
Centre d'Etudes Scientifiques et Techniques d'Aquitaine

The Centre d'Etudes Scientifiques et Techniques d'Aquitaine, (CESTA)*, one of the establishments of the "Commissariat à l'Energie Atomique" (C.E.A.), has been entrusted with the design, manufacture and testing of containers for air transportation of nuclear assemblies.

The purpose of this paper is to discuss :

- th[redacted]od used to ensure a satisfactory design for containers ;
- th[redacted]performed and the means used to prove the validity of such a design.

The four major parameters involved are the following :

- the nuclear assembly (configuration, weight, sturdiness) ;
- the transport mode (speed, carrying capacity, cargo hold geometry) ;
- the available timetable (design studies, construction, testing, and analysis of test data) ;
- the budget alloted.

The design of the container largely depends mainly on the sturdiness of the nuclear assembly (or conversely of its fragility) and on the stringeness of the specifications for testing procedures, which globally cover the cases of air accidents envisaged.

The ability of the container to withstand impacts at relatively high velocities (40 and 80 ms^{-1}) has been demonstrated on reduced scale models to minimize costs.

The testing facilities comprise :

- the accelerated fall tower at the C.E.S.T.A.
- the launching rail at the Centre d'Essais des Landes.

* The C.E.S.T.A. is under the authority of the C.E.A. and is devoted to scientific and technical research conducted in Aquitaine.

1. NUCLEAR ASSEMBLIES TO BE TRANSPORTED

Main features :

The dimensions of a nuclear assembly are shown in Figure 1.
Each assembly weighs 170 kg.

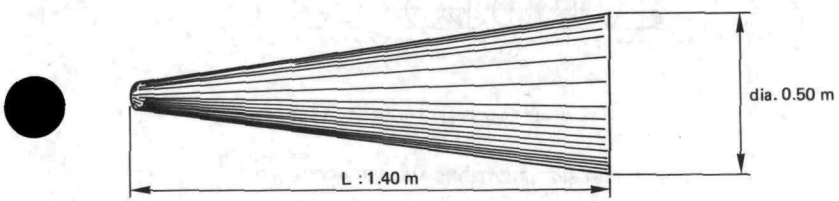


Fig. 1 : Configuration of nuclear assembly

Sturdiness :

The internal design of the nuclear assembly is such that it withstands any shock, of which the characteristics (γ , t) are within the shaded area.

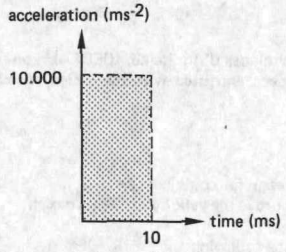


Fig. 2 : Strength zone (γ , t) of nuclear assembly

Assuming that a bundle consists of the nuclear assembly and its container, if there is no dislocation of the structures likely to result in dispersal of nuclear substances, then the bundle is said to resist.

2. TRANSPORT AIRCRAFT

Main features :

The proposed aircraft for transporting such nuclear assemblies is the TRANSALL C 160, a twin turbojet :

- cruising speed : 450 km/h (245 knots = 282 mph) ;
- payload : 16 000 kg (35.274 lbs) ;
- cargo hold dimensions
  - . length : 13.50 m
  - . max. width : 3.150 m { International railways
  - . max. height : 2.980 m { loading gauge

The bundles are arranged transversally in the cargo hold, one behind the other, spaced at intervals of 0.45 m.

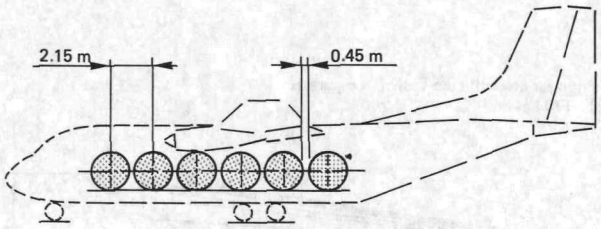


Fig. 3 : Layout of bundles in TRANSALL cargo hold

3. TEST SPECIFICATIONS AND AIR ACCIDENTS

Certification trials :

Prior to its operational use, a container should be certified, i.e. the bundle (container + nuclear assembly) must undergo the following tests :

impact test against a flat rigid target,
- punching test,
- fire test,
- leaktightness test.

NOTE : This paper will deal only with impact tests which are the only tests genuinely typical of air accidents.

Air accidents

Statistical analysis of transport aircraft accidents which occurred in France and in the United States, has led to taking into account the following breakdown schemes of impact velocities:

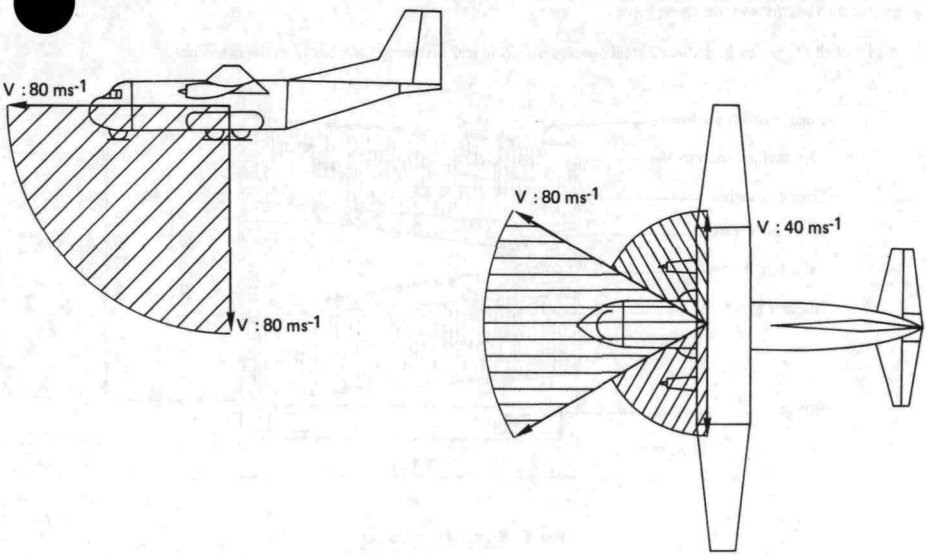


Fig. 4 : Breakdown of impact velocities

This breakdown covers nearly the whole of air accidents (97 %).

Specifications of impact tests

Each bundle must be so arranged on board the aircraft as to withstand a 80 ms^{-1} transverse impact and a 40 ms^{-1} longitudinal impact.

4. CONTAINERS

4.1. Design and main features of the container

The design is directly derived from successive certification tests.

The container is shaped like a cylinder (dia. 1.70 m ; length 2.10 m), the actual shape being slightly truncated. Its weight is 1 800 kg approx.

From the inside outwards, the container comprises successively :

- an adaptive hardware of rigid plastic material for shimming the nuclear assembly ;
- a rigid inner enclosure made of aluminum alloy, which constitutes the leaktightness barrier ;
- a shock absorber made of balsa which, at the time of the impact, absorbs the kinetic energy of the internal mass M_i:

$$M_i = \begin{cases} \text{Mass of inner enclosure} \\ + \text{ Mass of adaptive hardware} \\ + \text{ Mass of nuclear assembly} \end{cases}$$

- a thermal protective layer made of laminated cork;
- an outer impact-proof enclosure made of high deformation steel designed to contain the shock-absorber at th[illegible]f impact and to wisthstand punching forces.

Additionally, a frame of ordinary steel enables handling and fastening the bundle in the cargo hold.

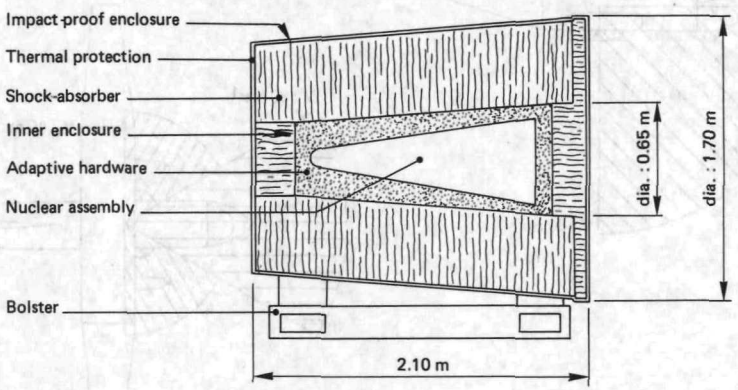


Fig. 5 : Sketch of container

4.2 Impact damping calculation (principle)

Kinematic approach

The kinematic calculation permits assessing the dimensions of the damping material.

Assuming a constant deceleration γ of the internal mass M_i after an impact, let V_o be the impact velocity, the minimum crashing course C_{vo} of mass M_i is :

$$C_{vo} = \frac{V_o^2}{2\gamma}$$

hence for $V_o = 80\ ms^{-1}$ and $\gamma = 10\,000\ ms^{-2}$ $C_{80} = 0.08$ m

for $V_o = 40\ ms^{-1}$ and $\gamma = 10\,000\ ms^{-2}$ $C_{40} = 0.03$ m

These minimum theoretical values must in fact be multiplied by a coefficient C_f (> 1) which depends on the shape of the inner enclosure with respect to the impact (plane, cylinder, flat cone point, sphere, etc).

Furthermore, the maximum crushing of whatever damping material reaching at the most 60 % of the initial thickness, e_o must therefore be as follows :

$$e_o > \frac{C_{vo} \times C_f}{0,6}$$

Dynamic approach

The dynamic calculation permits specifying the mechanical features of the damping material. There remains to seek such [redacted]al material, hoping that it is actually available on the market.

K[redacted]c energy of mass M_i :

$$E = \frac{1}{2} M_i V_o^2$$

If S is the projected surface area of mass M_i, kinetic energy E is absorbed by the damping material which is crushed by an amount C_{vo}

$$E = \frac{1}{2} M_i V_o^2 = p S C_{vo}$$

in which p = crushing pressure (or compression rupture stress) of the material. Hence :

$$P = \frac{M_i V_o^2}{25 C_{vo}} = \frac{M_i \gamma}{S}$$

For $V_o = 80\ ms^{-1}$ (and $S_T = 1\ m^2$) then : $p \sim 50$ bar

For $V_o = 40\ ms^{-1}$ (and $S_L = 0,32\ m^2$) then : $p \sim 150$ bar

The values found have led to opting for :

- low density balsa for lateral damper ($\rho \sim 150\ kg/m^3$)
- high density balsa for axial damper ($\rho \sim 250\ kg/m^3$)

5. DEMONSTRATION METHOD

Several models intended to be subjected to impact tests have been prepared to validate the design proposed in § 4. The purpose of these tests is to check the boundary characteristics of the measured impact on the dummy simulating the nuclear assembly, as well as the condition of the inner enclosure (conservation or not of leaktightness).

The extent of the existing testing facilities and the concern for limiting the construction costs of these models have led to carr[redacted]out these tests on reduced scale models. The reduction ratio R has been selected to equal 0.4.

5.1 Similarity rules

- the materials used for the models are the same as those selected for the actual containers ;
- all the dimensions are reduced by the same similarity ratio ;
- impact velocities are kept.

In these conditions, in which similarity ratio R = 0.4, the distinct parameters are as follows .

Magnitude	Real container	Model container
– Length	L	$L' = LR = 0{,}4\ L$
– Velocity	V	$V' = V$
– Mass	M	$M' = MR^3 = 0{,}064\ M$
– Acceleration	Γ	$\Gamma' = \Gamma R^{-1} = 2{,}5\ \Gamma$
– Stress	σ	$\sigma' = \sigma$
– Time	T	$T' = TR = 0{,}4\ T$
– Cost	C	$C' \sim CR^{1,5} = 0{,}25\ C$

For the record $C = 240\,000$ F
$C' = 60\,000$ F } (in French Francs 1980)

5.2 Test layout

The layout is strictly in accordance with the arrangement of the bundles on board the aircraft and the appropriate specifications for simulating air accidents.

Test 1 : "Foreward" longitudinal impact at $V = 40\ ms^{-1}$

Test 2 : "Rearward" longitudinal impact at $V = 40\ ms^{-1}$

Test 3 : Transverse impact at $V = 80\ ms^{-1}$

Test 4 : "Simultaneous" and transverse impact test at $V = 80\ ms^{-1}$ of two bundles in tandem, with their longitudinal axis perpendicular to the direction of the impact velocity.

Test 4 is intended to measure the chronology of successive impacts and the effects of the impact by container 2 on container 1.

6. TEST ARRAYS AND FACILITIES

Near Bordeaux are two large facilities enabling the hereabove mentioned tests to be conducted, namely :

- the accelerated fall tower at the C.E.S.T.A.,
- the launching rail at the Centre d'Essais des Landes

6.1 Accelerated fall tower

The facility comprises :

- a 30 metre high steel tower,
- a horizontal target consisting of a concrete slab weighing 200 metric tons, lined with a shielding plate, 120 mm thick,
- 2 guide cables,
- a propellant device including one or more powder propellant booster (s), fitted at the rear of the bundle to be tested so as to reach the desired impact velocity.

This installation has enabled unit tests of containers to be conducted (tests 1, 2 and 3) using the following arrays :

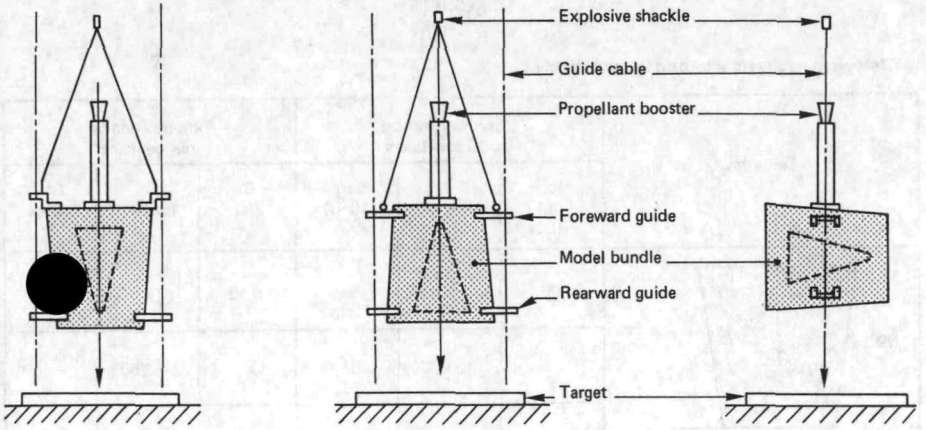


Fig. 6 : Unit impact tests

6.2 Launching rail

The facility comprises :

- a 300-metre long steel rail stopping 10-metre short of the target,
- a trolley, moving along the rail, equipped with a battery of powder boosters to speed up the device,
- a target weighing 60 metric tons made of concrete blocks, lined with a steel plate 120-mm thick.

Test array 4 is illustrated on Figure 7.

The two bundle models are arranged in tandem on a platform simulating the aircraft floor, this frame being fixed to the trolley.

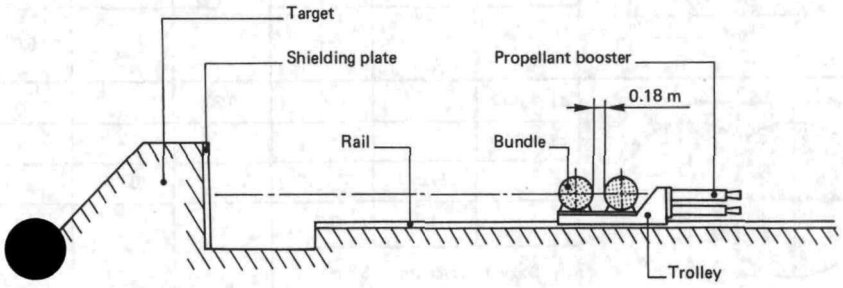
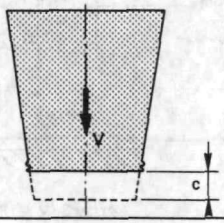
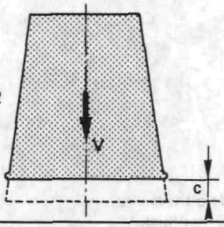
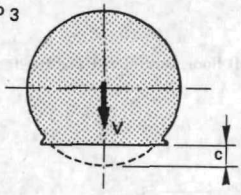
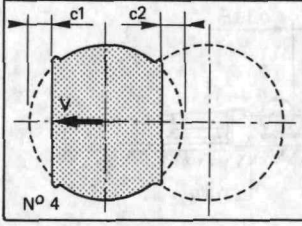


Fig. 7 : Simultaneous impact test

Each model container is equipped with accelerometric sensors ; acceleration is filmed by ordinary movie cameras (200 pictures/s), and the impact by extra fast movie cameras (8 000 pictures/s).

7. RESULTS

The overall results are shown on the table below :

Test array	Measures taken on model containers			Extrapolation on a real container		
	c (m)	γ (ms^{-2})	Δt ($10^{-3}s$)	C (m)	Γ (ms^{-2})	Δt ($10^{-3}s$)
N° 1 	0.05			0.12		[redacted]
		13 000			5 200	
			3.5			8.7
N° 2 	0.02			0.06		
		20 000			8 000	
			2.6			6.5
N° 3 	0.08			0.20		
		26 000			10 400	
			3.8			9.5
N° 4 	c1 = 0.12			0.30		
	c2 = 0.09					
		25 000			10 000	
			3.7			[redacted]
	Second impact at t = 5.3 ms					

Extrapolation on a real container of the results of measures taken on models shows that the desired impact resistance of ($\Gamma < 10\,000\ ms^{-2}$, $\Delta T < 10$ ms) has been achieved.

8. FILM

DISCUSSION

Question by A. H. Partridge, United Kingdom: In your test 4 you showed two packages one behind the other. Your diagram shows several packages transversely loaded in the aircraft. Thus in the real situation the front drum would be subject to crushing by all drums behind it. Would you care to comment on this aspect?

Answer: The six bundles are arranged in the aircraft cargo hold, one behind the other and spaced in such a way (i.e. 0.45 m) that the "flight" time required to cover this distance at the speed of [redacted] km/h (i.e. 20 ms) is longer than the impact time (i.e. 10 ms).

The transverse impact tests concerned two bundles only:
- on grounds of economy
- in view of its cylindrical shape, each container is capable of absorbing its own kinetic energy ("fore impact") and at least half of the kinetic energy of the following container ("aft impact").

It would be unrealistic to assume that the axis of the aircraft impact will strictly coincide with the axis of the aligned bundles.

SIMULATION OF AIRCRAFT CRASHES ON SPENT FUEL TRANSPORT AND STORAGE CASKS

F. Glaser, FR Germany

In the Federal Republic of Germany the authorities require that nuclear facilities are so designed that safety is ensured not only in the event of internal and external incidents but also in the event of an aircraft crash.

This requirement also applies to the casks in which spent fuel elements are to be transported and stored.

In the tests described below, it was shown that spent fuel transport and storage casks of CASTOR type retain their functional qualities even in the event of an aircraft crash.

At this point it should be noted, that tests of this type are model tests, whereby the methods and equipment must simulate an aircraft crash as realistically as possible.

A special test on the kerosene fire after the crash was no longer considered necessary as fire resistance was proved by experiments and finite-element calculations carried out during the procedure for approval as a transport cask (type B(U)) by the BAM, Berlin.

TESTING DETAILS

The simulation of an aircraft crash took place at a testing site in Emsland in the north-west of the Federal Republic of Germany. An acceleration facility installed there and sponsored by the Federal Ministry for Research and Technology was used, which can accelerate missiles with dimensions up to 600 mm diameter and maximum mass of 1000 kg to a velocity of up to 300 m/s.

Tests carried out up to now, impact on concrete walls of various construction, were performed using projectiles, hollow cylinders with graded wall-thickness without additional inner mass. Among other things, they help to corroborate the prescribed theoretical values of the collision-time behaviour on impact in practical tests.

In the tests described here, to obtain further simulation of the collision-time behaviour in the event of an aircraft crash, in particular impact of a jet engine, a projectile with additional mass was developed.

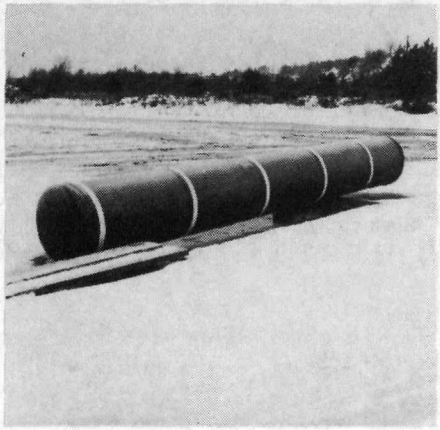


Fig. 1 View of the Projectile

The designed missile consists of a tube 5 m long, with a diameter of 600 mm, a wall thickness of 5 mm and a dome-shaped head of equal thickness. Inside, 1.64 m from the top, there is a boiler pipe centralised by welded plates; this boiler pipe is supported on a cross made out of 20 mm thick plate at the rear of the missile. The missile is sealed at the end by a 60 mm thick plate, which, by means of a circumferential copper sealing ring also serves as a cartridge-case base. The missile mass is approximately 1000 kg.

The 1.7 greater mass of a jet engine - a jet engine mass of 1.7 is taken as a basis in the case of a crash velocity of 215 m/s - is compensated for by selecting a higher impact velocity.

Furthermore, in comparison to the GRS (Gesellschaft für Reaktorsicherheit, Köln) which requires, as a minimum, that the crash energy of a jet engine is applied to the target, our tests showed a higher energy application. Moreover, we had an additional specific load per square unit due to the smaller cross-section of the projectile compared to the cross-section of the jet engine.

As, in addition, the collision-time behaviour on impact of such a projectile on the cask comes relatively close to the determined behaviour during impact of a jet engine, the projectile can be taken as idealised model of a jet engine and the tests may therefore be regarded as conservative.

Three tests have so far been carried out:

- impact of the missile side-on to the cask;
- impact of the missile on the lid region when the container is in an inclined position (line of fire diagonally through the centre of gravity of the cask);

- impact of the missile on the centre of the lid region at right-angles.

Cask parts in a ratio 1:1 but reduced in length, were used as test objects. For the side-on impact, part of a CASTOR Ia with auxiliary lid was used, for impacts on the lid region a CASTOR IIa part with original lid construction including all valves, attachments, testing devices, etc. The mass of the test objects was 20 t for test no. 1 and 48 t in the case of test no. 2 and 3. The impact velocity of the missile was measured and documented by high-speed cameras for all three tests. Apart from photographic documentation, after each test the casks were submitted to the following tests by the BAM, Berlin:

- ultra-sonic testing;

- testing for surface cracks;

- leaktightness testing.

TEST RESULTS

The three tests were carried out in December 1978, December 1979 and June 1980.

The distance of the test objects from the muzzle of the accelerator was 30-40 m.

During the tests sand banks and/or concrete walls were used as abutment to stop the cask from moving backwards, i.e. to ensure full energy input.

The individual impact velocities attained were:

- test no. 1: 305.9 m/s

- test no. 2: 304.4 m/s

- test no. 3: 292.8 m/s

In all three cases the missile either totally disintegrated or only a piece at the end about 0.7 - 1 m long was left.

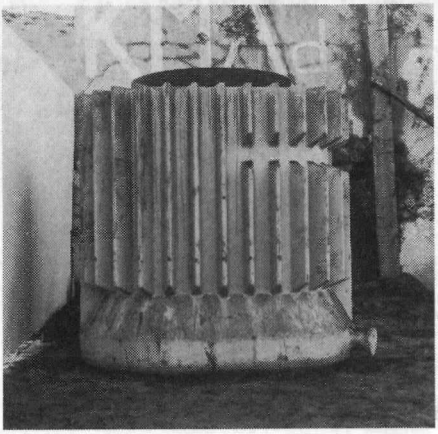


Fig. 2 Cask before the Test

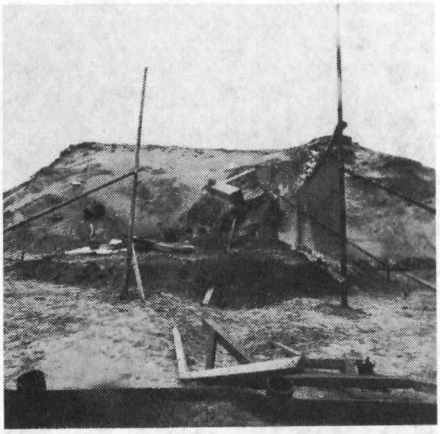


Fig. 3 Position of the Cask after the Test

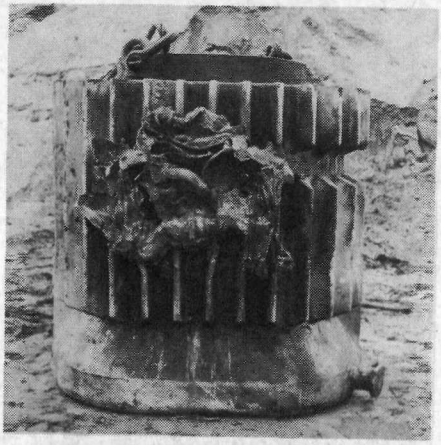


Fig. 4 Impact Point from the Front

Examination of the test object showed purely deformations in the rib region which received side-on impact; on the body itself, no cracks or flaws or other defects could be ascertained. The lid region was undamaged.

TEST NO. 2

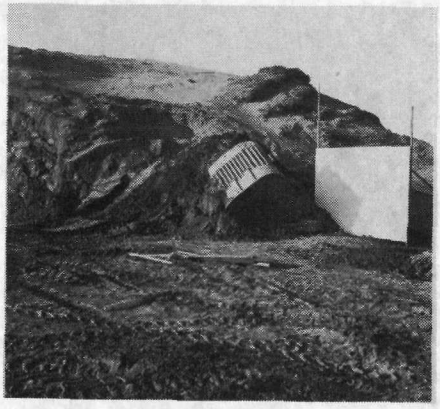


Fig. 5 Cask before the Test

Fig. 6 Position of the Cask after the Test

Fig. 7 Projectile Impact Point

Impact on the lid and/or valve region at a positioning angle of 31° did not reduce the shielding effect of the cask. No traces of damage to the cask were found here either. The leaktightness of the whole system was reduced but did not fail.

By means of a repair routine which was already established as part of the whole concept before the tests took place, full integrity could be restored within 90 minutes.

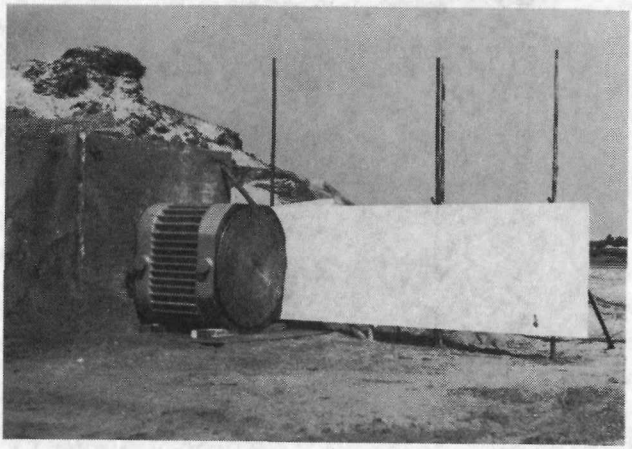


Fig. 8 Cask before the Test

Fig. 9 Cask after the Test

Fig. 10 Bending of the Outer Lid

The outer lid region was damaged (buckled), the main lid was undamaged. The leaktightness was reduced, loss of integrity was, however, within permissible limits. Complete repair was possible by replacing the outer lid.

In this test it should be taken into account that the probability of a stress load such as vertical impact on the lid region is arbitrary.

SUMMARY

The tests have confirmed that the CASTOR type transport and storage cask can withstand extreme stress loads without a great amount of damage and fulfills requirements by the authorities in the Federal Republic of Germany, that it is "resistant to a fast-flying military aircraft crash."

DISCUSSION

Question by M. Brown, United Kingdom: 1. How much energy went into motion of the cask? 2. Do you think the shaft of an aircraft turbine would provide a smaller frontal area through which the force of the impact could be applied?

Answer: This has not yet been investigated. It should, however, be noted that the container is only set in motion when the aircraft crash is over. The total energy therefore goes into the cask. 2. The aircraft turbine shaft is narrower and therefore has a [illegible]er impact surface. In the case of a projectile, it should be take[illegible]to consideration that the turbine is not hit first. The nose and the cabin are first buckled. Therefore, energy is already lost. Consequently, the test conditions show more extreme values than in reality.

Question by A. Müller, FR Germany: You mentioned the impact/time diagram. Did you determine an impact/time diagram from the experiment or to which impact/time diagram do you refer?

Answer: I refer to the given theoretically determined impact/time diagram. In the experiment we did not determine a time/impact diagram. We are trying rather to change the projectile so that we come as close as possible to the theoretical diagram.

RADIOISOTOPE PACKAGES IN CRUSH AND FIRE

C. B. G. Taylor
The Radiochemical Centre, Amersham

INTRODUCTION

Radioisotopes are now being transported on a very wide scale[1]. This year my own organization will send out more than 300,000 packages. Worldwide this traffic is now numbered in millions.

These are mostly small packages weighing between a few hundred grammes and several kilogrammes. Most carry radiochemicals, radiopharmaceuticals, or kits for clinical tests. Their activities are generally low, usually $10^{-3}A_2$ or less. Some carry radiation sources and may approach A_2 but these are a small minority. Very few require Type B packaging.

Mixed loads of several hundreds of these packages leave our laboratories every day and are taken by road to distribution depots or airports. What would happen if there were to be a serious accident, perhaps followed by fire? Our packages meet IAEA standards[2], which for Type A include tests for impact and penetration, but not dynamic crush or fire. How would these light-weight packages behave in a high-speed collision, in which much heavier packages might crush them severely?

We have thought it worthwhile to stage a series of full-scale crashes with real vehicles, loaded with typical packages. This could help us make better judgements about the safety of our operations, guide us in our choice of delivery vehicles, and give some foreknowledge of the problems we would meet when clearing up an accident. It would also be useful to those responsible for reviewing the transport regulations.

We used three small commercial vehicles which had been taken out of service. Vehicle A is shown in Figs 1, 2 and 3. With this open-backed truck we could see how the packages would move during a crash. Vehicle B (Figs 4 and 5) had the same chassis construction but had a closed van back. Both were loaded in the same way, with a mixture of Type A and Type B packages, half the former being in overpacks, half stacked loosely. Heavy Type B drums were loaded behind the Type A packages. After crashing vehicle B we poured fuel on the ground underneath it and staged a transport fire (Fig 4).

Vehicle C was a small conventional delivery van. It was loaded with partly filled Type A and Exempt packages to simulate a run between two of our production plants.

PACKAGES AND LOADINGS

Vehicles A and B each carried 300 Type A packages of the commonly used carton type. This has an outer cardboard case 140 x 110 x 110mm containing a sealed steel can Ø100 x 125mm in which are packing pieces of expanded polystyrene, a small lead pot or polystyrene container, and a glass vial holding a few

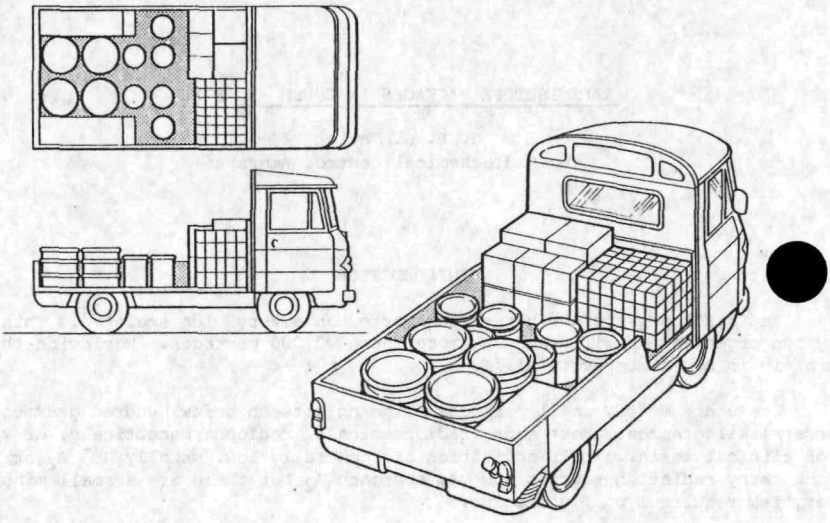


Fig. 1 Vehicle A loaded for crash test. 300 Type A packages are stacked at the front of the load area, those on the left being in overpacks

millilitres of radioactive solution. In the trials each vial held 10 ml of non-radioactive water. Half the vials were in lead pots 3mm thick, the rest being in polystyrene containers. These are the weakest inner components we use with this type of packaging.

The overpacks were rigid cases of strong corrugated cardboard closed by adhesive tape.

In both vehicles there were ten Type B steel drums behind the cartons, small ones weighing 50 kg and larger ones weighing 58 kg (Fig 1). In each drum was a lead pot weighing 40 kg, in a fire-shield of bonded cork. The drums were arranged so that a column of four drums, and beside this a single drum, would impact the overpacked and non-overpacked sections of the load of small cartons.

The smaller van C carried 72 partly-finished Exempt radiopharmaceutical kits in front of six small drums designed to carry Type A quantities of labelled compounds. Each kit was a moulding of expanded polystyrene 190 x 140 x 100mm containing 12 small glass bottles and weighing 0.3 kg. Each Type A drum held 80 glass vials and weighed 9 kg. The kits were packed in plastic trays, 18 kits per tray. Four trays were stacked one above the other at the front of the load compartment of the van. The Type A drums stood immediately behind them in parallel columns of three drums each. In a crash they would move forward and crush the kits in the bottom two trays.

TRIALS

The trials were done at a transport laboratory with a facility for impacting vehicles against traffic barriers, bridge parapets, etc. A vehicle can be

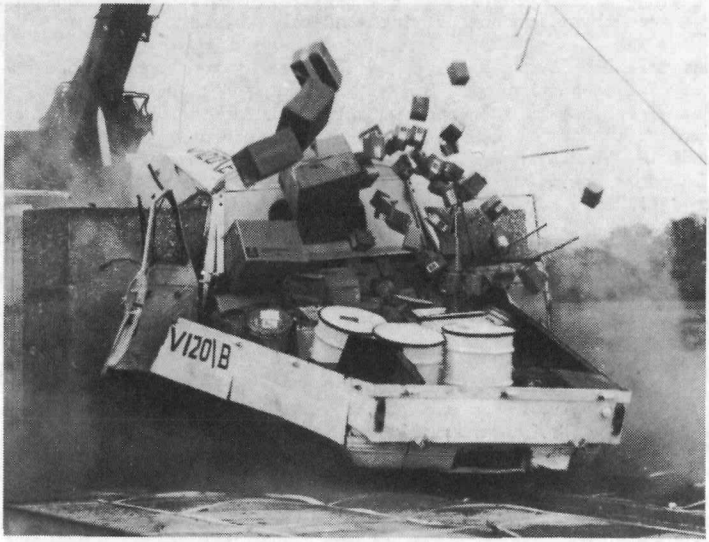


Fig 2. Impact of vehicle A at 30.8 m/s

Fig. 3 Damaged packages in vehicle A after impact at 30.8 m/s

accelerated along a track leading to the target (in this case 45 tons of concrete blocks) by an electrically powered winch-and-cable system.

We had intended to crash all three vehicles at the highest road speed allowed in Britain, 70 miles per hour (30.8 m/s), but our vehicles A and B, with their heavy loads, were close to the upper weight limit for the acceleration system and over-load protection cut in during our first two attempts. We achieved only 9.7 m/s with vehicle A and 19.8 m/s with vehicle B. Fortunately vehicle A was not greatly damaged by the first impact and we were able to run it again, this time reaching the full 30.8 m/s. There was no difficulty in reaching this speed with the much lighter van C.

In the event we were glad to have achieved this range of speeds. Impa[redacted]/s with a change of velocity 9.7 m/s is typical of many minor road accidents, [redacted]/s represents a serious crash, and 30.8 m/s into a rigid target is a very severe accident indeed.

The impacts, which were not instrumented, were recorded by high-speed cine-photography from above and from one side and by still photographs. The burning of vehicle B was also not instrumented (we had not expected it to burn as well as it did) but some information was gained from temperature-sensitive markers attached to the lead pots inside the Type B drums.

After the tests we took all the containers back to our laboratories for examination.

RESULTS OF TRIALS

During impact at 9.7 m/s the drums in vehicle A moved forward into the cartons and rebounded to near their original positions. There was little apparent damage to the cartons but many of the cans must have been dented. In the second impact, at 30.8 m/s, many cartons were severely damaged, the drums rebounded to the back of the vehicle, and two were thrown out onto the road. The upper layers of overpacks and cartons were propelled violently upwards, some reaching a height of 5m. Both pairs of small Type B drums in front of the larger ones were compressed to about three quarters of their original diameter and their lids were buckled and forced off.

Many of the cans from the damaged cartons were so crushed and distorted as to be hardly recognisable as cans. All were examined for failure of a seam or penetration of the metal at any point. Six had been pierced and 68 had failed seams, four of these also being pierced. The glass vial was found to be broken in 16 of these 70 potentially leaking cans. Of the 16 broken vials only three were in lead pots, the rest being in polystyrene containers. Even a thin-walled (3mm) lead pot gives excellent protection against crushing.

The impact of vehicle B at 19.8 m/s caused similar but milder damage to the packages. As they rebounded the Type B drums burst open the doors at the back of the van and one of them fell onto the roadway. In this case none of the dr[redacted] lost its lid. All the Type A packages were however severely affected by th[redacted] subsequent fire test.

The fire was started by igniting 40 litres of petrol poured on the ground under the van, to simulate leakage from a ruptured fuel tank. The flames took hold very quickly, setting fire to the tyres and rising into the driving compartment, from which they streamed out through the broken windshield (Fig 4). The cardboard of the cartons was set alight by heat transmitted through the metal floor of the load compartment and burned freely, air for combustion entering

Fig. 4 Vehicle B burning. It had previously been crashed at 19.8 m/s

Fig. 5 Damaged packages in vehicle B after fire test

through the open rear doors and over the wooden bulkhead behind the seats. Within 12 minutes reports could be heard as the air inside the cans expanded and their lid seams snapped open to release the internal pressure. The whole interior of the load compartment appeared to be filled with flames. The cork fire shields in the Type B drums contributed to the blaze, jets of flame bursting out as their lids bulged outwards and lid gaskets were destroyed. Eventually all the cartons and overpacks burnt away, leaving a disordered mass of cans and drums (Fig 5).

When the cans were examined after this fire test it was found that most had retained their lids, but the over-pressure had bulged them out until a seam opened, usually over a length of 10-20mm. All the lead pots in the cans had melted. Most of the glass vials were shattered, or were melted or distorted by heat. All had lost their contents.

About 15% of the can seals failed during the 19.8 m/s impact (as evidenced by the absence of bulging due to pressurization in the subsequent fire test) compared to 23% when vehicle A was crashed at 30.8 m/s. Only 2% of the cans were pierced, the same as in A. The maximum temperatures reached by the heavy lead pots in the Type B drums, as shown by the temperature sensitive labels, ranged from 70°C to 130°C.

When vehicle C was impacted at 30.8 m/s the drums demolished the two lower trays of Exempt packages, reducing the kits on these trays to broken pieces of polystyrene and scattered bottles. Nevertheless, of the 864 glass bottles in the kits 827 were recovered unbroken. This was a striking result of this test. In real kits of this type only one bottle in twelve contains any radioactivity, and the amount is small, 2μCi of I125. The missing bottles would thus have contained less than 10μCi of I125, and this would have been the total release of radioactivity in this violent accident.

PERFORMANCE OF OVERPACKS

It has been suggested that cans in cartons packed into overpacks may be more liable to crash damage than cans in separate cartons, because overpacking could hold the cans more firmly in the path of a heavy object. This was not confirmed by these tests. If anything the performance of the cartons in overpacks was slightly the better.

The use of overpacks always adds to safety when large numbers of small packages must be transported. For example, it prevents loss and reduces damage by cargo handling equipment, which is most likely to happen to a small package. Overpacked packages would also be much easier to recover after a road accident.

CONSTITUTION OF REAL LOADS

We have analysed the make-up of typical van loads leaving our plants and find the whole load is seldom more than one or two times A_2 even when as many as a thousand packages are being carried.

The majority of these packages are research chemicals containing a few millicuries of C14 or tritium, or medical kits containing a few microcuries of I125. Both these average less than $10^{-4}A_2$ per package. Even a high activity technetium generator, containing a curie of Mo99, amounts to only $10^{-2}A_2$. Some radiation sources may approach A_1 in a single Type A package, but would contribute nothing to the A_2 release fraction, even in a fire, because of their encapsulation. The most hazardous packages in this context would be bulk consignments of primary radioisotopes, or packages containing alpha emitters, both of which can approach A_2. However, these are very few in number.

In recent years there has been an increase in the number of radioisotope packages transported but there has also been a reduction in the activity carried per package, both absolutely and in terms of A_2. Fewer curies are carried because new analytical techniques need less radioactivity. Radiotoxicity is lower because nuclear medicine is using nuclides with shorter half life. It would be very difficult to quantify the benefits and hazards of this type of traffic in units which could be compared with each other, but a trend in their ratio can be discerned, and it is clearly favourable.

DISCUSSION

These trials simulated severe road accidents in which light-weight packages were badly damaged by the impact of heavier packages. Typically, the can in a crushed Type A package was crumpled and its seam might open over a few millimetres, but the glass vial inside was seldom broken. In a few cases the vial failed and there could have been some release of radioactivity, but this would have been slight. The release fraction 10^{-3} which underlies the IAEA regulations[3] still seems a valid assumption, at least for packages containing metal cans. The multiple barriers provided by carton, can, absorbent, spacers, pot and vial remain an effective containment system even when the package has been almost completely flattened.

There is more reason for concern about the effects of fire. Fire in a vehicle, although rare, may happen at any time - not only after an accident - because of an electrical fault, or friction between moving parts, or fuel leakage. As our tests showed, a fuel fire can spread very quickly and lead to the almost total destruction of a large load of Type A packages.

The probability of fire can be minimised by using diesel fuel instead of petrol, by all-metal bodywork, and by carrying effective fire extinguishers. There should be no windows in the walls or doors of the load compartment, as these can break in an accident and let in air for combustion. Doors should have an especially secure closure which will not easily burst open.

Although a serious fire would increase the fraction of material released it could also aid dispersion and so reduce the fraction available for ingestion. The fact that only low levels of contamination were experienced in the recent aircraft accident at Athens, reported elsewhere in this Symposium, illustrates that dispersion by fire can be helpful. The problem is however so complex, and so dependent on circumstances, that a quantitative understanding which will be generally applicable can hardly be expected.

In a road accident followed by fire the driver may be able to warn other people to keep away. If he is injured, their first action will be to help him, and the regulatory fire-proof label in his cab will then provide a warning. However, absolute safety cannot of course be assured as in the transport of other hazardous goods.

Responsibility for maximising safety remains, as it always must, with the operator of the transport service. Regulations and codes of practice provide guidelines but good equipment and well-trained operatives remain the final safeguards.

REFERENCES

1. Taylor, C. B. G., "Packaging and Transport of Radioisotopes", symposium paper IAEA SR 10/45, IAEA, Vienna 1976.

2. "Regulations for the Safe Transport of Radioactive Materials", 1973 revised edition, Safety Series No. 6, IAEA, Vienna, 1979.

3. Fairbairn, A., Dunning, N. J., "The classification of radioisotopes for packaging", Regulations for the Safe Transport of Radioactive Materials: Notes on Certain Aspects of the Regulations, Safety Series No. 7, STI/PUB/32, IAEA, Vienna (1961) 23.

AN ANALYSIS OF THE LONGITUDINAL CRUSH ENVIRONMENT FOR SMALL RADIOACTIVE MATERIAL PACKAGES INVOLVED IN AN AIRCRAFT CRASH*

J. D. McClure and W. F. Hartman
Sandia National Laboratories, Albuquerque, NM, USA

INTRODUCTION

The environment associated with severe transportation accidents has been studied in detail for relatively small packages.[1] An initial analysis of the crush environment in accidents involving US commercial aircraft has been performed.[2] The analysis which is described in this paper builds on the information in the earlier report and removes some of the conservatisms of the earlier analysis related to longitudinal crush through the use of information that has been obtained from a large commercial air freight operation, the Flying Tiger Line.[3] For the purpose of this analysis, a "small" package is any package which has a mass of 500 kg or less.

This paper describes the longitudinal crush environment for a light-weight, air-transportable, accident-resistant container (LAARC) that can be used to transport radioactive material (RAM). The analysis of the crush environment involves the determination of the loads imposed upon a RAM package during the crash of a large and heavily loaded cargo aircraft. The forces which act upon the RAM package in the aircraft crash environment are caused by deceleration of a defined cargo load column. The cargo load column analyzed in this paper is of unit cross-section area, it has the mass properties of the cargo and its length is the longitudinal extent of the cargo. The analyses described in this paper consist of a static and dynamic stability analysis of the cargo load column. In addition, a lumped-mass dynamic analysis of the load column was performed. The results of the analysis, while applied to the geometry of the LAARC packaging are, in fact, general and can be applied to other package geometries.

CARGO LOAD COLUMN DEFINITION

The crush of a radioactive material package has been considered to be divided into two basic modes, vertical crush and horizontal (or longitudinal) crush. The longitudinal and vertical crush modes were both examined in Reference 2 but emphasis was given to the analysis of the vertical crush mode. The rationale for the longitudinal analysis in Reference 2 was that it was to be an upper limit and, thus, assumed that the dynamic effects of all cargo aft of the hazardous material (or RAM) package location could be concentrated on a small hazardous material package. On-scene observation of actual cargo containers and cargo loading and unloading operations[3] has lead to the conclusion that this assumption is far too pessimistic. The cargo containers for aircraft cargo operations require the "containerizing" of many small packages of assorted sizes and shapes. These cargo containers are relatively thin-walled, are fabricated of sheet aluminum and are easily deformable. The industry-wide average density for air cargo is about 136 kg/m^3 (8.5 lb/ft^3). Relatively dense cargo in the industry, on the average, is approximately 320 kg/m^3 (20 lb/ft^3).[3] These average density values are used in the load-balance calculation

*Work sponsored by the US Department of Energy under Contract DE-AC04-76DP00789.

tions performed by the air-cargo carriers. The thin-walled configuration of the air cargo containers plus the low density of general cargo suggest that the relatively hard (and small) hazardous material package will deform the walls of an adjacent air-cargo container during longitudinal crush and be enveloped by the adjacent cargo.

The low stiffness provided by adjacent cargo allows large deformations around the hard, small hazardous material package. This produces the effect of cargo impinging upon cargo and cargo impinging upon and capturing (enveloping) a relatively hard hazardous material package. This suggests that it is inappropriate to consider that all of the cargo aft of a hazardous material cargo location will be concentrated upon a single small package during a longitudinal deceleration. In effect, the relative ease with which a small hard package can produce cargo deformation indicates that the dynamic crushing effects upon a hazardous material package most probably will be produced by a "longitudinal column of cargo" aft of the hazardous material package. This column is a longitudinal prism of cargo with a cross-sectional area equal to the projected area of the package and a length equal to the longitudinal extent of the cargo in the aircraft. The analysis which follows will examine the structural stability and dynamic loading of a cargo load column of unit cross-sectional area and the analytical results will be multiplied by the frontal (projected) area of the actual RAM package (for this report the LAARC package), but the results can be applied to other package sizes. This loading configuration can be contrasted with that of the earlier worst-case analysis[2] where it was assumed that the RAM package was sandwiched between a pair of nondeformable plates which transferred all of the inertial load from the aft cargo to the package. Closer examination of cargo loading practices indicated that this is an extremely unlikely (if not impossible) scenario.[3]

In the static and dynamic stability analyses which follow, the cargo load column is assumed to be elastic. It must be recognized, however, that the properties chosen to represent the mechanical characteristics of the load-column are assumed representative values and that the actual cargo composition is random and, in general, unknown.

The puncture of the hazardous material package by adjacent cargo is excluded from this analysis. Present day cargo operational procedures require that only frangible cargo be placed adjacent to the hazardous material cargo. If the composition of the cargo is such that it contains heavy equipment or machinery which has the capability of puncturing adjacent cargo, carrier operational procedures require that such cargo is to be removed a sufficient distance from the hazardous material (and with interspersed frangible cargo) in order to preclude puncture of a hazardous material package.

ANALYSIS OF CARGO LOAD COLUMN

Standard body, cargo aircraft contain a 9 g barrier net with a million pound capacity to contain the cargo and protect the flight crew from forward moving cargo in high deceleration situations such as aircraft crashes. In nominal crash loads, aircraft crash loads are represented by 9 g longitudinal decelerations.[4] Aircraft structural limits are given by a 20 g longitudinal deceleration.[4] Above 20 g deceleration the aircraft and the cargo load columns will structurally disassemble.

A simplistic analysis of the cargo load column can be accomplished by examining the free-body diagram of the cargo load column, Figure 1. For purposes of analysis, the cargo load column is assumed to have a unit cross-sectional area. Following the evaluation of the dynamic response of the load column the unit dynamic force on the load column is multiplied by the projected (frontal) area of the

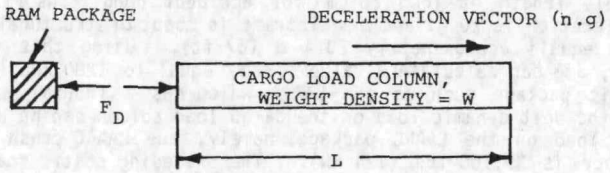


Figure 1. Free-body diagram of cargo load column.

radioactive material package to obtain the total force acting. Using Newton's Second Law, the magnitude of the dynamic load F_D on the load column for deceleration of magnitude ng and the cargo weight density per unit length W, is,

$$F_D = \frac{WL}{g} ng \tag{1}$$

If the cargo load column does not become unstable and buckle, the unit load on the column would be F_D. However, the length of the cargo can be on the order of 30 m (100 ft) and the question arises as to whether such a cargo load column can remain stable, that is, will it buckle? Observation of the combination of small packages in containerized and palleted (netted) cargo and their minimum lateral resistance to movement under extreme accident conditions produces the condition whereby it is assumed that the cargo load column has no lateral buckling resistance due to the presence of adjacent cargo. It is also assumed that the simplest structural configuration for the load column is a pinned-pinned column. The Euler static buckling load for such a column is,

$$P_{Cr} = \frac{\pi^2 EI}{L^2} \tag{2}$$

P_{Cr} = Euler buckling load
E = Modulus of elasticity of column material
I = Second moment of area about column cross-sectional axis
L = Column length

Combining equations (1) and (2) by equating F_D to P_{Cr} yields the equation

$$L^3 = \frac{\pi^2 EI}{nW} \tag{3}$$

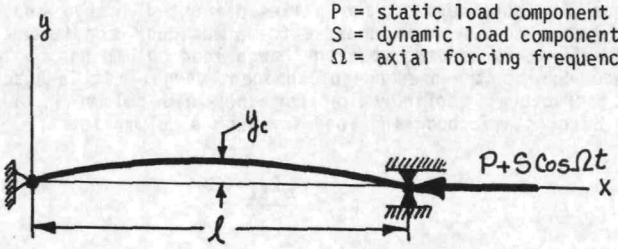
A comparison of density and modulus of elasticity values for common materials shows that, in general, the lower density materials possess a lower modulus. Based primarily upon the values for wood, density equal to 560 kb/m^3 (35 lb/ft^3) and modulus equal to 1.1×10^4 MPa (1.6×10^6 psi), a value for E of 6895 MPa (1×10^6 psi) was estimated to represent aircraft cargo. The analysis used a value of 320 kg/m^3 (20lb/ft^3) to represent the cargo density. With the unit dimensions

on the cross-section of the load column, equation (3) was used to calculate the maximum stable length of load column for accident conditions (i.e., the longitudinal deceleration is 20 g) and the aircraft is about to structurally disassemble. For this, L equals approximately 20.4 m (67 ft). Using this value for L in equation (1), one can calculate a value for F_D equal to 1280 kPa (26,800 lb/ft^2). For a specific package such as the LAARC which has a frontal area of .136 m^2 (1.46 ft^2), the unit dynamic load on the cargo load column can be used to evaluate the dynamic load on the LAARC package, namely, the LAARC crush load calculated in this manner is 39,100 lbs (174 kN). The preceding static stability analysis can be used to develop the concept that structural stability can be used to physically limit the magnitude of the longitudinal crush force on a RAM package. Additional stability arguments using dynamic stability analysis will also be presented.

Column Buckling Under Dynamic Loading Conditions

Since the forces generated by the cargo load column are basically dynamic in nature, the stability of the cargo load column shall be examined from a dynamic viewpoint[5] in this section. The results of this analysis shall be expressed in terms of the static buckling load P_{Cr}.

The geometrical description of the dynamically loaded cargo load column is shown in Figure 2. The cargo load column is assumed to be represented by a pinned-ended column configuration loaded with an axial load which consists of a constant term P and a cosine varying component.



P = static load component (axial)
S = dynamic load component (axial)
Ω = axial forcing frequency.

Figure 2. Column geometry - dynamic loading.

The differential equation of the dynamically loaded column can be transformed mathematically and solved. The form of the solution is given in equation (4),

$$S = \frac{b}{a} [P - P_{Cr}] \qquad (4)$$

where the terms in equation (4) are given in equation (5). The solution domain for equation (5) is shown

$\frac{q}{g}$ = mass per unit column length , $\rho = \frac{P}{P_{Cr}}$, P_{Cr} = Static Euler buckling load (pinned-pinned column)

$$\omega_o^2 = \frac{g}{g} \frac{\pi^4 EI}{\ell^4} \text{ (Fundamental frequency of vibration of column)} , \quad (5)$$

$$= \frac{\omega_o^2}{\Omega^2} (1 - \rho) , \quad b = \frac{\omega_o^2}{\Omega^2} s , \quad s = \frac{S}{P_{Cr}} .$$

The solution domain for Equation (5) is shown graphically in Figure 3. At certain values of a and b, the lateral vibration of the column grows with time and becomes unstable. The unstable condition is indicated by the unshaded regions in Figure 3. The shaded areas represent stable configurations of the column. The lowest unstable configuration of the column occurs at a = 1/4. Assuming some small value of b (i.e., b = 0.2) and further assuming that there is no static load component (P = 0), then S is equal to -0.8 P_{Cr}. Thus, the coefficient of the cosine varying term is less than the static buckling value, P_{Cr} and, therefore, the magnitude of the dynamic buckling load is less than the static buckling load. Thus, the static buckling analysis provides the largest stable cargo based on stability analysis of the cargo load column. A purely dynamic analysis of the cargo load column which does not incorporate stability analysis is presented in the next section.

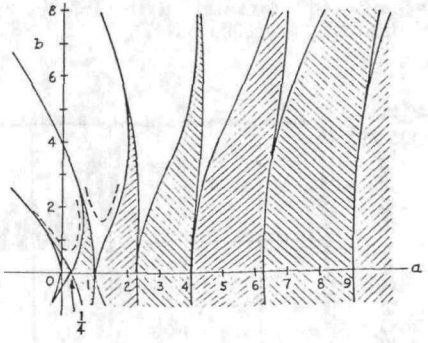


Figure 3. Solution Domain[5] - Equation (4).

Crush Analysis - Lumped Parameter Spring Mass Model

The third type of crush analysis presented in this paper involves a lumped parameter spring mass model of the cargo load column. Such a model is necessarily simplistic and does not incorporate the stability arguments presented in earlier sections of this paper. The configuration of the spring mass model is shown in Figure 4. The model was developed for a unit cross-sectional area cargo load column so that a unit dynamic loading on the RAM package could be evaluated. The crush loading on the RAM package is produced by the longitudinal crush of the

RAM package between the fixed mass and the cargo aft of the RAM package that is moving forward with impact velocity V_o. The pre-impact velocities, V_o, used in this analysis were 44.7 m/s (100 mph). The mechanical characterization of the cargo was accomplished by using a density of 320 kg/m^3 (20 lb/ft^3) (Reference 3) and hysteresis springs which have a loading and unloading modulus of 690 MPa (10^5 psi) with a yield strength of 6.9 MPA (1000 psi).

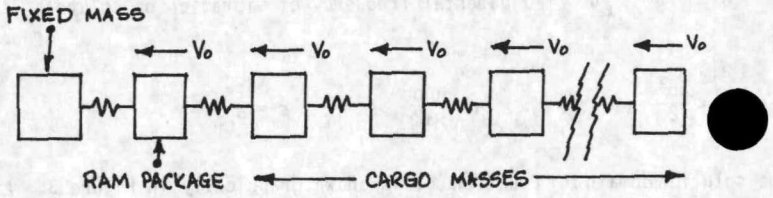


Figure 4. Spring-mass model configuration.

A typical output from the lumped parameter analysis is shown in Figure 5. Figure 5 indicates the force on the RAM package from one of its adjacent masses. Notice that the plateau in Figure 5 represents a force given by the yield strength of the cargo 6.90 MPa (1000 psi) acting on the RAM package. Such a load on the RAM package represents a unit dynamic load of 640 kN (144,000 lbs). The dynamic crush load applied to the actual dimensions of the LAARC package (.14 m^2) represent a total longitudinal crush load of approximately 940 kN (210,000 lbs). It must be noted that the maximum longitudinal load that a package experiences based on the dynamic analysis presented above is a function of the estimated yield strength of the cargo and is essentially independent of the impact velocity once the yielding of the cargo has been reached. Materials with a 150-300 kg/m^3 density typically have a yield strength of 6.895 MPa (1000 psi).

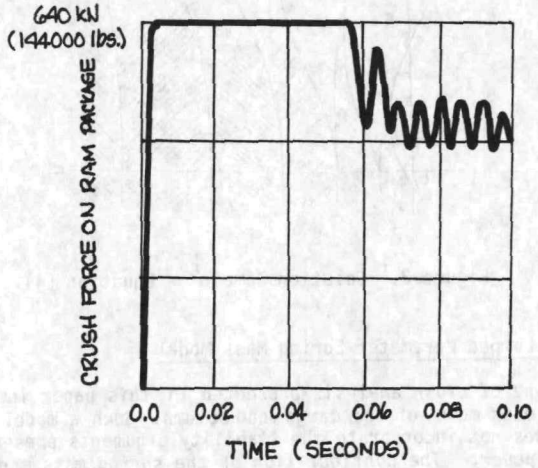


Figure 5. Typical output - lumped parameter analysis.

CONCLUDING REMARKS

This paper has analyzed, in three separate ways, the longitudinal crush load which can be applied to a small hazardous material (RAM) package which is located in a forward cargo position on a standard body or wide-body cargo aircraft. The load analysis was developed using the concept of a cargo load column. The main purpose of the analysis was to more accurately describe the dynamic crush loads which can be applied to a hazardous material (RAM) package positioned in the cargo aircraft as described above. In fact, one goal of the analysis was to evaluate these longitudinal crush loads and see if they were bounded by reasonable magnitudes of force. An attempt to determine such a bound was made by structural stability arguments concerning the cargo load column and the attendant structural disintegration of cargo aircraft in severe aircraft crashes. Another bounding mechanism was to limit the dynamic effect of aft loaded cargo on a forward loaded hazardous material package by allowing only the aft cargo in the volume of the cargo load column to impinge upon the hazardous material package. Using the LAARC package as an example problem, the stability arguments in combination with the relatively small dimensions of the LAARC design produced longitudinal crush loads even lower than the vertical crush 311 kN (70,000 lbs) forces determined in Reference 2. The dynamic loads on the cargo load column which did not incorporate stability arguments produced longitudinal crush loads of 939 kN (211,000 lbs) on the LAARC package which are approximately a factor of three larger than the vertical crush forces of 312 kN (70,000 lbs) determined in Reference 2.

While the disparate analyses presented above cannot yield a definitive upper limit on the longitudinal crush forces that might be experienced by a RAM package in a severe aircraft accident, it is concluded that they do provide a reasonable bound on the problem. The last of these analyses predicted a load on a LAARC-sized package in the 900-1000 kN range and while the analysis still includes several conservatisms; it is felt that these values for longitudinal crush would not be exceeded in a severe aircraft accident. In fact, it is most probable that the stability arguments in the earlier analyses would prevail due to the structural disintegration of the aircraft in severe aircraft accidents.

REFERENCES

1. R. K. Clarke, J. T. Foley, W. F. Hartman and D. W. Larson, Severities of Transporting Accidents, SLA-74-0001, Sandia National Laboratories, Albuquerque, NM (July 1976).

2. J. D. McClure and W. A. Von Riesemann, The Crush Environment for Small Containers Carried by U. S. Commercial Jet Aircraft, SAND76-0533, Sandia National Laboratories, Albuquerque, NM (August 1977).

3. J. D. McClure and W. F. Hartman, An Analysis of the Crush Environment for the Light-Weight Air-Transportable Accident Resistant Container (LAARC), SAND [illegible]0783, TTC-0079, Sandia National Laboratories, Albuquerque, NM, July 1980.

4. Cargo Aircraft and Spacecraft Forward Restraint Criteria, ASD-TR-7630, Aeronautical Systems Division, Air Force Systems Command, Wright-Patterson Air Force Base, December 1977.

5. S. P. Timoshenko and J. M. Gere, Theory of Elastic Stability, Second Edition, McGraw-Hill Book Co., Inc., New York, 1961.

COMPARISON OF POTENTIAL CRUSH AND IMPACT ENVIRONMENTS

C. M. Romander
J. D. Colton
SRI International
Menlo Park, CA 94025 USA

INTRODUCTION AND OBJECTIVE

A test to represent the potential crush environment for radioactive material packages under accident conditions is not currently required in the USA. However, the U.S. Nuclear Regulatory Commission does require an impact test, a 30-foot drop of the package onto an essentially unyielding surface, that results in some protection against crush. The objective of this analysis was to determine the level of protection against crush that is provided by the current qualification test for impact.

The most probable mechanism and also the mechanism that can produce the most severe crush environment occurs when a group or string of packages strikes another object or a barrier. The inertia of the aft packages increases the deflection of the front package beyond the deflection it would undergo in impact (i.e., if the string consisted of only one package). In the analysis, we compare package deflections produced in inertial crush with those produced in the 30-foot drop test.

APPROACH

The current impact test specifies that a package must meet certain acceptance criteria following a drop from 30 feet onto an essentially unyielding surface (barrier stiffness $K_b = \infty$). The resulting 30-mph impact produces a certain deflection in the package. In actual accident conditions, barriers may not be unyielding. Packages can impact compliant barriers at greater than 30 mph and have a deflection that equals that produced in the impact test. The first step in the approach then is to determine, for a range of package characteristics, the combinations of impact velocity and barrier compliance that produce the same deflection that occurs in a 30-mph rigid barrier impact when a single package impacts a range of compliant barriers. The accidents represented by these combinations of velocities and barrier compliance specify the level of protection against impact provided by the 30-foot drop test.

To compare the crush deflections with the impact deflections, we then analyzed the crush response of a string of packages for the identical accident conditions determined above (i.e., same velocity and barrier compliance). Comparison of maximum package deflection in crush with that in impact provides a measure of the level of protection against crush that is provided by the current impact test.

IMPACT ANALYSIS

We first determined the level of protection against impact provided by the 30-foot drop test. That is, we determined the combinations of impact velocities and barrier compliance that produce the same deflections as in the 30-foot drop test. These combinations depend on the barrier force-deflection characteristics

and the package force-deflection characteristics. Figure 1 shows the mathematical models used in the analysis. The barrier is modeled as a single spring with a stiffness K_b during loading and with no rebound during unloading. This simplified, linear, inelastic characteristic approximates the behavior of many crushable barriers ranging from massive concrete to fairly soft barriers.

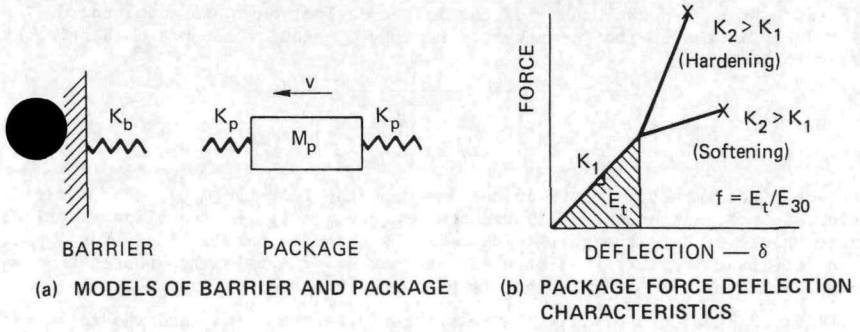


FIGURE 1 MODELS USED IN IMPACT ANALYSIS

The package is modeled as a lumped mass with springs on either end. Each package spring is characterized by the bilinear force-deflection characteristics shown in Figure 1(b). Force and deflection increase linearly with a slope K_1 until a transition point is reached; then the package either softens or hardens. If the package softens, the slope of the posttransition curve, K_2, is less than the slope of the initial loading curve, K_1; if the package hardens, the slope K is greater than the slope K_1. Softening in a package ($K_2 < K_1$) can result from buckling or crushing with decreased resistance once some transition force or deflection is reached. Hardening in a package ($K_2 > K_1$) could result from displacing the free volume inside the package, after which both the package and the contents are compressed at an increased modulus K_2.

When a single package impacts the barrier, only the barrier and the spring on one side of the package deform. For the package force-deflection characteristics shown in Figure 1(b), the deflections indicated by an X correspond to the deflection produced when the package impacts a rigid barrier at 30 mph (30-foot drop test). This package deflection is called δ_{30}.

In comparing different package force-deflection characteristics, one further parameter is useful. The parameter f is the ratio of the energy absorbed by the package at the transition point to the total energy absorbed at the deflection δ_{30}. If K_1 is the slope of the force-deflection relationship before the transition point and δ_t is the package displacement at the transition point, then the parameter f is given by the expression

$$f = \frac{K_1 \delta_t^2}{2Wh}$$

where W is the package weight and h is the drop height of 30 feet. The shaded portion of the package force-deflection characteristics shown in Figure 1(b) represents the fraction of energy absorbed before the transition point.

For most Type B packages, that is, packages that must meet the NRC qualification test requirements, the value of f is probably between 0 and 0.3. This range of values is rather arbitrary, but for packages with energy-absorbing overpacks, it is likely that the transition point will occur early in the response either because the crushable material deforms easily or because the package will buckle after a relatively small deflection.

The results of the analysis can be presented using the parameters f, K_1/K_2, and K_1/K_b. We have shown* that for the impact deflection δ_p to equal the deflection δ_{30} produced in the 30-foot drop test, the impact velocity depends on barrier stiffness as follows

$$v/v_{30} = \sqrt{1 + \frac{K_2}{K_1}\frac{K_1}{K_b}\left[\left(\frac{K_1}{K_2} - 1\right)f + 1\right]} \qquad (1)$$

where v_{30} = the impact velocity in the 30-foot drop test (30 mph). For a rigid barrier, $K_b = \infty$, and equation (1) reduces to $v/v_{30} = 1$. For compliant barriers (finite values of K_b), the ratio v/v_{30} will be greater than 1. Thus, for impact into a compliant barrier, a package must be moving at a velocity greater than v_{30} to produce the same deflection that is produced in a 30-foot drop test.

Figure 2 shows graphically the relationship between v/v_{30} and the ratio K_1/K_b for two values of f and for a range of values of K_1/K_2 and K_1/K_b. This figure summarizes the level of protection against impact provided by the 30-foot drop test. Note that maximum velocity of 60 mph ($v/v_{30} = 2$) is considered. Two features of these curves are of interest. First, in all cases, as the stiffness of the barrier decreases relative to that of the package (K_1/K_b increases), the velocity required to produce deflection δ_{30} increases (v/v_{30} increases). Second, if the percentage of energy absorbed by the package before the transition point is small (f is small) the effect of package posttransition force-deflection behavior is an important parameter. As the value of f increases, the influence of package posttransition behavi decreases (i.e., the curves are grouped closer together).

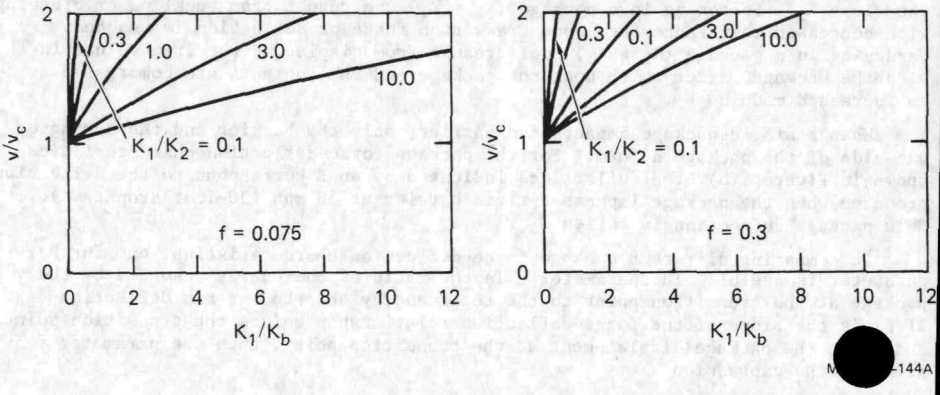


FIGURE 2 COMBINATIONS OF v AND K_b THAT PRODUCE THE SAME DEFLECTION IN IMPACT AS IN 30-FOOT DROP TEST

*J. D. Colton and C. M. Romander, "Potential Crush Loading of Radioactive Material Packages in Highway, Rail, and Marine Accidents," NUREG/CR-1588 (August 1980).

We can draw several conclusions about impact of a package into a compliant barrier and about the current level of protection afforded by the impact test. First, for packages that harden after they reach their transition displacement ($K_1/K_2 < 1$), a small decrease in barrier stiffness results in a large increase in the velocity that produces the displacement produced in a 30-foot drop test (δ_{30}). For example, consider a hard package (large K_1), such as a stiff steel structure impacting a stiff, but not rigid, barrier. Even if the barrier is made of concrete, the ratio K_1/K_b may be as large as 2. In this case, the velocity ratio v/v_{30} that produces the deflection δ_{30} may be as large as 2. Thus, the current level of protection afforded by the 30-foot drop test for hard packages that harden as they deform includes impact velocities as great as 60 mph ($v/v_{30} = 2$).

For packages that soften after reaching the transition point ($K_1/K_2 \geq 1$), large decreases in barrier stiffness are required for small changes in impact velocity. For example, consider f = 0.3 (a typical value for many Type B packages), packages that soften after transition ($K_1/K_2 \approx 3$), and soft barriers ($K_1/K_b > 2$). For this case, for the level of protection provided by the impact test, expressed in terms of velocity, to increase from about 42 mph ($v/v_{30} = 1.4$) to 60 mph requires that K_1/K_b increases from 2 to 6.

For packages that soften extremely as they deform ($K_1/K_b > 10$),* the current level of protection of the impact test is close to 30 mph for impacts into any barrier of interest.

CRUSH ANALYSIS

Current Level of Protection Against Crush

The next step in the analysis is to determine the crush response of a string of packages striking a compliant barrier. Figure 3 shows the barrier and package model. The barrier model is the same as in the impact analysis. The string of packages is made up of four identical packages each modeled as a lumped mass with a spring on either end. The same ranges of values of f, K_1/K_2, and K_1/K_b used in the impact analysis are used in the crush analysis. Only four packages are used in the analysis because additional packages in a string of packages do not increase the deflection of the most severely deformed package.† The governing equations were integrated numerically.

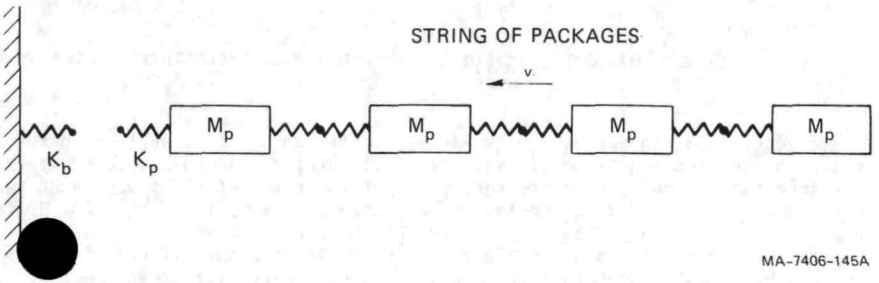


FIGURE 3 MODELS USED IN CRUSH ANALYSIS

*Probably a small fraction of Type B packages fall into this category.

†This phenomenon also occurs for a compliant barrier.

The accident conditions for a string of packages are taken as the same as those for a single package. Therefore, we calculate crush deflection for the same combinations of velocities and barrier stiffness for which we previously determined impact deflections, as summarized in Figure 2.

The results of these calculations are shown in Figure 4 as plots of $\delta_{crush}/\delta_{30}$ versus K_1/K_b. The point on each curve at $K_1/K_b = 0$ (rigid barrier) shows the ratio of crush deflection to the deflection δ_{30} resulting from impact with a rigid barrier at 30 mph. As the value of K_1/K_b increases (decreasing barrier stiffness), the impact velocity increases until, at the point where each curve ends, the impact velocity is 60 mph ($v/v_c = 2.0$). (The tick marks on the curves in Figure 4 are at velocities of 40, 50, and 60 mph.) This comparison shows the effects of the parameters f, package force-deflection characteristic K_1/K_2, and relative barrier stiffness K_1/K_b.

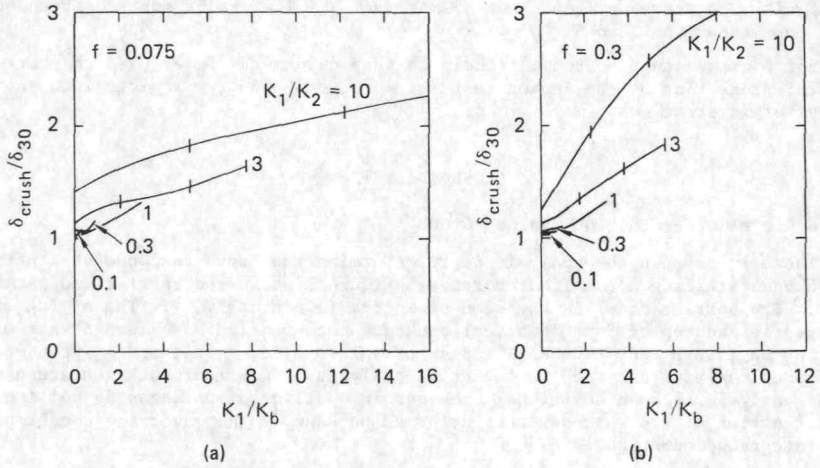


FIGURE 4 RATIO OF CRUSH DEFLECTIONS TO IMPACT DEFLECTIONS

In general, crush deflection is always greater than impact deflection for any particular accident condition (impact velocity, barrier stiffness, and package). It is important to note, however, that for certain ranges of package parameters, the difference between crush and impact deflection is small (less than 20% difference). For the general class of packages that harden as they deform (K_1/K_2 < 1), crush deflection is within about 20% of impact deflection for all velocities up to 60 mph. This result suggests that if a 20% difference between crush and impact deflection is acceptable, then the current impact qualification test provides an adequate level of protection for both crush and impact response of hardening packages in accidents up to 60 mph. It is likely that most current packages are of the hardening type and therefore have a level of protection against crush comparable to that against impact.

For packages that soften following the transition point ($K_1/K_2 > 1$), crush deflections can be significantly greater than impact deflections in a given accident. Even if crush deflection is allowed to exceed impact deflection by 20%, on

for slightly softening packages ($K_1/K_2 \approx 3$) can the impact velocity be as great as 40 mph. For packages with greater softening ($K_1/K_2 > 5$), the difference between crush deflection and impact deflection can be large. For these packages, the current impact test does not provide a level of protection against crush that is comparable to the level of protection against impact.

To ensure an acceptable level of protection against crush for softening packages requires that either (1) a qualification test for crush be required or (2) shipping procedures be modified so that crush deflection is limited to a level that is comparable to that produced in the current impact test. Such shipping procedures might include use of tie-downs or placement of packages to ensure gaps between them. To determine such requirements quantitatively, we investigated the effects of gaps between packages on crush response.

The Effect of Gaps Between Packages on Crush Response

The presence of gaps between packages is to be expected in any shipment of similar packages. While these gaps might be a small fraction of the size of the package, they can play an important role in the crush response of a string of packages. The purpose of this analysis is to determine if gaps between packages can reduce package deflection caused by inertial crush.

We begin with an example in which we calculate the response of a string of four softening packages ($K_1/K_2 = 10$, $f = 0.3$) with gaps between them when the string strikes a rigid barrier at 30 mph. Figure 4(b) shows that, for this case, the ratio of crush deflection to impact deflection is 1.25. As the gap between packages increases from 0% of the package length to about 2.4% of the package length, the crush deflection is reduced to a value that is only 5% greater than the impact deflection resulting from the same accident conditions. As the gap increases beyond the optimum gap of 2.4%, the crush deflection increases. This increase is attributed to the rebound of package 1 from the barrier. For initial gaps greater than 2.4% of the package length, package 1 impacts the barrier and rebounds before package 2 impacts package 1. Package 2 then impacts package 1, imparting a velocity to package 1. Package 1 then impacts the barrier again and undergoes additional deflection.

We now include the effects of a compliant barrier in the gap analysis. We calculated the crush response of a string of four softening packages as they strike barriers of various stiffnesses at velocities that would produce a deflection equal to δ_{30} in a single package impact into the same barrier. We analyzed extreme softening packages ($K_1/K_2 = 10$) because the crush deflection of such packages is the largest of those considered. In these calculations, we chose $f = 0.3$ and the gap between packages was varied from 0% to 20%.

Figure 5 shows the ratio of crush deflection to impact deflection as a function of barrier stiffness (K_1/K_b). The effects of five different gaps are shown. The case with no gaps (labeled 0% in the figure) is the same as shown in Figure 4(b) for $K_1/K_b = 10$. For this case, as K_1/K_b increases (decreasing barrier stiffness), the ratio of crush deflection to impact deflection increases monotonically. The largest velocity of interest, 60 mph, occurs at $K_1/K_b = 8$.

When gaps are present, the ratio of crush deflection to impact deflection initially decreases with increasing K_1/K_b (decreasing barrier stiffness). The optimum values of gap, of barrier stiffness, and of velocity are the combination of these parameters for which the ratio of crush deflection to impact deflection is a minimum. For a rigid barrier ($K_1/K_b = 0$) and a velocity of 30 mph, the optimum gap is 2.4%. For a critical gap of 10%, the corresponding barrier stiffness is $K_1/K_b = 2.3$ and the impact velocity is about 40 mph. The critical gap increases as velocity increases because, for minimum crush to occur, the time between subsequent impacts in a string of packages with gaps must be approximately constant.

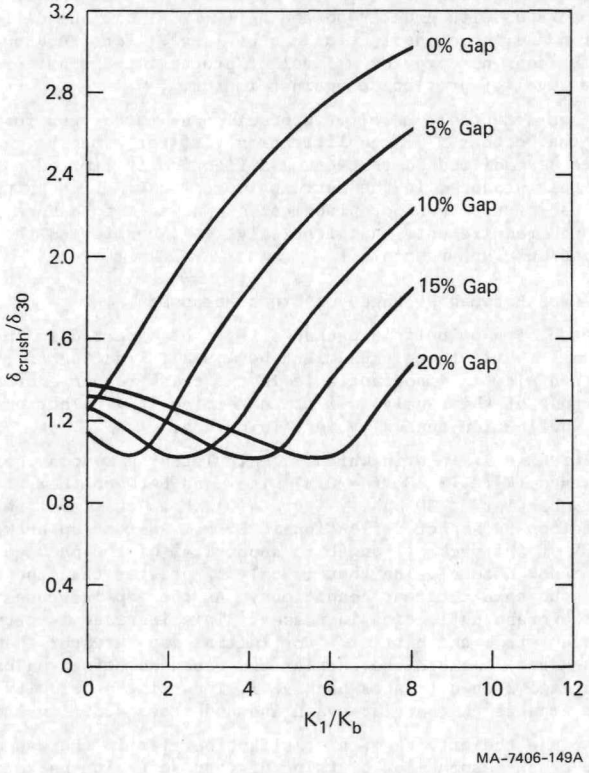


FIGURE 5 CRUSH DEFLECTION VERSUS BARRIER STIFFNESS FOR VARIOUS SIZE GAPS

Several conclusions can be drawn from the results presented in Figure 5. The most significant conclusion is that there is a dramatic reduction by up to a facto of 3 in crush deflection as a result of gaps between packages. For gaps of 20% of the package length and for velocities approaching 60 mph (and therefore for K_1/K_b < 8), the crush deflection exceeds impact deflection by no more than about 40%. For a range of barrier stiffnesses corresponding to K_1/K_b between 3 and 7 (and cor responding velocities between 43 mph and 57 mph), the crush deflection exceeds the impact deflection by 20% or less.

The crush deflections presented in Figure 5 represent an upper bound [redacted]he crush deflections on softening packages for several reasons. First, the deflectio shown are for extreme softening packages (K_1/K_2 = 10); for K_1/K_2 < 10 crush deflec tions are less than those shown in Figure 5. Second, in these calculations possib instabilities of a string of packages that would reduce crush deflections have not been taken into account. Actual packages have an initial misalignment or become misaligned from nonuniform crush, thus reducing crush deflections. This effect is particularly significant for extreme softening packages impacting soft barriers because the large displacements developed increase the likelihood of an instabilit Finally, the analysis ignores the effects of any tie-downs that would reduce the crush deflections.

DISCUSSION AND CONCLUSIONS

We have shown that for hardening packages impacting real barriers, the level of protection against crush provided by the current 30-foot drop test is adequate for velocities up to 60 mph. For soft packages, the current level of protection does not extend to such high impact velocities; for extreme softening packages, the velocity may be as low as 30 mph. However, the level of protection for soft packages increases significantly when there are relatively small gaps between packages. In the shipment of groups of packages some gaps between packages already exist. Gaps may result from loose tolerances in the packing arrangement or from the tie-down configuration. Finally, it is likely that the relative number of softening packages is small; most Type B package designs harden with increasing [redacted]tion.

From our analyses, we draw the following conclusions:

(1) The level of protection against crush provided by the current 30-foot drop test is comparable to the level of protection against impact for most present shipments. This may not be the case for packages with extreme softening.

(2) If the level of protection against crush of extreme softening packages must be increased, it can be accomplished by operational procedures to ensure gaps between packages.

(3) An additional qualification test is not required to ensure a level of protection against crush that is comparable to the current level of protection against impact.

DISCUSSION

Question by G. Malkoske, Canada: During the last IAEA advisory group meeting, concern was expressed about the crush that light weight packages might experience. I would be interested in packages with a low mass/volume ratio (low density packages). Have you considered the effects of volume of the package in your studies?

Answer: The effects of volume of the packages were taken into account through the stiffness parameters that describe the package. Softening type packages tend to have low density (large volume) and therefore are crushed more than hardening type packages (high density, low volume). Of particular concern for small, low density packages, but not addressed in this study, is mixing large, hard cargo with tne small packages. In this case, severe crush environments can result.

Question by J. D. McClure, USA: Has your sponsor speculated as to [redacted]her existing packagings are "hardening" or "softening?"

Answer: Yes, they speculate that most type B packages are of the hardening type.

Question by M. Brown, United Kingdom: I am not clear as to how the packages in your analysis behaved when unloaded. Could you clarify this?

Answer: The packages were unloaded along a very steep slope. This represents permanent deformation in the packages.

SESSION XXII
EMERGENCY PREPAREDNESS AND RESPONSE

THE NEED FOR EMERGENCY PREPAREDNESS IN TRANSPORT OF RADIOACTIVE MATERIALS

B.C. Bernardo
Division of Nuclear Safety and Environmental Protection
International Atomic Energy Agency

Introduction -

An estimated 8 million packages of radioactive materials are transported each year throughout the world, either by air, land or sea. The radioactivity in those packages varies from negligible to very large. In every case, however, the transportation of the material, which may be in the form of uranium ores, luminous dials for clocks or watches, smoke detectors, medical isotopes, radioactive waste or irradiated nuclear fuel, is essential to some operation or activity. All such shipments have to meet safety requirements based on regulations which are designed to provide protection of the public and transport workers. Compliance with the regulatory requirements insures a very low probability of occurrence of any release of radioactive material during transport that could have unacceptable severe radiological consequences. It cannot be completely excluded, however, that a package could, because of improper assembly or for some other reason, fail to retain its containment or shielding in an accident. It is only, therefore, prudent to prepare in advance, arrangements for dealing with emergency situations during transport.

Transportation accidents involving packages of radioactive material, present potential risk for radiological exposure to transport workers and to members of the general public although the expected values of the annual radiological impact from such potential exposure are very small.

The Need for Preparedness -

The reported recent transport accidents indicate that there is a real need for emergency preparedness. In March 1977, in a train wreck in North Carolina, several cars left the track and a variety of freight was scattered around the scene including four Type A cylinders containing Uranium Hexaflouride source material which was enroute for enrichment. A fire broke out and the report indicated that one uranium hexafluoride cylinder had possibly been breached. It was also reported that an individual who had inadvertently fallen, put his hand in the leaking material from the ruptured cylinder. He was, however, later found not to be contaminated. There was no loss of containment integrity and no radiological consequences that resulted from this incident.

Again in March 1979, another incident was reported involving uranium ore concentrates that were being transported by a tractor/trailer in Kansas. The accident occurred when the tractor/trailer left the roadway and overturned in attempting to return to the portion of the highway. It was raining at the time of the accident which added to the complexity of the incident. The steel drums were ejected through the trailer roof and 22 sustained damage sufficient to

result in leakage of the radioactive contents. The steel shipping drums were scattered over a widespread area resulting in the closure of the portion of the highway. The leaking containers were covered with plastic sheeting to prevent waterborne dispersal. There was no person that received any significant contamination as a result of this incident although complete clean-up took about two weeks. It is also to be noted that the health hazard related to this accident is not significant with regard to the radioactivity of the uranium, rather the health hazard was related to the chemical toxicity of inhaled or injested uranium and the resultant kidney damage. Cost of clean-up was, however, significant.

Again, in November 1979, a Canadian Pacific Railway train was derailed in an industrial area near the centre of Toronto. Although this incident involved only hazardous material other than radioactivity, the event triggered to what wa[redacted] result in the largest peacetime evacuation in recorded history. The incid[redacted] involved a propane car which blew to pieces and this was followed by severa[redacted] other explosions which resulted in 10 other propane cars being ruptured and burned. This also threatened the tank car that was containing liquid chlorine as it began to leak. It was the possibility of the chlorine tank car exploding that necessitated the evacuation.

Transport accidents have not been limited to train and truck alone. In October 1974, an air incident occurred from Gatwick to Dusseldorf in Germany. This involved a package containing Yttrium 90 which was crushed by a vehicle while being loaded at Gatwick Airport. In spite of this, the ramp attendant loaded the package on a cargo aircraft and when it landed at Dusseldorf, it was noted that the package had leaked and the liquid was spilt on the floor of the aircraft. The aircraft was ordered to return to England, where decontamination was carried out. There were no personal injuries attributed to this incident, although decontamination took place and some parts of the aircraft flooring replaced. It may be said that the impact of the vehicle followed by poor handling practice was the cause of the incident.

One incident at sea occurred in December of 1975 in the North Atlantic during a severe storm. A lifeboat which was on board a ship struck a Type B(U) package containing 10 kilo curies of Cobalt 60 in special form capsules. In the incident, the therapy device and skid got loose and the outside packaging was lost overboard. Another box containing accessory equipment was also smashed. When the ship was put into harbour at Savannah, Georgia a radiation check indicated that there was no increase in radiation level. No personal injuries were recorded and one may assess the incident as no loss of integrity, no increase in radiation level and the cause of the incident was tie-down failure in rough seas.

While the above incidents did not give rise to any significant radiological consequences the possibility of transport accidents happening is rising as the traffic increases and the need to prepare is therefore recognized.

Emergency Plans on National Basis -

Many national authorities have established detailed plans for the rep[redacted]g and handling of emergency situations during the transport of hazardous mate[redacted]s. In many cases these plans are tied up into the national emergency schemes. Advice on emergency planning for handling accidents is provided in existing Agency documents. A manual is to be published soon which will specifically cover accidents involving radioactive materials during transport. Some countries have prepared emergency procedures to cope with transportation accidents involving not only hazardous materials, but also radioactive materials.

In these procedures, organizations such as police and fire departments are informed well in advance of the basic actions required and these organizations have access to radiation experts that are equipped to assist at the scene of the accident. Most national emergency procedures are organized on a two stage basis, the first stage being to have the place of the accident investigated by a radiation expert equipped with radiation survey instruments that are capable of identifying any hazard present and the second stage to provide trained personnel and equipment to remove unshielded radiation sources and contamination should this be present as a result of the accident.

Accidents resulting in the escape of radioactive substances into the environment or in the irradiation of people and other living things may occur due t[redacted]eakage of packages involved in a crash, wreck or fire or due to leakage from [redacted]s causes. Emergency plans should be prepared well in advance and be sufficiently broad to cover unforeseen or improbable accidents.

In case of transport accidents, the importance of prompt notification of the authorities who can give advice and assistance and those who have been assigned responsibilities for resolving radiological consequences, cannot be overemphasized. If possible, the first-on-the-scene responders should not spend valuable time completing these notifications, rather the more important information, pertinent information, should be relayed from the accident scene to a despatcher at the home station, who in turn will make the notifications required for regulatory or information purposes after needed assistance has been arranged. In the United States, the area personnel, if not injured, will probably notify the responsible company personnel, who in turn, will notify the Department of Transportation, the Shipper and the Nuclear Regulatory Commission. The Department of Energy's (DOE) Radiological Assistance Programme provides for the department radiological advice and assistance to minimize personal injury, minimize loss of property, cope with radiological hazards and protect the public health and safety. For implementation, the DOE has divided the United States into 8 regions, each with a regional coordinating office.

The US Nuclear Regulatory Commission as the Federal Agency designated to enforce regulations concerning nuclear material can provide valuable assistance in the event of a radiological incident. The Inter-Agency Radiological Assistance Plan, otherwise known as the IRAP, was developed as a means whereby the participating Federal Agencies may coordinate their activities with those of state and local health, police, fire and civil defence agencies to provide rapid and effective assistance in handling radiological incidents. These agencies are, the Department of Agriculture; the Department of Commerce; the Department of Defence; the Department of Energy; the Department of Health, Education and Welfare; the Department of Labour; the Department of Transportation; the Defence Civil Preparedness Agency; the Environmental Protection Agency; the Inter-State Commerce Commission; the National Aeronautics and Space Administration; the Nuclear Regulatory Commission; the Postal Service and the DOE is the coordinating agency. The Joint Nuclear Accident Coordinating Centre, JNACC, is a joint operation of the Defence Nuclear Agency and the Department of Energy with head[redacted]ers located in New Mexico. Its mission, is to provide a centralized agency [redacted]changing and maintaining information concerned with radiological assistance ca[redacted]ilities and coordinating assistance activities in connection with accidents involving radioactive material.

The Radiation Emergency Assistance Centre Training Site, REACTS, is operated by the Oak Ridge Associated Universities of the US Department of Energy and is located in Oak Ridge, Tennessee. It provides the most modern multi-purpose facilities available for handling radiation emergencies. Sometimes, the transport company itself, would have its own emergency plan. In the case of the incident in

Mississauga, Canada it was noted that as soon as the Canadian Pacific Railway Despatchers Office was notified of the derailment through the train radio system, the C.P. Emergency Plan was immediately implemented. Its technical experts were alerted through fan-out and the technical crews were instructed to converge at the scene of the accident. The procedure for a chemical emergency such as this one included calling local hospitals, police, fire departments, the Environmental Protection Service of Environment, Canada, Transportation Emergency Assistance Plan, TEAP, and the Canadian Representative of the Bureau of Explosives of the Association of American Railways. Even the local and regional police have their own disaster plan and these helped a lot in putting the emergency under control. Records will show that the well prepared plan and the well documented instructions all add to the success of the evacuation that followed.

The IAEA Plans and Programmes -

Since 1959, the IAEA has had an action plan by which it will, upon request, arrange to provide Technical Assistance to Member States following an accident involving radioactive materials. The plan has been reviewed at intervals by a consultant. Information on the type of assistance that might be made available, as well as the individual or office to be contacted for request for assistance, is issued periodically in a report, WP.35 - "Mutual Emergency Assistance for Radiation Accidents".

The World Health Organization (WHO), United Nations Food and Agricultural Organization (FAO) and the International Labour Organization (ILO) have participated in the recent revision to this publication. It is now being revised with the additional participation of the Office of the United Nations Disaster Relief Coordinator (UNDRO), with which the IAEA has entered into an agreement to coordinate and cooperate in disaster-related activities.

The IAEA maintains the capability through a Duty Officer Roster to have a Senior Technical Staff available in the event of any request for assistance. Usually, one of the 20 people involved in the Duty Officers' list is available at the emergency control post within one hour of the initial request. Also special arrangements have been made internally with Radio Austria so that telex or cable requests for assistance will receive immediate attention. The programme also includes the capability for the IAEA to send a small group of observers or consultants to the scene of the emergency when required. Kits are available with radiation survey instruments and protective clothing suitable to support the IAEA group in any radiological situation. In addition, the facilities of the Agency's Laboratory can arrange for the radiochemical analysis of environmental samples or for bio-assay.

Each Duty Officer, has in his possession, a copy of the IAEA Emergency Assistance Handbook containing approximately 50 pages of detailed instructions or reference material to assist in handling an emergency request. This handbook describes the internal arrangements by which the Agency would render assistance and also defines the responsibility of each staff member who would be involved in the response.

In order to achieve rapid and accurate response, simplicity has been the main concern in preparing the operating procedures. In the 18 year history of the IAEA Emergency Assistance Plan, there have been only two requests for medical assistance and one for radiological assistance in case the Sky Lab would fall in the territory of one of the Member States. It is necessary to test the programme and train the Duty Officers in test exercises. The internal arrangement is shown in the attached schematic flow chart. A request for assistance could come to the

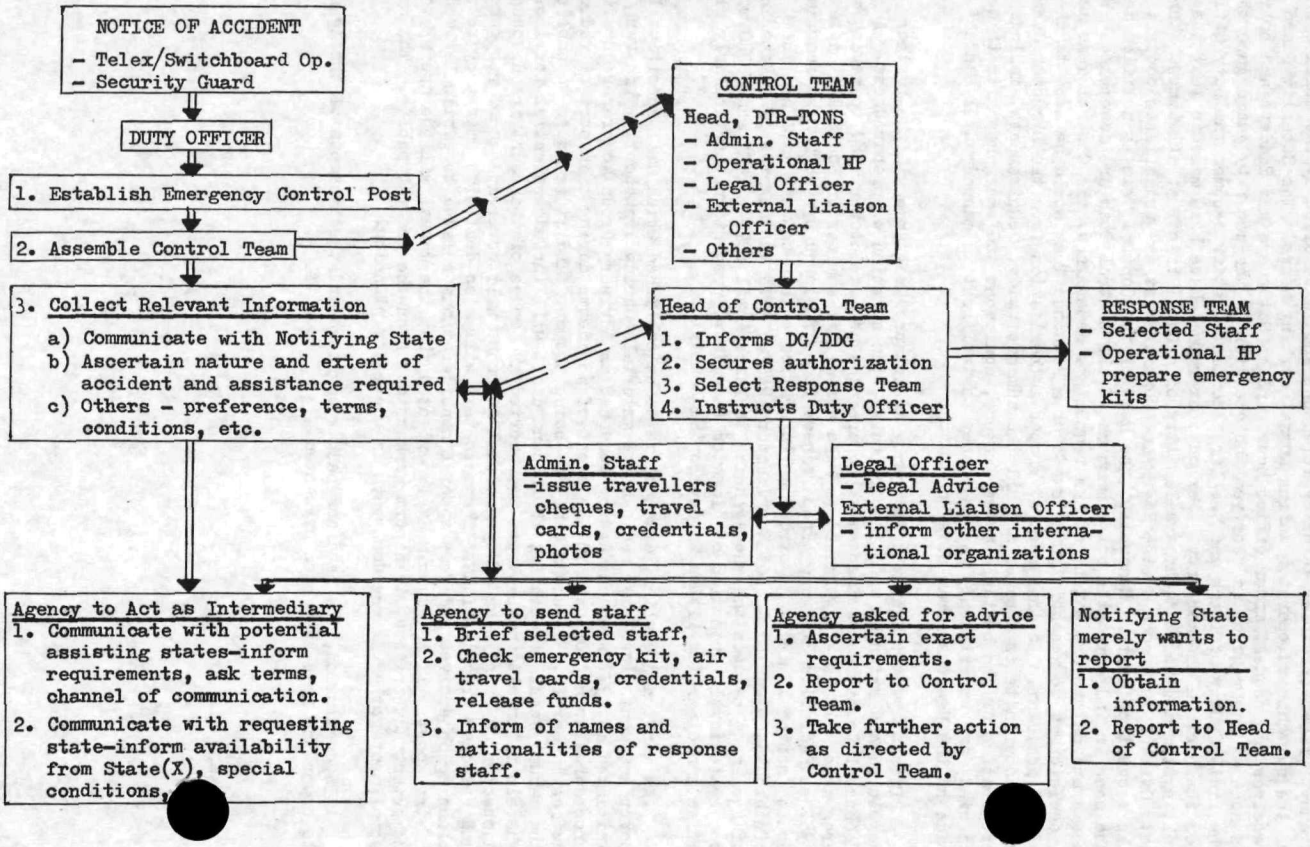
Agency, either by cable, telex or telephone and is received by the telex operator or the switchboard operator during normal working hours. The Security Guard receives the request after office hours and on Saturdays and Sundays and holidays. As soon as the request is received the operator or the security guard goes through the list of Duty Officers and the first contacted staff becomes the Duty Officer for the particular emergency. Periodic exercises have been made to test the on-call system and to find the availability of a Duty Officer at all times. The Duty Officer then goes to the Emergency Control Post and establishes contact with the requesting state. Meanwhile, the Head of the Control Team is notified and if the need is indicated the other members of the Control Team are assembled. When the exact nature and extent of the request is determined, the Control Team makes a decision as to whether Agency staff or a Response Team would be sent to the site of [redacted] accident. In this regard, proper authorization from the Director General or [redacted] designate is secured and all the administrative requirements, including the preparation of the portable survey meters, equipment and emergency kits to be carried, are undertaken. The Emergency Control Post is manned until contact is made by the Response Team after despatch.

Presently, the Agency's Emergency Assistance Programme will be further developed and extended and, in particular, the question of preparedness. A Technical Committee on Emergency Preparedness met in October 1980 to review the programme. It is also planned to hold a regional seminar in South East Asia in 1981 on Radiation Emergency Preparedness to be followed by another seminar in South America in 1982. A strong collaboration exists between IAEA and the Office of the U.N. Disaster Relief Coordinator (UNDRO) with the signing of a Memorandum of Agreement between the two Agencies in March/April 1977. IAEA and UNDRO will cooperate in the promotion of disaster (accident) prevention, particularly as it affects nuclear facilities in developing countries.

As mentioned earlier, a manual is to be published which will specifically cover accidents involving radioactive materials during transport. This was initiated in 1977 when the Agency convened an Advisory Group to prepare draft procedures to be followed in the event of accidents during transport of radioactive materials. A second draft is being prepared and it is hoped to publish the manual soon. Presently, the second draft will include potential for incidents wherein materials and packages transported and modes of transport are discussed. A chapter on principles of operational response will cover stages of responses and response organizations. The main text will also deal with emergency response planning objectives and some special considerations related to particular modes of transport will be given. It is expected that the manual will be useful to competent authorities, transport workers and even the general public. The manual compliments other Agency documents in emergency planning.

In conclusion, the need has been recognized and the attempts of the Agency to cover the need have been presented.

IAEA RADIATION EMERGENCY ASSISTANCE PROCEDURES



References -

1. SS.6 - Regulations for the Safe Transport of Radioactive Materials 1973 Revised Edition (As amended)

2. SS.37 - Advisory Material for the Application of the IAEA Transport Regulations

3. Response to Radioactive Materials Transport Accidents DOT/RSPA/MTB-79/8 Interim Edition

4. Mississauga Train Derailment and Subsequent Evacuation by Denis Amyot, Regional Director, Emergency Planning Canada, for Ontario

5. IAEA-TECDOC-225 - Planning for Off-Site Response to Radiation Accidents in Nuclear Facilities

6. IAEA Bulletin - Volume 21, No.6, December 1979

7. NUREG 0170 - Vol. 1 - Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes.

EVALUATING DANGEROUS GOODS EMERGENCY RESPONSE WITH TIME/LOSS ANALYSES

Elwood T. Driver and Ludwig Benner, Jr.
National Transportation Safety Board

The National Transportation Safety Board is an independent agency of the United States of America. The Safety Board has the responsibility to investigate accident and report publicly on the facts, conditions and circumstances of each accident investigated. The Safety Board also issues periodic reports recommending action and advocating meaningful responses to reduce the likelihood of recurrence of transportation accidents similar to those investigated. In the hazardous materials safety field, in addition to investigating accidents involving hazardous materials, the Safety Board has the duty to evaluate the adequacy of safeguards and procedures concerning the transportation of hazardous materials and the performance of other government agencies charged with assuring the safe transportation of such materials.

These duties created a problem for the Safety Board. How should the adequacy of safeguards and procedures and the performance of other government agencies during emergency responses be evaluated? What criteria should be used for judging the adequacy of the performance observed?

A transportation accident involving dangerous goods occurs. The accident creates an emergency because the dangerous goods have escaped from their containers or they threaten to do so. Firefighters, law enforcement officials, technical experts, and others respond to the emergency. After the emergency has run its course, how should the response be investigated and analyzed to determine if the performance of those who responded might be improved?

When the Safety Board was created, no commonly used method for evaluating emergency response performance was available. No generally accepted evaluation principles were being applied to emergency response performance. The evaluations consisted of small groups of participants getting together and subjectively critiquing their own performance. However, these informal methods were highly subjective, they were not generally circulated, and they showed no promise for producing technically consistent and objective evaluations.

To meet the Safety Board's need for technically consistent and objective evaluation standards, the Board set out to find a way to meet its needs. The result has been the development of a method based on analysis of losses versus time during the emergency. This time/loss analysis approach provides a general way to examine the results achieved during responses to dangerous goods emergencies, and to identify safety emergency response problems. The method is also useful for evaluating safeguard performance.

EVALUATION CONCEPTS

Two common elements have been observed in past dangerous goods transportation emergencies. These elements are "time" and "loss". Time is uniform, measurable, and always present. Every emergency has some finite duration. It begins and it ends. The duration of an emergency can be measured with customary time measurement. For analytical purposes, the beginning of an emergency should be identified with the beginning of the accident or incident which created the emergency. It should

not be identified with the arrival at the scene of the first responding emergency personnel.

For convenience, we will call the beginning of a dangerous goods emergency "time zero" (t_o) and the end of the emergency "time end" $t_{(end)}$. A second ingredient present in all emergencies is "loss." Loss if viewed as the full-range of injury to people or other animated objects, and damage to property or inanimate objects, plus the economic consequences of system disruption. Loss also appears to be quantifiable. Direct losses and costs are often reported, but indirect costs of losses are not customarily collected or reported. To illustrate the evaluation method, use numbers to count fatal or other injuries, and use monetary units to count property loss.

During an emergency, losses change with time. Sometimes losses increase and sometimes they are reduced with time. For example, as a fire continues to consume the cargo in a vessel, cumulative losses increase with time. On the other hand, as firefighters gain control of the fire, the rate of increase declines, and when the fire has been extinguished, the direct fire losses stop increasing. On the other hand, as injured persons recover from accidental exposures to dangerous goods releases, the ultimate loss level from the emergency may actually decrease with time.

A time loss plot helps to visualize these outcomes. Loss during an emergency begins at or after t_o and typically increases in some manner as time passes during an emergency. At the end of the emergency some total loss will have occurred. This total loss is the outcome of the emergency. Every emergency will eventually run its course and come to an end, even without intervention. Fire consumes all accessible combustibles, and goes out. Gas is released from a container, mingles with the air and drifts into the atmosphere. Dust and debris are carried aloft temporarily, but then settle to earth after an explosion. When no one intervenes, this natural course of events will have produced loss.

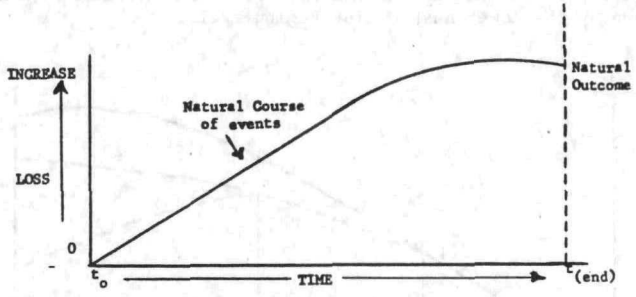


Figure 1. Time/Loss Analysis Plot.

On the other hand, when some intervenes in an emergency, the normal course of events might be changed. Intervenors rarely arrive until after t_o. This can be shown as in Figure 2 (next page).

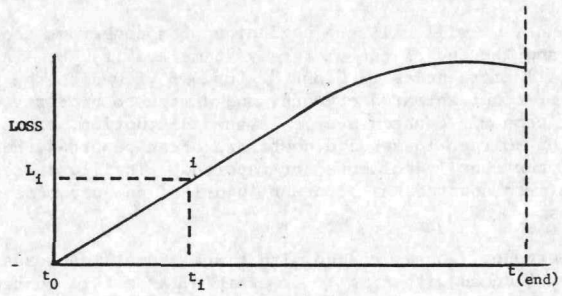


Figure 2. T/LA Plot: Intervenor arrives.

The first-arriving firefighter or police officer is faced with loss that has already occurred by time t_i. The only losses that can be controlled after arrival are those which have not yet occurred. Intervention should change the course of events and natural outcome. Ideally, intervention will reduce the losses that would have occurred naturally.

Ideally, each intervenor will contribute some net reduction in the loss.

The desired relationships can be shown in Figure 3. In that Figure, the solid line represents the net losses that would have occurred without any intervention. By intervening in that course of events at time t_1, intervenor I_1 reduced the losse in our illustration. The change in losses over time is shown by the dashed line beginning at I_1 (Figure 3). If a second party subsequently intervenes, say at time t_2, that intervention should further reduce the already reduced harm expected to occur, as shown by the lower dashed line beginning I_2.

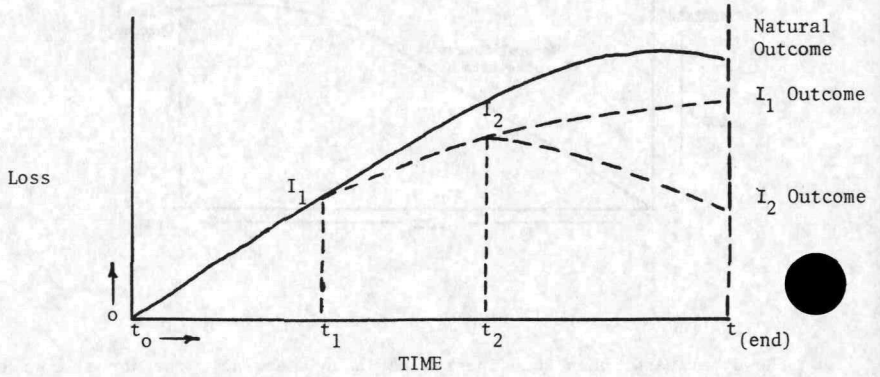


Figure 3. Emergency Time Loss Plot With Intervention.

The point of intervention by each intervenor constitutes a "triple point" The slope of the "natural" time/loss analysis curve can be increased, unchanged or decreased following the intervention. This can be seen in Figure 4.

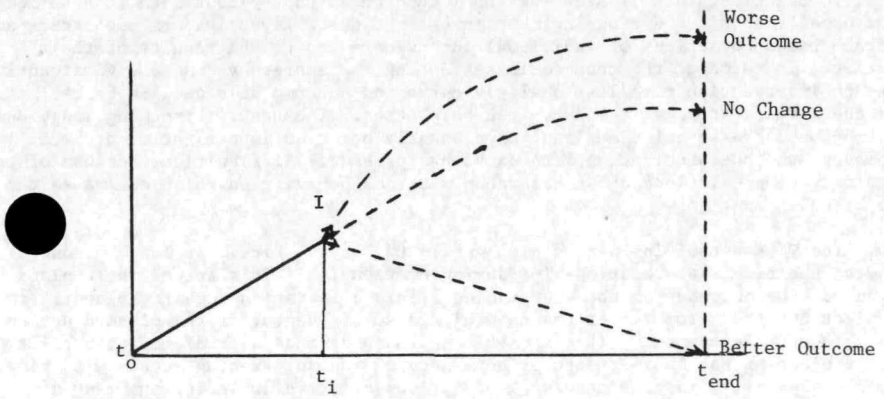


Figure 4. T/LA Plot "Triple Point"

If the response aggravates the net losses, as when responding firefighters or other officials are injured without offsetting gains, the response must be judged ineffective. When the loss curve remains unchanged after the intervention, it must be judged indifferent. When the loss curve declines after intervention below the "natural" curve, the response must be judged successful. The amounts of change in loss measure the degree of success or additional loss produced by intervention.

THE TIME/LOSS ANALYSIS

Based on these concepts, a time/loss analysis method has been developed to implement the approach. The method consists of tracking intervenors' actions, the times they acted, the actual losses, and the times they occurred. Then the predicted natural losses during the time t_o to t_{end} during the course of the emergency are estimated. These data are then displayed to produce a time/loss analysis (T/LA) display in the graphic format shown in Figures 5 and 6. The events supporting the analysis are all the response and loss events that occur between t_o and t_{end}.

This method requires investigators to identify the time when losses occurred, to measure or estimate the losses, and to relate these losses to the actions taken by response personnel during the emergency.

The method can be expanded to evaluate performance of safeguards and procedures, too. Control of the course of events before arrival of the first intervenors is dependent on (1) the safeguards and procedures designed and built into the system operating in the system at the time of an incident, and (2) the adaptive reactions of exposed persons and properties to the threats posed by the dangerous goods. Thus, the time/loss curve between t_o and t_1 on Figure 5 can show the performance of these "built-in" safeguards and procedures, and exposed persons who reacted to the emergency before t_1. For example, a pressure relief device or installation on a container may delay a violent rupture for intervenors to take successful preventive action. Alternatively, the small quantity or non-pressurized form of a material may also delay or prevent violent rupture before intervenors can act.

The T/LA method influences investigative tasks, planning, training, and operational objectives during emergency responses.

In practice, this T/LA evaluation method requires accident data in a format not usually acquired during accident investigations. Investigators must track and attach times to actions by individual intervenors and to the results of their actions, in terms of the changed losses during the emergency. More significantly, the no-intervention time/loss analysis curve and outcome also need to be estimated by the investigator, so the curve can be plotted. The task of tracking individual actions and the actual time/loss curve usually poses no investigative problem. However, the no-intervention T/LA curve has presented difficulties, because of the scarcity of predictions about hazardous material behavior on which estimates can be based.

The T/LA method influences planning in two ways. First, it leads planners toward the time/loss relationships during emergencies. This forces their plans to focus on the outcomes of the emergencies. This orientation requires planners to project outcomes from either the natural course of events, or the planned actions to change those events. Likewise, this approach demands a predictive capability for projecting the loss effects of both natural and intervening events with time. Such a capability is presently weak or infrequently available for the many dangerous goods in transportation. Yet without this capability, planning for control of emergencies by responding personnel often relies more on luck than reason.

The second influence on planning relates to the division of planned task assignments among intervening personnel. Only those who can contribute to control of the emergency should respond. The potential contribution to improvement of the course of events constitutes a test for each planned task. This test consists of a measure of the contribution versus time involved. Thus, a practical tool for task planning becomes available for use by planners. A way now exists to judge the value of each person's presence on scene by the specific contribution of each to the net reduction in losses at successive time intervals.

If one make no contribution, one should not respond. If one plans a contribution, that contribution can serve as an objective during a real response.

Training can also be influenced by the T/LA evaluation method. The method suggests that intervenors should be able to predict (1) the remaining natural course of events and their effects on the expected losses versus time, and (2) the effects of intervenors' actions on those events. Both predictions are needed before intervenors can make an informed decision to intervene or not to intervene. Training in this diagnostic and predictive task is only recently receiving significant attention in the United States. When liked to the first principle - - if can't contribute, don't intervene - - the need for predictive study of time/loss behavior of dangerous goods shipments in transportation emergencies becomes very evident. Training of emergency response personnel in these predictive and diagnostic skills becomes an imperative aspect of their curriculum.

The evaluation method can influence operations during emergencies, too. The potential value of the intervention of each successive intervenor should be evaluated by the reduction in losses achieved by that intervenor. If the official in charge of the emergency operation determines that a person or organization can make no contribution, their presence serves no immediate purpose and their involvement should be carefully reconsidered. Emergency operations are sufficiently stressful on those performing response tasks that extraneous persons who might detract from these tasks should not be allowed to become involved.

During the emergency, each intervenor's goal should be to reduce the losses

or time duration of the emergency. If intervenors do not change the "natural" course of events, or if they increase the net losses, intervention practices or assumptions, safeguards and procedures must be re-examined to find ways to improve the operational effectiveness and emergency outcomes.

EXAMPLES OF T/LA PLOT FOR TWO ACCIDENTS

Two examples of the method help to show its value. Figure 5 is a T/LA plot using data from a 1975 highway accident. The loss values used are best estimates because of the uncertainties surrounding the time of death for the victims and the actual amount of the legal settlements during subsequent litigation.

[redacted]his accident involved the violent rupture of a bulk liquified flammable gas c[redacted]tank during an accident, followed by the massive release of the entire cargo, formation of a large vapor cloud and sudden ignition of the escaped gas. A large tank section rocketed into nearby mobile home residences.

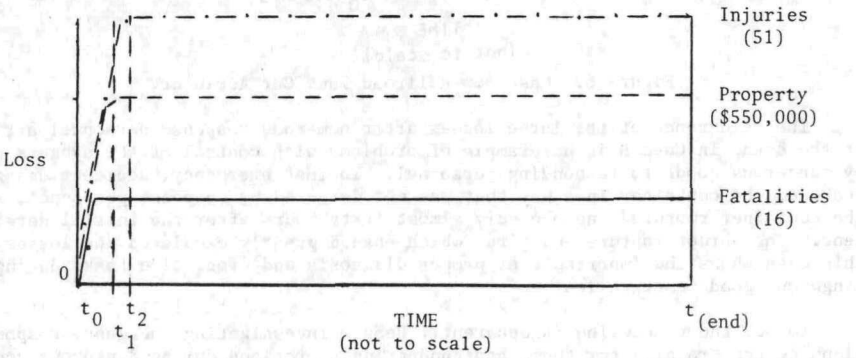


Figure 5, Case A. Cargo Tank Accident.

While the time is not precisely to scale, most of the losses in this accident occurred before any emergency response personnel arrived on scene at time t_1. The sixteen fatalities and 51 injuries were almost all burns. Property losses were estimated at $550,000 U.S. Losses before the arrival of emergency personnel at the scene illustrate how the T/LA plot can show the importance of built-in safeguards on the dangerous goods container. The elapsed time between t_0 and t_1 was less than five minutes. Fires were extinguished quickly and the biggest problem was caring for the burn victims.

[redacted]A second illustration reflects a major hazardous materials railroad accident, in[redacted]ng the same kind of product.

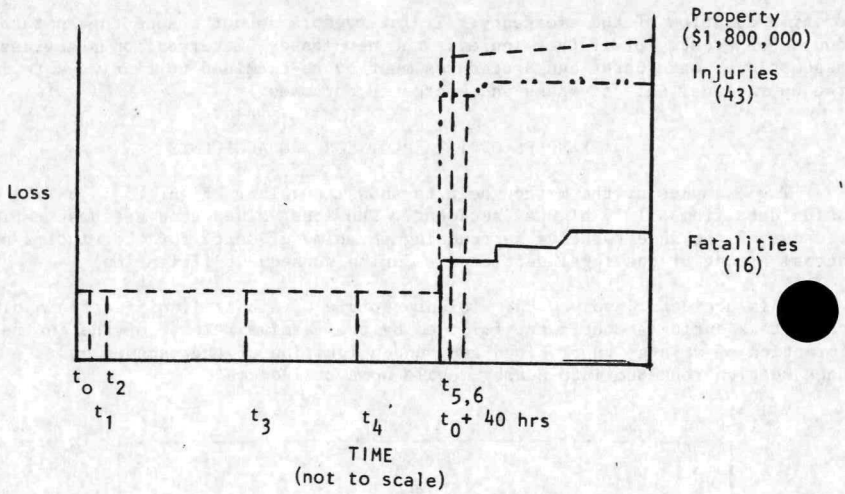


Figure 6. Case B. Railroad Tank Car Accident.

The occurrence of the large losses after numerous response personnel arrived at the scene in Case B is an example of problems with control of the dangers posed by dangerous goods to responding personnel. In that emergency, accident damage had weakened the container in a way that was not detected by response personnel, and the container ruptured unexpectedly almost forty hours after the initial derailment. The abrupt rupture and fires which ensued greatly escalated the losses. This case shows the importance of proper diagnosis and predictive tasks during dangerous goods emergencies.

To me, the conclusion is apparent. People investigating emergency responses, planning and training for them, and conducting operations during dangerous goods emergencies should become familiar with and utilize the time/loss analysis method and share results of its application. The potential safety benefits should be significant.

DISCUSSION

Question by E. C. Hardin, USA: Doesn't the time/loss analysis technique provide nuclear transporters with an opportunity to demonstrate the superior built-in safety features of RAM packages and systems? This should be particularly evident in the construction of productive T/L curves determining essential response players.

Answer: I agree. As indicated in my presentation it is critical that safety be built into the device so that the initial intervener will have a chance to effect damage/injury control. Time can be of the essence.

Question: Does the time axis (t_1, t_2, etc.) refer to the initiation of action or the arrival of emergency response forces?

Answer: Both. It refers to any action which affects or effects the accident outcome in terms of duration or extent of loss.

Question by J. DiStefano, USA: Has the NTSB's experience in responding to accidents involved supervision by the NTSB or the local authority? If the chief responsibility is that of the local authority, has the NTSB found a successful way to prepare the local authority for a smooth coordination with the federal authority and expertise?

Answer: First of all, NTSB has investigative authority and responsibility only in case of accidents. In this respect the NTSB has the prime role. This is most evident in aircraft accidents, least evident in marine accidents where the coast guard and the NTSB act collaterally. The NTSB does not supervise the accident control or clean up operations. The NTSB allows the person in charge of the accident scene, usually the on-scene commander, and usually the fire control people, to run the show until control is achieved. At that time we step in for accident data collection. We do our best to advise the local authorities that our interest accident investigation not damage/injury control. This is necessary because in some cases, there is a tendency on the part of local communities to assume that the federal authority (NTSB) is going to take charge. I do feel that there needs to be identified, one person or authority in charge. This should be firmly established in disaster plans or whatever else they are called and then followed and adhered to. Perhaps the biggest value, at least initially, of having established viable, written procedures is to establish authorities and responsibilities. This concept in no way would prohibit the cooperation and even codirection of the control of an accident involving possible release of nuclear material by the local commander (e.g. fire chief) and representatives of the nuclear interests (FEMA, NRC). This needs to be included as part of the disaster plans though for it to be effective.

The NTSB is prime in the accidents it investigates in terms of determining cause and making recommendations for correction of the causes. This is by law and statute. The same does not apply to accident damage/injury control. In all cases we contact the local authorities via our local representatives, advise them of our interests and needs and work with those authorities in terms of identifying those needs and acutally recovering necessary accident data.

Comment by A. L. Kaplan, USA: As a further comment on the question raised concerning pre-planning for transportation emergencies in the United States, licensees of the US Nuclear Regulatory Commissio (NRC) must have approved emergency plans. Also, the NRC requires that agreement states and all states in which nuclear power plants are located must have emergency plans, which must include plans for transportation incidents involving radioactive materials. The NRC also requires periodic drills to assure that personnel involved are properly trained and to provide continual evaluation of the emergency plans. In the state of North Carolina, personnel from the state government agencies and local agencies together with emergency personnel from our facility have already held one exercise in which a [redacted]cal traffic accident involving a truck carrying radioactive materi[redacted]as hypothetically involved. More such joint exercises are being p[redacted]ned for the future.

Answer: Good. I have seen cooperation of this type, and trainir of this type among communities exposed to aircraft and train disasters - and it is proving successful. Joint exercises are critical - and more critical is the objective evaluation of those exercises.

EMERGENCY RESPONSE CAPABILITIES DEVELOPED IN THE UNITED STATES TO DEAL WITH NUCLEAR MATERIALS TRANSPORTATION ACCIDENTS*

S. G. Vandevender and R. T. Reese
Sandia National Laboratories, Albuquerque, NM, USA

A. H. Schilling
Battelle Human Affairs Research Center, Seattle, WA, USA

INTRODUCTION

No nuclear transportation incidents in the United States have caused loss of life or massive property damage resulting from radiological contamination. Nor have there been serious radiation injuries or exposures. Thus the civilian emergency preparedness and response capabilities have not yet been subjected to severe radiation emergency conditions. The readiness of federal, state or local agencies and the transportation industry to respond to accidents involving radioactive material is the subject of this paper.

NEED FOR EMERGENCY RESPONSE PLANNING

The nuclear transportation incidents in which radioactive releases have occurred have all involved Type A packages; that is, packages containing radioactive materials having limited activity. Those incidents involving packagings of radioactive materials having greater activity did not result in releases. About 600 incidents have occurred during transportation of nuclear materials, compared to some 86,000 reported hazardous materials incidents.[1] Thus, the emergency planner is probably more concerned with appropriate response to chemical and other non-radioactive hazards. However, the responders perception of the hazards involved often does not correlate well with the actual hazards which may exist as a result of an accident. Untrained response personnel tend to overestimate the hazards at an accident scene, often leading to incorrect response actions and possible overreaction by the public.

The value of emergency response preparedness can be seen in the following case history.[2,8,9] Four 14-ton uranium hexafluoride cylinders were involved in the wreckage when a 102-car train derailed near Rockingham, North Carolina. All of the cylinders remained intact and no radioactive material was released. Seventeen government agencies, including thirty radiological experts, responded to the accident because the dispatcher called all agencies considered appropriate. Most of the experts arrived much later than would have been desired if there had been a radiation hazard. The state team arrived more than three hours after the accident, while the federal government's radiological assistance team (Energy Research and Development Administration (ERDA), now Department of Energy (DOE) arrived seven hours after the accident. There was substantial confusion over the severity of the situation. As late as four hours after the derailment, it was uncertain whether four or six cylinders were involved; the conditions of the cylinders had not been determined. No one had established command over the response effort, and conflicting reports of radiation leakage had been issued. A reporter was

*Work sponsored by the US Department of Energy under contract DE-AC04-76DP00789.

burned by a substance thought to be radioactive; seven hours later it was determined to be a thermal burn from molten fertilizer. The ERDA (DOE) team, looked upon as the dominant authority, informed the state team that ERDA (DOE) had no authority but could provide back-up support. Communications at the scene were hampered by lack of equipment. Onlookers, including news media personnel, were not kept out of the area of suspected radiation hazard. Over five hours were required to determine the number of cylinders on the train. Authorities and responsibilities were not fixed until eight hours after the accident.

This case illustrates: (1) lack of coordination and duplication of effort between various agencies, (2) failure to have a pre-designated agency and mutually agreed upon on-scene coordinator appointed to serve as the local authority, (3) lack of involvement of shipper and carrier organization in state and local emergency response planning, (4) inadequate communications both at the accident site and between the site and the emergency response agencies and (5) lack of proper news media interface.

In developing a methodology for evaluating emergency response, the Battelle Human Affairs Research Center (HARC) identified the following components as being essential for successful response plans:[3]

- Responsibility for planning and operations
- Notification methods and procedures
- Accident assessment
- Public information
- Protective response
- Radiological exposure control
- Emergency response training and exercises

The following conclusions evolved from the study:

1. Plans should focus upon relationships among emergency organizations and personnel to maintain flexibility and generality. Training of personnel and testing of operational systems should be emphasized to reduce implementation problems.

2. Integrating jurisdictional responsibilities may be a problem in emergency response. There is redundancy in response capability at various levels of government. This redundancy can be either helpful or harmful.

3. First responders (e.g., police or fire agencies) are important elements. They must know both how to respond and when to summon additional help.

4. Differing responsibilities can make federal response difficult. Some confusion may be inevitable, as in the case of a newly created agency such as Federal Emergency Management Agency (FEMA).

5. Carriers display a lack of knowledge about the roles of various federal agencies in emergency response situations.

6. There appears to be confusion regarding federal agency responsibilities.

7. It is essential for members of a response system, especially first responders, to have up-to-date information on communication procedures.

8. State and local response plans are generally lacking in provisions for investigating accidents and taking corrective action.

9. There is a perception that training opportunities for first responders are too limited and could be improved in quality.

10. Documents containing information on the contents of shipments appear quite vulnerable to fire. Loss of this information could compound the seriousness of an accident.

11. It is essential that emergency response plans be tested.

EMERGENCY RESPONSE JURISDICTIONS

Federal agencies have both planning and response roles for radiological emergencies. The planning function is formalized in the Federal Interagency Regional Advisory Committees.[4] One Headquarters Advisory Committee and ten Regional Advisory Committees have been established to support radiological response planning by state and local governments. The federal agencies involved are the Environmental Protection Agency (EPA), DOE, Department of Health, Education and Welfare, Department of Transportation (DOT), Defense Civil Preparedness Agency, Federal Disaster Assistance Administration and Federal Preparedness Agency. Complicating the situation in the immediate future is the recent establishment of the Federal Emergency Management Agency (FEMA). At this point the relationship between FEMA and the Headquarters Advisory Committee is not fully established. The objective of the Headquarters Advisory Committee is to provide a common focal point for direction and coordination of field assistance operations. The two principal objectives of the Advisory Committees are: (1) to provide a regional forum for exchange and consolidation of information on radiological emergency response plans (RERP), capabilities and needs in support of field assistance for state and local governments and (2) to make state RERP's as effective as possible through provision of guidance in their development. Planning assistance is primarily aimed at nuclear power station locales; however, transportation emergencies are also within the jurisdiction of the Regional Advisory Committees. This assistance is provided only upon request because the committees have no authority to impose plans on the states.

The Interagency Radiological Assistance Plan[7] (IRAP) defines the response role of federal agencies in radiological emergencies. The major responsibilities are assigned to DOE, EPA, and the Nuclear Regulatory Commission (NRC). Other participating agencies are the Defense Civil Preparedness Agency, Interstate Commerce Commission (ICC), National Aeronautics and Space Administration (NASA), Postal Service and the Departments of Agriculture, Commerce, Defense, Labor, Transportation and Health, Education and Welfare. IRAP assigns to DOE the lead responsibility for providing equipment and expertise to protect the health and safety of individuals, the public and the environment. DOE has established eight radiological assistance regions, with a radiological assistance team (RAT) based in each region. The RAT responses for commercial shipments are coordinated by the National Coordinating Office at DOE Headquarters, who delegates the responsibilities to the appropriate region. The response to emergencies involving defense shipments is coordinated by the to the Joint Nuclear Accident Coordinating Center (JNACC) in Region 4, located in Albuquerque, New Mexico. EPA has been assigned the responsibility for providing monitoring teams to measure environmental radiation, to evaluate the extent of contamination and to provide advice as to appropriate protective actions. NRC's assigned role is to collect information and evaluate accidents involving NRC a licensees. All IRAP assistance is provided only upon request from a state or local agency. There is no federal authority to assume command of a civilian accident response effort.

The primary responsibility and jurisdiction for peacetime transportation accident response rests with the state or local governments. In a recent study[5] commissioned by NRC to survey emergency preparedness at state/local levels, Indiana University found two types of emergency response planning and organization: (1) centralized systems in which all authority and technical capability resides in the state organization and (2) decentralized systems in which local jurisdictions have a degree of authority and some technical capability. Only eight states employ a decentralized response mechanism wherein the local jurisdictions have a well developed capability to respond, although seventeen states reported that all local jurisdictions have autonomy. The objective of decentralization is to avoid deploying the state organization, which may be hours removed from the scene, except in the case of serious incidents. In 43 states the centralization places jurisdiction with a state organization. First-on-scene respondents administer first aid and control crowds. They then call for help from the state, usually through the state police dispatching system. Fourteen of these states have established field offices to assist in the response. Within the centralized states, some large cities have also developed a response capability.

The study also found that most states adapt their response plan and system to the existing organizational framework and utilize existing resources where possible, so that responsibility is frequently dispersed among several agencies. In the event of serious accident, the resulting multi-agency response must be well coordinated at the scene. Participation of local first-onscene responders and assistance from federal teams add to the coordination problems and make predesignation of a state command authority a necessity. Thirty-four states have designated a state official as on-scene coordinator, twelve have no one designated and one places authority with the county sheriff. (Three states did not respond to the question.)

STATUS OF EMERGENCY RESPONSE PLANS

Although it is the states' responsibility to plan for response to transportation emergencies, NRC was required under Federal Register Notice 40FR59494 (December 24, 1975) to develop guidelines for state planning. This requirement was recently transferred to the new Federal Emergency Management Agency (FEMA). In January of this year, a task force of federal and state agency representatives* began drafting the Guidance for Developing State and Local Radiological Emergency Response Plans for Transportation Accidents. This document will provide general information on the types of radioactive materials transported, how they are transported and what the consequences are likely to be in the event of an accident. It will provide general guidance for state and local agencies on the content of emergency plans for responding to accidents. It will also contain a brief discussion of some typical transportation accident scenarios.

In the study of state** preparedness, Indiana University found that one state has no generalized plan for radiological emergencies but had a specific Transportation Radiological Emergency Response Plan (TRERP), while thirty have made provisions for handling transportation emergencies within, or as annexes to, their general RERP. Of those, twenty also have Emergency Operations Plans (EOP), and only three handle radiation emergencies with entirely distinct plans. Of the remaining states, twelve have RERP's without transportation emergency provisions, one has only an EOP and no information was available for eight states.

*Two Transportation Technology Center (TTC) staff members are also participating in this task force.

**Puerto Rico and the District of Columbia are included here.

The study noted that two states are experimenting with the development of a specially trained cadre of state police officers who will respond to and assess all hazardous materials accidents. These officers are trained in biology, chemistry, physics and emergency management. They are provided patrol cars equipped for hazard assessment and are responsible for determining the level of response required, if any, by highly trained agency personnel.

While 38 states have designated response teams, only twelve have multidisciplinary teams, although specialists are usually available to assist the designated team. The remaining states depend on a variety of volunteer experts, field staff or specified radiological health employees. However, only eighteen states maintain directories of physicians trained in radiological emergency care.

TRAINING STATUS AND PROGRAMS

The states have begun to provide training for first-on-scene respondents to radioactive materials transportation accidents. Indiana University found that most states have made a serious effort to train state police to identify the hazards involved in a radiological accident, some to the extent of estimating the seriousness of the hazard. However, training for local police, firemen and road maintenance personnel is not as thorough except in some larger cities. While the states generally assume responsibility for training public safety officials in emergency management, they rely on federal programs for training in the more technical aspects of response. Several states have sent a few police and fire personnel to federal training programs as preparation for conducting local training sessions. Radiological health and emergency service personnel are largely trained through federal programs.

The federally sponsored training programs are listed below:[6]

1. DOE, through Oak Ridge Associated Universities (ORAU) offers --

   - "Medical Planning and Care in Radiation Accidents," a one week course for physicians, training about 48 participants per year.
   - "Health Physics in Radiation Accidents," a one week course for health physicists, training about 36 participants per year.
   - "Handling of Radiation Accidents by Emergency Personnel," a 2-1/2 day course for emergency room surgeons and nurses, training about 54 participants per year.

2. DOT, through ORAU, offers --

   - "Response to Radioactive Materials Transport Accidents," a self-training manual.

3. [redacted]T, through Canyon Research Group, Inc. --

   - "Radioactive Materials Transportation Accident Training," a package that is not yet routinely offered.

4. DOE, through National Fire Protection Association (NFPA) --

   - "Handling Hazardous Materials Transportation Accidents, a 20 hour course for first-on-scene response personnel with 8-hour supplement dealing with radioactive materials.

- "Handling Hazardous Materials Transportation Emergencies," instructors seminars.

5. FEMA --

   - "Radiological Emergency Preparedness," a one week training course intended for local community officials.

6. NRC, through ORAU --

   - A ten week program for state health physicists, training about 20 participants per year.

7. NRC, through Reynolds Electrical and Engineering Company --

   - "Technical Operations for Radiological Emergency Response Operations," a 8-1/2 day course for state or local government response team members, training about 400 participants per year.

8. NRC, through Federal Interagency Task Force on Training and Exercise --

   - "Radiological Emergency Response Planning," a 5 day course for state or local response planners.

   - "Technical Management for Radiological Emergency Response Coordinator and Staff," a 7 day course.

9. Colorado Training Institute --

   - A 3 day seminar and a 2 week course on all phases of hazardous materials transportation incident response, including radioactive materials. Originally funded by a grant from DOT, but now an independent state-run program.

CONCLUSION

The non-existence of emergency response programs is frequently stated as a reason for restricting the movement of radioactive materials through states or local jurisdictions. Yet, studies discussed here indicate that emergency response capability, while not in the best condition, is getting more money, interest and attention, and in most states response networks exist which could be effective in responding to radiological emergencies. Awareness of such capabilities by the public is an important feature in increasing the public's confidence in the ability of federal, state and local officials in controlling hazards. One aspect of this awareness program could be in broader availability of radioactive emergency techniques for possible first responders to emergencies. This training, public awareness and more emphasis on workable emergency plans will help to assure reliable and workable emergency response plans.

REFERENCES

1. J. D. McClure, Emergency Response Planning Guidance, Draft, TTC-0123, Sandia National Laboratories, Transportation Technology Center, Albuquerque, NM, June 27, 1980.

2. H. E. Lipzek and G. R. Schuller, Legal, Institutional and Political Issues In Transportation of Nuclear Materials at the Back End of the LWR Nuclear Fuel Cycle, PNL-2457, Battelle Human Affairs Research Center, Seattle, WA, 1977.

3. A. H. Schilling et al., Emergency Response In Transportation of Radioactive Materials: An Evaluation Methodology, B-HARC-311-044, TTC-0045, Battelle Human Affairs Research Center, Seattle, WA, to be published.

4. Emergency Response to Nuclear Materials Transportation Accidents, BDM/TAC-80-573-TR, TTC-0170, BDM Corporation, Albuquerque, NM, to be published.

5. Survey of Current State Radiological Emergency Response Capabilities for Transportation Related Incidents, NUREG/CR-1620, Indiana University Bloomington, IN, to be published.

6. C. Thompson, Preliminary List of Radiological Emergency Response Training Available to First-On-Scene Personnel, ORAU-167, Oak Ridge Associated Universities, June 1980.

7. Interagency Radiological Assistance Plan (IRAP), DOE/EV-0010, US Department of Energy, Washington, DC, March 1978.

8. Special Investigative Report: On Scene Investigation Among Agencies at Hazardous Materials Accidents, NTSB-HEM-79-3, National Transportation Safety Board, Washington, DC, September 1979.

9. Critique of Seaboard Coastline, Derailment South of Rockingham, North Carolina on March 31, 1977, Radiation Protection Branch, Department of Human Resources, State of North Carolina, May 6, 1977.

DISCUSSION

Question by J. Hamard, France: 1. In the USA is there, at a federal level, a legal obligation for training of personnel involved in transportation of radioactive materials? 2. Many agencies, both federal and regional, are involved in case of accident. Who is coordinating their actions and who takes the decisions?

Answer: 1. No, although this requirement will probably occur within the next few years. 2. This is a complicated question because the first responder is in charge but the capabilities of many first responders are limited because of the very infrequent occurrence of accidents involving nuclear materials. The first responder and those more qualified from state or federal agencies must resolve how they will divide responsibilities and who will be in charge. Therefore, who is in charge depends on where the accident occurs, how severe the accident is, whether federal help was requested, and many other factors. The decisions will be made, but who actually makes them will likely vary among different accidents.

Question by R. Kortright, Canada: What has been done to ensure that response procedures for radioactive material are consistent with those for other hazardous material? Is FEMA the agency at the federal level which is responsible for this?

Answer: The planning for response procedures is designed to be thorough and, as such, will result in a similar level of consistency for all hazardous materials. To my knowledge there is no planned effort to be consistent nor do I know of any criteria that exist to measure consistency. FEMA is the agency which is responsible.

RESCUE CONTAINER TO ACCOMMODATE DEFECTIVE TRANSPORT CONTAINERS FILLED WITH URANIUM HEXAFLUORIDE

Heinz Beyen
Reederei und Spedition "Braunkohle" GmbH, Wesseling
Federal Republic of Germany

Introduction:
In the past years, the storage quantities of uranium hexafluoride have increased and the periods of storage have inevitably become longer. This le[redacted] to a by far higher probalitiy of possibilities for incidents. The s[redacted]ising authorities and in particular responsible storekeepers are a[redacted]s to offer the best possible safety and to take preventive measures for the case of incidents, not only during storage but, under certain circumstances, also during transport. Following these interests, the rescue container to be described here was developed.

Construction: (principles)
The construction is based on the following principles:

1. Simple handling of a heavy, defective 48" container and 30" container.
2. Greatest possible freedom in choosing the means of transportation.
3. Easy decontaminability.
4. The number of openings should be minimized.

As to 1.: When handling the 15-t UF_6 container, the heavy weight and the fact, that work has to be done by using a respirator, render things extremely difficult. Hence the requirement here for the simplest possible handling for the personnel.

As to 2.: The rescue container has to be transportable by any means of transportation having the relevant carrying capacity (rail, road, sea and air transport) - so that it can be used in many places and at any time.

As to 3.: The easy decontaminability is a requirement which is self-evident for all objects etc., which come or may come into contact with radioactive material.

As to 4.: This represents a basic principle for all containers etc. for the storage and transport of hazardous goods. Each opening contains in itself the danger of leakage rates, even if these are possibly very small.

[redacted]The listing reveals that it is not immediately possible to make all requirements comform to one another; it involves compromises in order to reach the appropriate optimization. An additional aggravation is that the competent authorities imposed the following requirements:

- the rescue container must have an installation to empty the defective UF_6 container into an undamaged container at the site, i.e. without prior transport,
- it must further be possible to empty the defective container also after transport from one site to another, without removing this container from the rescue container.

Because of these two latter provisions simple wrapping systems could no longer be thougt of.

Set-up:
In describing the construction we follow the set-up of the overall [redacted]m from the inside to the outside. The core part is a horizontal pressu[redacted] tank - designed as such for safety reasons - although during handling and decanting no relevant internal pressure occurs. The design pressure amounts to 10 bar; the underpressure which may eventually occurs has been taken into account in the construction. This pressure tank can be opened on side, i.e. on bottom side can be hinged entirely; to allow better handling, the whole circular cross section is set free.

In the pressure tank there is a sled which can be moved on rails in longitudinal direction of the container. The defective container is put on this sled - the 30" container with adapters - the construction has been designed in such a way that, when the pressure tank cover is closed, the defective container including sled is locked in position. It is fixed in such a manner that forces occurring during transport are absorbed; however, it is not possible to meet all conditions of standard packing. But since measures for an incident - thus not for normal transport - are concerned, additional organizational or operational measures have to be taken, if necessary. For transport, for instance, this could be a speed limit.

Apart from the already mentioned big opening, this pressure tank only has supply and outlet tubes for UF_6, N_2 and the corresponding measuring systems. It goes without saying that much attention was given to the sealing of the big opening.

For the purpose of emptying a defective container, a heating system is installed around the pressure tank. This heating system is regulated by thermostat. The temperatures can be set or limited, respectively. The maximum temperature is adapted to the admission conditions of the 48" containers, i.e. the highest admissible temperature is 120° C. As already mentioned, an overflow tube for UF_6 gas completes the equipment.

The pressure tank is installed in a 20' container. The one container wall, namely the one to be associated with the tank opening, can be opened. An additional pair of rails - as extension of the rails in the pressu[redacted]ank is arranged on a frame which is hinged down and in this hinged-dow[redacted]siti lines up with the rails in the inside of the pressure tank. In orde[redacted]o allow the opening of the pressure tank, it must be possible to partly hinge the rail system on the frame.

On the frame system of these rails, there is also a hydraulically operated winch which allows the insertion and removal of the defective cylinder. The rail frame can balance out possibly existing soil unevennesses or, in case of unfavorable soil conditions, reduce the relevant soil pressures to a tolerable degree.

Between the pressure tank and the supply part there is a sealing wall, the purpose of which is to prevent contaminations of the supply systems and to facilitate the decontamination of the pressure tank side, respectively.

In the zone of the pressure tank including the appertaining intermediate w[illegible], the container is heat-insulated in order that the energy input for [illegible]ng up can be kept within limits and that the desired temperatures [illegible]de the pressure tank can be reached, respectively. On the other side of the container there are, in different separated cells, first the connection possibilities for the overflow tube and further the connection for the nitrogen input. This part is especially separated since in this zone a contamination is possible during the connecting and disconnecting, respectively, of the tubes. Other compartments house the necessary regulating and control instruments, further the hydraulic control unit, electric switching systems and a battery. In addition, tools and protection equipment can be accommodated here. On the outside, the container has the same connection possibilities as a standard container for cargo-handling equipment or fixation in the means of transport, respectively, and can also be handled and transloaded accordingly. The net weight of the container amounts to 13 t, so that it reaches a gross weight of 28 t when a useful load of 15 t is added. The pressure tank may be manufactured either of boiler plate H II DIN 17155 A St 41 DIN 17135 or of refined steel 1.4541 DIN 17440, (austenitic Cr, Ni, TI). While considering which material to choose it should be remembered that the equipment in question has to be ready for operation at any time and that consequently maintenance should be kept to a minimum.

On the other hand, however, if there is a possibility of having maintenance operations carried out cheap, it may very well be more economical to use the regular boiler plate instead of the high-quality material. In this case, though, attention is to be paid to frequent maintenance and *appropriate protection against corrosion. If refined steel is used, also* other hazardous goods can, under certain circumstances, be inserted and transported safely, as far as for example corrosive, toxic or similar agents are concerned.

The system has been submitted to the "Bundesanstalt für Materialprüfung (BAM)" and to the "Physikalisch Technische Bundesanstalt (PTB)" for approval. The running-up test for the railroad transportation license has been applied for. Furthermore, patent applications have been filed for the Federal Republic of Germany and other countries world-wide. To co[illegible]lete the picture it should be mentioned that the manufacture takes [illegible] in accordance with the ASME provisions.

Handling of a defective cylinder and mode of operation of the rescue container:

The rescue container is prepared to accommodate the defective cylinder; in case of a 30" container a corresponding adapter has to be installed. The pressure tank is open, the sled is on the extended rail system in

front of the container, the winch has been set in such a manner that it can draw in a defective container immediately after it has been put on. Depending on the extent of the damage of the transport container, the following operations have to be carried out using a respirator.

By means of a crane, - eventually a mobile crane -, the defective container is put onto the accommodation system, then follows the drawing into the pressure tank. The traction rope is taken off the winch and put into the container (it is attached to the sled). Subsequently, the pressure tank itself is locked and sealed. After these operations, the use of a respirator is no longer necessary, the system is safe and, after locking the container, ready for transport. Then, nothing hinders the transport to the decanting station anymore. The unlaoding at the decanting station takes place in opposite sequence, again using th[redacted]- sonal protection measures.

The decanting with the heating installation of the rescue container requires additional measures or considerations, respectively, which are also influenced by the extent of the damage of the transport container. It is recommended in any case - as far as this is possible - to seal the leakage. There are several possibilities to do this, for instance stoppers, bandages, good results have also been obtained with glass fiber reinforced synthetic plasters. In this case, one first takes a quick-hardening material, polyester for example, on top of which comes a second plaster with better adhesion, perhaps epoxide.

For the decanting operation, it is not at all sufficient to just heat up the rescue system. On top of the solid UF_6 mass, there is a gas mixture of different composition and, if there is a leakage, with atmospheric conditions of course. This light gas volume can - again depending on the extent of the damage - encompass the whole free space of the pressure tank; it must be led off before the decanting operation can start. From this fact follows that, the bigger this gas quantity the bigger also the decanting equipment, under certain circumstances. You will understand that only basic characteristics can be handled here.

If the container is properly sealed or if there is only a damage which did not penetrate the transport container, it may possibly be sufficient to evacuate it down to about 100 mm Hg pressure. If, however, large quantities of gas have to be led off or, for whatever reasons, the UF_6 should, or has to be extracted, several cold traps have to be connected in series, or adequate filters with pumps or compressors, respectively. In these compressors, however, already slight contaminations by UF_6 or HF, respectively, can lead to lubrication problems and, under certain circumstances, to the relevant damages. These few considerations show what importance these gas quantities have and how important a sealing - if at all possible - is. However, these thoughts may not lead to the opinion that the heating system of the rescue container is not necessary. Nevertheless, only this heating system renders the decanting possible w[redacted]t handling the unprotected defective transport container once more. [redacted] safety technical reasons it is more appropriate to leave the 48" or [redacted]" container in secured condition and to interpose the addionally required equipment - between rescue system and receiving container - and to heat it subsequently. With this procedure, the transport to a decanting station or the like will, of course, have a positive effect since expert personnel auxiliary equipment etc. will be available there to a greater extent. As a matter of principle, the overflow operation should take place very

slowly, i.e. with relatively insignificant temperature rise. For this incident, one can very well accept relatively long overflow times. Whether the receiving container is additionally cooled or whether this is not necessary because of the surrounding conditions, is of minor importance.

When the decanting operation has been accomplished, the pressure tank is opened again, the defective container, under certain circumstances using a respirator, is removed, decontaminated and brought to repair, then the relevant decontamination operations on the rescue container can be carried out.

If the rescue container is used for the safe transport of containers with [redacted] agents, measures have to be taken to ensure that these containers are fixed in the pressure tank. This fixation has to withstand loads occurring during transport. When size and construction of these containers are known, the corresponding installations can be prepared on the insertion sled. Furthermore, in the latter case it is to be examined whether the material of the pressure tank is suited for the operation. Inflammable agents can additionally be made safe by injecting inert gas, other agents can be stabilized by additions.

DISCUSSION

Question by W. Heudorfer, FR Germany: How is suction carried out if the suction pipe is damaged by an accident?

Answer: The overflow piping has two connections in the pressure vessel:
1) connection to the valve
2) connection to the ambient space.
If the valve is defect suction takes place from the ambient space.

Comment by E. Steinebach, The Netherlands: There are, for example, two cases where this rescue container will not work:
1) Determination of 40" Y or F container, so that it is not possible to put it into the rescue container;
2) A big hole in the container and heavy hydrolisation to UO_2F_2 will form a plug of UO_2F_2 inside the container so that the container valve will not work.
We think it is more useful and cheaper to use a"plaster."

Answer: Concerning your first point, the diameter of the pressure vessel already is relatively large. There are always "conceivable [redacted] accidents" but the probability is low and does not justify that therefore nothing is attempted or no measures are taken which remove a large percentage of the probable accidents.

Concerning your second point, by means of the rescue container, the transport container can be safely moved to a facility where it can be treated. The plug must then be removed under the appropriate safety measures, if necessary with special tools, e.g. by boring.

It is hardly conceivable that a defect container with plug can be moved to a decanting station or similar or that approval will be given.

Question by G. Cohendy, France: To what extent could this rescue container be useful in view of dealing with the case of a 48Y UF_6 container involved in a fire and releasing its contents?

Answer: For this case there is no solution, or rather after extinguishing the fire the defect, cracked, possibly contaminated transport container can be put in the rescue container and transported

TEAM - AN EMERGENCY RESPONSE PLAN

Calvin Brantley and Ermes DeMaria
New England Nuclear Corporation
Boston, Massachusetts

[redacted] the Fifth International Symposium held in 1978 in Las Vegas, Nevada, one of us described a new system to respond quickly to incidents involving packages of New England Nuclear's radioactive products. At that time the TEAM system was in the process of being implemented. As our company had grown, the number of telephone calls concerning transportation incidents had increased, although few of these were accidents causing release of radioactivity. However, we had no reliable method to provide advice nor to get expert help to the caller. TEAM was designed to remedy both of these situations.

TEAM, an acronym for Transportation Emergency Assistance Member, starts with a TEAM label applied to all outgoing packages of radioactive material. This label provides a 24-hour telephone number with instructions to call in case of emergency. The first response to a TEAM call is by an Initial Response Member, who is either a member of the Order Entry or Customer Service Departments. These persons evaluate the call, determine what shipment was involved, and either handle the call directly or send it on to one of the Support Team. This Support Team is made up of persons highly experienced in packaging, radiation safety, and regulations affecting radioactive packages. They can usually evaluate the required response, advise the caller directly, and propose remedial action. However, in two cases during the last two years, it was necessary to arrange for expert, local help to pick up badly damaged packages or clean up contaminated soil. Each call is thoroughly documented in a TEAM report after the incident is resolved.

We believe this program to be unique in allowing 24-hour 7-day per week response capability along with the reports, analyses and feedback to guide corrections to packaging, shipping procedures and response mechanisms. It has, in several cases, prevented undue media response by providing on-the-spot evaluation of the seriousness of an incident. When an "accident" is properly and quickly shown to be an incident in which no radioactivity is released nor radiation levels increased, newspaper and TV reporters usually put the story into correct perspective.

For purposes of this paper, we have analyzed two time periods: the first covering July, 1978 through June, 1979; the second from July, 1979 through June, 1980. In this period of time about 740,000 packages were shipped by truck and air[redacted]. In the first period, 355,000 packages were shipped with the number inc[redacted]g to 385,000 packages in the second period. However, company sales in the product areas covered by this analysis increased by approximately 23% while the number of packages increased only about 8%. This differential appears to be due to the effects of inflation and on somewhat larger orders of products per shipment. All the packages covered by this analysis were Type A or limited Quantity packages.

TEAM reports were received on 271 package incidents for the total period. About 14% of these did not involve transportation but were useful to us in other

operational areas. Of the remaining 234, 65% of the reports were in 1978/79 and 35% in the next period of 1979/80. The number of packages involved was 0.04% of the total packages shipped for the 1978/79 period and 0.02% for 1979/80.

In the first three months of the operation of TEAM, it was found that small packages of limited quantity amounts were being lost at the rate of 15 per month. Because these packages contained extremely small quantities of Carbon-14 and Tritium, they represented no risk to the public, but they were indicative of a problem. Changes in labeling, carrier control and delivery control were started. Since that time there have been no lost packages in 11 of the months, seven months with only one lost, one month with two lost, and two months with three lost. One month, February, 1979 was marked by extremely bad weather across the United States so the month ended with ten packages lost or at least delayed by two or th[redacted] weeks. None of these packages were of more than a few millicuries of eith[redacted]w hazard materials such as Carbon-14 or were of very short half-life materials.

Although there has been much emphasis on damaged packages and their potential effects on the environment by regulators and the press, damaged or wet packages were 14 in the first year and 16 in the second year. This represents only 0.004% of the packages shipped in the two years. Or to put it another way: out of 740,000 packages shipped in that period 739,970 arrived at their destinations without damage. Out of the 30 damaged packages only four were damaged so severely that any radioactivity was released to the environment - or 739,996 arrived at their destinations without releasing radioactivity to the surroundings, in spite of crushing, falling and other effects. The four packages contained only about 15 millicuries of radioactivity. In only one case was there sufficient radioactivity to require clean-up of soil.

Most of the damaged packages were caused by dropping from trucks or were smashed by passing trucks and freight carts. Four packages were involved in vehicle crashes or accidents. Two of these four packages were damaged so slightly that they were shipped to the customers after being checked. The other two were in a small airplane that crashed on takeoff and burned. Of the twelve millicuries of radioactivity in these packages most was recovered by cleaning the soil and picking up contaminated metal. The total cost of the clean-up was $4,700.

Although these are extremely useful statistics and have directed much of our planning for distribution of radioactive packages, the principal value has been in the direction and guidance of corrections to the system. It became apparent early in the operation of TEAM that damaged packages and packages to be returned to New England Nuclear by customers for any of a dozen reasons, constituted a major problem. Few customers or carrier personnel are qualified to prepare a radioactive package for either safe or legal carriage. A task group was established to find a solution to this problem using the information acquired through TEAM calls. The solution was a return system, called PAC BAC. For any package to be returned, we send complete directions for packaging, boxes, labels, and shipping papers. This system has been used by customers, carriers, police and regulatory agencies to return packages. In the first year of operation of PAC BAC we have averaged 55 packages per month through the procedure. Each such p[redacted]ge has been one less potential problem to the transportation system.

Another useful result of TEAM has been better identification of package components that cause weakness in package strength or integrity. In one four month period, damaged packages in Cleveland and Baltimore resulting in lost radioactive containers and/or increased radiation levels. Analysis of the reports showed a deficiency in the internal bracing. This led to a complete redesign of the package.

The usefulness of TEAM in accidents and their coverage by the media can be shown by the events of one evening. On that one evening a New England Nuclear package fell from a trailer at Logan Airport in Boston and was crushed by the wheels. The TEAM label led to a quick call, a quick trip to the airport to recover the package and determine that no contamination had occurred with no newspaper stories or inconvenience to the public.

In Miami at about the same time, a package fell from a freight truck on the main roadway of the airport. Although this particular package contained no radioactive liquid, it did hold sterile saline vials for eluting the radionuclide generator in the package. The fall broke some of these vials and liquid leaked out to the exterior cardboard. The first police at the scene roped off a 1500 foot exclusion area which also closed the main road to the airport. For two hours while police tried to determine what should be done this main road remained closed to hundreds of incoming passengers and cars. Newspaper stories appeared all over the United States covering this "accident" which could have been resolved in less than a half-hour by getting directions from some knowledgeable person. The solution was for someone to pick up the box and walk to the nearest truck to carry it away.

The most severe test of TEAM was started in the early morning of January 24, 1980 when a Cessna 411 aircraft crashed and burned on takeoff near Columbus, Ohio. On board were two Type A packages from New England Nuclear. These packages were destroyed. The Nuclear Regulatory Commission is aware of TEAM so an early report to us was made. The State of Ohio Disaster Assistance Agency, the Nuclear Regulatory Commission and New England Nuclear kept in constant touch to evaluate the problems. By noon, Batelle Laboratories in Columbus had been retained to start the clean up. Contaminated debris was separated and removed from the site. Some soil was removed to get the soil to background. The newspapers reported the incident and mentioned the presence of radioactive packages but also reported that they had been removed with no problems.

TEAM has accomplished what we expected. It is not perfect and changes are being made constantly to improve it. More police forces need to be educated on how to use TEAM, how to recover packages and how to handle radioactivity. Such programs have been developed by New England Nuclear and are being used to train law enforcement officers and fire personnel.

DISCUSSION

Question by E. Driver, USA: What is the means by which the initial intervenors or disaster personnel can get access to the "TEAM" telephone number in those instances in which the package and the "TEAM" telephone numbers are involved in the crash and possible conflagration and unavailable at the time? Is it possible to integrate the "TEAM" number and information into the CHEM TRAC system?

Answer: TEAM has been registerd with CHEM-TRAC recently. Copies of TEAM brochures have gone to all state radiation offices. It is described to state police being trained at the FAA safety school in Oklamhoma.

PANEL SESSION

COMMUNICATION BETWEEN EXPERTS AND THE PUBLIC

Moderator:

R. Neider
Bundesanstalt für Materialprüfung, Berlin (West), FR Germany

Panel Members:

W. Baier
Scientific Journalist, Berlin (West), FR Germany

Y. Sousselier
Commissariat à l'Energie Atomique/Commission de Sureté des Transports
Fontenay-aux-Roses, France

J. C. Allen
Department of Transportation, Washington DC, USA
(who kindly participated instead of M. A. Carlson, City of Spokane, WA, USA, who was unable to attend)

P. I. Ek
Swedish Nuclear Power Inspectorate, Stockholm, Sweden

R. M. Jefferson
Sandia National Laboratories, Albuquerque, NM, USA

R. F. Barker
International Atomic Energy Agency, Vienna, Austria

M. Blechschmidt
Physikalisch-Technische Bundesanstalt, Braunschweig, FR Germany

Introduction by R. Neider

The subject of communication between experts and the public is in fact a completely new problem, one unknown to us during the first 50 or 60 years of this century. Why is this subject so topical today?

The extremely fast developments in technology have led to an estrangement between the general public and this technology. Admiration of technology and its achievements during the first 60 years of this century has been replaced by considerable scepticism and even a certain fear. An important reason for this lies in the estrangement of man from himself, which can be put down to the rationalisation and materialisation of our life. Man's life today is not even thinkable without the many attainments of science and technology. In spite of this, it seems that many people now want no more to do with these attainments. The urge to return to a simple way of life can be felt everywhere. How can those of us who are familiar with science and technology, namely the experts, win back the confidence of the public, in order to convince them that technology is necessary? In this panel discussion we want to try to answer this question for the nuclear energy field and in particular the transportion of radioactive materials.

Statement by W. Baier

This panel consists of eight experts - seven for the packaging and transportation of radioactive materials, one for communication between these experts and the public. During the course of the discussion this may lead to difficulties. For, on the one hand, numerous scientists and engineers have had bad experiences when interviewed by journalists about their work. On the other hand, there are no fewer journalists who would gladly cross over to the other side of the street when they see a scientist or engineer approaching.

We must of course distinguish between scientific journalists and those who write for the general public. The scientific journalists rarely reach the general public, but there is a more important difference between these two types of journalist, which is, in an exaggerated sense, that the journalist writing for the general public must have understood what he is writing about, for the scientific journalist this is not necessary.

I believe such comments to be important, as for several years in the question of nuclear energy self-critical remarks have been heard from experts, that too little attention has been paid to explaining to the general public what is going on. Lately there has been a sinister conceit behind this. Only a few experts really want to give any explanations. They are not trained for that. There have always been well-meaning attempts by experts to provide information. However, because these attempts stem only from good intentions and not from communication knowledge, they may even have encouraged the so-called environmentalists. People are afraid of things they cannot understand.

There would be no point in our discussion if it only served for experts to reassert their readiness to inform and explain. More important, it should be recognised that opponents of the nuclear industry have gained valuable ground. Factual explanations alone are no longer sufficient. They persuade at the best the fencesitters.

To show this, I would like to quote an example from the French town of Orléans, where, several years ago, there were rumours that there was a white-slave trade taking place in the fashion boutiques of the town. Pretty customers were abducted and carried away at night in submarines. When the police publicly explained that there was no evidence of any missing person, they only succeeded in confirming the rumours that the police were in league with the white slave traders. The experimental psychologist, Watzlawick, drew the conclusion from this and other examples, that reality is always only what reality is believed to be.

Wünschmann pointed out that opponents of the nuclear industry cleverly use so-called archetypal images such as legends like those of Luzifer and Prometheus who brought light to men and were consequently brought down by the gods and unmercifully punished. These opponents use another archetype if they promise paradise to their believers. The influence of these ancient beliefs by mankind is so great, that there is no possibility of communication between opponents of nuclear energy and those responsible for it.

Under such circumstances, according to Watzlawick, the only chance which remains is to penertrate into the reality of others. The discussion about this cannot be concluded during this panel session.

Statement by Y. Sousselier

In the field of communication between experts and the public, one point should be dealt with with particular care, and that is the explanations which must be given to people involved in an accident by experts called to the scene.

In such an event, the questions asked by people in the immediate vicinity of the accident will be: "Have I been irradiated? Should I take any measures with regard to my health?"

It is a certain fact that the transportation of radioactive materials is extremely safe, that in the case of transportation in type B packagings the risk of radioactive release is exceedingly small. It is equally certain that the consequences of a radioactive release caused by damage to or opening of a type A packaging are very mild and that they are negligible in cases where the materials transported were of low specific activity.

It is nevertheless very difficult to reply with scientific precision in the event that radioactive materials are dispersed or a packaging opens, that the person has not been irradiated and there is no risk of contamination.

There is a risk that the problem becames slightly more complicated when the expert who arrives at the scene of the accident, generally some time after it has taken place and often without having been able to take any precise measures or sometimes even without knowing exactly what the packagings contain, finds himself confronted with such questions.

In some cases, a certain number of measures can be taken (e.g. when there has been a release of materials of low activity to check that the people involved have not been contaminated). It is obvious that particularly in this case, it is important to emphasise that the checks carried out are only over-precautions and show that in the transportation of radioactive materials, no risk whatsoever is [redacted]en.

It is, however, also obvious that at this point there can be a risk that there is a lack of comprehension by the people involved.

Statement by J. C. Allen

I would like to relate some experiences that the Department of Transportation (DOT) in the United States has had recently in attempting to communicate with the public.

DOT has had relatively limited experience in communicating directly with the general public on nuclear transportation matters, that is until recently. During the spring of 1980, DOT conducted a series of public hearings on its proposed rule-making on the highway routing of radioactive materials. The hearings were held at seven sites throughout the US and the purpose of the hearings was to obtain comments from the public on the highway routing proposals.

Communication can be defined as a meeting for exchange of information, and I would say that DOT's efforts during the public process were something less that a complete success. If communication can be loosely defined as a meeting of the minds, the public hearing were close to being a complete disaster. What we learned during this effort to communicate with the public was somewhat surprising to the DOT representatives at the hearings and somewhat unsettling, although I am sure it is not surprising to many of those who have been involved in the nuclear field for some years and are used to this type of negative public reaction.

I would say there are three basic distingushing characteristics of the public hearings. First, the highly emotional nature of the public debate and comments. Comments were often irrational and totally unrelated to the actual routing proposals upon which the hearings were based, and I would say that the emotionalism was carried to its ultimate in the last hearing in New York City when the DOT hearing officers had to receive a police escort to leave the building in which the hearing was being conducted.

Secondly, there is a severe lack of knowledge on the part of the public, not only a lack of knowledge of the specific routing proposals and the DOT's existing safety program for the transportation of nuclear materials but also a lack of basic technical knowledge necessary to discuss the important issues. We found that, not sur-

prisingly, the public is totally unfamiliar with the technical jargon such as transport index, nuclear criticality, numbers like 10^{-10} and usually use of these terms turns the public off.

The third characteristic I would call a credibility gap between the general public and the federal government. I believe this actually results from the lack of knowledge and the corresponding fear of the unknown. This credibility problem seems to be much greater than DOT has experienced in public hearings related to other hazardous materials, e.g. for compressed gases, explosives or inflammables. DOT efforts to correct testimony or respond to misinformation often met with doubt and outright disbelief by many members of the public.

Another phenomenon that we noted was that the public often came into the hearings with prepared comments, but were misinformed on the exact nature of the routing proposals. Most members of the public do not read government publications such as the federal register, notice of proposal-making, etc. They get most of their information from newspaper accounts which, though not necessarily incorrect in all cases, are often misleading in the manner in which information is presented.

On the other hand, I do not want to be too critical of the public participation in this formal hearing process that DOT participated in. We did find a sizeable number of people who could not easily be described as your basic anti-nuke but who are average citizens seriously concerned and relatively open-minded about the issue of transportation safety. Many of these people and, in many cases, state and local officials were understandably concerned, not so much with the existing structure of the nuclear transportation regulations and packaging requirements etc., but they were more concerned with relative factors which they felt struck closer to home, such as the selection of highway routes, emergency response, questions of reliability, etc. I think it is very important to communicate with this particular group of citizens and not dismiss them casually as anti-nuclear advocates just because they have questions and doubts about nuclear transportation safety.

In short, DOT's experience in attempting to communicate with the public during this hearing procedure points out again that a greater educational effort on the part of both government and industry is needed to give the public a better technical understanding of nuclear transportation and the risk involved, and also to help reduce the existing credibility gap between the public and government.

Statement by P. I. Ek

Is there a lack of communication between nuclear experts and the general public? If so, why?

In my opinion the situation has improved over the last few years. There is, however, a need for further improvement.

In order to obtain a better understanding of the present situation, I think it is necessary to look back to the beginning of the nuclear era.

What happened then was that those working in this field were highly influenced by the fact that most of the work until then had been done in the non-peaceful field. As a result, they were working under the assumption that every piece of information was classified. Consequently, this had to be kept away from those who had no clearance to receive such information.

Unfortunately in my opinion, this "tradition of keeping information classified" was transferred to the civilian side as that side started to develop. A lot of information was kept classified without any proved need for doing so.

Furthermore, I think that at the time it became evident t[redacted] there was no need to keep all these pieces of information away [redacted]m the general public. Scientists, engineers and not least the ma[redacted]ge-ment side and perhaps also politicians had found an advantage in itself in keeping the knowledge inside a restricted group of people. Why?

Well, we can just guess but I would assume that it was easier to take decisions without being questioned about the justification fo these decisions and that it was thus easier to obtain money for the work they wanted to carry out, etc.

I think that in many cases these factors were somehow linked together so that all parties involved in the development of the peaceful uses of nuclear energy found something positive in considering themselves to be members of this "unique club."

Being a member of this club gave at the same time a particular status to its members which was felt to be important for them and for their work. This is quite natural behaviour, I believe.

Well, this way of thinking continued for a long time and the amount of information on nuclear matters given to the general public was kept to a minimum.

Later on, when nuclear energy became more developed and started to play a role in the production of electricity, members of the general public began to ask temselves questions like: what is nuclear energy? Is it safe? Then, due to lack of knowledge, all sorts of speculations started.

At that time "the members of the nuclear club" felt that the ground on which they were standing was starting to shake. They there fore suddenly started making all sorts of statements about how safe the use of nuclear energy was. All sorts of safety measures were taken: there was no risk or at least the risk was so small that it was negligible, etc. Too late, these "experts" found out tha[redacted]me had overtaken them. Things started to happen that showed the [redacted]eral public that maybe nuclear energy was not as safe as the exper[redacted]had told them? Maybe they had been fooled? Therefore confidence in the nuclear experts decreased very rapidly.

So, when the experts finally understood what corner they had put themselves into - it was too late. The confidence was simply not there.

I sincerely think that these reasons are some of the reasons, but of course not all, why nuclear matters have been, and to a great extent still are, so controversial.

Now, what can be done to improve the situation? Among other things I think that:

- we must face the fact that there is very little information which really needs to be kept away from the general public;

- we must find a way to communicate our information to the general public which includes acceptance of the fact that being a nuclear [redacted]rt does not automatically mean that we are qualified to com-[redacted]cate our thoughts and knowledge to "non-experts." We should therefore make fullest possible use of experts on matters of communication;

- we must ignore the risk that information given is being misused - that will always happen. What we can hope for is that more information leads to more knowledge, better understanding and hopefully acceptance of nuclear power as a source of energy on the same terms as other sources of energy;

- finally we must learn to listen to the general public to be in a better position to understand their concern and to be able to deal with it.

Having made all these critical points, I must however confess that I think we are in a far better position today than we were only a few years ago concerning matters of communication.

Of course I realise that what I have said is somewhat controversial and to a great extent an oversimplification and therefore I will stop here even if the same sort of things could and should be said about both the general public - who should be blamed for their "little interest" in looking for information already available to them - and there is a lot of it - and the press, radio and television which have often used information given to them only with the purpose of making a "scope" to sell more papers or to increase the popularity of a particular programme. They also have something to learn.

Statement by R. M. Jefferson

First let me say that my remarks are strictly my own and do not necessarily represent those of Sandia National Laboratories or the US [redacted]artment of Energy. Further, let me say that my remarks assume th[redacted]ere exists within the industry a genuine interest in the continued assurance of public and industrial safety in the transportation of radioactive materials.

One of the realities that we in the nuclear business have had to learn to live with is the constant, although not universal, opposition to the nuclear power option. For quite some thime now we have faced a concerted, well-organised, and articulate minority who would have the world believe that we who are involved in the nuclear

option were willing agents of the devil. But, their continual cries of impending disaster arising from reactor accidents are falling on progressively less receptive ears. The unrelenting cry of unimaginable horrors is less effective today because of a number of factors; among them the Three Mile Island event. The effectiveness of the anti-nuclear movement has topped out and is now declining (although by no means disappearing). Now that movement sees the same signs. They too are aware of their fading impact and are concerned about it. In the last several years there has been a change in direction. Unable to convince the public that power reactors harbored the potential to destroy society, they have turned to other fronts. Initially it was wastes. What is the industry going to do with the wastes? Here again the public is realising that the problem is not tec[redacted]al but political in nature. Solutions are at hand when those in [redacted]ge choose to exercise the options available. But, another front [redacted] appeared, at least in the US.

To use the terms of the anti-nuclear movement, transportation is the "weak underbelly" of the nuclear power cycle. Here at last is an activity where they can convince the public that literally everyone is affected.

The public is reminded that "these incredibly dangerous material are moved daily on the same roads and rails upon which we travel". Further, these roads and rails pass nearby to our homes, schools and work places. The public is told that "this awful material is even moved through our most densely populated cities where a simple accident could kill thousands and cost billions." Quite a bleak picture as painted by the objectors. Further, they are not just word pictures anymore. Very professional television advertisements have been produced and shown. Legal battles have been won. Legislation is being promulgated. At last a new rallying point has been found for the "Movement." Further, if transport can be halted the whole nuclea option ceases to operate. You see, we are indeed an attractive targe Here at last is the method of mounting the fifth horseman and modern society has now accepted the addition of fear to war, pestilence, famine and death. The anti-nuclear movement feels that at last they have found a fear around which everyone can rally.

Now I have been talking in the context of the United States but I am afraid that my remarks will find some acceptance elsewhere. Unfortunately, we in the US seem to export our troubles more effectively than our products. But, that aside, if we in the radioactive materials transportation business are to be effective in providing safe, cost effective services, we must understand this attack and learn how to counter it.

I believe there are four factors which contribute to the problems of transporting radioactive materials. First is the fact [redacted]at we face aggressive antagonists. You do yourself and your ind[redacted] a disservice if you underrate these people. They make great u[redacted]of the technique of simplification. The public is hard pressed to under stand the complexities of studies or test programs so these people provide simplifications. In translating for the public they frequent drop the important details, misinterprete the data, color results or even lie to make their point. Those data which cannot be represented or distorted to support their point of view are understandably not presented in their argument.

A second factor is the truism that it is easier to generate the emotion of fear than it is to foster the attitude of trust. I do not feel the need to defend that statement but it is important to follow it to its conclusion. Emotion, any emotion, and fear is one of the strongest as is indicated before, is sufficient motivation for participation. Thus, the anti-nuclear protester gathers others who support that point of view out of the fear thus generated. To the politician this appears to be an interest group with which the political system must deal. The proponents have no constituency expressing their viewpoint, thus the small but active antagonist group develops a disproportionate voice in the politics of decision-making. Evidence of that concern among politicians was included in remarks made during the opening session of this symposium.

A third factor contributing to this problem or dealing with the opposition is the fact that in spite of claims to the contrary, the opposition is well-funded. There is enough money available that the agitator of today is employed full time as his trade. In the US at least they are viewed as public spitited citizens, thus eligible for grants from various philanthropic organisations, tax exempt trusts and some private sources. Further, a substantial proportion of their backing is derived from the US government under the guise of studies of various sorts. In the past year or so there has been pressure in the US Congress to fund these people at a level in some proportion to the agencies to which they object.

The fourth and possibly one of the most significant factors is the lack of uniform, understandable information for use by the public and the policy makers. In the field of transportation it seems that anyone with a driver's licence is an expert. So this abundance of experts not surprisingly offers a wide diversion of opinions ranging from well-informed to malicious. But, even opinion by knowledgeable experts is sometimes at odds with those of their colleagues. To the public this conflict implies either lack of competence on the part of the professionals or even the placement of private interests ahead of public safety. This public confusion places political pressure upon the regulators to take actions based upon public perception even when that perception runs counter to technical fact.

How does an activity such as radioactive materials transportation counter these factors? Let me retrace each factor and couple it with what my experience dictates as a possible solution.

To counter the lack of uniformity in information given to the public by at least those actually involved in this activity there are two considerations. For one we must offer unbiased information. By that I mean that we do not try to hide the real problems when they exist. Not only does that mean admitting the technical problems that exist but it also means that we, or the regulators, must not tell the public that political decisions are the solution to technical problems. The second aspect of uniform information involves understanding the public forum. Information for the public, to be useful, must be understandable by a non-technical public (without the benefit of interpretation). This also means that we must answer the real questions faced by the public, not the ones we think they face. Involved in this is the communication to the average person of how he or she benefits from the transportation of radioactive materials.

The second counter force involves funding. National or for that matter private firms must recognise that success in maintaining the ability to transport radioactive materials depends upon adequate funding of technical programs to address the technical questions and adequate funding or effort to communicate the technical effort to the public. Somehow, the risk of transporting radioactive materials has to be communicated in terms useful to the public. Searching for that methodology takes time, attention and money.

The third counter force I would propose is that the emotional bent of the objectors should be faced by unemotional professionalism by those chosen to represent our activity to the public. To adequate portray the realities of our business the representative must [redacted]st of all be well-prepared. Such preparation can enable one to [redacted] being confused technically but a good debater must avoid being rattle or thrown off balance by personal attacks as well. Further, in speaking for the continued safe transport of radioactive materials one must be willing to admit limits of knowledge. It is very dangerous to speculate since the activist may lead you to say things which destroy your credibility.

Lastly, it is important to counter the aggressive nature of the opponents with aggressive actions on your part. Before speaking in public you should know your opponent. Know his technical arguments and their weaknesses. Know your opponent's personality and those weaknesses as well. When before the public be aggressive. Contrast the half truths offered by the opposition with the whole truths offered by technology. But do not spend all your time defending. When the opportunity arises attack the goals, motives and reasoning of avowed opponents. On the other hand avoid attacking the interested member of the public. The public and the press have honest questions that deserve honest answers. Deciding which tactic to use when is an important part of the communication effort. Finally be aggressive about getting your story out to its intended audience. Be freely available to the press. Take the time to explain to them much more detail than you can hope to transmit to the public. Their education is often a slow and difficult process. Be aggressive in contacting the public instead of waiting for them to ask you. You may have the best story in the world but until someone hears and understands it you haven't any story at all.

Let me close with one very important point. When it is all said and done you must remember that the public is right. When it comes to a showdown between technical fact and political fact - politics will win every time. Our task is to make sure that in the field of transportation of radioactive materials the political facts are in harmony with the technical facts. If you make that your goal and are successful at that one point, all else will fall into place.

[redacted]

Statement by R. F. Barker

The general public is concerned over real and perceived hazards in transportation of radioactive materials. The best response to such concerns is to set out the facts for everyone to see; but the facts must be definitive and understandable. Studies show that the

risks from transportation of radioactive materials are small, much smaller than the risks from many other activities which are readily accepted. Nevertheless, more effective communication with the general public is needed to allay their fears, real or perceived.

But assuring safety in transport and convincing the general public that it has been achieved are two different things. There must be free communication among members of the many groups involved - shippers, carriers, receivers, regulators, transport workers, police, emergency crews and members of the general public.

The shipper must assure that the material is prepared so that it can be transported safely. The basic responsibility for the material rests with the shipper. He originated the material, he knows what the material is and he knows its properties. And he must communicate to the carrier, and others, the necessary information about the materials - any special properties, unusual hazards, what to do in an emergency, special precautions (if any) - so that whoever is handling the material is informed. The regulators, shippers and carriers share the primary responsibility to inform the public - to provide the factual information in understandable terms with perspective.

Because the shipper knows most about the material he is shipping, he does most of the communicating. Regulators communicate with the public in promulgating regulations and attending to compliance and enforcement. Shippers communicate with the public - notification, labels on packages, package appearance, responses to inquiries, preparing for and handling emergencies. Carriers communicate with the public - placarding, personnel, appearances, responses to inquiries, preparation for and handling of emergencies.

Communication - Risk. *The number of shipments is increasing* steadily and the number of persons involved in transportation is necessarily increasing rapidly. In spite of every practical and reasonable precaution being taken, the occurrence of some incidents is unavoidable - absolute safety in any operation is not possible. As a result, the number of incidents may be expected to increase.

And, because it is so unusual to have any excessive radiation exposure in transportation, each incident that does occur is considered a major breach of safety. It is not only misleading, but adds to this problem to tell members of the general public that nothing can happen in transport. However, the public is not reassured by a heavily qualified statement such as it is "unlikely" that anything major will happen.

Certainly, great care must be exercised to communicate facts about safety in transport. Choosing terms meaningful to the public to express risk from an activity is difficult. It is especially difficult for those involving radiation because the effects of radiation have been so thoroughly studied that we can identify and quantify so many different health effects. The public has few other hazards about which as much is known for comparison.

Method

It is important to be aware that communication occurs other than by words. As important as selection of the words is the way in which

they are said, the tone and attitude of the one saying the words. Also the speaker's appearance and his body language adds to the message:
Calm, careful statements are reassuring, whereas the same words offered in a hasty, emotional manner can be badly misunderstood and misinterpreted.

It is also necessary to remember communication involves two parts speaking and listening. What the listener hears may be quite different from what the speaker says. It is affected by his knowledge and experience on the topic, everything else that is going on at the time in his mind, in the vicinity, in the world as well as past and recent events to which he has been subjected - family affairs, job, friends, club, newspaper stories, movies, television programs, etc. It is also affected by the number of other items which the listener is receiving at the same time.

Communication with the general public occurs in many ways:

1. One-to-one, person to person, which is usually the easiest and most effective.

2. With small groups, the circumstances range from friendly to adversary groups where there can be a useful exchange with questions and answers.

3. With large groups or the media (television and radio) with no opportunity for exchange.

There also are other times when communication with the public is secondary:

- a formal hearing where one provides very specific answers to specific questions in dealing with a legal issue;

- an appearance before a legislative body where one provides a carefully thoughtout, positive (but brief) picture of the topic with direct, honest answers;

- a media interview (radio or television) where one offers his story prepared well in advance, whether long or short, in response to any question, whether or not it is pertinent.

Statement by M. Blechschmidt

This 6th International Symposium on the Packaging and Transportation of Radioactive Materials differs quite considerably from other five symposia from several points of view. The previous symposia took place in the United States of America. They were mostly held in places where there was a large nuclear research centre in the immediate neighbourhood. The present symposium is taking place in Europe, in the centre of Berlin and this means that the public is directly involved in the questions which are being dealt with here.

Of course public organisations, politicians, lawyers, trade union delegates were also represented at earlier symposia. There

were also press conferences and radio and television interviews. However, this time there is evidence of more intensive public interest, proved by this panel discussion, the subject of which is communication between experts and the public.

I believe the main intention of the following discussion is to find out whether communication up to now - even regarding the sixth symposium - is felt to be sufficient or insufficient. Should criticism arise, then suggestions should be made for improvement. It should be born in mind that standards exist for packaging and transportation, standards which are based on world-wide experience and which are implemented in the regulations for the individual modes of transport. If it is thought that communication, with regard to this sometimes rather dry subject, could be improved, then constructive criticism is needed.

Questions and Comments by Panel Members

Question by R. F. Barker: I would like to raise the question to Dr. Baier as to whether there is a reason why journalists present just the bad news and also why is it always, or seemingly always, that, very clear as our statements are, the journalist seems to misquote them?

Answer by W. Baier: There are various reasons. The first is the following: I cannot imagine a newspaper with the early morning headlines "The ICC is still standing." The second reason is that there are hardly any journalists writing for the general public who understand more than a little about what they are writing. I have already said that there are a number of journalists who would rather cross the street when they see an expert coming. This is a problem we must solve. It would be possible to show films to journalists who are unsure of the facts, and of these there are quite a few, films as they have been shown here, in which, for example, containers are brutally hurled against a concrete wall. They should see this happening. Perhaps in this way there would be better results if journalists could be shown what is actually going on, rather than trying to understand explanations. For we all know, that with the little word "if" catastrophes can be invented which will never really happen.

Comment by W. Baier: I would like to make a comment about the statement by Mr. Ek. In Germany the atom bomb has never been developed. In this respect it is somewhat difficult for the German people to think in this direction, as he said. As a comfort, perhaps it should also be pointed out that there was no less a man than Robert Jungke, who discovered "good" plutonium. If you have read his "Atomstaat", you will know that "bad" plutonium is reactor plutonium. In Germany there are 35 tonnes of bomb plutonium, which are "good" as they do not endanger citizens' rights. If I might add a small piece of information to this general moral uplift, encouraged by Mr. Jefferson and what he has said, our well-known friend Armury Lovins is, or was recently, lecturer at a Holistic Life Seminar at which, as far as I know, not only courses on "soft energy" but also on psi and astrology were, and still are, held.

Comment by Y. Sousselier: I would like to add something to what has been said by Dr. Baier and which, I believe, considerably complicates the journalist's task. I hope you will allow me to adopt a little of the journalist's viewpoint. It is a certain fact that when nothing happens, the public is not interested. I think the same phenomenon must exist in other countries as in France, but we have often tried to organise "information days" or "information sessio in order to provide the public with information about nuclear energy, transport, etc. There are always difficulties in finding people willing to take part, except of course some members of the "Commissariat à l'Energie Atomique" or "Electricité de France". People are only interested in the problem when there is an incident or an accident and at that point, the emotion which they feel makes it difficult for them to properly understand the information which is given to them. I do not want to say that we should not try to provide information. On the contrary, I believe that we should not hesitate to preach in the wilderness even if no fruit is borne immediately.

Questions and Comments from the Auditorium

Comment by A. L. Kaplan, USA: My remarks are applicable to the situation in the United States. Technical experts appear always to try to be absolutely technically correct when communicating with the public, rather than to speak in generalities which might be more easily understandable to non-technical persons. Being as absolutely technically correct as possible is, of course, very important especially in performing technical work.

However, I believe that a technical person does not like to speak in general terms which might be more understandable to the general public, especially concerning his technical expertise, probably because there are always exceptions to general statements and therefore such general statements might be subject to technical criticisms by colleagues.

Also, I believe that many technical persons are not skilled in communicating with the general public in the general, non-technical terms which can be most easily understood by non-technical persons, particularly when translating technical matters vital to public interest into understandable political terms.

My experience in the area of communicating with non-technical persons, particularly in public forums, indicates to me that we technologists are not the only intelligent people in the world, that most non-technical people are anxious to better understand technologies, and finally that when non-technical people receive technological explanations in terms which they understand, they are generally positive in their responses.

Therefore, I believe that technical experts need to learn how to communicate with the public in understandable terms. Also, I believe that, because of past experience by non-technical people in being exposed to highly technical topics in terms not easily understood, restoration of public confidence in technologies will take a long time to accomplish and will take a large effort by technical experts with good communication skills.

Comment by B. Wright, United Kingdom: We must recognise that the problem is not technology; nuclear power is safe, fuel transport is safe. The problem is power, political power, power of groups. Most people, however, perceive a technology problem.

The problem therefore is communication. Communication barriers can be removed by choice of speakers, simplifying facts and content. The general public does not want to know about reactor design. My wife does not want to know how the car works. She wants to know that it will stop when she puts her foot on the brake. So keep it simple.

It is important not to speak for too long, to allow time for questions and to listen carefully. The facts and hypotheses of the antinuclear groups should be challenged, and possibly also their motives. The hazards must be explained and compared to other hazards. The benefits should be explained as well as the consequences of doing without the benefits of technology.

Comment by H. W. Hübner, FR Germany: Communication between experts and the public needs the medium of language. Therefore, the language must be clear and simple, in order to be also understood by "non-experts". This does not mean that clear and simple explanations by experts should be regarded as non-scientific.

If experts express themselves clearly and simply, then they can also ask that opponents of the nuclear industry speak clearly and simply.

Comment by W. Baier: I would like to say something about this comment. I believe this is where the problem really lies, but it is the experts who are affected. A few days ago, during a press conference at the nuclear research facility in Jülich, the controversy with the Instiute of Ecology in Freiburg was propagated. The reporter from the local radio station Rias told his partner at the microphone over the air: "I haven't understood that!" That is exactly the problem.

Now concerning the second point you mentioned. It is with pleasure that I quote Prof. Kind, who some time ago here in Berlin was addressed during discussions by experts on safety and replied coolly smiling: "I know no discussion between experts!" And that was the end of the question. This requires, however, considerable training and that is very uncommon.

Comment by J. A. Gatley, United Kingdom: It is necessary to resolve the conflict between the need to communicate in simple terms to the public and the need to be factually correct. Unfortunately, the facts are complicated. We can perhaps resolve this conflict by thinking in terms of different languages: the language of ordinary communication, and technical language.

The difference can be illustrated by an example drawn from the Common Law (traditional law) of England. An English judge in a trial by jury must instruct the jury not to convict the accused unless they are sure "beyond reasonable doubt" that he is guilty. As there can be no absolute certainty, "reasonable doubt" must be interpreted as meaning a chance of less than a few percent. The law claims that the "reasonable man," speaking ordinary language, would regard such risks as negligible.

If we make clar, then, that we are speaking ordinary language, we are fully justified in saying that the risks are negligible. When using technical language, we must be precise and use quantified probabilities. This distinction must be very definite; otherwise we shal confuse both the public and ourselves.

Comment by W. Baier: Perhaps I may lodge a complaint. My complaint is that it has already twice been stated that it is impossible to express technical facts in simple language. It is possible! That is my job in fact. That is why I know what I am talking about. Mr. Wright said of course quite rightly that his wife is not interested how the car works. She only wants to know what she should do when she wants to stop. That is quite justified and is also import[redacted]ate for such a discussion. The public is not interested in the i[redacted]ate details of a technical process. They just want to know approxi[redacted]tely what happens, and that can be explained in everyday language.

Question by C. P. Haigh, United Kingdom: Most of the panel of speakers asked for a wider dissemination of a uniform set of information. It has been my experience that the "anti-nukes," and particularly the anti-transportation activists, choose very selectively from the information that the nuclear industry releases. They are therefore able to build a totally misleading picture which they then communicate very effectively to the general public by a variety of methods. Can the panel suggest any method in which the information we wish to give to the general public can reach that general public in an intact state?

Answer by Y. Sousselier: I think that is a very good remark and anyone who is involved in this problem of communicating information is faced with the possibility of this information being misused I know very well that in my country, it was sometimes the most important reason why certain politicians responsible did not give too much information. But I think this stage has now passed.

I would like to recount how I personally was induced, along with my colleagues at the "Commissariat à l'Energie Atomique" to try to reconcile this information with the possibility of it being misused.

I believe that first of all, the information to be given should be prepared. Secondly, we must try to see and determine to what misuse this information could be put. Obviously, this is the most difficult step but it is by trying to obtain criticism of one's own information that we succeed. In some cases it is necessary to amend information. In particular, we should avoid quoting figures, facts which could be misinterpreted if quoted alone. In other words, and perhaps I am being rather vague, I think we should be most careful about the use to which simply cutting part of the information could be put. For example, care should be taken that a sentence wh[redacted] might be cut short would seem cut short. It is a question of [redacted]t. I think that by doing this we will succeed in the majority of [redacted]ases, perhaps not in completely avoiding misuse of information but in considerably lessening the risk. In this way we have, I think, in Franc considerably reduced the misuse of the information which we give.

Answer by R. M. Jefferson: One of the techniques that is used by the advertising industry is to very selectively present information to those people who will handle it the way you want it handled. I find the same technique to be useful in handing out information to

the media. That requires that you study the individual reporters with whom you deal and find out what their prejudices are, how they will likely present your story and to utilise those who will present your story the way you want it presented and simply not talk to the rest of them. I have found that using that technique, those who have been very critical of what we have done at Sandia have indeed, because they want to be given the information, been very willing to change their viewpoint on many occasions and present information in a more realistic manner than they had originally set out to do.

Concluding Remarks by R. Neider

We have seen from the remarks by the panel members that there are at least three areas in which communication between experts and the public, or rather laymen, is needed. The first is the transportation of radioactive materials. The driver of the vehicle transporting the radioactive materials should be informed, so should the railway official and the airport employees, all of whom have no basic technical knowledge but who are worried. This is one area and perhaps the most simple.

The second area is communication with the general public, people who are to a large extent, I might say, fairly indifferent and therefore quite willing to digest information and to listen.

Finally, the third area, the most difficult area, is contact, relations with nuclear industry opponents. Those of our citizens who from the very beginning because of certain opinions or prejudices are always against, many of whom can, as we have heard, organise very well, are very skilled in discussion and choose their methods very carefully. What is the best way of dealing with this group of people? How can we restrain them? We have heard that we should attack from our side as well. Of course this should be done carefully and with a certain aim in view. We should not always remain on the defensive.

The second area concerns language. What language should be used, in which way should we communicate with these various groups? That is certainly different for each of the three areas. But in every case we should try to speak simply with no scientific terms which the layman will not understand. We must express ourselves with confidence and be confident. We should not speak from above, as from a platform. That is unfortunately customary, but I can fully understand the remark which was made in this respect. This is often a hindrance but on the other hand, it is only possible to speak to a large number of people when placed a little higher.

We must encourage understanding for the man in the street. The official information given out must be very well prepared so that there is no way it can be misinterpreted or misused. At the back of our minds we must always remember what Mr. Jefferson said, that what the public thinks, the politicians will in the end do. It is our job to convince the public that the transport of radioactive materials constitutes no danger so that the politicians draw up the right regulations.

Finally, we should learn from the experiences gained from this symposium, from this panel discussion, from the Public Evening, from our policy used in dealing with the press, and put this experience into use at future symposia, not only PATRAM symposia but all conferences on nuclear energy problems. An attempt should be made, probably a much greater attempt than we have made, to include the general public and to enable this public to understand what is going on.

I am fully aware that what I have said covers of course only some of the points. It should provide an impetus. I think every one of us who sees himself as an expert in one of the fields discussed here, should feel the duty to convey the information and knowledge he has to the general public. That begins with the family, then friends, then further. Everyone should practise simply selling his science and I agree with Dr. Baier that everyone should understand his occupation. When things are understood, then they can be simply explained. It is certainly not easy.

PUBLIC EVENING

RADIATION PROTECTION DURING THE TRANSPORTATION OF RADIOACTIVE MATERIALS

R. Neider
Bundesanstalt für Materialprüfung (BAM)
Unter den Eichen 87, D-1000 Berlin 45

Introduction

The application of radioactive materials in science, technology, and medicine and their operational function in research reactors and nuclear power plants presuppose the land, water and air transportation of these materials, including irradiated and non-irradiated nuclear fuels.

Radioactive materials are used in science just as tracker dogs in police work as tracers for investigating physico-chemical and biological processes; they have made a substantial contribution in recent decades towards increasing our understanding of such processes. In the technological field, radioactive materials are applied to investigate technological processes in, for example, process chemistry, or to identify the causes of friction-induced wear in mechanical engineering. They furthermore serve as a source of radiation for radiographic purposes, i.e. for the non-destructive testing of all types of construction components for hidden internal flaws, a function similar to the X-ray examination undertaken in medical work on human beings. In modern medicine, radioactive materials are used extensively for both diagnostic and therapeutic purposes. The functioning of almost all internal body organs can now be monitored with the aid of radioactive medicines. It is also possible to use the ionizing rays emitted from radioactive materials in the treatment of carcinomas.

There are at present 14 nuclear power stations in operation in the Federal Republic of Germany; their power output covered ca. 14 percent of federal German electricity consumption in 1979. In addition, some 30 research reactors are being operated for the production of radioactive substances and for a wide range of research projects.

All the applications of radioactive materials described above require that such materials be transported to a greater or lesser extent on public travel routes. Since the available statistical data on the nature and volume of radioactive freight are unfortunately not comprehensive, estimation is a necessary part of all work in this field. Table 1 gives the estimated figures for freight items with a radioactive content transported in 1975 in the USA and the Federal Republic of Germany re-

spectively. [1,2] Similar figures are available for other industrial countries. Yet a comparison of these figures with those for all types of dangerous freight reveals an interesting point: as regards the USA, it appears that radioactive materials and nuclear fuels account for only 2 percent of all dangerous freight. This ratio is believed to be likewise representative of other countries. Moreover, the potential danger represented by the individual items is found to cover a considerable range, the large majority of all radioactive goods being those with a relatively low degree of radioactivity destined for application in medicine.

The deleterious effect of radioactive materials on the [redacted]n body has been known for many decades and has been much more tensively researched than the effects of many other noxious materials. The ionizing rays emitted from radioactive materials are known to be able to cause biological-medical damage in the event of body penetration from outside or incorporation of a source of radiation by physico-chemical interaction within the body. In view of this fact, when radioactive materials are transported on public travel routes, precautions as to the design of the packaging and the transport procedure have to be taken to ensure that no damage is inflicted on the transportation workers concerned, the general public, or material goods such as unexposed film. The importance of such precautions was acknowledged as early as 1961 in a series of regulations to govern the transportation of radioactive materials published by the International Atomic Energy Agency (IAEA) in Vienna. Since traffic in radioactive materials is in many instances border-crossing traffic, it was exceedingly important that a uniform regulation for the safe transportation of radioactive freight be enforced at international level. The IAEA recommendations have been reviewed on many occasions over the past twenty years and updated to meet the requirements of the latest scientific and technological advances. A further revision is taking place at the present time. They have been extensively adopted in regulations introduced subsequently at national and international level, the result being that uniform regulations for the transportation of radioactive materials are now in force on a worldwide basis.

We shall now proceed to examine in detail the potential deleterious effects of radioactive materials, and the IAEA recommendations for effective protection against such effects. The paper will conclude with an examination of the findings of risk analyses relating to the transportation of radioactive materials.

Potential Deleterious Effects

The personnel of transportation enterprises (e.g. rail[redacted], ro haulage, airline, and shipping companies), and individual members of the general public may be exposed to radiation as a result of the transporation of radioactive materials. Under normal transportation conditions, such exposure would in principle result from the continued emission of ionizing rays from the individual freigh items and from incorporation of leaked radioactive material. In th event of an accident, the radiation exposure may be greater. How c

ionizing rays affect the human body?

Figures 1 and 2 illustrate the biological chain reaction which takes place in the event of irradiation and the various types of damage which ionizing rays can inflict on the human body. [2] The primary physical processes which lead to the ionization and excitation of atoms and molecules in living tissue are followed by molecular changes in the basic elements of life, e.g. proteins, enzymes, and nucleic acids. These molecular changes in turn exert an influence on the metabolic and division processes which take place at cell level and represent an essential part of life. Depending on whether this influence is exerted on body cells or germ cells, the nature of the ensuing damage is either somatic or genetic. Whereas somatic damage is discernible on the individual exposed to the radiation, genetic damage finds expression in that individual's progeny. Four types of radiation damage are known:

1. Acute or short-term damage which becomes evident shortly after radiation exposure (e.g. acute radiation sickness);
2. Non-malign, long-term damage which becomes noticeable in the individual years after radiation exposure (e.g. fibrotic tissue changes, increasing opacity of the lens of the eye, diminished fertility);
3. Malign, long-term damage which becomes manifest in the exposed tissue only after a period of years or decades of latency (e.g. leukemia following irradiation of the bone marrow, tumours in other body tissues);
4. Genetic damage resulting from specific mutations in germ cells after irradiation of the gonads (testicles or ovaries) which emerges in the progeny of the exposed individual (e.g. skeletal anomalies, retarded mental development, mutations of eye colouring, hereditary diseases).

The first three types of damage listed above can be classified as somatic damage, their manifestation taking place in the individual exposed to radiation. Damage resulting from prenatal radiation exposure is also classified as somatic damage.

The carcinogenic and genetic effects are referred to as stochastic or random effects of radiation, since it is the probability of their radiation-induced incidence and not the gravity of the damage which depends on the radiation dose. In contrast, in the case of acute damage and non-carcinogenic long-term damage, it is the gravity of the damage which depends on the radiation dose. These types of damage are classified under the general term non-stochastic effects."

One decisive factor in the evaluation of the health damage caused by ionizing radiation is the risk of a radiation-induced shortening of life-span. This risk is known as the mortality risk. In order to determine the permissible threshold values for the radiation to which personnel handling radioactive materials and the general public may be exposed, it is essential that the relation between the radiation dose absorbed by a tissue and the

probability of a manifestation of radiation-induced damage in that tissue can be expressed as a known value. This dosage-risk ratio differs according to the type of radiation damage in question, as is indicated in Figures 3 and 4. In the case of stochastic damage, such as carcinoma and genetic damage, there is no threshold below which no effect can manifest itself. Indeed, the probability of damage increases as the dose increases: under the least favourable conditions, one could speak here of a linear c relation, that is to say, if the dosage is doubled the probability of damage will be doubled, or, in other words, if the dosage is doubled the proportion of individuals suffering adverse effects among the total number of irradiated individuals will be do[illegible]ed. In the case of non-stochastic effects, there is a lower thr[illegible]ld at around 100 rem below which no radiation-induced health d[illegible]ge ensues. The corresponding values for the cancer risk in the event of radiation exposure are given in Table 2. These should be compared with the mean cancer risk figures statistically calculated for the population of the Federal Republic of Germany which are presented in Table 3. The IAEA regulations, which will be examined in greater detail below, proceed from the assumption that the maximum radiation dosage to which an individual may be exposed in the event of an accident with a radioactive substance is 3 rem.

The reference value of 125 for the overall cancer risk in the event of equal-intensity whole-body radiation indicates that if one million individuals were exposed to a whole-body radiation dose of 1 rem, 125 of those individuals would develop cancer and die. Using this value, it is possible to calculate as follows the risk incurred by an accident with radioactive freight as a result of which, say, 10 individuals are exposed to a radiation dose of 3 rem:

$$\text{radiological risk: } \frac{125 \times 3}{1{,}000{,}000} \times 10 = 0.00375$$

This figure permits the calculation of the number of accidents required before, according to the laws of probability, one individual would die of cancer. The number of accidents thus arrived at is 267. It will be shown below that the incidence of accidents in which the packaging is destroyed and 10 individuals are exposed to a radiation dose of 3 rem each is extremely improbable.

IAEA Regulations for the Transportation of Radioactive Materials

The basic principle underlying the IAEA regulations [4] is that the design of packagings for radioactive materials with a radioactivity exceeding specific threshold values is to be such as to withstand the strain of an accident, a[illegible] in the case of that for radioactive materials with an acti[illegible] below these values, such as to ensure that the radiation effects ensuing after destruction of the packaging remain minimal and do not seriously endanger human life or health.

This principle is translated by the following practical precautions. Two basic types of packaging are admissible (see Table 4). The A-type packaging may be used only for radioactive materials having an activity not exceeding the value prescribed in the regulations, this value being determined on the basis of the radiotoxicity (toxicity relating to radiation and chemical properties) of the radioactive substance in question. It is known that radioactive materials are not all equally toxic: radium, plutonium, and strontium for example, are very much more toxic than radioactive hydrogen or carbon having the same incorporated activity levels . If the materials are considered to be exceptionally safe, for example, if they are contained in a sealed rust-free steel capsule, the threshold value for the admissible radioactivity in an A-type packaging relates instead to the external radiation. A-type packagings must be so designed and constructed that they withstand the normal strains and stresses of transportation. Here, account is taken of the fact that in the event of a major accident, the packaging will be totally destroyed: the threshold value for permissible radioactivity is determined on the basis of the assumption that if such activity is released, no undue damage will be inflicted on human health. The basic thesis is as follows: the packaging is, say, crushed by a moving heavy vehicle and totally destroyed. On destruction, 1/1000 of the powdered content will be released into the surrounding area; the greater part of the radioactive material will remain in the immediate vicinity of the destroyed inner container. Of the radioactivity released, not more than 1/1000 will be incorporated by an individual present at or near the site of the accident. Accordingly, the threshold value for the radioactivity content of A-type packagings is so determined that the radiation dosage caused by the incorporated one-millionth part does not exceed 3 rem. This value was adopted on the basis of the recommendation made by the International Commission on Radiological Protection (ICRP) to prescribe the threshold value for the maximum permissible dose within a period of three months at 3 rem. [8] The health danger which this dose represents has already been examined above. And as was also pointed out above, a distinction is made between radioactive materials of special form and so-called "open" radioactive materials. A number of conditions have to be fulfilled by materials alleged to be of special form and fulfilment of these conditions has to be sanctioned by an official test which requires that, even in the event of a major accident, the radioactive material will not be released from the packaging and, accordingly, no danger of incorporation exists. In the Federal Republic of Germany, these tests are conducted by the Bundesanstalt für Materialprüfung (BAM) which is at the same time the certification agency.

If the radioactive materials to be transported have an activity exceeding that permissible for A-type packagings, the use of B(U)-type packagings is mandatory. This type of packaging must be able to withstand strains and stresses of a magnitude conceivable in the event of a major accident. Observance of this regulation is controlled by way of an official construction-type registration procedure, this being carried out in the Federal

Republic of Germany by the Physikalisch-Technische Bundesanstalt working in cooperation with the Bundesanstalt für Materialprüfung. Type-B items for transportation have to be accompanied by the official certificate sanctioning the packaging before they may be surrendered for transportation by road, rail, or air. Institutions which undertake their own transportation must ensure that the construction of the packagings used complies with the relevant specifications. The activity leakage rates permissible after testing are so low that the radiation dose to which individuals incorporating 1/1000 of the total released radioactivity via inhalation are exposed does not exceed 3 rem. Higher radioactivity leakage rates are admissible in the case of B (M)-type packagings, f[redacted]hich multilateral approval must be secured, since additional admi[redacted]ra-tive measures prevent unauthorized individuals from approachi[redacted] the possible site of any accident.

The IAEA regulations furthermore contain requirements as to the maximum permissible dose rate on the outside of the undamaged packaging and the maximum permissible leakage rate. This is a further precaution to ensure that the radiation to which transport staff and the general public are exposed is kept below the threshold value of 500 mrem/a recommended by the International Commission on Radiological Protection (ICRP). This level of radiation exposure involves risks such as those found elsewhere in life at home and at work. Additional regulations govern the labelling of freight items. Such items must bear labels specifying the content of the package and the dose rate at 1 distance from the outside surface (see Figure 5), so that those transporting the items can handle them accordingly and not, for example, choose to use such a package as a breakfast table.

Risks Involved in the Transportation of Radioactive Materials

Experience gained over the past two decades has shown that the above IAEA-prescribed procedure operates efficiently: nowhere in the world has there been a grave accident in which individuals have suffered any discernible health damage as a consequence of escaping radioactivity. In the majority of the transport accidents in which radiation was released, it was found that the regulations had not been duly observed. Data on accidents in which radioactive materials were involved have been collected and evaluated as systematically as possible in a number of countries for many years now. A series of lectures to be held within the framework of this Conference will examine the conclusions to be drawn from the accidents which have happened and will report on research being undertaken on the risks involved in the transportation of radioactive materials. J. D. McClure and E. C. Emerson of the USA have surveyed the accid[redacted]ts which have occurred during the transportation of radioactive[redacted] materials in the USA during the past ten years.[2] They report 85 accidents during transportation, 110 accidents during the handling connected with transportation, and 465 associated incidents. These associated incidents incurred no damage to packagings and no release of radioactive material. As regards the 85 transport accidents, of the total of 711 freight items involved only 41 had their packaging damaged so severely that radioactive

material was able to escape. Of these 41 items with damaged packaging 38 had an A-type packaging, that is to say, the radioactivity of their content did not exceed specified threshold levels. Yet these figures must be viewed in the light of the total number of freight items with a radioactive content. The total number of such items transported in the USA in the past ten years has been calculated at ca. 15 million. From this it can be deduced that, on average, three out of every million such items were involved in transport accidents accompanied by a release of radioactive material. There is no evidence of any noticeable physical damage having been suffered by any individual as a result of radiation exposure in connection with these accidents and only probability calculations can be made regarding health damage possibly incurred by the transport workers concerned and individual members of the general public. And whereas it is possible to make fairly accurate forecasts as to the number of fatalities and injuries anticipated in any given year from normal traffic accidents on the basis of the reliable statistics available for previous years, the damage to human life and health which will be caused by the transportation of radioactive materials cannot be forecast with the same facility. It is, however, possible to make fairly accurate estimations regarding the upper limits to such damage. For example, taking the number of transport accidents which occurred in recent years in which radioactive materials were involved as the basis for conjecture, if one assumes that ten such accidents were to occur each year in the Federal Republic of Germany and that in each accident one person were to receive a radiation dose of 3 rem, this would be equal to a situation in which three persons each receive a radiation dose of 1 rem. From here, the radiation risk factor stipulated by the International Commission on Radiological Protection (ICRP) presented in Table 2 permits us to calculate that loss of life would be inflicted on four persons in 1,000 years. It should, however, be pointed out that in view of the actual number of accidents occurring, for example, in the USA, the assumption of ten accidents per year in the Federal Republic of Germany is an extremely conservative point of departure and a gross overestimation.

Yet in making such calculations, attention should also be paid to the effects of normal, i.e. accident-free transportation since it must be assumed that a certain number of individuals among the transport workers and the general public are nevertheless exposed to some radiation from the freight items. If one computes the total of such radiation doses and multiplies this figure by the ICRP risk factor of 1.25×10^{-4} rem^{-1}, one arrives at a figure which indicates the probable number of fatalities from cancers induced by such radiation. A very comprehensive study has been conducted recently in the USA on the risks involved in the transportation of radioactive materials, the findings of which are presented in Table 5. However, these figures must be compared firstly with those for fatalities and injuries resulting from conventional causes in such accidents and, secondly, with those for cancer deaths which are statistically attributable to natural or background radiation exposure.

It has been found that the damage to life and health resulting from conventional causes in accidents involving radioactive materials is some 50 times greater than that induced by radiation. Likewise, if one applies the calculation formula for cancer deaths from exposure to natural radiation (Table 6) using a mean radiation exposure value of 110 mrem per person per year, one arrives at a figure of 825 cancer deaths per year in the Federal Republic of Germany which can be attributed to natural radiation exposure.

A study of particular interest which was commissioned by the relevant US authorization agency and which has been the subject of two accounts during this Conference, [8,7] examined the situation which would arise as a consequence of the regular transportation of radioactive materials in general and of irradiated nuclear fuels in particular through densely populated urban areas. Higher values were found to obtain in this case. For New York, the figures given were slightly less than one fatality per year and one case of genetic damage every two years. All these figures should be considered as purely statistical values which were arrived at on the basis of known statistical data on accident incidence and its consequences, assumptions regarding the performance of packaging materials, and the above-described dose-effect ratios. Although it was found that isolated accidents in urban areas could, under certain circumstances, result in up to 11 fatalities, the probability of such a grave accident is extremely remote, namely, twice in 300 million years.

Risk calculations of this type and discussions on accident and mortality statistics are inclined to give the impression that those who deal with such figures count human lives as they count inanimate objects and fail to display the necessary respect for each individual human life. Yet it is precisely in order to be able to guarantee each such human life a sufficient degree of protection that the risks involved in specific activities must be calculated, and this preferably before the figures are substantiated by the statistics of real-life accidents. For then there is still time to take action to reduce the risks involved in the transportation of radioactive materials, although, as has been shown, these risks are already extremely small.

References

1. Hartwig, S. et al.: Weak Point Analysis and Risk Assessment in the Transportation of Radioactive Materials in Germany. Proc. 5th International Symposium PATRAM '78, Las Vegas, vol. II, p. 968.
2. NUREG - 0170 vol. 1: Final Environmental Statement on the Transportation of Radioactive Materials by Air and Other Modes, Dec. 1977, p. 1.18.
3. Deutsche Risikostudie Kernkraftwerke, Hauptband, Verlag TÜV Rheinland, 1979.

4. Regulation for the Safe Transport of Radioactive Materials, 1973 Revised Edition (as amended), International Atomic Energy Agency, Vienna, 1979.

5. McClure, J. D. and Emerson, E. C.: A Review of U.S. Accident/Incident Experience Involving the Transportation of Radioactive Material 1971 - 1980, PATRAM '80 Compacts, p. 356.

6. Finley, N. C. et al.: Environmental Impacts of the Transportation of Radioactive Materials in Urban Areas, PATRAM '80 Compacts, p. 317.

7. Eisenberg, N. A.: Transport of Radionuclides in Urban Environs, PATRAM '80 Compacts, p. 321.

8. Recommendations of the International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation (1959), Pergamon Press, p. XXXI.

Tables and Figures

Country	Radioactive materials	Nuclear fuels
USA	1,100,000	400,000
Fed. Rep. Germany	140,000	800

Table 1

No. of transportations of radioactive materials and nuclear fuels in 1975 (estimated figures)

Organ/Tissue	Radiation-induced cancer risk/Equivalent dose[1] (Incidence per 10^6 persons per rem)	
	Anticipation range[2]	Reference value ICRP 26 (1977)
Red bone marrow (leukemia)	15 - 40	20
Breasts	15 - 40	25
Lungs	10 - 30	20
Bone surface (bone cancer)	5	5
Thyroid gland	5 - 10	5
Digestive organs, aggregated	20 - 50	50
Others, aggregated	10 - 30	
Total cancer risk from uniform whole-body irradiation	80 - 200	125

1/ Mean dose equivalent in respective tissue
2/ Values adjusted in accordance with age structure of federal German population

Table 2

Radiation risk for additional cancer mortality

Site of manifestation (tissue)	Cancer incidence per 10^6 persons per year	Relative incidence 1/ in %	Mortality risk 2/ (cancer incidence per 10^6 persons)
Respiratory tract	400	3.3	29,000
Stomach	350	2.9	25,000
Intestine	350	2.9	25,000
Thymus gland 3/	180	1.5	13,000
Lymphatic and hemopoietic [redacted]	150	1.2	11,000
[redacted]kemia	70	0.6	5,000
Pancreas	75	0.6	5,400
Kidneys	70	0.6	5,000
Liver	45	0.4	3,200
Bone 4/	10 - 15	ca 0.1	700 - 1,100
Thyroid gland 4/	5 - 10	ca 0.05	350 - 700
Other tissues, including unidentified types	ca 800	6.6	ca 56,000
Total body (aggregate risk)	2,400	20	173,000

1/ in relation to total number of deaths

2/ assuming average life-span of 72 years

3/ both sexes

4/ estimated values based on data from Saarländisches Krebsregister

from "Deutsche Risikostudie Kernkraftwerke"

Table 3

Mean cancer risk (mortality) of federal German population 1974/75

Type of packaging	Resistant to /under	Activity limitation	Official approval
A-type	normal transport conditions and minor incidents	yes	no
[redacted] B-type	major accident conditions	no	yes

More lenient regulations apply for materials with a low specific activity and equipment comprising low-activity materials

Table 4

Types of packaging for radioactive materials

	Normal transport	Accidents
Transport workers	0.7 long-term damage/a	5×10^{-3} long-term damage/a
Members of the public	0.5 long-term damage/a	1×10^{-3} short-term damage/a

Table 5

Transport risk in USA, according to NUREG 0170

Cosmic radiation at sea level	ca 30 mrem/a
External terrestrial radiation	ca 50 mrem/a
Incorporated natural radioactive materials	ca 30 mrem/a
Total	110 mrem/a

Table 6

Genetically significant natural radiation exposure of man in Fed. Rep. of Germany in 1976

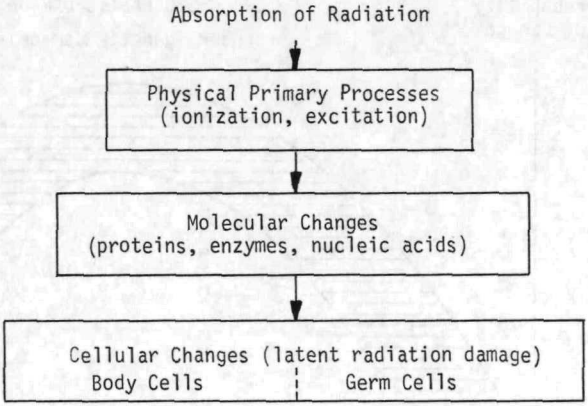


Figure 1 Biological Chain Reaction from Irradiation to Induction of latent Radiation Damage

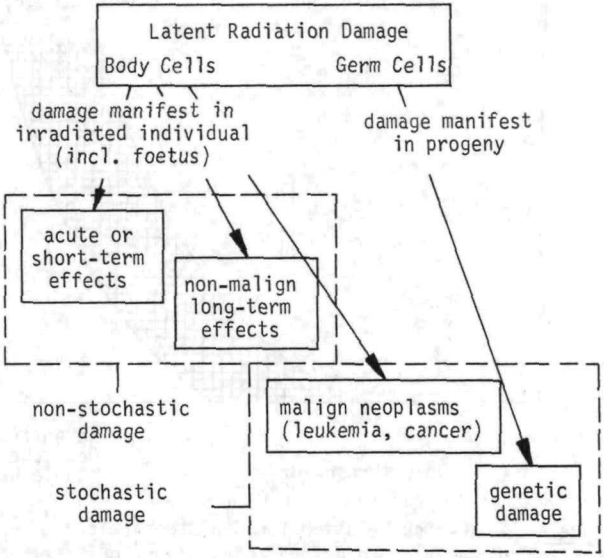


Figure 2 Types of Radiation Damage

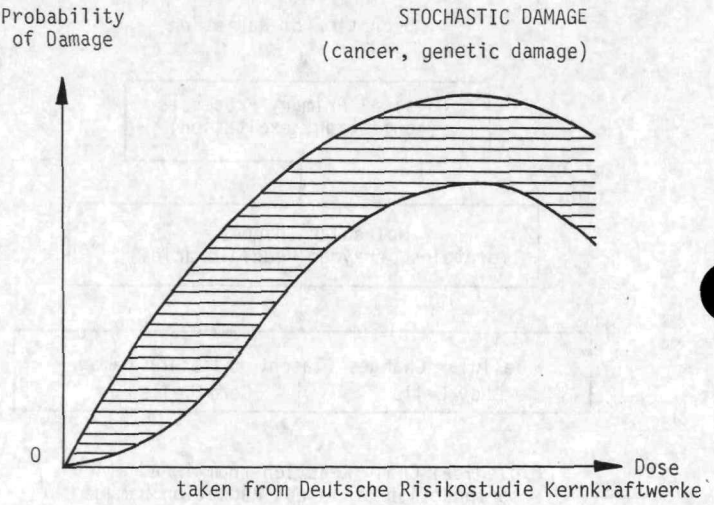


Figure 3 Dose-risk Relation for Stochastic Damage (cancer, genetic Damage)

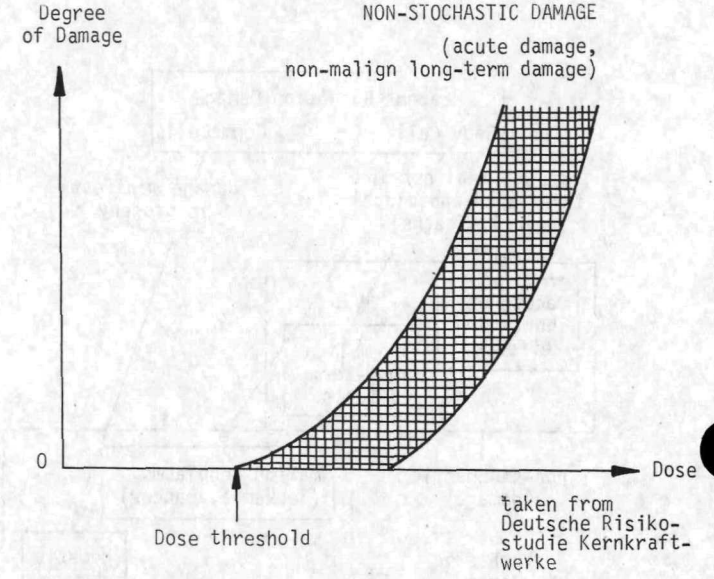


Figure 4 Dose-risk Relation for Non-Stochastic Damage (acute damage, non-malign long-term damage)

Figure 5

Warning Labels for Radioactive Freight Items

THE TRANSPORTATION OF RADIOACTIVE MATERIALS IN THE FEDERAL REPUBLIC OF GERMANY

B. Schulz-Forberg
Bundesanstalt für Materialprüfung (BAM)
Unter den Eichen 87, D-1000 Berlin 45

Introduction

This public evening provides a valuable occasion on which to examine the work being undertaken in connection with the packaging of radioactive materials and inform the general public on a number of important points.

Firstly, a few observations on the recently publicized risk analyses pertaining to the transportation of radioactive materials. Such risk analyses enable technicians and scientists to calculate a numerical value to quantify the risks involved. Yet it is of decisive importance that these numerical values are properly understood, are seen in the context of other risk factors, and finally, that all members of society realize that there will never be such a thing as absolute or 100 percent safety. On the contrary, we should all work conscientiously towards making the risks more transparent and more quantifiable, and, in the final analysis, towards accepting that a certain degree of risk exists. Reference should be made in this connection to the much-cited example of the dangers of road traffic, to which we all subject ourselves more or less willingly each day. Fifteen thousand traffic accident fatalities and more than 300,000 injured in the Federal Republic of Germany per year is an expression formulated in unambiguous terms. Even the oft-quoted concept of coal-fired power generation is not without its risks. An accidental death rate of some 16,000 fatalities in 31 years illustrates the dimensions of those risks. Even the installation work required for the exploitation of solar energy has already involved the sacrifice of human life: workers installing sunlight collectors on rooftops have been known to fall and in some cases suffer fatal injuries.

There is thus an urgent need for us all to formulate a clear picture of the risks involved in our daily lives, and, on the basis thereof, to arrive at a consensus as to what risk we are all willing to tolerate.

This paper is intended to provide information on past and present endeavours to develop safer transport containers for radioactive materials.

Packagings for Radioactive Materials

There are two basic types of packaging for the transportation of radioactive materials, Type A packagings and Type B packagings. Type A packagings are so designed as to be incident--proof but not accident-proof. Accordingly, they are considered unsuitable for anything more than a very small amount of radioactive material. A typical Type A packaging consists of a metal drum lined with cavity-filling and shock-absorbing material, a containment system made of glass ampoules set in lead containers that fit tightly around the radioactive material itself. Between the lead containers and the glass ampoules are quantities of absorptive material with an absorptive capacity of twice the volume of liquid contained in the glass ampoules. This ensures that, even in the event that the packaging is destroyed, the escape of radioactive liquid is prevented. Since the Type A packaging is used for limited quantities of radioactivity only, any damage incurred, even in the event of a serious accident, still remains within the bounds of the tolerable.

The situation is different where transport packagings for considerably larger amounts of radioactivity are concerned. These are the so-called Type B packagings or accident-proof packagings. The relevant regulations require that in the event of an accident no undue loss of radioactive material occurs and that the shielding function of the wall casing material is not so reduced that unduly high doses of radiation may escape into the environment.

Type B packagings must of course first of all conform with the general regulations governing packagings. They are then required to meet the standards prescribed for Type A packagings, and, finally, the additional regulations pertaining to accident--proof packagings.

The tests used to simulate accidents are designed as follows:

1. Mechanical stress is simulated by a free fall from a height of 9 m onto an unyielding surface and a fall from a height of 1 m onto a bar of 15 cm diameter.
2. Thermal testing takes the form of 30 minutes spent in an 800°C fire.
3. Water immersion is simulated by immersion under a 15 m water head for a period of 8 hours.
4. Leakage testing, the final and a very important criterion in the assessment of test performance, has to be undertaken to an extremely high degree of sensitivity, depending on the type and properties of the radioactive material to be transported. The normal procedure is a helium leakage test during which it is virtually possible to detect any isolated helium atoms which are able to escape from the packaging subsequent to submission to the other tests.

The most exacting tests for packagings are the 9 m free fall and the thermal test.

The effects of these tests on the constructional design of containers can be seen from the accompanying illustrations.

Figure 1 shows the constructional development of a packaging for which the lid fastening was initially to be ensured by very short external bolts. After consultation with BAM, the applicant modified the design in favour of long, external bolts which were able to absorb considerably more thrust in the event of mechanical strain. The container was finally constructed with internal [redacted]s which were inserted as so-called necked-down bolts and thus a very high shock-absorption capacity. Figures 2 and 3 show the approved version of the container which had been intended for the transportation of high-intensity cobalt sources for use in hospitals.

The drop tests, one of the most demanding tests, are conducted by BAM on test equipment located on the detonation ground of the Berlin (West)police authorities in Grunewald forest. (See Figure 4.) During tests on a container for fresh fuel which had been cooled to -40°C, the lid construction was found to disintegrate completely, as Figure 5 shows. Figure 6 shows the corrected version of the container which, when likewise cooled to a temperature of -40°C, conformed with the test specifications. Figure 7 shows both versions side by side. Figure 8 shows the thermal testing equipment now in use; thermal testing is likewise useful in detecting unsatisfactory construction elements. Figure 9 shows a container after it had been subjected to the thermal test: its heat-insulating foam had proved not to be fire-proof, a defect which caused the container to fail the tests. Figure 10 shows a container for which a fire-proof foam had been selected which retained its heat-resistant properties throughout the duration of the test.

These were a few examples from the practical work carried out at BAM. To date, more than 100 different construction types have been tested. Yet in only few instances do such design defects become manifest as late as in the testing stage: the majority can be eliminated during the design stage on the basis of the practical experience gained at BAM.

Relation Between Testing and Actual Accident Incidence

Figure 11 shows a transport container for uranium hexafluoride, one of the basic materials in the manufacture of fuel elements. This material is transported in steel pressure containers which are embedded in an external packaging. An accident occurred with such containers near Hamburg in 1973. Two containers were involved in a rail accident while being transported at a speed of ca. 100 km/h. The damage suffered by the containers was so insignificant that they would also have withstood any fire which might spontaneously have ensued. Even in the accidents involving uranium hexafluoride and a container for the transportation of spent fuel elements which occurred in the USA the containers

have not failed to fulfil their function. Such results illustrate the positive effect of the very restrictive regulations governing the transportation of radioactive material.

One method of estimating how the testing can be related to genuine accident incidence is calculation based on risk analyses. Figure 12, for example, shows the probability of a fire igniting in terms of transport miles covered by heavy goods vehicles on American highways. Only once every 30 million miles is there the probability of a fire of a magnitude exceeding the half-hour 800°C fire of the test. Assuming that one were to require, for example, that the test article be subjected instead to a fire of 1000°C for a duration of one hour, the horizontal line between the two "fire points" indicates the additional investment required for relevant insulation and other constructional design precautions. The vertical distance between the two points represents the benefit to be gained thereby, namely, the anticipated reduction in the probability of a fire exceeding the magnitude of the test fire. The cost-benefit ratio for this example is already unfavourable. It can be seen that, if one doubles the investment, the benefit gained therefrom is extraordinarily insignificant. There exists no linear or proportional relation between additional cost and additional benefit. A limit thus emerges which marks the ultimate point at which a construction remains purposeful. Figure 13 illustrates the probability of collision accidents, showing that the velocity at the point of impact does not exceed 20 km/h in 95% of all accidents. The test fall from a height of 9 m gives the container a velocity at impact of 50 km/h. Figure 14 shows the results of extensive work being carried out in the USA where actual impact surfaces have been taken into consideration in addition to the mandatory unyielding or real surfaces. The figure shows that the more flexible the surface, the higher the tolerable impact velocity. For example, the impact velocity of a container falling onto soft ground would have to be multiplied by 7 in order to exert the same strain as is exerted on a container falling from a height of 9 meters onto an unyielding target. In terms of kilometres per hour, a velocity of 350 km/h would be required to subject the container to stress equal to that undergone in the test fall.

However, compliance with the test requirements does not imply that, if real conditions were more exacting than test conditions, the containers would be likely to fail in their function. On the contrary, the large majority of the packagings used in the Federal Republic of Germany for the transportation of radioactive materials have been found to have a safety margin which extends well beyond the test specifications. One example, shown in Figure 15, is a container for the transportation of plutonium nitrate solution which BAM subjected for a considerable period of time to a fire fuelled with heating oil. After 75 minutes in the fire, no temperature increase was discernible inside the container. Figure 16 shows that even after 75 minutes in the fire, only parts of the fire-proof foam had carbonized and that around the vessel proper was not even discoloured, a fact which proved that the high temperatures represented no major stress factor for the packaging.

Figure 17 shows the preparations for a test at Berlin-Gatow airport, where BAM had arranged for British Army Air Corps to cooperate in dropping packagings from a height of ca. 200 m. The appearance of a 100 kg packaging after being dropped onto a runway from a height of 200 m can be seen in Figure 18.

Yet packagings of even greater weight have likewise undergone testing in the form of a helicopter drop from a height of ca. 200 m. Figure 19 is an example of a packaging weighing 2 t which had been dropped onto hard ground from a height of 200 m. By using a height of 200 m, it has been possible cover the eventualities involving even the highest bridges in Europe and the fastest collision velocities in land travel. Figure 20 shows the same packaging again after being dropped onto a reinforced 60 cm-deep solid mass. The packaging had withstood both falls without suffering any major damage: the internal seal had remained intact. A packaging with a weight of 4 t is featured in Figure 21. This is a model of a fuel element transportation container on a scale of 1:2. Figure 22 gives an insight into the test preparations: the packaging is to be taken to a height of 200 m by helicopter. Figure 23 shows the same packaging after a drop from a height of 200 m onto the 60 cm-deep reinforced solid surface. Although the exterior of the packaging is deformed, in actual fact only the heat dissipation fins are affected. A subsequent leakage test showed that the package was still leaktight.

A report prepared by BAM on the reserve safety margin of packagings for the transportation of radioactive materials in the Federal Republic of Germany [1] reveals that very wide margins of safety remain beyond the prescribed statutory requirements which are considered by experts in this field without exception to be already high. Packagings can be classified into a series of groups in this connection.

The first group is comprised of gammagraphy and similar equipment, all of which has been found to have a very wide margin of safety over and above the prescribed standards. The second group is comprised of the containers for cobalt sources, some of which having very wide and some having wide additional safety margins. The third group is made up of the containers for the transportation of plutonium compounds, this being the only group in which some items have an only slight safety margin over the statutory norm. A substantial increase in the safety margin is anticipated from planned additional security measures: the Americans have already developed the ICV-concept, the integrated container vehicle concept in which the transporter vehicle and the transported container are viewed as one integrated unit. This integrated unit has been found to have an extremely wide margin of safety over and above the already very strict statutory requirements. The last group is made up of the containers for the transportation of spent fuel elements, these again having wide and very wide margins of safety.

[1] Schulz-Forberg, B. and Hübner, H. W. Packagings for the Transport of Radioactive Materials - Classification and Margins of Safety, Bundesanstalt für Materialprüfung, Unter den Eichen 87, D-1000 Berlin 45, August 1980

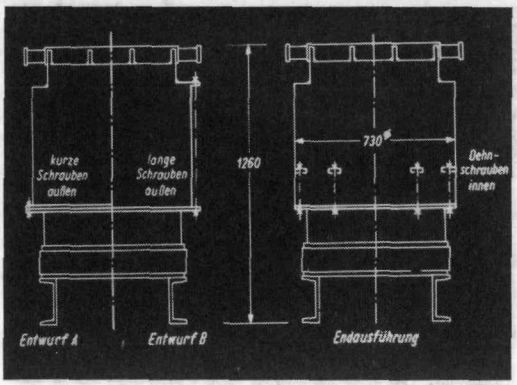


Figure 1:

Example of a Constructional Development on Type B Packaging D/DB - 0003B

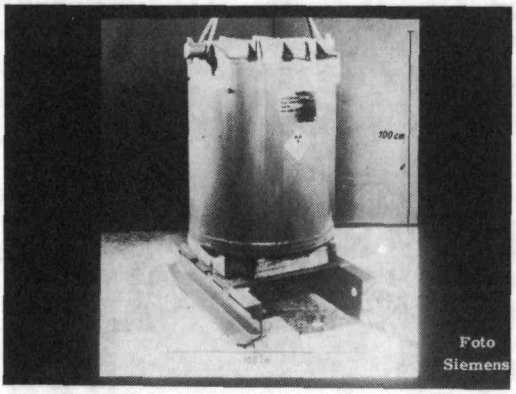


Figure 2:

Approved 2 t Transport Container for Therapy Capsules (Type D/DB-0003B) for max. 17 KCi Co-60

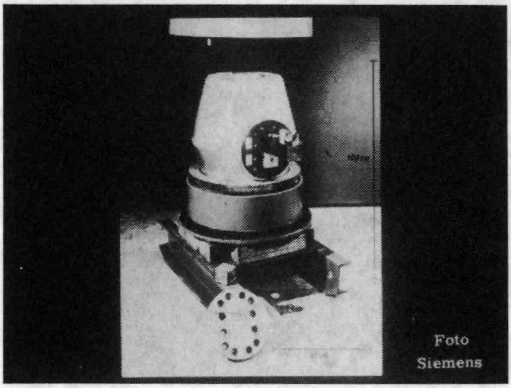


Figure 3:

Type B Packaging (D/DB-0003B)
Elevated Guard Cant and Open
Lead-Shielded inside Container

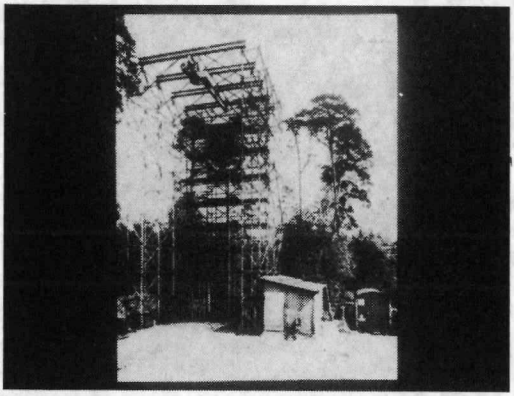


Figure 4:

Test Equipment for Collision Tests
Load 10 t Height 13 m

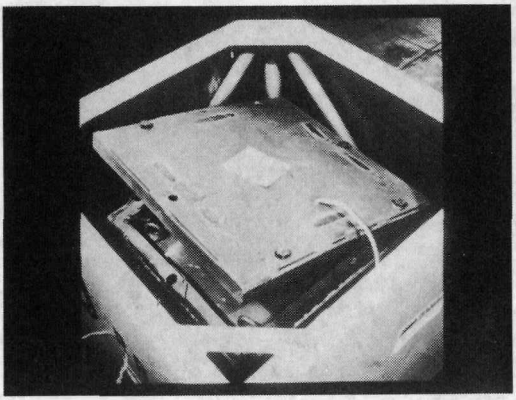


Figure 5:

Fissile Class II Packaging
for Fresh U-Pellets.
Failure after 9 m Drop at 233 k

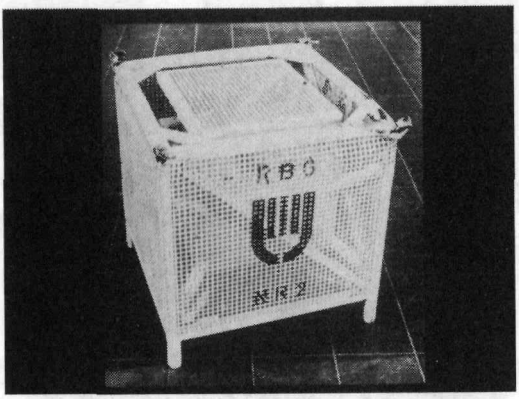


Figure 6:

Fissile Class II Packaging
for Fresh U-Pellets. Improved
Design after 9 m Drop at 233k

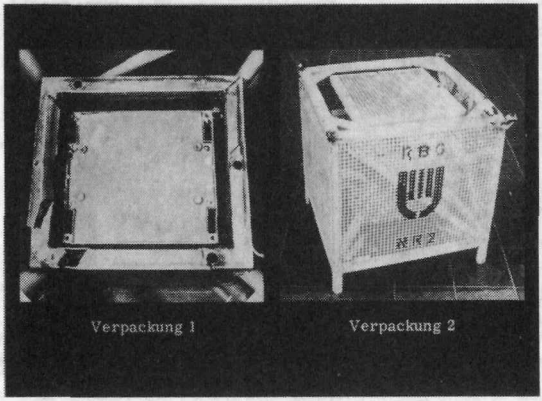


Figure 7:

Transport Container, Type A
Packaging after 9 m Drop Test at -40°C

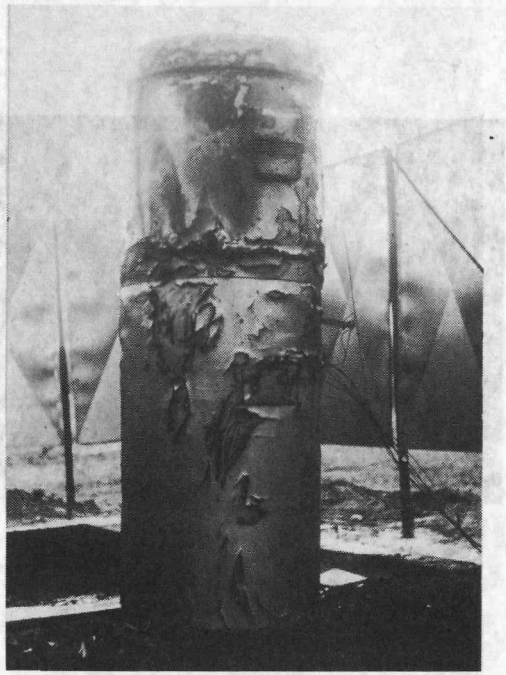


Figure 8:

Pu-Container after 90 min. Fire Test

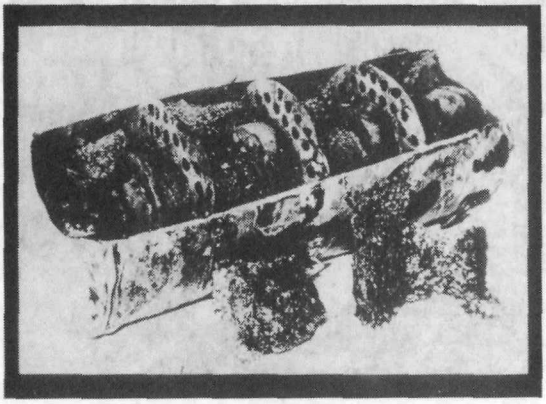


Figure 9:

Packaging after Tests. Packaging Had Non-Fire-Resistant Phenolic Foam as Insulating Material

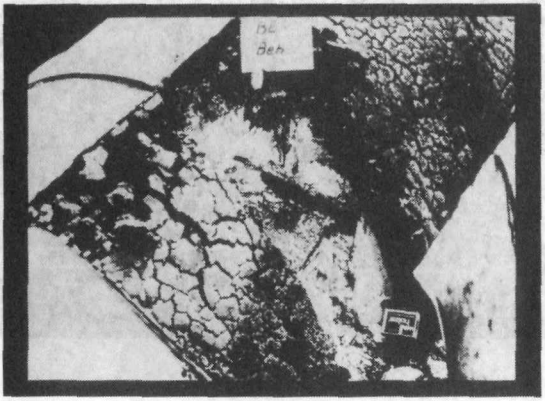


Figure 10:

Packaging after Tests. Packaging Had Fire-Resistant Phenolic Foam

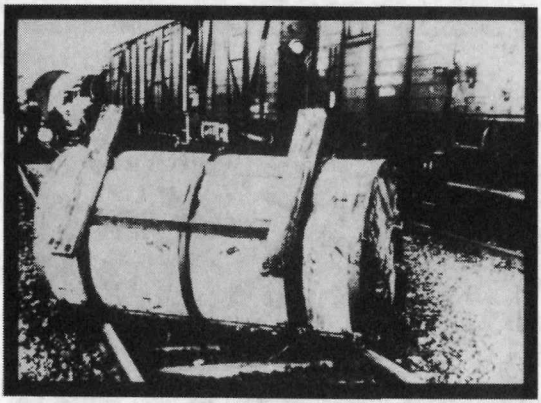


Figure 11:

UF_6-Package (Serial No. 62)
after Rail Accident

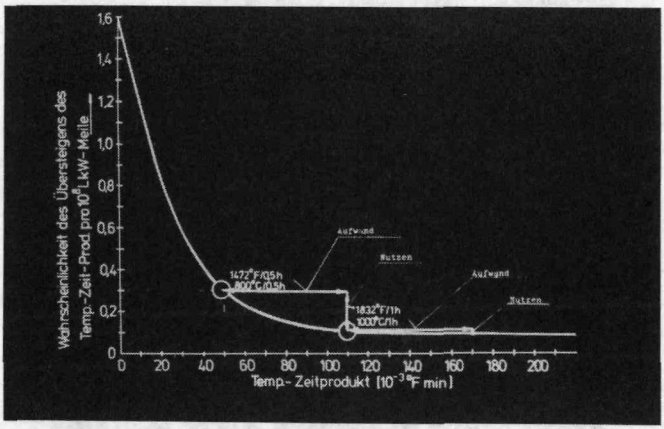


Figure 12:

Clarke, R. et al.
Probability of Post-Accident Ignition
in Street Traffic in USA. Cost-Benefit
Ratio for Stricter Test Requirements

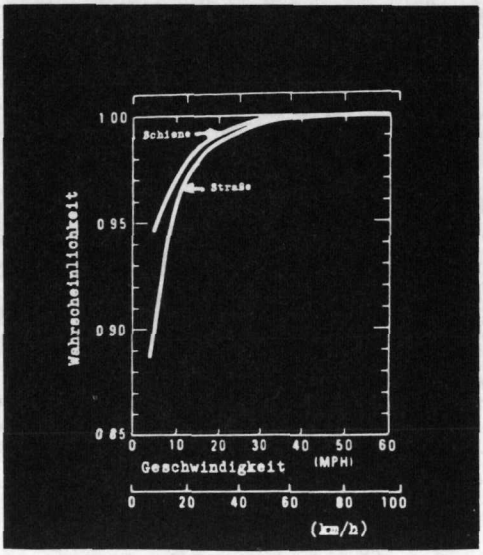


Figure 13:

Dennis, A. W.
Probability of a Mechanical
Accident as a Function of Velocity

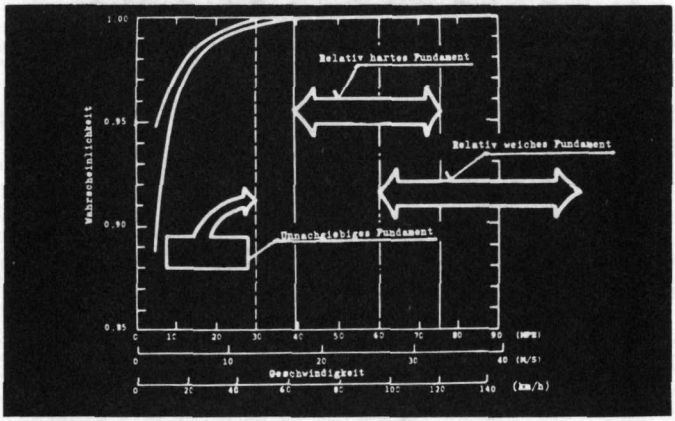


Figure 14:

McClure, J. D. et al.
Influence of Consistence of
Impact Surface on Container Damage

Figure 15:

BAM Fire Test Facility in Operation

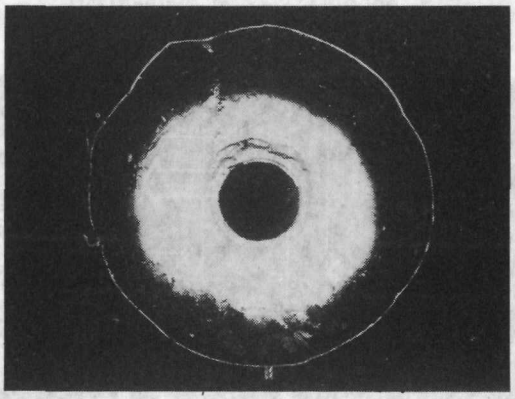


Figure 16:

Cross-Section of Plutonium
Nitrate Container after
Exposure to Fire

Figure 17:

Helicopter with 6M4 Pu-Container

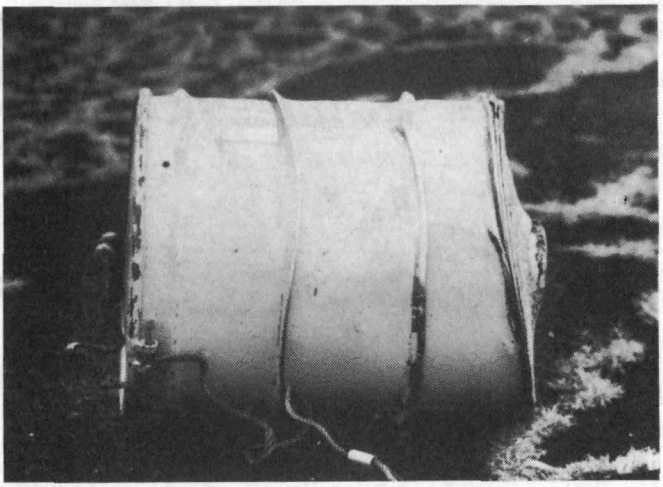


Figure 18:

6M4 Container after 200 m Drop

Figure 19:

BRECO KR 100/200 Cask for Spent Globular Fuel Elements - Recovery after 200 m Drop

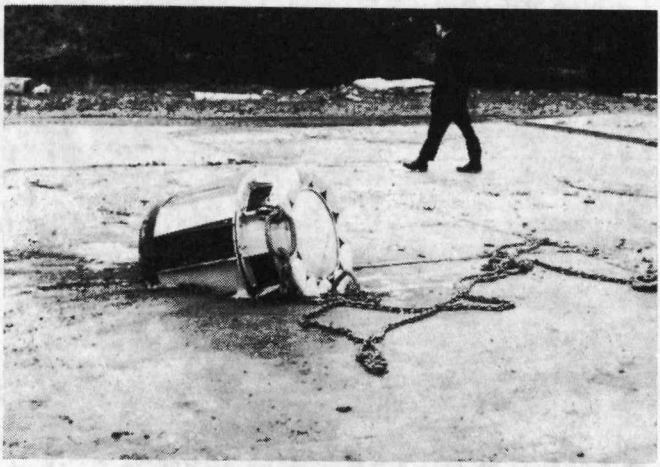


Figure 20:
BRECO KR 100/200 Cask Impact on Reinforced Concrete Target after 200 m Drop

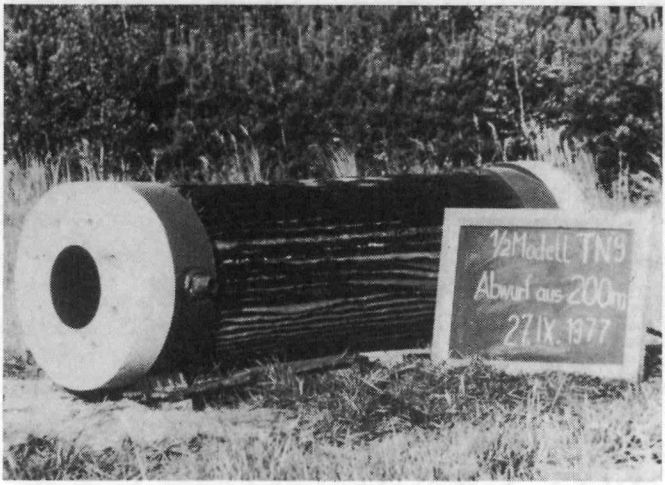


Figure 21

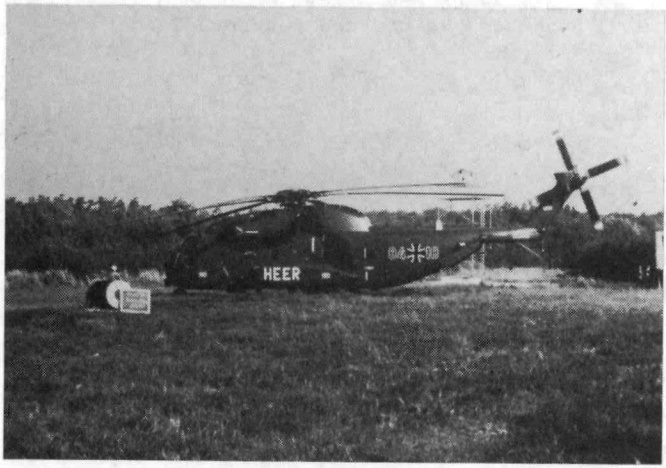


Figure 22

Figures 21 and 22: 1:2 Model TN 819 Ready for 200 m Drop Test

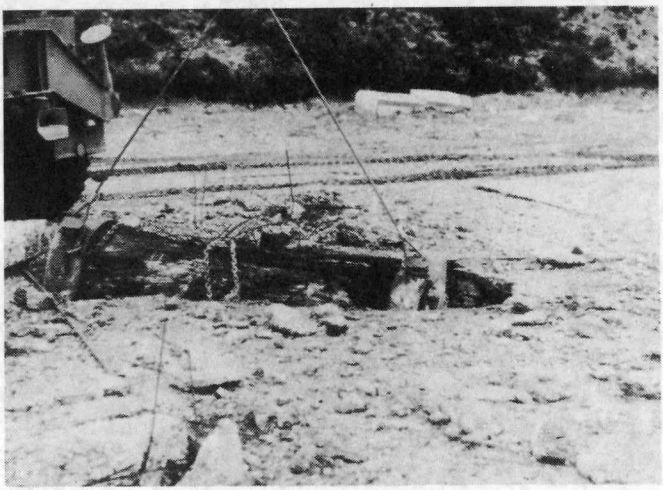


Figure 23:

Recovery after 200 m Drop
on to a Reinforced Concrete Target

DINNER SPEECH

THE NUCLEAR BOX

William A. Brobst, President

The Transport Environment, Inc.
Kitty Hawk, North Carolina 27949
United States of America

Today is the fifth day of August, 1982. Just 30 days ago, on July 6, 1982, late on Tuesday afternoon, the first major transportation accident involving a nuclear spent fuel shipment occurred in the United States. The accident happened in the rail yards of Newark, New Jersey, just about 35 kilometers southwest of downtown New York City.

A train carrying four heavily shielded spent fuel casks collided with another train carrying heavy construction equipment and steel structural beams. The train with the nuclear fuel was also carrying 14 tank cars of liquefied petroleum gas, separated from the nuclear cars by just a few cars of general and common freight. The tank cars immediately caught fire, and several of them exploded. The nuclear casks were entrapped in the middle of this fire. A nearby gasoline storage tank also caught fire and exploded.

The scene of the accident was one of almost complete destruction of nearby buildings and rail cars. Some low levels of radioactivity were detected in the smoke clouds and on the ground downwind of the accident. The wind was blowing toward New York City at the time, and there was some thought to evacuate the entire city of New York. The radioactivity levels were low at that time, and city officials decided to wait to see if the levels would become higher before they would order an evacuation. They feared that a massive evacuation might have caused many thousands of deaths and injuries from traffic accidents. However, people in the immediate area of the accident were evacuated out to a distance of a few kilometers along the path of the smoke. Accident response teams were called in from the city and state public health departments and from the Government nuclear laboratory at Brookhaven. The emergency teams were on the scene and ready within a few hours to examine the spent fuel casks, but the heat of the fire prevented access for nearly twelve hours. Finally, when they entered, they found the casks damaged but intact. Three of the four casks had been struck by the steel beams. The water neutron shields had been penetrated and the water had escaped. One of the casks was severely damaged, with some puncture of the outer shell of the cask and leakage around the cask closure device. The cooling fins on the casks were warped by the heat of the fire, which also destroyed all identifying marks. Maximum radiation levels two meters from the casks were about 5 millisieverts* per hour.

* equivalent to 500 millirem per hour

Just as recovery operations were about to begin, local officials refused to allow the recovery crews into the area for fear that moving the casks might release more radioactivity. The Federal Atomic Regulatory Agency immediately took charge, brought operations to a halt, and began a study into the risks and potential environmental impacts of recovery of the spent fuel casks. No work of any kind has been allowed in the area until the study is completed, which is expected to be soon.

Food and fuel for eastern consumers are still in short supply because the entire eastern seaboard rail system has been shut down. Shipments of nuclear materials all over the country have been stopped by local officials responding to anti-nuclear activist groups. The Atomic Regulatory Agency has suspended approval of all spent fuel cask designs. Many hospitals have not been able to obtain supplies of nuclear medicines. Several nuclear power plants have been forced to close until they perform a complete reexamination of their nuclear transport operations.

Well, we could go on with this story. With not too much imagination, we could think of many other things that could happen as a result of such an accident. But let's stop here and compare this hypothetical accident with Three Mile Island:

1. There was a serious accident, more serious than pro-nuclear people ever thought would really happen, and almost as serious as the "gloom and doom" fears of the anti-nuclear people.

2. The amount of radioactivity released to the environment was very small. It killed no one and injured no one. The long-term radiological consequences were not significant. The spent fuel casks performed exactly as they were designed to do.

3. There was some precautionary evacuation of nearby residents and workers.

4. The coverage by the news media was massive and detailed, generally emphasizing the potential dangers rather than effectiveness of the safety features of the equipment and the performance of the people.

5. The local officials and regulatory agencies both overreacted and underreacted, but they were unable to do anything but to wait until they had more information.

6. The Atomic Regulatory Agency required that similar spent fuel casks be removed from service until the Agency's staff could examine the design and quality assurance procedures in detail. The Agency also imposed a 12 month moratorium on nuclear spent fuel shipments.

7. The U. S. Congress, the regulatory agencies, local officials, and the nuclear industry all set up separate and independent investigation teams, most of them consisting of people who don't know much about nuclear transportation.

8. Technical experts want to go in and clean up the mess; political officials argue about who has the auchority to allow cleanup.

We could go on and on with comparisons, but I expect that for months after such an accident, the spent fuel casks would still be sitting there in the rail yard waiting for someone to do something besides investigate and study the matter. This would occur, in spite of pressure from the railroad company and nearby industries to get on with the cleanup.

Why? With all we know about radiological hazards, and with so much technological information and exotic equipment, why do we find ourselves in such a box full of emotion, misinformation, and fear?

Let's examine for a few moments tonight (1) the doorway that led us into that box, (2) the interior of the box itself, and (3) the possible doorway out of the box. We'll look first at the energy box, then the nuclear energy box, then the nuclear energy transport box.

Let's start with the energy box.

As a practical matter, it's no longer possible or even desirable for a nation to consider in isolation what its energy policy should be. Energy supply and demand have become global issues. The world's demand for oil today is based on social ideas, technical progress, and fuel supply. The world's demand for nuclear energy tomorrow will be based on the same factors. But today, nuclear power has its own problems. Safety, nuclear *weapons proliferation, waste management, even transport.* The focus of the problems is not proportional to the reality of the problems.

Most of us here are already convinced that these issues are not technological ones, but institutional - political - ones. The political resolution of these problems cannot be effective if it is attempted on a purely national basis; look at the United States as an example of the ineffectiveness of an isolationist attitude toward the use of oil. Also, because our nuclear problems have not been resolved on an international basis, our nuclear future is more clouded than the future of other energy sources. These uncertainties impact on suppliers and users in different ways, but the net effect is the same - the people of the world are being denied the effective use of nuclear power.

The prestige of the U. S. has declined sharply in many ways during the past few years. One reason is a self-centered energy policy, at least ten years old, that has ignored the

legitimate needs and interests of the rest of the world. For example, Americans use more oil per person than any other country! Another factor is our giving up of world leadership in nuclear energy. Our efforts to halt nuclear proliferation, while having a desirable goal, have produced undesirable results. In losing our nuclear leadership, we have also lost much of our ability to influence the nuclear energy policies of other countries.

Energy policy involves complex social and political issues that are not so well understood as the technical issues. Some of the social and political considerations could be better understood through research on the issues of public, official, and industrial perception of the problems. But there will remain an element of conflicting social and political values that cannot be resolved except in the political arena. Unfortunately, there is all too little real communication of ideas between technical and political people.

Now, how about the nuclear energy box?

The International Nuclear Fuel Cycle Evaluation was recently completed. One evident conclusion is that, of the 57 nations that participated in that study, the United States has the strongest doubts about the extent to which it should rely on nuclear power to reduce its dependence on foreign oil. It was the U. S. who suggested that the study be done and encouraged a broad international participation. The participating nations found, as the U. S. had hoped, that there were risks from nuclear proliferation. Even so, those other 56 nations became only more convinced that an increase in nuclear power still is necessary. Only the U. S. seems to resist that logical conclusion, especially with regard to the breeder reactor and the use of plutonium. The other nations have also become convinced that the solution to proliferation problems are institutional - social and political - while the U. S. officials still seem too much focused on technological controls. This displacement of focus is a very clever defense mechanism to avoid having to deal with the real issues of nuclear power.

In the U. S., the National Academy of Sciences has recently done a nuclear energy study combining, as best they could, the technical and the sociopolitical ideas and facts. Their conclusion is that nuclear power growth is necessary if we are to survive as a nation or group of nations, and that the people must be convinced of this by the political leaders of those nations. The key is money: profits, living expenses, energy costs, price of gasoline and heating oil. Not until the public is convinced that not having nuclear power will take large sums of money out of their pockets will we see a major turnabout in public attitudes toward nuclear power.

On the positive side, we have already begun to see some beginnings of a public awareness of the need for nuclear power. Political leaders are beginning to understand themselves the potential dangers of not going nuclear. The economic and safety hazards of cutting back on nuclear power are frighten-

ing, once you can get the facts out in the open. Those political leaders themselves must not only become convinced, but they must become carriers of the nuclear flag. They must become evangelists for balanced and sensible energy programs. They must be willing to make those hard decisions now that will provide for our energy needs in the future, even at the expense of some sacrifices now.

How did we let ourselves be trapped in these boxes? Actions of the anti-nuclear activists over the past 10-20 years have done much to destroy the confidence of the public in nuclear power safety. But we are often misled by the degree of noise we hear. The anti-nuclear people make much more noise than the pro-nuclear people, and pro-nuclear statements are lost in the background. The news media - newspapers, radio, television - plays up the negative image and makes nuclear power appear very frightening. When did you last see a front-page story supporting nuclear power? Even in the face of this gruesome imagine, the majority of the public in the U. S. favors nuclear power and supports the building of more nuclear power plants. And this is happening even considering the accident at Three Mile Island. The reason? A simple and logical one - the public is finally beginning to be aware of the growing severe energy shortage in the next 10-20 years. The public is noticing that the horrible effects of nuclear power which are claimed by the anti-nuclear people are not happening. The anti-nuclear people said that if a nuclear power reactor lost its cooling water, the core would melt and sink into the earth - the China Syndrome. They predicted that such an accident would kill thousands of people and contaminate hundreds of square kilometers of land. Well, Three Mile Island lost its cooling water, uncovered its core, and nothing melted. Nobody died and nothing got contaminated. The Kemeny investigation report showed that even if it had melted, there still would not have been anyone hurt by nuclear radiation. For the first time, the public had some proof that the nuclear industry might be right and the anti-nuclear activists might be wrong. But is the public still vulnerable to the attacks of the critics? How can we bring about this change in public attitude more quickly?

Here in Germany, you have reelected Helmut Schmidt, just a month ago. Like the U. S., you Germans are heavily dependent on foreign oil. More than half your energy needs are supplied by oil, and you have large stockpiles to carry you through Middle East wars. But you are also beginning a large program to increase domestic coal production and to strengthen your nuclear program. And you have one of the very first and best plans to get rid of nuclear waste for all time - a salt mine. The French and the Russians are going ahead with great speed on breeder reactor programs. Nuclear power programs are beginning to expand all over the world. Why? Not because the public has suddenly become aware that nuclear power is safe, but because the public is beginning to understand the need for nuclear power.

In the U.S., we have just elected a new President - Ronald Reagan. I see a strong likelihood that there will be a major change in the nuclear policy of the U. S. over the next four years. Mr. Reagan is a strong supporter of nuclear power as a major building block for our national energy and defense programs as well as for the world. Four years from now, I expect that our nation will have made major decisions to go ahead and make final disposal of nuclear waste, to reprocess spent nuclear fuel, and to move forward on the breeder reactor program. I believe that the highest officials in our government will recognize just how important nuclear power really is, and set a policy for nuclear power growth. The public will follow and support that policy. That public support will start a groundswell that will carry us into the 1990's, although we'll still be ten years behind where we should be.

The war in the Middle East between Iran and Iraq has turned some people toward nuclear power, but I don't suggest starting more wars just to change the mind of the public. Nor do I seriously suggest that we should have more nuclear reactor accidents, although I do think that if there had been more serious accidents with no serious consequences, it might be quite helpful in the long run in shaping the public attitude. We have had some serious transport accidents involving nuclear materials, but there have not been any significant nuclear consequences. Can we take advantage of this kind of information to change the public's attitude? I think so. In one way, I'm glad we had the accident at Three Mile Island! It had to happen someday; now let's take advantage of it. I think it has already helped us, and will help us even more in the future. The accident demonstrated the remarkable desgree of safety built into nuclear powerplants which allows them to withstand such an accident without releasing dangerous amounts of radioactivity. Shipping casks for nuclear spent fuel have similar high degrees of safety built in, and we need some more demonstrations of that. The crash tests done at Sandia Laboratories in the U. S. in 1977 and 1978 were very useful, but what we need now is a "good" transport accident - a real one to show just how safe nuclear transport really is.

Another result of the lack of good perspective on nuclear hazards is the amount of attention being given to emergency planning and response. We are spending far too much time setting up complex and expensive emergency resonse and evacuation plans aimed at nuclear hazards when those plans would be better directed toward other more common and more serious hazards. Local officials have found a new status symbol - their emergency plan. There almost seems to be a competition as to who can evacuate the most people the fastest! I understand how difficult it is to deal rationally with accidents that have a very low probability of occurence, but which have a high consequence if they do occur. But I find it difficult to understand why we should become so preoccupied with planning for accidents that have a low probability and a

low consequence. Does yellowcake really present such a hazard that a special emergency plan is necessary for it? Again, it's too bad in a way that we have not had more transport accidents to show that this level of public concern is not justified. If there had been more accidents, and if we had been given a green light to go in and clean them up, the concern would be lower. But there are no green lights these days, only red ones. Those of us in the nuclear industry have great confidence in our ability to deal effectively with common accidents, but we have not radiated this confidence to local officials or even to national regulatory agencies.

Just [redacted]o months ago, in Vienna, a worldwide group of experts in [redacted]ear transport met to consider what changes, if any, are needed to the IAEA regulations. These regulations have been in use for many years and have produced the excellent safety record we have been talking about. The experts agreed that the regulations provided a very high degree of safety, and recommended no major changes. They produced a revision of the regulations, but the changes are more of form than of substance. They are very good regulations. But does the public give much credit to the nuclear industry for good regulations? No. The public instead finds great delight in stories about how dangerous is uranium hexafluoride or yellowcake or uranium mill tailings.

The "overkill" by the news media of insignificant hazards is part of the problem. For example, the deliberate controlled venting of krypton-85 from Three Mile Island amounted to about 1-2% as much krypton as was released during the accident itself, which hurt no one. But the news media still made headlines over the controlled venting.

Another part of the problem is regulations and reporting requirements that put too much emphasis on small things. It's difficult for the public to distinguish between small problems and large ones; they both get a lot of press coverage.

In the nuclear field, we report <u>everything</u> to someone, and every report requires an action by the regulators. For example, the existing waste ash piles from coal power plants in the U. S. generate 10 times as much radon as the existing uranium mill tailings. Millions of dollars are being spent on cleaning up the mill tailings, mostly because of the publicity, not the hazard. And nothing is being spent on cleaning up the coal ash piles. A transport accident involving yellowcake gets as much attention for the regulatory agencies as one involving [redacted]t fuel.

The [redacted]ceptance by the public of the transport of nuclear materials is tied very closely to the acceptance of nuclear power itself. The public often rejects nuclear transport, not so much because the public thinks that nuclear transport is not safe, but because they think it's not <u>necessary</u>. Technological activities which are much more dangerous than nuclear transport are accepted because the public is convinced that society cannot do without them. Let's look at some examples.

Just one year ago, 240,000 Canadians - a quarter of a million people - were evacuated from their homes because of a rail accident near Toronto. The accident released dealy chlorine and phosgene gases. The story of this accident made the headlines in the U. S. and Canada for a day or two and then was quickly forgotten. The Three Mile Island accident resulted in a few hundred people being evacuated and no harmful releases, but has been in the headlines of the newspapers for 18 months. This illustrates the strange and puzzling way in which society responds to the many risks of industrialized living. Why?

What accounts for the phenomenon that produces so little concern over chemical accidents and instant headlines about even the hint of nuclear danger? It's not just a reaction to the ghostly qualities of radiation. Many chemicals - carbon monoxide, for example - are colorless, odorless, tasteless, and deadly. It's also not just that nuclear power is new and unfamiliar. Commercial nuclear power is over twenty years old, whereas the relationship between many chemicals and cancer is much newer than nuclear power.

Society reacts differently to risks of different types, even though a mathematician might say the risks are equal. Coal, for example, is much more dangerous than nuclear power when one considers the combined risks of mining accidents, black lung disease, air pollution, acid rain, and carbon dioxide buildup. But how many anti-coal demonstrations have we seen? We live happily with one technology - the automobile - which causes 50,000 deaths each year in the U. S. alone.

In the U. S., there are at least 300,000 people who die each year just from the effects of smoking cigarettes. Everybody knows that, and we get tired of hearing it (especially the smokers). That's not news, and the news media isn't interested in making an issue of it anymore. Cigarettes cause more preventable illnesses world-wide than any other single man-made influence. Here we have no deaths from nuclear power over a period of many years, and fear of nuclear power is still a number one news item. Why? 300,000 people each year is like 1000 Boeing 747 jumbo jets crashing each year, killing everyone aboard. A thousand crashes a year! The public would not stand for such a series of disasters. But the public is willing to pay the price of that many deaths from smoking because the public wants its cigarettes. The public is willing to pay the price of gasoline and tens of thousands of highway deaths each year because the public wants automobiles.

Airline flight attendants receive about 200 times as much ex[illegible] radiation each year as the people received who live near Thr[illegible] Mile Island from the accident. Those people insist on the ri[illegible] to keep on flying even in the face of increased radiation exposure and even if they happen to be pregnant! Why? Because they are willing to pay that price to continue their career..

People are naturally more willing to accept the risks of a voluntary activity - especially one from which they receive a direct and obvious benefit - than the risks of an involuntary one. But a larger part of the explanation lies in a general public confusion. Recently it seems that just about everything is dangerous to your health - nuclear power, chemical wastes, pesticides, antibiotics, bacon, tampons, eggs, sugar substitutes. Just about everything seems to cause cancer - what you do, what you eat (or don't eat), where you live. In the face of such a systems overload, it's very difficult to be very rational. It's as difficult as for an American to correctly pronounce "fünf and fünfzig".

Are risks really increasing, or are they just being advertised better? On one hand, we are running out of empty space when chemical and biological wastes can be dumped. We're able to measure smaller and smaller amounts of chemicals - amounts that were not detectable at all just a few years ago. So we are becoming more aware of risks that have been around for a long time. But on the other hand, more industrial activity and fewer natural resources combine to produce new dangers.

It will be very important in the 1980's to deal with the combination of a highly developed technical capacity to generate and measure risks with a very poorly developed social capacity to perceive, control, and manage those risks.

So the excellent record of nuclear transportation over more than 30 years doesn't count for much in the eyes of the public. No deaths, no injuries, and no interest in such a good record. The public will respond only to a perceived risk, and the acceptability of that risk is directly a function of how much the public thinks it needs the activity that produces the risk. When the public is convinced that it needs nuclear power in order to maintain its standard of living, then the safety questions will become of minor importance.

We have looked enough, certainly for tonight, at the insides of this crazy nuclear energy box. What is the way out of it? How can we bring a better perspective on nuclear power and nuclear transport? How can we turn off the influence of the anti-nuclear activists and turn on the public to our own views?

Advertising! Information, facts, visibility. We've got to be seen before we can be believed. And we've got to change our message. It's pretty obvious that the message we've been carrying these past twenty years isn't doing the job. I believe we must redirect our efforts. We've spent too much time and money trying to impress the public with how safe nuclear power is. And it hasn't really sold well at all. We've allowed

ourselves to be put into a corner by the anti-nuclear activists and we've ended up fighting from a defensive position, with the other side calling the shots and setting the rules.

The history of man shows that few wars have ever been won just defending oneself. It is the strongest sttack that wins, in wars, in sports, and in public opinion. In World War II,

England was losing when she was on the defensive. When she came up to speed and took the offense (with a little help from her friends), the momentum shifted and the Germans ended up on the defense. We fought defensive wars in Korea and Viet Nam, and lost.

It's time for nuclear advocates to take the offensive. We should stay away from the safety issues, and we should ignore the anti-nuclear activists. We'll never convince these anti-society people of our own beliefs; let's stop trying. Let's stop making it so easy for them to be heard. They have only a bark; they have no bite other than what we give them. It's time for us to bark.

Albert Einstein was one of the greatest scientists ever. But he was also one of the greatest friends of mankind. He believed very strongly in the future of nuclear power, and told anyone who would listen that the facts of nuclear energy "must be taken to the village greens".

There will always be a small group of noisy anti-nuclear, anti-society, anti-progress, anti-technology people. Let's shake their dust from our shoes, and march into the word war with our own public relations campaign. Let's advertise nuclear power. Let's convince the public that they can't do without nuclear power. Let's convince them that the transport of nuclear fuels and medicines is vital to their well-being and to their pocketbooks. Let's motivate the public to fight for nuclear programs rather than against them. Let's convince the public that they should become pro-nuclear activists, just as they are pro-automobile activists and pro-cigarette activists. If you try to take away their automobiles or their cigarettes or their televisions, you'll have a real fight on your hands. When they feel that way about nuclear power, we won't have to argue safety anymore.

It won't be easy to turn this thing around, but the alternative is worse - to keep on as we are going now. It's worth the effort. As nuclear power goes, so goes nuclear transport.

LIST OF AUTHORS

LIST OF AUTHORS

	Page
Aamlid, H.	87
Allen, G. C.	299, 374, 1001, 1020
Allen, J. C.	1405
Alter, U.	737
Andersen, J. A.	897
And[redacted]on, D. R.	1020
Ar[redacted]s, W. B.	180, 206
An[redacted]h, W.	383, 401, 931
Aoyagi, S.	1138
Aoki, S.	589
Armbrust, G.	642
Armiriotis, J.	826
Asano, R.	842, 1147
Auerbach, T.	1222
Baatz, H.	235
Baier, G.	771
Baier, W.	1405
Bailey, W. J.	1113
Baker, A. R.	126
Baker, W. E.	1321
Balmert, M. E.	762
Baran, A.	383
Barker, R. F.	1, 1405
Barnes, R. W.	993
Baybutt, P.	762
Bazelaire, B.	1329
Beach, S. A.	31
Behrendt, V.	778
Benner, L.	1378
Bennerstedt, T.	87
Berg, J. D.	111
Berger, J. D.	111
Bernard, H.	479
Bernardo, B. C.	1371
Beyen, H.	1395
Biringer, B. E.	161
Blauvelt, R. K.	291
Blechschmidt, M.	1405
Blum, P. T.	437, 887
Boch[redacted], C.	881
Bon[redacted] D.	463, 850
Boyd[redacted]. D.	1222
Brackenbush, L. W.	206
Brantley, C.	1401
Brassell, G. B.	687
Bredael, L. U. B.	921
Brobst, W. A.	1445
Brown, M. L.	905
Brown, O. F.	1050
Brücher, H.	495
Bruel, H.	605
Burgoyne, R. M.	255, 625
Burian, R. J.	1105
Burnett, R. A.	206
Butterfield, R. S.	454
Cashwell, J.	374
Chatelet, J.	1176
Cheresh, M.	568
Chicken, J. C.	914
Chitwood, R. B.	967
Christ, R.	120, 800, 931
Cohendy, G.	9, 697, 1073
Colton, J. D.	1362
Conti, M.	99
Costa, M.	143
Crawford, H. L.	1275
Cross, H. D.	673
Cuillerdier, C.	517
Curren, W. D.	463, 850
Curtis, H. W.	583
Daniel, S. L.	161, 723
DeMarco, J.	558
DeMaria, E.	1401
Denk, W.	1172
Devlin, R. L.	977
Dicke, G. J.	419
Diem, H.	129, 240
DiStefano, J.	324
Dobremelle, M.	143
Doty, J. W.	291
Draulans, J.	921
Driver, E. T.	1378
Droste, B.	708
Duffey, T. A.	267, 1214
Duill, V.	283
Dupree, S. A.	152, 299, 367
Duret, B.	1073
Eakes, R. G.	152, 267
Easterling, E. M.	1238
Ebisu, T.	1147
Edling, D. A.	291
Eggers, P. E.	1275
Eisenberg, N. A.	728
Ek, P. I.	1405
Elbestawi, M. A.	1283

Page

Elliott, J. M. 334
Emerson, E. L. 811
Emmerton, A. 126
Engelmann, H. J. 283
Ericson, D. M. 723
Erlenwein, P. 747
Evrard, G. 921

Fabry, J. M. 921
Faloci, C. 703, 1033
Fields, S. R. 1302
Finley, N. C. 723
FitzPatrick, V. F. 793
Flothmann, D. 410
Forasassi, G. 99
Fore, C. S. 1067
Forest, J. W. 1283
Fradin, J. 982
Freathy, R. G. 274
Fretter, E. F. 1222
Fujise, Y. 550
Fukuda, S. 392, 540, 589, 1163
Fullwood, R. R. 197
Furudera, M. 1147

Gahm, J. W. 625
Geffroy, J. 143
Geiger, H. 105
Germain, M. 517
Gibson, W. 1205
Glaser, F. 1338
Goldfinch, E. P. 48
Goldmann, K. 82, 479
Greenborg, J. 206
Grella, A. W. 57
Grenier, M. 43
Gustafsson, B. 1008

Hadjiantoniou, A. 826
Hamann, J. E. 1081
Hamard, J. 9
Hankins, D. A. 367
Hansen, L. J. 111
Haon, D. 872, 881
Hara, Y. 842
Harmon, D. W. 1027
Harter, C. A. 531
Hartman, W. F. 1355
Hartwig, S. 410
Heiskell, M. M. 1067
Heudorfer, K. 410
Heumos, K. 472
Hillsman, E. L. 359
Hoehne, M. R. 255
Hoffman, C. C. 636
Hokanson, J. C. 1321

Page

Hudson, B. J. 344, 351, 1238
Hübner, H. W. 708, 1230, 1254
Humphrey, R. C. 291

Ichikawa, Y. 392
Imamasa, J. 550
Irino, M. 550
Ishida, Y. 135
Israel, F. 1292
Iwasaki, T. 1147

Jais, M. 1183
Janberg, K. 235
Jefferson, R. M. 1247, 1405
Johanson, N. W. 255
Johnson, P. E. 359
Joly, J. J. 141
Jones, R. H. 82
Joseph, B. J. 299, 1230, 1267
Joy, D. S. 344, 351, 374
Jung, W. 939
Jungclaus, D. 737

Kage, K. 83
Kaikkonen, H. T. 21
Kajishima, Y. 83
Kanae, Y. 115
Kanai, Y. 116
Kaspar, G. 93
Kawahara, I. 114
Keese, H. 93
Keith, D. A. 123
Kikuchi, S. 135, 512, 1251, 1254
Killian, C. B. 97
Kirchner, B. 43
Kishi, N. 113
Kleimann, H. 61
Klein, D. E. 108
Klingensmith, R. W. 64
Klöpper, H. 95
Knackstedt, H.-G. 80
Knox, W. 123
Köster, R. 28
Kok, K. D. 76
Konno, K. 13
Kortright, R. 99
Kotler, J. 120
Kowalewsky, H. 446, 471
Krieg, D. M. 80
Krieger, E. 41
Krupar, J. J. 11
Kubo, M. 512, 1251
Kudo, E. 540, 1163

	Page
Kunzendorf, U.	105
Kuwashima, K.	833
Kwong, K. C.	1222
Lafontaine, J.	921
Lahs, W. R.	90
Lake, W. H.	1090, 1199
Lambert, R. W.	1001
Lamoreaux, G. H.	267, 1214
Lamprey, A. S.	1238
Lan[redacted]F.	526
Lar[redacted]ff, D. C.	1113
Lark[redacted] D. J.	1001
Laug, R.	240, 249
Lauterbach, U.	66
Lazarevitch, S.	887
Leblais, R.	1176
Leinio, A.	87
Leisher, W. B.	152, 575, 1230
Lemercier, M.	143
Lewis, J. R.	206
Loewen, T.	1283
Lohmann, D. H.	818
Lowry, L. M.	762
Lührmann, A.	240, 249, 955
Luna, R. E.	309
Macdonald, H. F.	48, 189
Mackenzie, A. C.	531
Maeno, Y.	1138
Maki, Y.	392
Malmström, H.	955
Manaranche, J. C.	872
Mangin, D.	872
Mangusi, J.	754
Mancioppi, S.	558
Masslowski, J.-P.	961
Matthias, M. R.	993
Maubert, L.	872, 881
May, R. A.	1267, 1321
McCarthy, J. D.	806
McClure, J. D.	811, 1247, 1355
McCreery, P. N.	1238
Menc[redacted] W. R.	374
Meth[redacted] D.	682
Meyer[redacted]S. H.	40
Michoux, M.	143
Milchert, T.	1008
Mishima, J.	488
Moore, R. L.	625
Mori, N.	1147
Morin, P.	887
Morris, F. A.	316
Moscati, M.	558
Motz, J. M.	655
Moulin, M.	143
Müller, W.	526
Murphy, D. W.	206
Muskovszky, J.	742
Nacfaire, H.	1033
Nagakura, T.	392, 1138
Nakata, M.	501
Nakazawa, M.	540, 1163
Naqvi, S. J.	993
Neider, R.	1405, 1423
Nickell, R. E.	82
Niephaus, D.	495
Niomura, N.	842, 1147
Nishikawa, K.	842
Nommensen, O.	495
Nomura, M.	589
Ohashi, M.	550
Olivieri, A.	1121
O'Malley, L. C.	367
Onodera, A.	842, 1147
Opp, O.	613
Orsini, A.	703
Ortman, J. M.	152
Owzarski, P. C.	488
Ozaki, S.	842
Peclier, R.	715
Peehs, M.	939
Peterson, B. E.	359
Petrasch, P.	283
Piermattei, S.	558, 1033
Pochini, G.	1121
Pope, R. B.	152, 299, 367, 374, 673, 1081, 1230, 1247, 1275
Poullot, G.	872
Pounder, J. O.	126
Portal, M.	143
Preston, M. K.	1238
Price, M. S. T.	274
Prulhière, J. P.	1292
Pryor, W. A.	597
Quirk, P. M.	531
Rack, H. J.	299, 568, 666, 673, 687
Raghuram, S.	762
Rämö, E.	219
Rao, P. K. M.	993
Rawl, R. R.	72

Page

Reese, R. T. 197, 227, 309, 316, 1387
Rhoads, R. E. 180
Rhyne, W. R. 197
Riddle, J. H. 673
Rittscher, D. 235
Roberti, M. 1033
Romander, C. M. 1362
Romesberg, L. E. 267, 1214, 1267, 1321
Roos, G. 410
Rosenfield, A. R. 762
Ross, K. E. L. 334

Sandoval, R. P. 227
Sasaki, T. 1155
Sata, K. 1138
Scarpa, G. 558
Schaich, R. W. 1192
Schilling, A. H. 1387
Schilling, F. 682
Schlemmer, F. 939
Schlich, E. 931
Schnatz, G. 410
Schneider, K. 105, 383, 401, 800
Schroeder, G. 955
Schüler, R. 742
Schulz-Forberg, B. 1183, 1230, 1254, 1438
Schweer, H.-H. 866
Schwendiman, L. C. 488
Seagren, R. D. 351, 1247
Sekiguchi, A. 501, 540, 1163
Seran, R. 1146
Shappert, L. B. 374, 1247
Shaw, K. B. 31
Shepherd, E. W. 1041
Shieh, R. C. 1311
Shikata, H. 392
Shimamura, S. 1155
Shiojiri, H. 1138
Simmons, J. A. 197
Smugeresky, J. E. 687
Sofo, G. 703
Sousselier, Y. 9, 43, 1405
Souther, D. W. 993
Staake, T. R. 642
Stadnikoff, V. 1073
Suchsland, K. E. 1222
Sutherland, S. H. 211, 267, 1214
Sutter, S. L. 488

Page

Taig, A. R. 428
Takagi, S. 392
Takano, H. 392, 1138
Takeda, T. 589
Takeuchi, K. 540, 1163
Tanaka, J. F. 993
Taylor, C. B. G. 1347
Taylor, J. M. 161, 723
Taylor, J. M. 793
Thielemann, K. 613
Timm, U. E. 1058
Tomassetti, G. 99
Tomita, N. 550
Trapp, R. 961
Treffner, H. 682
Trujillo, A. A. 575, 673, 1275
Tveten, U. 168

Ulbak, K. 87
Ueki, K. 501, 540, 1161
Underwood, B. Y. 531

Vandevender, S. G. 1387
Vernon, M. E. 1020
Viebrock, J. M. 63
Vilkamo, S. 21
Vivien, M. 14
Vuori, S. 21

Walker, G. 85
Warniez, P. 69
Watson, C. D. 1231
Weber, E. T. 11
Wehner, G. 1
Weise, H.-P. 47
Welles, B. W. 31
Wiedemann, G. 1231
Wieser, K. E. 708, 1181
Williams, R. J. 53
Wright, B. 27
Würdemann, G. 41

Yamada, T. 540, 1161
Yamakoshi, H. 501, 541, 1161
Yamashita, M. 84
Yoshimura, H. R. 1081, 1222, 1231, 1247, 1261, 1275, 1311
Young, R. L. D. 1091
Yuki, A. 58

Zannos, A. 81

LIST OF PARTICIPANTS

LIST OF PARTICIPANTS

AUSTRALIA

J. M. Rolland
Australian Embassy
Mattiellistrasse 2-4
1040 Vienna, Austria

AUSTRIA

G. Arendasy
Simmering-Graz-Pauker AG
Mariahilferstrasse 32
1071 Vienna

O. Artner
Simmering-Graz-Pauker AG
Mariahilferstrasse 32
1071 Vienna

+R. Barker
International Atomic Energy Agency
P.O. Box 100
1400 Vienna

G. Burian
Vereinigte Edelstahlwerke AG
2630 Ternitz

H. Jesser
Simmering-Graz-Pauker AG
Mariahilferstrasse 32
1071 Vienna

M. Komurka
Österreichisches Forschungszentrum
Seibertsdorf GmbH
Institut für Reaktorsicherheit
Sobieskigasse 21
1090 Vienna

J. Neubauer
Österreichisches Forschungszentrum
Seibersdorf GmbH
Lenaugasse 10
1082 Vienna

W. Stolz
Bundesministerium für Verkehr
Karlsplatz 1
1010 Vienna

BELGIUM

+J. Draulans
Belgonucleaire
Europalaan 20
2480 Dessel

M. Jurisse
Ministère de la Santé Publique
Quartier Vésale
1010 Brussels

+I. Lafontaine
Belgonucléaire
Europalaan 20
2480 Dessel

+H. Nacfaire
Commission of the European Communities
rue de la Loi 200
1049 Brussels

K. Pflugrad
Commission of the European Communities
rue de la Loi 200
1049 Brussels

E. M. Schank
Compimo Belgie NV
Ahlers House
Noorderlaan 139
2030 Antwerp

CANADA

+J. G. Dicke
Ontario Hydro, H6F4
700 University Avenue
Toronto, Ontario, M5G 1X6

G. L. Dooley
Atomic Energy of Canada Ltd.
P.O. Box 6300 Station "J"
Ottawa, Ontario K2A 3W3

M. Gavin
Ontario Hydro, H14 B23
700 University Avenue
Toronto, Ontario M5G 1X6

+Author

+W. Gibson
Atomic Energy of Canada Ltd.
P.O. Box 6300 Station "J"
Ottawa, Ontario K2A 3W3

C. G. Jack
Atomic Energy Control Board
P.O. Box 1046 Station "B"
Ottawa, Ontario, K1P 5S9

+R. Kortright
Ontario Hydro H7 E17
700 University Avenue
Toronto, Ontario M5G 1X6

+J. Kotler
Atomic Energy of Canada Ltd.
P.O. Box 6300 Station "J"
Ottawa, Ontario K2A 3W3

+T. Loewen
Ontario Hydro Research
800 Kipling Avenue
Toronto, Ontario, M8Z 5S4

G. Malkoske
Atomic Energy Control Board
P.O. Box 1046 Station "B"
Ottawa, Ontario K1P 5S9

+D. W. Souther
Ontario Hydro, H14 A24
700 University Avenue
Toronto, Ontario, M5G 1X6

W. R. Taylor
Atomic Energy of Canada Ltd.
Chalk River Nuclear Laboratories
Chalk River, Ontario, KOJ 1JO

J. Williamson
Atomic Energy of Canada Ltd.
Whiteshell Nuclear Research
Establishment
Pinawa, Manitoba

DENMARK

F. R. Morais
Nordic Liaison Committee for
Atomic Energy
4000 Roskilde

J. S. Qvist
Risø National Laboratory
4000 Roskilde

+K. Ulbak
National Institute of Radiation
Hygiene
Frederikssundvej 378
Copenhagen-Broenshoej

FEDERAL REPUBLIC OF GERMANY

G. Achilles
Reederei und Spedition "Braunkohl
GmbH
Postfach 52
5047 Wesseling 1

B. Ahrens
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+U. Alter
Gesellschaft für Reaktorsicherheit
(GRS) mbH
Glockengasse 2
5000 Köln 1

D. Althaus
Interatom
Postfach
5060 Bergisch-Gladbach

C. Amberg
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

F. Ammentrop-Schmidt
Ammersham Buchler GmbH & Co. KG
Gieselweg 1
3300 Braunschweig

G. Andreae
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+W. Anspach
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

+G. Armbrust
Kernforschungsanlage Jülich
Postfach 1913
5170 Jülich

Author

D. Aurich
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+H. Baatz
Gesellschaft für Nuklear-Service
(GNS) mbH
Rosastrasse 15
4300 Essen

R. [redacted]
Re[redacted]i und Spedition "Braunkohle"
GmbH
Postfach 52
5047 Wesseling 1

K. Bätke
Kernforschungszentrum GmbH
Postfach 3640
7500 Karlsruhe

+G. Baier
Dornier System GmbH
Postfach 1360
7990 Friedrichshafen

+W. Baier
Hähnelstrasse 8
1000 Berlin 41

A. Baran
NUKEM GmbH
Postfach 110080
6450 Hanau 11

M. Bauschke
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

G. W. Becker
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Becker
Bu[redacted]nstalt für Materialprüfung
Un[redacted]n Eichen
1000 Berlin 45

T. Beckmann
Deutsche Gesellschaft für Wieder-
aufarbeitung von Kernbrennstoffen
Hamburger Allee 4
3000 Hannover 1

+V. Behrendt
Dornier System GmbH
Postfach 1360
7990 Friedrichshafen

H. Beisswenger
Technisher Überwachungsverein Baden
Postfach 2420
6800 Mannheim

H.-U. Berger
Kernforschungszentrum GmbH
Postfach 3640
7500 Karlsruhe 1

H. Betzold
Dornier System GmbH
Postfach 1360
7990 Friedrichshafen

W. Bewarder
Stahlwerke Peine
Am Laubberg 5
3320 Salzgitter 51

+H. Beyen
Reederei und Spedition "Braunkohle"
GmbH
Postfach 52
5047 Wesseling 1

P. Beyer
Rheinisch-Westfälisches
Elektrizitätswerk AG
Bauabteilung
Kruppstrasse 5
4300 Essen 1

+M. Blechschmidt
Physikalisch-Technische Bundesanstalt
Bundesallee 100
3300 Braunschweig

F. Bosten
Technischer Überwachungsverein Baden
Postfach 2420
6800 Mannheim

W. Botzem
NUKEM GmbH
Postfach 110080
6450 Hanau 11

H. Brandenburg
Kraftwerk Union AG
Hammerbacher Strasse 12
8520 Erlangen

+Author

M. Briesenick
Elbe Jeetzel, Zeitung Lüchow
Pallasstrasse 12
1000 Berlin 30

H.-U. Brinkmann
Kernforschungsanlage Jülich
Postfach 1913
5170 Jülich

B. Brock
Technischer Überwachungsverein
Rheinland e.V.
Postfach 101750
5000 Köln

B. Broszies
Badenwerke AG
Postfach 1680
7500 Karlsruhe 1

+H. Brücher
Kernforschungsanlage Jülich
Postfach 1913
5170 Jülich

H. Brütt
Technischer Überwachungsverein
Norddeutschland e.V.
Grosse Bahn Strasse 31
2000 Hamburg 54

W. H. Buchler
Buchler GmbH & Co.
Harxbüttler Strasse 3
3300 Braunschweig

A. Budäus
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+R. Christ
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

R. Christensen
Gemeinschaftskernkraftwerk Neckar
Postfach
7129 Neckarwestheim

N. Collin
Physikalisch-Technische Bundesanstalt
Bundesallee 100
3300 Braunschweig

+Author

K. E. Crefeld
Birkhofstrasse 8
4040 Neuss 22

N. Czaika
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

U. Dachroth
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

M. Daichendt
Bayer AG
INAP Ing.-wiss. Abt.
5090 Leverkusen

H. Deipenau
Deutsche Gesellschaft zum Bau und
Betrieb von Endlagerung für Abfallstoffe
Postfach 1169
3150 Peine 1

+W. Denk
Gesellschaft für Reaktorsicherheit
Forschungsgelände
8046 Garching

H. Derp
Thyssen Giesserei AG
Postfach 011220
4330 Mülheim/Ruhr

+H. Diem
Dürerstrasse 13
8755 Alzenau

P. v. Dobschütz
Bundesministerium des Innern
Husarenstrasse 30
5300 Bonn 1

+B. Droste
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

W. Eckhardt
Chemische Fabrik von Heyden
Donaustaufer Strasse 378
8400 Egensburg

H. Eick
Leibergweg 6
4300 Essen 15

W. Eisenlohr
Rheinisch-Westfälischer Technisher
Überwachungsverein e.V.
Abt. Aus- und Fortbildung
Steubenstrasse 53
4300 Essen 1

+H.-J. Engelmann
Kernforschungszentrum GmbH
Postfach 3640
7500 Karlsruhe

R[redacted]elstädter
K[redacted]rschungsanlage Jülich
Abteilung AP
Postfach 1913
5170 Jülich

+P. Erlenwein
Gesellschaft für Reaktorsicherheit
GmbH
Glockengasse 2
5000 Köln 1

A. Elsner
Siemens AG
UB Med., Abt. TV
Henkestrasse 127
8520 Erlangen

K. Ermisch
Nordwestdeutsche Kraftwerke AG
Schöne Aussicht 14
2000 Hamburg 76

H. Ernst
Bleiwerk Goslar KG
Im Schleeke 8
3380 Goslar

P. Faulhaber
Kernforschungszentrum Karlsruhe
Hauptabt. EKM
Postfach 3640
7500 Karlsruhe

J. Fischer
Rh[redacted]isch-Westfälisches Elektrizitäts-
w[redacted]
K[redacted]trasse 5
4300 Essen 1

T. Fischer
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

J. Fleisch
Deutsche Gesellschaft für Wieder-
aufarbeitung von Kernbrennstoffen
Hamburger Allee 4
3000 Hannover

J. Frêcaut
Quartier Napoléon
Kurt Schumacher Damm
1000 Berlin 51

J. Friedrich
Hahn-Meitner-Institut für Kern-
forschung Berlin GmbH
Glienickerstrasse 100
1000 Berlin 39

G. Fuchte
Technischer Überwachungsverein
Norddeutschland e.V.
Grosse Bahn Strasse 31
2000 Hamburg 54

G. Gaede
Buderus AG
Postfach 1220
6330 Wetzlar

+H. Geiger
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

H. Geiser
Gesellschaft für Nuklear-Service
(GNS) mbH
Rosastrasse 15
4300 Essen

H.-D. Gembus
KEWA GmbH
Hamburger Allee 4
3000 Hannover 1

H. Gerhard
Westerwälder Eisenwerk, Gerhard GmbH
Postfach 20
5241 Weitefeld/Sieg

+F. Glaser
Dammstrasse 26
4470 Meppen

E. Gollner
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+Author

K.-H. Grek
Max-Planck-Strasse 38
6056 Heusenstamm

L. Gülicher
Siempelkamp Gesellschaft für Guss- und Reaktortechnik mbH
Siempelkampstrasse 45
4150 Krefeld

R.-J. Hart
Röttgenerstrasse 140
5300 Bonn 1

R. Heet
Kernforschungsanlage Jülich
Zentralabteilung Strahlenschutz
Postfach 1913
5170 Jülich 1

K. Herberholz
Nukleare Transportleistungen GmbH
Postfach 110050
6450 Hanau 11

E. Herkommer
Gesellschaft für Reaktorsicherheit
Glockengasse 2
5000 Köln 1

R. Helms
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+W. Heudorfer
Battelle Institut e.V.
Am Römerhof 35
6000 Frankfurt/Main 90

+K. Heumos
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

A. Hoegel
Siemens AG
ZT StS
Postfach 3240
8520 Erlangen

O. Hoffmann
Luftfahrt-Bundesamt
Lilienthatplatz 6
3300 Braunshweig

H. Holtz
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

G. Horny
Kraftwerk Union AG
Berliner Strasse 295-299
6050 Offenbach/Main

G. Hottenrott
Hamburgische Elektrizitätswerke AG
Überseering 12
2000 Hamburg 60

+H. W. Hübner
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

J. Jahn
ALKEM GmbH
Rodenbacher Chaussee
6450 Hanau 11

+K. Janberg
Gesellschaft für Nuklear-Service (GNS) mbH
Rosastrasse 15
4300 Essen 1

+D. Jungclaus
Gesellschaft für Reaktorsicherheit mbH
Glockengasse 2
5000 Köln 1

A. Karabateas
Technischer Überwachungsverein Stuttgart e.V.
Abt. KS
Postfach 1380
7024 Fliederstadt 1

+G. Kaspar
Kraftwerk Union AG
B222
Postfach 3220
8520 Erlangen

+H. Keese
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

M. Kemper
URANIT GmbH
Postfach 1411
5170 Jülich

+Author

A. Kersting
Kraftwerk Union AG
Abteilung R 453
Hammerbacherstrasse 12
8520 Erlangen

D. Kind
Physikalisch-Technische Bundesanstalt
Bundesallee 100
3300 Braunschweig

K. [redacted]chhoff
N[redacted]sächsisches Landesversorgungsamt
Postfach 107
3000 Hannover

F. E. Kirk
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

A. Kleinen
Domplatz 1-3
4400 Münster

K.-H. Kleinschroth
Kraftwerk Union AG
Postfach 962
6050 Offenbach/Main

+H. Klöpper
Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen
Hannoversche Strasse 28
3006 Burgwedel

+H.-G. Knackstedt
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

R. Kokai
Mitsui & Co. Europe GmbH
Königsallee 92a
4000 Düsseldorf 1

+W[redacted]
Ph[redacted]alisch-Technische Bundesanstalt
Bundesallee 100
3300 Braunschweig

+H. Kowalewsky
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+Author

+R. Köster
Kernforschungszentrum GmbH
Postfach 3640
7500 Karlsruhe

S. Kreuz
Gesellschaft für Reaktorsicherheit mbH
Glockengasse 2
5000 Köln 1

F. Krieger
Gg. Noell GmbH
Schweinfurter Strasse 28
8700 Würzburg 2

H. Krug
Gesellschaft für Reaktorsicherheit mbH
Glockengasse 2
5000 Köln 1

U. Kunzendorf
Reaktor-Brennelemente Union GmbH
Rodenbacher Chaussee 6
6450 Hanau 11

W. Kuthe
Amersham Buchler GmbH & Co. KG
Gieselweg 1
3300 Braunschweig

+F. Lange
Gesellschaft für Reaktorsicherheit mbH
Glockengasse 2
5000 Köln 1

+R. Laug
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

J. Lauterbach
Deutsche Presse Agentur
Savigny Platz 6
1000 Berlin 12

+U. Lauterbach
Physikalisch-Technische Bundesanstalt
Bundesallee 100
3300 Braunschweig

F. Lehr
Amersham Buchler GmbH & Co. KG
Gieselweg 1
3300 Braunschweig

J. Leistikow
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. W. Levi
Hahn-Meitner-Institut für Kernforschung
Berlin GmbH
Glienickerstrasse 100
1000 Berlin 39

M. Lichy
Ausstellungs-Messe-Kongress GmbH
Messedamm 22
1000 Berlin 19

+A. Lührmann
Deutsche Gesellschaft für Wieder-
aufarbeitung von Kernbrennstoffen
Hamburger Allee 4
3000 Hannover

+H. Malmström
STEAG Kernenergie GmbH
Bismarckstrasse 54
4300 Essen 1

J. Manzke
Havariekommissariat K. Gassner GmbH
Savigny Platz 1
1000 Berlin 12

H.-J. Marcinowski
Redaktion Gefährliche Ladung
Schumannstrasse 38
4000 Düsseldorf 1

+J.-P. Masslowski
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

W. Matthees
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Matto
Nuklearer Versicherungsdienst
Rodenbacher Chaussee 6
6450 Hanau 1

A. Max
URANIT GmbH
Postfach 1411
5170 Jülich

A. Meaton
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Meder
Gefährliche Ladung
Stahltwiete 7
2000 Hamburg 50

+D. Methling
Gesellschaft für Nuklear-Service (GNS)
mbH
Rosastrasse 15
4300 Essen

K. Metscher
Kraftanlagen AG
Im Breitspiel 7
6900 Heidelberg

H. Meyer
Kernforschungsanlage Jülich
Zentralabteilung Strahlenschutz
Postfach 1913
5170 Jülich 1

H. Miggenrieder
Bayrisches Staatsministerium für
Landesentwicklung und Umweltfragen
Rosenkavalierplatz 2
8000 München 81

S. Mika
Hochtemperatur-Reaktorbau GmbH
Abt. TK 1
Gottlieb-Daimler-Strasse 8
6800 Mannheim 1

W.-D. Mischke
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H.-U. Mittmann
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. J. Möller
Firma G. Bessau
Geo-Plate-Strasse 5
2850 Bremerhaven-K.

H. Moll
Energie Versorgung Schwaben AG
Postfach 158
7000 Stuttgart 1

+Author

+J. Motz
Thyssen Gießerei AG
Abt. Forschung und Entwicklung
Postfach 1767
4650 Gelsenkirchen

A. Müller
Technischer Überwachungsverein
Hannover e.V.
Loccumer Strasse 63
3000 Hannover 81

K. [redacted]ller
Uhde [redacted]mbH
Abt. TB V
Postfach 262
4600 Dortmund 1

A. Naseband
Interatom
Postfach
5060 Bergisch-Gladbach 1

K.-H. Naumann
Bleiwerk Goslar KG
Im Schleeke 8
3380 Goslar

C. Nehmert
Bundesanstalt für Materialprüfung
Unter den Eichen
1000 Berlin 45

+R. Neider
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+O. Nommensen
Kernforschungsanlage Jülich
Institut für Chemische Technologie
Postfach 1913
5170 Jülich

J. Obladen
Reederei und Spedition "Braunkohle"
GmbH
Po[redacted]h 52
50[redacted]sseling 1

H.-J. Obst
Gg. Noell GmbH
Schweinfurter Strasse 28
8700 Würzburg 2

W. Ohmann
Reederei und Spedition "Braunkohle"
GmbH
Postfach 52
5047 Wesseling 1

+O. Opp
Gesellschaft für Strahlen- und Umwelt-
forschung
Institut für Tieflagerung
Theodor-Heuss-Strasse 4
3300 Braunschweig

J. Palmowski
Kraftanlagen AG
Im Breitspiel 7
6900 Heidelberg

+P. Petrasch
Nuklear-Ingenieur-Service GmbH
Lyoner Strasse 22
6000 Frankfurt/Main

U. Pfeifer
Hoesch Belfa AG
Werk Attendorn
5952 Attendorn

H. Phillipp
Fachinformationszentrum Energie,
Physik, Mathematik GmbH
Kernforschungszentrum
7514 Eggenstein-Leopoldshafen 2

H.-M. Pietz
Interatom
Postfach
5060 Bergisch-Gladbach 1

H. Pirk
NUKEM GmbH
Postfach 110080
6450 Hanau 11

U. Probst
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

M. Quintin
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+Author

H.-J. Reinecke
Isotopen Labor der Universität Tübingen
Auf der Morgenstelle 24
7400 Tübingen

H. Richter
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

P. Riem
Gemeinschaftskernkraftwerk Neckar
Postfach
7129 Neckarwestheim

+D. Rittscher
Gesellschaft für Nuklear-Service (GNS) mbH
Rosastrasse 15
4300 Essen

J. D. Rollins
Gesellschaft für Nuklear-Service (GNS) mbH
Rosastrasse 15
4300 Essen

O. Rosenbaum
Der Minister für Arbeit, Gesundheit und Soziales des Landes Nordrhein-Westfalien
Landeshaus
4000 Düsseldorf

P. Rothenbacher
Schwäbische Hüttenwerke GmbH
Postfach 3280
7080 Aalen-Wasseralfingen

M. Rühmkorf
COC-Kongressorganisation GmbH
Mühlenstrasse 58
1000 Berlin 46

U. Sach
STEAG Kernenergie GmbH
Bismarckstrasse 54
4300 Essen 1

M. Sack
Senator für Arbeit
Postfach 101527
2800 Bremen 1

H. Schacky
Rheinisch-Westfälisches Elektrizitätswerk AG
Postfach 27
4300 Essen 1

M. Scheffler
Senator für Arbeit und Soziales
An der Urania 4-10
1000 Berlin 30

R. Scheunemann
Deutschlandfunk
Radeberggürtel
5000 Köln 51

+F. E. Schilling
Siempelkamp Gesellschaft für Guss- und Reaktortechnik mbH
Siempelkampstrasse 45
4150 Krefeld

R. Schimmel
Gurlittstrasse 53a
4000 Düsseldorf

G. J. Schimmele
Badenwerk AG
Postfach 1680
7500 Karlsruhe 1

K. Schimmelpfennig
Zerna, Schnellenbach und Partner
Postfach 102869
4630 Bochum 1

E. Schlich
NUKEM GmbH
Postfach 110080
6450 Hanau 11

D. Schmidt
Allianz-Versichungs-AG
Königinstrasse 28
8000 München 44

G. Schmidt
NIS Nuklear-Ingenieur-Service GmbH
Hausmannstrasse 23
6450 Hanau

W. Schmidt
KEWA GmbH
Hamburger Allee 4
3000 Hannover 1

+Author

W. Schmitz
Domplatz 1-3
4400 Münster

H. Schneider
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+K. Schneider
Transnuklear GmbH
Po[redacted]h 110030
64[redacted]nau 11

R. Schneider
Technischer Überwachungsverein Bayern
Postfach 210 420
8000 München 21

+R. Schüler
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

K. Schulz
Dornier System GmbH
Postfach 1360
7990 Friedrichshafen 1

+B. Schulz-Forberg
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Schulze
Rheinisch-Westfälischer Technischer
Überwachungsverein
Langemarckstrasse 20
4300 Essen

W. Schulze
Gemeinschaftskernkraftwerk Neckar
Postfach
7129 Neckarwestheim

+H.-H. Schweer
Physikalisch-Technische Bundesanstalt
Bu[redacted]lee 100
330[redacted]aunschweig

C. Segebade
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+Author

H. Siebel
Der Minister für Arbeit, Gesundheit
und Soziales des Landes Nordrhein-
Westfalen
Landeshaus
4000 Düsseldorf

R. Sieber
GANUK
Castellstrasse 1
6451 Hammersbach

H. Siwek
Ministerium für Arbeit, Gesundheit
und Sozialordnung, Baden-Württemberg
Postfach 1250
7000 Stuttgart 1

B. Spielmann
URANIT GmbH
Postfach 1411
5170 Jülich

H. Spilker
Kernbrennstoff-Wiederaufbereitungs-
GmbH
Hildesheimer Strasse 47
3000 Hannover 1

+T. Staake
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

J. Stellpflug
Die Tageszeitung
Wattstrasse 11/12
1000 Berlin 65

P. Stipanits
Kernforschungsanlage Jülich
Abt. Dekontamination
Postfach 1913
5170 Jülich

R. Stippler
Gesellschaft für Strahlenforschung
Institut für Tieflagerung
Theodor-Heuss-Strasse 4
3300 Braunschweig

S. Storch
Kernforschungsanlage Jülich
Abt. AP
Postfach 1913
5170 Jülich

L. Sütterlin
Gesellschaft für Reaktorsicherheit mbH
Glockengasse 2
5000 Köln 1

W. Syborg
COC-Kongressorganisation GmbH
Mühlenstrasse 58
1000 Berlin 46

K. Tägder
Technischer Überwachingsverein
Loccumer Strasse 63
3000 Hannover 81

T. Tamberg
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

M. Tennie
Kraftwerk Union AG
Abt. R 331
Berliner Strasse 295-299
6050 Offenbach/Main

R. Theenhaus
Kernforschungsanlage Jülich
Postfach 1913
5170 Jülich

W. Thömel
Kraftwerk Union AG
Berliner Strasse 295-299
6050 Offenbach/Main

M. Thomany
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

+U. Timm
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

+R. Trapp
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Uhlenbruck
Gesellschaft für Reaktorsicherheit mbH
Glockengasse 2
5000 Köln 1

A. Ulrich
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Unger
Kraftwerk Obrigheim GmbH
Postfach 100
6951 Obrigheim

H. Vaidyanathan
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

H. Veith
Bundesanstalt für Materialprüfung 87
Unter den Eichen 87
1000 Berlin 45

H. J. Viecenz
Gg. Noell GmbH
Schweinfurter Strasse 28
8700 Würzburg 2

B. Vogel
Buderus AG
Postfach 1220
6330 Wetzlar

H. Vornhusen
KEWA GmbH
Hamburger Allee 4
3000 Hannover 1

A. J. Vox
Scharnhauserstrasse 73
7302 Ostfildern-Ruit

P. Vygen
Transnuklear GmbH
Postfach 110030
6450 Hanau 11

I. Warnecke
Physikalisch-Technische Bundesanstalt
Bundesallee 100
3300 Braunschweig

H. Weber
Stahlwerke Peine
Sonnenblick 4
3320 Salzgitter

+G. Wehner
Bundesministerium des Innern
Postfach 170290
5300 Bonn 1

+Author

G. Weinhold
STEAG Kernenergie GmbH
Bismarckstrasse 54
4300 Essen

+H.-P. Weise
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

E. Wildner
Bl[redacted]burgerstrasse 6
80[redacted]nchen 2

M. Wermann
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

F. Werner
Siempelkamp Gesellschaft für Guss-
Reaktortechnik mbH
Siempelkampstrasse 45
4150 Krefeld

K. Werner
Deutsche Bundesbahn
Zentralamt Minden
Postfach 2960
4950 Minden

+K. Wieser
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

B. Wilke
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

E. Wörner
Gemeinschaftskernkraftwerk Neckar
Postfach
7129 Neckarwestheim

D. Wolters
Thys[redacted]Giesserei AG
Pos[redacted]011220
4330[redacted]heim/Ruhr

W. Wurtinger
Gesellschaft für Reaktorsicherheit
mbH
Glockengasse 2
5000 Köln 1

Author

H. Wust
Gesellschaft für Reaktorsicherheit
mbH
Glockengasse 2
5000 Köln 1

W. Zander
Bundesanstalt für Materialprüfung
Unter den Eichen 87
1000 Berlin 45

M. Ziesler
Reederei und Spedition "Braunkohle"
GmbH
Postfach 52
5047 Wesseling 1

U. Zimmermann
COC-Kongressorganisation GmbH
Mühlenstrasse 58
1000 Berlin 46

W. Zimmermann
Kernforschungszentrum GmbH
Postfach 3640
7500 Karlsruhe 1

. FINLAND

+A. Leinio
Institute of Radiation Protection
P.O. Box 268
00101 Helsinki 10

A. Sihvola
Imatran Vioma Oy
Power Station Department
P.O. Box 138
00101 Helsinki 10

V. P. Törmälä
Imatran Vioma Oy
P.O. Box 138
00101 Helsinki 10

O. Usvasalo
Kutojantie 8
02630 Espoo

+S. Vilkamo
Institute of Reactor Safety
P.O. Box 268
00101 Helsinki

+S. Vuori
Technical Research Centre of Finland
Nuclear Engineering Laboratory
P.O. Box 169
00181 Helsinki 18

L. E. Willberg
Institute of Radiation Protection
P.O. Box 268
00101 Helsinki

FRANCE

+B. Bazelaire
Commissariat à l'Energie Atomique
CESTA
Le Barp
33830 Belin Beliet

R. Bertonneau
53 Avenue de Neufville
91540 Mennecy

+P. Blum
Transnucléaire S.A.
11 rue Christophe Colomb
75008 Paris

+C. Bochard
Robatel Slpi
B. P. No. 20
69740 Genas

Mr. Bouby
Electricité de France
SEPTEN
B. P. No. 8
92080 Paris Cedex

R. Brissier
Commissariat à l'Energie Atomique
CESTA
Le Barp
33830 Belin Beliet

+H. Bruel
Robatel Slpi
B. P. No. 20
69740 Genas

+J. Chatelet
Laboratoire National d'Essais
1 rue Gaston Boissier
75015 Paris

+Author

G. Chevalier
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de Saclay
DEMT
B. P. No. 2
91160 Gif-sur-Yvette

+G. Cohendy
Commissariat à l'Energie Atomique
Commission de Sureté des Transports
B. P. No. 6
92260 Fontenay-aux-Roses

+C. Cuillerdier
Commissariat à L'Energie Atomique
Centre d'Etudes Nucléaires
B. P. No. 6
92260 Fontenay-aux-Roses

+M. Dobremelle
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de Saclay
DEMT/SYST/TCR
B. P. No. 2
91160 Gif-sur-Yvette

+B. Duret
4 rue Dio Dore Rahoult
38000 Grenoble

G. Feidt
Commissariat à l'Energie Atomique
CESTA
Le Barp
33830 Belin Beliet

+J. Fradin
Compagnie Générale des Matières
Nucleéaires
Section des Transports Spéciaux
La Boursidière
92357 Le Plessis Robinson

R. Gaucherot
Compagnie Générale des Matières
Nucléaires
La Boursidière
92357 Le Plessis Robinson

+J. Geffroy
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de Saclay
DEMT/SYST/TCR
B. P. No. 2
91190 Gif-sur-Yvette

R. Genevray
Compagnie Générale des Matières
Nucléaires
La Boursidière
92357 Le Plessis Robinson

+M. Grenier
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
DSN/SASICC
B. P. No. 6
92260 Fontenay-aux-Roses

+J. [redacted]rd
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
DPR/SPS
B. P. No. 6
92260 Fontenay-aux-Roses

+D. Haon
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
B. P. No. 6
92260 Fontenay-aux-Roses

G. Jeanpierre
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
B. P. No. 6
92260 Fontenay-aux-Roses

+J. Joly
Novatome
149 cours la Fayette
69006 Lyon

J. Kerjean
13 ave. du Golf
44800 St. Herblain

+B. Kirchner
Transnucléaire S.A.
11 rue Christophe Colomb
75008 Paris

+S. Lazarevitch
Nuclear Transport Limited
c/o [redacted]agnie Générale des Matières
Nucl[redacted]s
B. P. No. 270
50107 Cherbourg

G. Le Bastard
Compagnie Générale des Matières
Nucléaires, BR/GIC
La Boursidière
92357 Le Plessis Robinson

Author

B. Lenail
Compagnie Générale des Matières Nucléaires
Branche Retraitment
La Boursidière
92357 Le Plessis Robinson

+D. Mangin
Commissariat à l'Energie Atomique
Centre d'Etudes de Valduc
SEESNC
B. P. 21
21120 Is-sur-Tille

+L. Maubert
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
DSN/SEC
B. P. No. 6
92260 Fontenay-aux-Roses

+M. Moulin
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de Saclay
B. P. No. 2
91160 Gif-sur-Yvette

H. Murano
Nissho Iwai France
2 rue des Pyramides
75001 Paris

J. Olivier
Organisation for Economic Cooperation
and Development
Nuclear Energy Agency
38 Boulevard Suchet
75016 Paris

+R. Peclier
Compagnie Générale des Matières
Nucléaires
Etablissement de Pierrelatte
B. P. No. 10
26700 Pierrelatte

+R. Portal
NERSA
177 rue Garibaldi
69219 Lyon Cedex 1

+G. Poullot
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires
DSN/SEC
B. P. No. 6
92260 Fontenay-aux-Roses

+J.-P. Prulhière
Commissariat à l'Energie Atomique
CESTA
B.P. No. 2
Le Barp
33830 Belin Beliet

+R. Seran
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de Saclay
B. P. No. 2
91190 Gif-sur-Yvette

+Y. Sousselier
Commissariat à l'Energie Atomique
Commission de Sureté des Transports
B. P. No. 6
99260 Fontenay-aux-Roses

+V. Stadnikoff
Commissariat à l'Energie Atomique
CESTA
B. P. No. 2
Le Barp
33830 Belin Beliet

R. Talbot
Gondrand DATS
60 rue du Faubourg Poissonnière
75010 Paris

J. Teulon
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de
Cadarache
B. P. No. 1
13115 Saint-Paul-Lez-Durance

E. Tomachevsky
Commissariat à l'Energie Atomique
Centre d'Etudes Nucléaires de Saclay
DEMT
B. P. No. 2
91190 Gif-sur-Yvette

J. Vassard
Le Clos Salibert
3 les Bruyères
78860 St. Nom le Breteche

ITALY

R. Bertini
Enel
Viale Regina Margherita 137
00198 Roma

E. Celant
Commission of the European Communities
Joint Research Centre Ispra
21021 Varese

+M. Conti
Comitato Nazionale per l'Energia
Nucleare
RAD - RSI
CNS Casaccia
C.P. 2400
00100 Roma

+M. Costa
Costa Armatori S.p.A.
Via G. D'Annunzio 2/20°Floor
16121 Genova

+I. Demarco
Comitato Nazionale per l'Energia
Nucleare
Via Le Regina Margherita 125
00198 Roma

+C. Faloci
Comitato Nazionale per l'Energia
Nucleare
Via Le Regina Margherita 125
00198 Roma

+G. Forasassi
Istituto Impianti Nucleari
Via Diotisalvi 2
56100 Pisa

A. Gandellini
Agip Nucleare
Corso di Porta Romana 68
20122 Milano

+A. Olivieri
Agip Nucleare
Corso di Porta Romana 68
20122 Milano

Mr. Ottati
Agip Nucleare
Corso di Porta Romana 68
20122 Milano

+S. Piermattei
Comitato Nazionale per l'Energia
Nucleare
Divisione Protezione del l'Ambiente
Servizio Trasporti
Viale Le Regina Margherita 125
00198 Roma

+Author

JAPAN

+S. Aoki
Research Laboratory for Nuclear Reactors
Institute of Technology
2-12-1 Ohokayama
Meguro-ku
Tokyo

+Y. Fujise
D[redacted]pment Department
M[redacted]ishi Heavy Industries Ltd.
Ma[redacted]ouchi, Chiyoda-ku
Tokyo

+S. Fukuda
Central Research Institute for Electric Power Industries
Otemachi Building
1-6-1 Otemachi-Chiyodaku
100 Tokyo

K. Fukudaome
Kobe Steel Ltd.
1-3-18 Wakihama-cho
Kobe-shi
Ashirya-ku

Y. Futamura
Japan Nuclear Ship Development Agency
1-15-16, Toranomon
Minato-ku
Tokyo

A. Hayashi
Nuclear Fuel Cycle Engineering Dept.
Mitsubishi Heavy Industries Ltd.
Marunouchi, Chiyoda-ku
Tokyo

K. Itoh
Japan NUS Co. Ltd.
6-2, Ohtemachi, 2-chome
Chiyoda-ku
Tokyo

T.[redacted]hiro
Nu[redacted] Transport Service Co. Ltd.
1-1-[redacted], Shiba, Daimon
Minato-ku
Tokyo

+K. Kage
Nuclear Transport Service Co. Ltd.
1-1-3, Shiba, Daimon
Minato-ku
Tokyo

+H. Kajishima
Nuclear Transport Service Co. Ltd.
1-1-3, Shiba, Daimon
Minato-ku
Tokyo

H. Kasukawa
Nuclear Safety Research Association
1-2-2 Osaka Building
Uchisaiwai-chu
Chiyoda-ku
Tokyo

K. Kataoka
Hitachi Express & Tokyo Monorail Co.
3-6-3 Shibuya
Shibuya-ku
Tokyo

M. Kobayashi
The Japan Steel Works Ltd.
4 Chatsu-machi
Muroran-shi
Hokkai-do

+K. Konno
Fuels and Material Group
FBR Development Project
9-13, 1-chome
Akasaka, Minato-ku
Tokyo

H. Kuranaga
Nissho Iwai Corporation
4-5 Akasaka, 2-chome
Minato-ku
Tokyo

I. Maruyama
Yoshizawa Kiko Co. Ltd.
2-17-7, Kyobashi
Chuo-ku
Tokyo

T. Meguro
Nuclear Fuel Services Commission Ltd.
2-2, 2-chome, Uchisaiwaicho
Chiyoda-ku
Tokyo

+R. Mori
Nippon Nuclear Fuel Development Co.
2163, Narita-cho, Oarai-machim
Higashiibaraki
Ibaraki-ken

+Author

T. Nakajima
Tokyo Engineering Corporation
Chipa Office
Shoko Building
1-10-10, Hon-cho-Funabashi-shi
Chiba

Y. Nakatani
Mitsui Engineering & Shipbuilding Co.
6-4, Tsukiji, 5-chome
Chuo-ku
Tokyo

Y. Nakazato
JGC Corporation
Nuclear Division
1-14-1, Besho, Minami-ku
Yokohama-city
Kangawa- en 232

+A. Onodera
Hitachi Shipbuilding & Engineering Co. Ltd.
1-3-40, Sakurajima
Konohana-ku
Osaka

+A. Sekiguchi
Dept. of Nuclear Engineering
Faculty of Engineering
University of Tokyo
3-1, Hongo, 7-chome
Bunkyo-ku
Tokyo

+S. Shimamura
Mechanical Engineering Laboratory
Agency of Industrial Science and Technology
Ministry of Internal Trade & Industry
1-2, Namiki, Sakura-mura, Niihari-gun
Ibaraki-ken

+H. Takano
Central Research Institute of Electric Power Industry
1646 Abiko, Abiko-shi
Chiba

T. Tamaki
Japan Nuclear Fuel Services Co. Ltd.
2-2-2, Uchisaiwai-cho
Chiyoda-ku
Tokyo

+Author

+K. Tanaka
Nuclear Safety Research Association
1-1-2 Osaka Building
Uchisaiwai-cho
Chiyoda-ku
Tokyo

T. Yagi
JGC Corporation
1-14-1, Besho, Minami-ku
Kanagawa-ken 235
Yokohama-city

MALAYSIA

F. Mohd Amin
Tun Ismail Research Centre
No. 4, Jalan 114
Bandar, Baru Bangi
Selangor

THE NETHERLANDS

K. H. Cloes
Commission of the European Communities
Joint Research Centre
Postbus 2
1755 Petten ZG

J. A. Kaihatu
Weth. Baltussenweg 9
Driel GB

H. Nieuwehhuizen
Neth. Energie
3 Westerduihweg
Petten NH

J. Ostermann
URENCO Nederland Operations B.V.
Planthofsweg 81
7600 Almelo

H. A. Selling
Ministry of Health and Environmental Protection
Radiation Protection Division
P.O. Box 5811
2280 Rijswijk HV

E. Steinebach
URENCO Nederland Operations B.V.
Planthofsweg 81
7600 Almelo

NORWAY

+H. Aamlid
State Institute of Radiation Hygiene
P.O. Box 55
1345 Osteraas

G. A. Ronneberg
Institute for Energy Technology
P.O. Box 40
2007 Kjeller

+U. [redacted]en
Ins[redacted]ute for Energy Technology
P.O. Box 40
2007 Kjeller

PAKISTAN

M. Ahmad
Pakistan Atomic Energy Commission
P.O. Box 1356
Islamabad

POLAND

T. Glebocki
The Radioisotope Production and
Distribution Centre
Institute of Nuclear Research
OPIDI-1BJ, Swierk
05-400 Otwock

T. Musialowicz
Central Laboratory for Radiological
Protection
Konwaliowa 7
01-194 Warsaw

SOUTH AFRICA

D. Sykes
ESCOM
P.O. Box 1091
Joh[redacted]sburg

B. [redacted]inkler
Private Bag X256
Pretoria 0001

SPAIN

M. Lizzaraga
Transnuclear S.A.
Hermosilla 57
Madrid 1

T. San Segundo
Junto de Energin Nuclear
Departmento de Seguridad Nuclear
Avda. Complutense, s/n
Madrid 3

SWEDEN

P. Behrenz
Asea-Atom
P.O. Box 53
72104 Västeras

B. Dufva
Swedish Nuclear Power Inspectorate
P.O. Box 27106
10252 Stockholm

+P. I. Ek
Swedish Nuclear Power Inspectorate
P.O. Box 27106
10252 Stockholm

A. M. Ericsson
Kemakta Konsult AB
Linnegatan 52
11454 Stockholm

H. O. Fransson
Svendk Kärnbränsleförsörjning AB
P.O. Box 1746
11187 Stockholm

E. Grimsedt
Studsvik Energitechnic AB
61182 Nyköping

J. F. Gustaffson
Svendk Kärnbränsleförsörjning AB
P.O. Box 1746
11187 Stockholm

S. Johnson
Studsvik Energitechnic
61182 Nyköping

+T. C. Milchert
Salen Technologie AB
10609 Stockholm

+Author

B. Persson
National Institute of Radiation Protection
P.O. Box 60204
10401 Stockholm

B. G. Pettersson
Swedish State Power Board
16287 Vällingby

C. Sjökvist
The Swedish Plant Inspectorate
P.O. Box 49306
10028 Stockholm

SWITZERLAND

B. Balmer
Nuclear Safety Division
5303 Würenlingen

H. Bay
Nordostschweizerische Kraftwerke AG
5401 Baden

A. Clausen
Nordostschweizerische Kraftwerke AG
5401 Baden

P. Doroszlai
Elektrowatt Eng. Co.
Bellerivestrasse 36
8022 Zürich

V. Kohut
Elektrowatt Eng. Co.
Bellerivestrasse 36
8022 Zürich

H.-J. Nebiker
Kernkraftwerk Union Däniken AG
Postfach 55
4658 Däniken

J. Stobbs
Nuclear Assurance Corporation
Weinbergstrasse 9
8001 Zürich

UNITED KINGDOM

S. C. Alexander
United Kingdom Atomic Energy Authority
AERE Harwell, Building 528/9
Harwell, Didcot, Oxon.

P. Alton
Rolls-Royce and Associates Ltd.
P.O. Box 31
Derby DE2 8BJ

K. L. Ashman
Gravetom Industries Ltd.
Gosport, Hants.

A. Barker
United Kingdom Atomic Energy Authority
Risley Nuclear Power Development Establishment
Central Technical Services
Room E428
Risley, Warrington, Cheshire WA3 6AT

D. Bibby
Nuclear Transport Ltd.
Delenty Drive
Risley, Warrington, Cheshire WA3 6AS

+M. Brown
United Kingdom Atomic Energy Authority
DNPDE
Dounreay, Caithness

+R. S. Butterfield
United Kingdom Atomic Energy Authority
Risley Nuclear Power Development Establishment
Risley, Warrington, Cheshire WA3 6AT

+J. C. Chicken
Culham Laboratory
United Kingdom Atomic Energy Authority
Abingdon, Oxon.

+W. D. Curren
Atomic Energy Establishment
Building A52
Winfrith, Dorchester, Dorset

+H. W. Curtis
Nuclear Transport Ltd.
Delenty Drive
Risley, Warrington, Cheshire WA3 6AS

L. Curtis
British Nuclear Fuels Ltd.
Risley
Warrington, Cheshire WA3 6AS

J. T. Daniels
United Kingdom Atomic Energy Authority
Safety & Reliability Directorate
Wigshaw Lane
Culcheth, Warrington, Cheshire WA3 4NE

+Author

B. W. Emerson
H. M. Nuclear Installations Inspection
Tithebarn Street
Liverpool L2 2LZ

A. Emmerton
Atomic Energy Establishment
Building A31
Winfrith, Dorchester, Dorset

J. Gatley
Br[redacted] Nuclear Fuels Ltd.
Ri[redacted]
War[redacted]gton, Cheshire WA3 6AS

+E. P. Goldfinch
Central Electricity Generating Board
Safeguards Branch
Nuclear Health and Safety Dept.
Courtenay House, Warwick Lane
London EC4

J. E. Gore
Central Electricity Generating Board
Scientific Services Dept.
Timpson Road
Wythenshawe, Manchester M23 9LL

P. Haigh
Central Electricity Generating Board
Sudbury House
15 Newgate Street
London

R. F. Halliburton
Babcock Power Ltd.
165 Great Dover Street
London SE1 4YB

J. D. Hart
Central Electricity Generating Board
Generation Development and Construction Division
Barnett Way
Barnwood, Gloucester GL4 7RS

M. [redacted]ey
Nu[redacted] Transport Ltd.
Dele[redacted] Drive
Risley, Warrington, Cheshire WA3 6AS

B. Holyoak
Health & Safety Executive
25 Chapel Street
London NW1 5DT

L. James
Nuclear Transport Ltd.
Delenty Drive
Risley, Warrington, Cheshire WA3 6AS

E. Livesey
British Nuclear Fuels Ltd.
Risley
Warrington, Cheshire WA3 6AS

+D. H. Lohmann
The Radiochemical Centre
Product Finishing Dept.
Amersham HP7 9LL

+H. F. MacDonald
Central Electricity Generating Board
Berkley Nuclear Laboratories
Berkley, Glos. GL13 9PB

W. G. Milne
British Nuclear Fuels Ltd.
Risley
Warrington, Cheshire WA3 6AS

A. H. Partridge
Dept. of the Environment
Dangerous Goods Branch
2 Marsham Street
London SW1P 3EB

M. C. Peters
Babcock Power Ltd.
165 Great Dover Street
London SE1 4YB

D. R. Poulter
United Kingdom Atomic Energy Authority
Safety & Reliability Directorate
Wigshaw Lane
Culcheth, Warrington, Cheshire W13 4NE

+M. Price
Atomic Energy Establishment
Building A32
Winfrith, Dorchester, Dorset

+K. B. Shaw
National Radiological Protection Board
Harwell, Didcot, Oxon.

G. Swindell
33 Tetbury Street
Minchinhampton, Glos. GL6 9JH

+Author

R. Smith
Lead Development Association
34 Berkeley Square
London W1X 6AJ

+C. B. G. Taylor
The Radiochemical Centre
Isotope Production Unit
Building 10.23
Harwell, Oxon. OX11 ORG

+B. Y. Underwood
United Kingdom Atomic Energy Authority
Risley Nuclear Power Development
Establishment
Risley, Warrington, Cheshire WA3 6AT

H. Wardelmann
IMCO
101-104 Piccadilly
London W1V OAE

G. B. Williamson
Nuclear Operations Branch
24 Cathedral Place
London EC4

S. Williamson
British Nuclear Fuels Ltd.
Risley
Warrington, Cheshire WA3 6AS

+B. Wright
NEI Projects Ltd.
Cothbert House
All-Saints, Newcastle-upon-Tyne
NE1 2DP

C. N. Young
Department of Transport
2 Marsham Street
London SW1P 3EB

+R. L. D. Young
United Kingdom Atomic Energy Authority
Risley Nuclear Power Development
Establishment
Risley, Warrington, Cheshire WA3 6AT

UNITED STATES OF AMERICA

+G. C. Allen Jr.
Sandia National Laboratories
Division 4552
Albuquerque, NM 87185

+J. Allen
US Department of Transportation
Washington DC 20590

+J. A. Andersen
Sandia National Laboratories
Division 1725
Albuquerque, NM 87185

+W. B. Andrews
Pacific Northwest Laboratories
P.O. Box 999
Richland, WA 99352

A. Anselmo
Eg & g Idaho Inc.
P. O. Box 1625
Idaho Falls, ID 83415

M. L. Bell
Rockwell Hanford Operations
MO-043, 200 East
Richland, WA 99352

+J. D. Berger
Westinghouse Hanford Company
W/A-98
P.O. Box 1970
Richland, WA 99352

W. G. Bevan
5812 W. Yellowstone Avenue
Kennewick, WA 99336

+J. C. Brantley
New England Nuclear Corporation
549 Albany Street
Boston, MA 02118

W. H. Brinkmann
Stearns-Roger Manufacturers Inc.
P.O. Box 5888
Denver, CO 80217

+O. F. Brown
US Department of Energy
1000 Independence Avenue SW
Washington DC, 20585

+Author

+W. A. Brobst
The Transport Environment Inc.
SR 285 Old Squaw Drive
Kitty Hawk, N. Carolina 27949

+R. Burgoyne
General Atomic Company
P.O. Box 81608
San Diego, CA 92122

+R. Chitwood
US [redacted]artment of Energy
T[redacted]rt Division
ITM[redacted]107
Washington, DC 20545

M. Clay
12633 Inkster Road
Livona, Michigan 48150

+J. D. Colton
SRI International
333 Ravenswood Avenue
Menlo Park, CA

C. R. Davis
General Electric Company
Advanced Reactor Systems Dept.
MC-524
P.O. Box 508
Sunnyvale, CA 94086

+J. DiStefano
US Department of Energy
1000 Independence Avenue SW
Washington, DC 20585

+E. T. Driver
National Transportation Safety Board
800 Independence Avenue SW
Washington, DC 20594

+P. E. Eggers
Ridihalgh, Eggers and Associates Inc.
2112 Juka Avenue
Columbus, Ohio 43201

+N. [redacted]isenberg
US [redacted]ear Regulatory Commission
Of[redacted] of Standards Development
Rockville, Maryland 20812

+N. C. Finley
Sandia National Laboratories
Division 4413
Albuquerque, NM 87185

+Author

K. Gablin
Private Consultant
6749 Towne Lane Road
McLean, Virginia 22101

+K. Goldmann
Transnuclear Inc.
One North Broadway
White Plains, NY 10601

+A. W. Grella
US Nuclear Regulatory Commission
Office of Inspection and Enforcement
Mail Stop EW-359
Washington, DC 20555

E.,C. Hardin
US Department of Energy
Albuquerque Operations Office
P.O. Box 5400
Albuquerque, NM 87115

+D. Harmon
Edlow International Company
1100 17th Street, N.W. Suite
Washington, DC 20036

+M. M. Heiskell
US Department of Energy
Oak Ridge Operations
P.O. Box E
Oak Ridge, TN 37830

P. F. Highberger
Allied General Nuclear Services
P.O. Box 847
Barnwell, S. Carolina 29812

+R. M. Jefferson
Sandia National Laboratories
Department 4550
Albuquerque, NM 87185

C. Johnson
Nuclear Assurance Corporation
24 Executive Park West
Atlanta, Georgia, 30329

+N. Johanson
General Atomic Company
P.O. Box 81608
San Diego, CA 92138

+B. Josephs
Transnuclear Inc.
One North Broadway
White Plains, NY 10601

+D. Joy
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

A. Kaplan
General Electric Company
M/C J26
P.O. Box 780
Wilmington, N. Carolina 28405

G. Kiesel
Japan Air Lines
1710 Gilbreth Road
Burlingham, CA 94010

+C. B. Killian
New England Nuclear Corporation
549 Albany Street
Boston, MA 02118

+R. W. Klingensmith
Battelle Columbus Laboratories
505 King Avenue
Columbus, OH 43201

+K. Kok
Battelle Columbus Laboratories
505 King Avenue
Columbus, OH 43201

+D. Krieg
Rockwell International
Rocky Flats Plant
P.O. Box 464
Golden, CO 80401

L. Macklin
Transnuclear Inc.
One North Broadway
White Plains, NY 10601

+J. Mangusi
Transnuclear Inc.
One North Broadway
White Plains, NY 10601

+J. D. McClure
Sandia National Laboratories
Division 4551
Albuquerque, NM 87185

+P. N. McCreery
Allied General Nuclear Services
P.O. Box 847
Barnwell, S. Carolina 29812

+Author

+J. Mishima
Pacific Northwest Laboratories
Battelle Boulevard
Richland, WA 99352

+R. B. Pope
Sandia National Laboratories
Division 4552
Albuquerque, NM 87185

+W. A. Pryor
US Department of Energy
P.O. Box E
Oak Ridge, TN 37830

+H. Rack
Sandia National Laboratories
Division 5835
Albuquerque, NM 87185

+R. R. Rawl
Department of Transportation
Materials Transportation Bureau
Washington, DC 20590

+R. T. Reese
Sandia National Laboratories
Division 4551
Albuquerque, NM 87185

+R. E. Rhoads
Pacific Northwest Laboratories
P.O. Box 999
Richland, WA 99352

+J. H. Riddle
400 Bradmoor Street
Richland, WA 99352

D. Ryskiewich
Ciba Geigy
410 Swing Road
Greensboro, N. Carolina 27409

+R. Schaich
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

+L. B. Shappert
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

+J. M. Taylor
Pacific Northwest Laboratories
P.O. Box 999
Richland, WA, 99352

B. Teer
Transnuclear Inc.
One North Broadway
White Plains, NY 10601

W. Terry
411 Quinn Road
K[redacted]lle, TN 37921

+R. H. Yoshimura
Sandia National Laboratories
Division 4552
Albuquerque, NM 87185

LATE ENTRIES

+Mr. D. A. Edling
Mound Facility
P.O. Box 32
Miamisburg, Ohio, 45042
USA

+J. Muskovszky
Transnuklear GmbH
Postfach 110030
6450 Hanau 11
Federal Republic of Germany

+Author