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

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VOLUME I

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OF RADIOACTIVE MATERIALS

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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.

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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

014 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

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	[redacted]
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	[redacted]
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
[illegible]6	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
[illegible]25	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	3
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)

097	*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
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	61
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)

[illegible]	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 .....

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 .....

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
[illegible]2	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
[illegible]6	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	99[illegible]
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[redacted]
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[redacted]
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
[redacted]	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	1423
	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

PLENARY SESSION

SAFETY IN TRANSPORTATION OF RADIOACTIVE MATERIALS

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Introduction

Transportation is the act of carrying something from one place to another. At some stage, transportation is involved in, and therefore a critical aspect of, every operation. Any operation in which radioactive material is used requires transportation of radioactive material. The operation and continued growth of the nuclear industry depend on safe transport of radioactive materials, in most cases by the transportation industry. It is recognized as the vital link for all phases of the entire nuclear fuel cycle (1).

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; but the facts must be definitive and understandable. Studies (2,3) 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.

I will discuss that along with some other problems in the transport safety area and identify some of the work ongoing to deal with those problems.

The Transportation Industry

The transportation industry is very large and exists everywhere. It moves several billion tons of cargo in the world every year (4), handles items of every size and shape, operates anywhere on earth, and uses every means of transport known to man. A sizable fraction of that cargo, millions of tons, is made up of hazardous goods. The estimated 8 million packages containing no more than a few tons of radioactive material transported in the world each year is not very important by comparison. Furthermore, studies (5,6) indicate that less than 5% of those shipments (or less than 400,000 packages) contain appreciable amounts of radioactivity.

Nonetheless, as the transportation people observe, those few shipments which provide insignificant additional income to the transportation industry attract lots of attention. Furthermore, packages of radioactive material require special handling and a disproportionate amount of additional effort in such areas as paperwork, placarding, special training and instruction of workers, segregation from people, insurance and preparation for emergencies.

The Nuclear Industry

Radioactive materials in use in industry, medicine, research and power generation are of many types, forms and quantities. They present a wide range of hazards from two characteristics uniquely associated with radioactive materials: emission of ionizing radiation and nuclear criticality. In transportation, protective packaging is provided so that these special characteristics add little to the hazards normally associated with transportation.

The record of safety in transportation did not occur by chance. Early on, the need to control the hazards was recognized. Shipments of radioactive materials are made by persons familiar with their characteristics. Most shipments of appreciable

quantities are prepared for shipment by professionals in laboratories operated by large companies or by governments, persons who are conscientious, well-funded, aware of the need to assure safety in their activities and trained in radiation protection. Extra care has been exercised and more than adequate safety measures taken by competent persons out of respect for the hazards.

Nevertheless, transportation is vulnerable. The conditions encountered in transportation are largely uncontrolled, although predictable. And transportation takes place in the public domain. The transportation of spent fuel from a nuclear reactor involves taking highly radioactive segments of the reactor from a fixed location, within a containment shell, surrounded by an exclusion area and protected by multiple safety measures and transporting them in a simple container on a railcar or vehicle on the railway or public highway, through cities and towns, past homes and farms, in close proximity to people and subject to stresses of the transportation environment, including handling by unsophisticated personnel and accide

The number of shipments is increasing steadily and the number of persons involved in transportation is necessarily increasing rapidly. At the same time, budgets are being cut so that taking extra safety measures is not being encouraged. Further, strict standards have been imposed and the technology has been developed to meet those standards. The extra margins of safety are no longer being included to provide for lack of adequate quality control. The principle of keeping radiation exposures as low as reasonably achievable, economic and social considerations being taken into account, applies equally to eliminating precautions that are not justified and to requiring additional precautions if they are justified. All of these factors tend to increase the number of incidents that do occur.

In spite of every practical and reasonable precaution being taken, the occurrence of some incidents are unavoidable - absolute safety in any operation is not possible. And because it is so unusual to have any excessive radiation exposures in transportation, each incident that does occur is considered by some to be a major breach of safety and indicates lack of control. It is not only misleading, but adds to this problem to tell members of the general public that nothing can happen in transport of radioactive materials. However, the public is not reassured by a statement that 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 aspects. The public has few other hazards about which as much is known for comparison.

Sweden is carrying out a research project for the Agency on transport risk assessment (see the Environmental Risk Assessment discussed later in this paper) which is to include providing some perspective for judging the results. In a joint project with the International Institute of Applied Systems Analysis, the Agency is examining the basis for public health acceptance of risk. But, as yet, the concept of acceptable risk is not agreed on and what is acceptable in terms of risk is largely a matter of individual preference. Acceptability is very often influenced by what one perceives to be of direct personal benefit to him.

Regulations

Although safety results partly from attitudes and application of common sense, regulatory standards (i.e., regulations and requirements, criteria and codes of good practice, guides, advisory material and explanatory documents) also contribute to safety.

The International Atomic Energy Agency was established in 1957 to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". In the late 1950's, in view of the expanding use of radio-

active materials for peaceful purposes, the Agency undertook to develop harmonized safety rules for transportation of those materials on as wide a basis as possible and for all means of transport. Based on existing good practices and the few early, simple but excellent regulations already in effect, the Agency began a synthesis of such rules in 1958 working closely with other international bodies with competence and experience in the field of transport and utilising experts in the field from Member States from the East and West with advanced and developing nuclear programmes.

The first edition of the Agency's recommended Regulations for the Safe Transport of Radioactive Materials was published in 1961 (7). Soon after, the Agency published "Notes on Certain Aspects of the Regulations" (8), in which ideas underlying the Regulations were explained as part of a continuing programme to communicate the bases for and adequacy of the measures taken to achieve safety in transport of radioactive material.

The Regulations were revised and updated in 1964, 1967, 1973 and 1979 to reflect experience in their application and changes in the methods and technology.

By 1969, the Agency's recommended Regulations had been adopted by or used as the basis for regulations of almost all of the international organizations concerned with transportation and by most of the Agency's Member States (9). As a result, the Regulations are applicable to transportation of radioactive materials almost anywhere in the world.

The last comprehensive revision of the Agency's Regulations was published in 1973. Since that time, several minor amendments, mainly of an editorial nature, together with a number of changes of detail, approved under the "90-day" amendment procedure authorized by the Board of Governors, were promulgated by the Director General on 14 May 1975 and 16 December 1977. These, with some editing, were incorporated into the present text of the Regulations, which is referred to as the "1973 Revised Edition (As Amended)".

The Agency is in the process of carrying out the third comprehensive review and revision of the Regulations. An Advisory Group convened in July 1979 set out a number of basic principles underlying the transport safety regulations. A Technical Committee on Package Test Standards, which met in December 1977 and in August 1979, and the Agency's Standing Advisory Group for Safe Transport of Radioactive Materials (SAGSTRAM) at its meeting in Paris in April 1980 made specific recommendations for the comprehensive review.

In response to an invitation from the Agency, the Agency's Member States and interested International Organizations submitted more than 200 pages of comments and proposals for consideration in the review.

From 1-12 September 1980, the Agency convened an Advisory Group to make a Comprehensive Review of the Regulations and prepare proposals for necessary changes for which there was shown to be adequate justification. After a detailed and critical review of all but the nuclear criticality safety aspects, which are to be addressed by a group of specialists in 1981, the Group decided that a number of minor changes should be made, but only these significant changes were needed:

1. It was proposed that low specific activity materials be limited to those having a radiation level of no more than 1 rem/hr at 3 meters from the surface of the unshielded and uncontained material. Also, the packaging requirements for such materials were upgraded;

2. The following criteria for developing segregation tables were proposed for inclusion in the Regulations:

- for critical group of passengers	25 mrem/yr
- for workers not subject to radiation control	500 mrem/yr
- for film	10 mr
- for occupied working spaces and living quarters of seagoing vessels	0.75 mrem/hr;

3. It was proposed that the requirement be added to the accident test conditions that irradiated fuel packages containing more than 10^6 Ci be capable of being immersed under 200 meters of water without rupture of the containment system.

Other minor modifications were proposed including converting to SI units, clarifying and simplifying some of the language, allowing for the use of "overpacks", limiting the loss of shielding for Type A packaging after the normal test conditions to a 20% increase in the surface radiation level, prohibiting continuous venting of Type B(M) packages, requiring labels and placards to be removed or masked when not applicable, and "grandfathering" all packages manufactured to a design approved under the 1973 Revised Edition of the Regulations, that is, allowing them to continue to be used without review to determine if they meet the revised requirements. Several controversial issues were addressed. The Regulations were amended to provide greater recognition of other types of hazards associated with radioactive materia[l] segregation of radioactive materials from other hazardous goods and call attentio[n] to the need to avoid compromising safety when imposing controls on radioactive materials for other purposes. Also, a statement was added that routing restrictions for safety purposes are generally not necessary but "when such restrictions are imposed, account must be taken of all risks". Consideration was given to adding a crush test for lightweight packages but it was decided that sufficient data was not available at this time to determine whether there was adequate justification, so it could not be supported.

The draft revised Regulations will be circulated to Member States for review and comments about the end of this year. A second meeting of the Advisory Group will be held in 1982 to consider the comments and a final draft revision submitted for approval of the Board of Governors prior to publication of the revision currently planned for late 1983.

Based on results of this review and recommendations of the SAGSTRAM, it is unlikely that a comprehensive review would be undertaken again in less than 10 or 15 years.

Practical difficulties in international transport are caused by the fact that various editions of the Regulations with differing requirements are in effect at the same time. Application of the Regulations depends on adoption by many regulatory authorities and not all can act promptly or at the same time to adopt revisions. Therefore, keeping the regulatory requirements stable for as long as they are known to be adequate is highly desirable.

Objectives of the Regulations

To be effective, regulations must be practical, expressed in language that is readily understandable and free of detail that doesn't significantly add to safety. Regulations should be designed to achieve an acceptable level of safety. Understanding of the regulations and practicability of their application are important to compliance and enforceability. Communication between the regulators and the regulated is essential to compliance. Nevertheless, industry, not the regulatory bodies, bears the primary responsibility for assuring safety in transportation as in other regulated activities.

The Agency's programme in transport safety, while continuing to evolve and refine the standards, is being expanded to promote and assist in developing adequ[ate] compliance and enforcement programmes in Member States.

The Regulations were developed to protect the public, transport workers and property from radiation in the shipment of radioactive materials. Protection is achieved by a combination of limitations of the contents of a package according to the quantity and type of radioactivity, the package design, and certain simple handling, stowage and storage precautions which are to be followed during transport. The Regulations include provisions concerning containment of the radioactive mate-

rial, protection against radiation emitted by the material, safe dissipation of heat generated in the process of absorbing the radiation, and prevention of criticality when the material is fissile. While some provisions concern control by the carrier, for example segregation to limit the radiation levels in occupied areas, primary reliance for safety is placed on provisions relating to package design.

The overall safety of an operation is the summation of the safety inherent or "built-in" to the design of the shipment and facilities and that provided by the operator. In transport, the shipper is responsible for the safety built into the consignment and its preparation for carriage, and the carrier is responsible for operational control during transport. As for other hazardous goods, identification of radioactive materials in transport is provided by distinctive labels on each package and placards on trucks and railcars. A basic concept underlying the development of the trans ort regulations is to maximize the consigner's contribution to [redacted]erall safety, hereby reducing any requisite contribution by the carrier to a [redacted]imum.

There are some standard forms for communicating with the public. For example, the labels on packages convey meaning through the use of warning symbols and giving some specific information about the contents. The "radioactive" label advises members of the general public who might be in the vicinity that a package of radioactive material is present. It also informs the carrier of the characteristics of the package to the extent that is necessary. The label is placed on two sides of the package so that one is likely to be visible.

Also, for road or rail vehicles carrying packages with other than negligible radiation levels, a placard is required on front, back and both sides of the vehicle. The size and shape of placards vary somewhat and the information on the placard is not completely standardized. But, again, one of the purposes is to inform the public of the presence of radioactive material in the vehicle.

But, it's not easy to provide precautionary instructions without arousing undue concern for some people. Those interested in avoiding identification of shipments for reasons of physical security may consider that they are being compromised by a label on package of fissile material which is necessary for nuclear criticality safety controls. As a principle, however, safety must be kept paramount, i.e., other areas of control must not be permitted to compromise safety.

But assuring safety in transport and convincing the general public that it has been achieved require understanding that comes only with good communication among members of the many groups involved - shippers, carriers, receivers, regulators, transport workers, police, emergency crews and members of the general public, to name a few, as well as those concerned with allied fields, such as physical security.

Let's describe some of those groups and talk a little about their responsibilities.

Whoever prepares the material for transport is the shipper or consignor; whoever transports the material is the carrier and, of course, whoever receives the material is the receiver or consignee. The shipper and carrier often are the same person.

The shipper must assure that the material is prepared so that it can be trans[redacted]rted safely. This includes determining that each shipment complies with the ap[redacted]cable regulations and has the necessary approvals, the appropriate persons have [redacted]en notified, necessary insurance and indemnity coverage is in effect and plans for dealing with accidents or unusual circumstances have been made.

In any case, the basic responsibility for the material rests with the shipper. He originated the material, he knows what the material is, he knows its properties and he must assure that the material is prepared so that it can be transported safely. The radioactive material is his responsibility until it has been delivered to the consignee. And the shipper 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 principle of packaging for containment, shielding and nuclear criticality safety are applied so that the carrier is not required to exercise special radiation control practices while the package is in transit. From the time a package is picked up at the shipper's loading dock to the time that it is delivered to the consignee, routine monitoring of the package is not considered necessary and is normally not carried out.

The carrier must exercise due care in transporting the package from consignor to consignee. In an accident, he is responsible for contacting the shipper and proper authorities and dealing with the emergency, including segregating the public from any spill and cleaning up the area involved.

In addition to accepting the consignment, the receiver or consignee is expe[illegible] to determine if leakage or loss of radioactive material has occurred in transit a[illegible] to ascertain that carrier personnel and transporting vehicles are not contaminated. He must communicate the information quickly and in a manner that can be understood and is not likely to be misinterpreted.

The regulatory authority is responsible for setting minimum standards of safety. Safety regulations should be written such that compliance with the requirements is essential to protection of people and property (the environment); but demonstrating compliance may not be sufficient to avoid liability. Good safety practices or good common sense must supplement the regulatory requirements to provide adequate safety of people and property.

The regulators and the shippers share the responsibility to inform the public about the material being transported - to provide factual information in understandable terms with some perspective. Because the shipper knows most about the material he is shipping, he does most of the communicating.

Some Problems

Acceptance of Approved Packages. The Regulations suggest that each State designate one person or group as the "competent authority" for implementing the Regulations including issuing approvals of package design (11). The IAEA publishes a list of National Competent Authorities (12) annually. The levels of interest and competence vary dramatically and their authority may not be clearly understood or carefully defined.

In many cases, the Competent Authority refers requests for approval of package design to technical bodies within the State, some of whom carry out tests for the safety analysis report. In some States, cities' and local jurisdiction also must be consulted, usually only for very large shipments.

Certain package designs, Type B(U) for example, when approved by the competent authority in the country of origin of design, are expected to be accepted by the competent authorities in other States without further review. Some States refuse to accept such packages, sometimes causing serious delays in shipments. To help resolve this, the Agency is compiling a list of valid packaging approval certificates (13) and offering assistance in establishing adequate programs for implementing the Regulations.

Environmental Risk Assessment. Broad interest has developed in protecting t[illegible] environment. The Agency has a program underway to encourage and assist in evaluating the risk from transportation of radioactive material.

As the first phase, a request was issued to all Member States to gather information on shipments of radioactive material for the year 1980. Some of that data is already being submitted (13).

A model for assessing the risk including necessary parameters and a calculative

model are under development. The model with instructions will be circulated in 1981; during 1982-1983, each Member State will calculate the risk in his State and submit the results to the Agency. The Agency will estimate the risk from transport worldwide based on the individual results.

The acceptability of risk is a function of the perception of the hazard which is often quite different from the reality of the hazard. The public reacts more from fear and emotion rather than logic and information. Lack of a balanced public information contributes to this. Reliable information on shipments and associated risks can be used to allay public concerns.

Transport Data Collection. Need for public understanding to foster acceptance makes clear the importance of providing the public with all of the facts. Also, variations in the shipping documents leads to misunderstandings and delays in international movements. Work towards standardization is underway in the Agency and other international groups (14). To make more complete information available, the Agency has an ongoing program of transport data collection, compilation, analysis and reporting (13). Data on shipments, competent authority package approval certificates and accidents and incidents are being collected now and put into computer programs for analyses and reporting. The Agency will soon be in a position to respond to inquiries regarding technical details of those data. In some cases, this can supplement or even replace systems currently being used by individual countries for recording such data.

Conclusion

The transport regulations in effect for many years appears to be adequate. The record of safety is good and the estimated risk is small. Yet, the public is concerned. Perhaps we haven't communicated very well how safely we transport radioactive materials, or perhaps the public doesn't know what it wants.

REFERENCES

1. Proceedings of the Fifth International Symposium on Packaging and Transportation of Radioactive Materials, May 1978, Las Vegas, Nevada
2. Transportation of Radioactive Materials in Sweden, A Risk Study; A.M. Ericsson, September 1979
3. Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, NUREG 0170, Vol. I, December 1977
4. 1978 Statistical Yearbook, United Nations; NY 1979
5. Survey of Radioactive Material Shipments in the United States, BMWL-1972; April 1976
6. The Transportation of Radioisotopes in Italy, L. Failla (5th Int. Symp.), May 1978
7. Regulations for the Safe Transport of Radioactive Materials, IAEA Safety Series No. 6, STI/PUB/40, Vienna, 1961
8. Notes on Certain Aspects of the Regulations, IAEA Safety Series No. 7, Vienna, 1961
9. International Regulatory Control of the Transport of Radioactive Materials, G.E. Swindell, IAEA Bulletin, Vol. 21, No. 6, December 1979
10. Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition (As Amended), IAEA Safety Series No. 6, STI/PUB/517, Vienna 1979 (English version published in December 1979; French, Russian and Spanish in preparation)

11. Advisory Material for the Application of the IAEA Transport Regulations, IAEA Safety Series No. 37, 1973

12. National Competent Authorities List No. 12, July 1980, IAEA unpriced publication

13. Scope and Purposes of the IAEA Transport Information Collection System, D.M. Pal, IAEA Bulletin, Vol. 21, No. 6, December 1979

14. Inventory of the Regulations and Administrative Procedures Concerning the Carriage of Radioactive Materials in CEC Countries, prepared by Mrs. S. Piermattei for CEC (Draft Report, May 1980)

DISCUSSION

Question by A. R. Baker, United Kingdom: It seems to be implied that the external dose limits might be made more restrictive in future versions of the IAEA regulations. Are the regulations for impact and fire resistance likely to be altered in the future?

Answer: It is unlikely that the requirements for impact or fire resistance of type B packagings will be altered in the near future. Based on consideration of the small risk from transportation accidents, the Advisory Group for the comprehensive review of the Regulations which met in September 1980 decided there was not adequate justification for changes in those requirements.

EVOLUTION OF THE CONCEPT OF RISK IN RADIOACTIVE MATERIALS TRANSPORT

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Risk can be defined as the product of the consequences of an accident by the probability of this accident. It can be reduced either by acting on potential consequences, or by lessening the probability of occuring.

When were to be laid down the first regulations of IAEA, the experts did not practically have at their disposal any statistical data on the accident probabilities.

The traffic of radioactive material was indeed very low and at that time there had been, perhaps incidents, but at least no noticeable accident.

Of course, it could have been possible to found upon the statistics of other accidents, for instance in transports of other dangerous goods, which are submitted to regulations too. But it must be seen that these rules took in account accidents having already occured, when the ambition of the initiators of our regulations was, on the contrary, to lay down rules such that they prevent accidents. So it was difficult to found them upon such statistics and therefore, from the beginning, the basic idea was to ground the transport safety on the packaging : as soon as the shipped activity goes beyond a value such as the consequence of the external or internal irradiation caused by a spreading of the material out of its packaging would be unacceptable, this packaging must withstand severe accidental conditions, represented in the regulation by a certain number of well-known tests.

Besides the risk brought upon the workers in handling and transport operations was taken in account, by limiting the irradiation and surface contamination values.

The first issues of the regulations were essentially aimed at :

- Protecting the workers and public from any unacceptable risk, coming from an accident (obviously the first objective).

- Limiting the irradiation of involved workers in normal conditions.

It is to this end, that, since the origins, the irradiation doses were the subject of limitations : 200 mrem/h at the surface of the package and 10 mrem/h at 1 m of its surface.

It is to be noticed that, generally, the restraint comes from the limitation at a distance and not from the surface one. Therefore, the latter is more often, very lower than the limit. The regulations foresaw always in the same purpose limits to residual contamination, fixed and non fixed ; the first, because it is an irradiation source and able, in the long run to disperse itself ; the second, because obviously its dispersion during handling or normal transport conditions (wind, rain, etc...) must not infer engaged doses unacceptable for the concerned people, i.e. essentially the workers.

Nevertheless, the regulations were not taking in account the collective detriment i.e. the entire dose "distributed" to the whole public.

In the same way, it was not made mention of optimization, i.e. to aim at lessening as much as reasonably possible this detriment by appropriate steps ; for instance, in avoiding the highly populated areas, either in space (by deviated routes) or in time (transporting preferably by night).

In the same way, possible consequences on environment itself were not taken in account ; for instance the spreading into a river of radioactive materials being not in special form.

The result of implementation of regulations has been, on the whole, excellent. The integrated doses, in normal conditions, remained very low, and to our knowledge, they was no real problem for professionnally exposed people. No accident involving radioactive material entailed consequences other than negligible, and for spreading of contamination, only low specific activity materials (Uranate on a road in half life radiopharmaceutical Siuthern France in February 1975, or in Colorado in September 1977) or short were concerned.

It seems however possible to go further in optimization of regulations, in particular by taking in account the last Recommendations of International Commission of Radiological Protection (1976) founded upon the three principles of justification, optimization and respect of dose limits.

The respect of dose limits in an already solved problem as this prescription is absolutely imperative and was always taken in account. Experience showed, besides, that, in practice, this should not infer any difficulty.

As far as justification is concerned, the issue is no more stated ; either the use of nuclear energy is justified, and as a consequence, the transports of radioactive materials are justified too, or it not, and the transports of RAM, if they do not disappear entirely-how to put down, for instance, radiogamma-graphic devices and radiopharmaceuticals at least change completely their order of magnitude (assessing their importance by the activity or the mass of the shipped packages).

The essential problem remains therefore optimization. We have just said in what it consists ; it is not easy, if not to seize, at least to quantify and the specialist of cost/benefit studies have got there a large subject of reflexions and analysis.

This more especially as taking on account the consequences is needed either in normal conditions or in accidental ones.

The present basic theory ground upon the principle that, when the shipped activity is higher than A_1 (or A_2) the risk of release becomes unacceptable, and the containment must be assured, even in a severe accident.

It is in implementation of this principle ans in terms of the criteria ch were retained in order to define the severe accident, that were determined e tests for type B packages. The accidental condition optimization is therefore founded upon the resistance of these packages to the tests above.

It is, nevertheless, well known that these tests are not supposed to cover all the possible cases of accident. However, actually, we can see (as, in particular, the real size tests in United States have shown) that they go further than that is strictly foreseen by the regulations. This is due to the fact that the designer must, in the practical manufacture, take some margins. The fire test we carried out in France is to consider in the same menner. Optimization could, for a part, consist in considering whether it is not favourable to establish different sets of tests, as functions of transport mode. As attractive as could seem this approach, it appears to us, not only, unconsistent with the spirit of the regulation, but somewhat hazardous ; so, indeed, we should end at the concept of specialized packages, of which we should have to be assured they remain in what they are design for. Without speaking of the unavoidable complexities in the regulations themselves, it is strongly to be feared that, considering the risk of errors, such a specialization be going exactly at the opposite of what is aimed at i.e. : increasing the collective detriment and not lessenning it. In the same way, if optimization can also turn on the determination of A_1 and A_2, we must be aware of the need of avoiding to lessen them systematically ; doing that, indeed, would increase the number of shipments, and, very probably, the detriment.

On the contrary, it will be necessary to draw a special attention to transport of FAS and SFA materials. It is, indeed, a case where the basic theory is no longer applied. Without pronouncing the word, optimization was made in advance, balancing alleviated conditions of the transport of these materials with consequences of an accident involving them.

It seems sure to us that, in the field of these materials optimization must be thorougly considered in order to examine whether there is no need to proceed to any ajustment in corresponding regulations.

In normal conditions, optimization concerns essentially irradiation doses. ere again, it must be seen that possible decrease of the present limits does t lead, an through increase of the number of shipments to a growth final of the integrated doses for the workers.

This seems obvious when considering large packagings for irradiated fuels, of which the mass is limited by traffic difficulties, essentially in rail transport ; in this case, it is clear that in order to limit the irradiation dose, there is no other possibility than increasing the thickness of biological shieldings , to the detriment of the useful capacity. (that, besides, also

goes towards a diminution of the dose), and the increase of the number of shipment is unavoidable.

It is perhaps a little less obvious when looking at small packages.

It could be thought it is not very important to add, for instance, a further protection of 1 cm of lead. But, most of the time , it is the useful capacity of the vehicule, in mass and volume, which will be lessened, and, here too, the number of shipments will be increased.

The correct implementation of ICRP principles will not certainly infer changing the present values.

Optimization of collective must not infer, too, a diminution of these values, but, overall a study and comparison of transport modes.

As an example - they are numerous- it seems that, from that point of view, rail is better that route ; trains, indeed circulate, practically a part from the public ; and the marshalling yards are less "visited" than the parking areas on highways.

However, the analysis is very much to be deepened ; for instance, whether, a team of rail workers could not be tempted, in winter in a mashalling yard, to take advantage of the cover and warmth of irradiated fuel packages for having a picnic under the wagon.

At last, we must speak of optimization in the field of impact on environment, the impact can be assimilated to a risk, although, most of the time, it is not a direct consequence. In fact the first regulations took into account the risk of atmospheric contamination (obviously localized) that could infer a risk of internal or external irradiations. The risk of water contamination was not taken in account for it is always possible, practically, to avoid in this case direct consequences on the public other than negligible. From this point of view, this way of thinking is correct ; but not at all longer when considering the collective detriment. When about 200 kg of uranate were spread out on a road, in France, there was no difficulty to recover the most part of it ; if it had been uranyl nitrate, we would have practically recovered nothing and uranium and daughter radionuclides dispersed in nature, would have kept on, for more than five years, delivering microrems, or even much less, perhaps far from the accident location.

It would still more the case, in the wreck of a ship transporting irradiated fuels, which would not entail direct irradiations but could later induce contamination transfers, which, as far as collective detriment is concerned, would not be negligible.

Optimization will consist in investigations whether it is more pertinent to reduce the probabilities of the accident, by further reinforcing the packaging (i.e. going beyond type B , for example is making more stringent the conditions of immersion test), or by improving the safety of the ships, which would allow to keep package tests for all different mods of transport.

It is on purpose that we did not say anything of risk of accidents going far beyond the regulatory tests. Their probability being, by definition, extremely low, we must assume this risk, whatever could be the consequences, which, in transport of radioactive materials, can with difficulty be conceived as dramatic.

There has been no actual change in the concept of risk, but, as a matter of course, as we had experience and new recommendations were published by ICRP, the transport regulations are improving, especially in order to optimize the prescriptions and to make than as homogeneous as possible, in such a way as all transports of radioactives materials can still have the same very high safety.

DISCUSSION

Question by B. Schulz-Forberg, FR Germany: Concerning the two possibilities mentioned, i.e. to go (far) beyond the type B test or to look for greater safety in the sea-going vessel, I would like to know something about the latter possibility. In particualr in Japan, people are thinking of creating a double-hull ship for the transport of spent fuel. What measures should be taken for additional safety for sea-going vessels in France?

Answer: Maritime transport is certainly a delicate part of the field of transportation of radioactive materials, in particular the degree of safety against potential accident is certainly higher than it is for other modes of transport.

Increasing the resistancy of the sea-going vessel is certainly a good measure. But we also think in France that it is important to reinforce the crew because the modern navigational devices can cause, as a consequence, a relative decrease in surveillance.

For the cask itself, we consider an immersion test would be useful and not very costly.

Comment by J. T. Daniels, United Kingdom: In your presentation you referred to the possible hazard from the loss of radioactive material to the marine environment. The symposium may be interested in the results of some of our work. In the Safety Reliability Directorate of the UKAEA we have made a number of studies on this topic. One of them was to have been the subject of a paper by A. Taig but unfortunately it has not been put through the necessary clearance procedures in time for presentation. Hopefully it will appear at the next symposium. This work, which entailed a detailed quantitative examination of the possible methods of release of radioactive material from a package of irradiated fuel lost at sea, the transport of radioactivity through the marine environment and the possible path to man, led to the conclusion that any doses to man would be quite trivial.

Another paper by J. C. Chicken summarises a risk assessment of the transport of plutonium nitrate by sea which included a quantitative assessment of the effects of the total loss of 250 l plutonium nitrate to the shallow waters of the Irish Sea. Again, the ensuing risks to man were found to be trivial.

Answer: Of course we would be very interested in receiving these reports. According to reports already published, it is certain that the consequences of a possible release would be very low in terms of irradiation of individual members of the public. But it seems that the consequences would not be completely negligible for the global dose commitment, expressed for instance in man rem and we certainly have to study this point very carefully.

SAFETY PHYLOSOPHY AND ACCEPTABLE RISK

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1. SAFETY PHILOSOPHY OF THE TRANSPORT REGULATIONS

The regulations for the safe transport of RAM (RAdioactive Material) developed by the International Atomic Energy Organisation (IAEO) are based on the following safety principles: /1,2/

a) During normal transport the radiation exposure of persons should not exceed 500 mrem/a[+)] for transport or storage personnel and 25 mrem/a[++)] for other critical groups of the general public. This is achieved by prescribing dose rates, segregation distances and a limit for the total transport index.

b) In an accident no person should recieve a dose of more than 3 rem. This is achieved by the requirement of different packaching types according to the potential danger of the radioactive material. The following packaging types are used: No packaging, normal, strong industrial, type A and type B packaging. According to the risk of external radiation, inhalation and criticality the radioactive materials are attached to the different packaging types.

c) An approval of the design is only required for Type B and Fissile Class packages. In general this approval is unilateral, that means it is given only by the competent authority of the country of origin of the package. But

+) By arrangements with the competent authority this value may be raised in special cases to 5 rem/a

++) This value is not yet in the regulations. It was proposed in July 1979 by an IAEO Advisory Group for the next revision of the regulations.

...

for Type B (M) and Fissile Class III packages a multilateral approval is required. That means that also each country through or into which the consingment is transported has to approve the package.
An approval for the shipment itself is required for special arrangements and for Fissile Class III and certain Type B (M) packages. This approval is of course multilateral.

d) The activity limits required for the packages and shipments are listed in table 1 for the different packaging types. It has to be noted, that an upper activity limit does not exist for each shipment.

e) The test procedures prescribed for the different packagings are not explained here because lack of time. The same is valid for other important safety principles-as for instance quality control and maintenance of packagings, labelling and marking of packages, general accident provisions - that are not needed in this paper.

2. RISK IN TRANSPORT OF RAM

Without going in detail in the different risk theories it is only stated here, that the risk is defined as the product of the consequences of an unwanted event with the probability that this event occurs. In transport of RAM the consequences of an unwanted event may be

1. somatic effects of radiation to the person at risk
   1.1 early death (III)
   1.2 late death (I, II, III)
   1.3 diseases (I, II, III)

2. somatic effects of radiation to the fetus or embryo of a pregnant women (I, II, III)

3. genetic effects of radiation (I, II, III)

4. effects on property
   4.1 contamination of cargo, vehicle, ground and other property (II, III)
   4.2 losses of premises, salaries and other business costs (II, III).

On the other hand, the radiological risks involved in transportation of RAM may be grouped as follows

I risks associated with normal conditions of transport,

II risks resulting from an accident for which the packaging is designed (design basis accident),

III risks resulting from an accident for which the packaging is not designed (extreme accident).

The groups which may contribute to the different consequences are given in brackets in the list of unwanted events. In this paper only the unwanted events 1.1 and 1.2 (early and late death) are considered. It is further assumed, that both events are accepted in the same manner not only by the individuals at risk but also by society. This simplification is only allowed in a short overview paper like this. In reality there are differences in the acceptability of both events. While there may be a direct relationship between a certain accident and early death, this connection is normaly lost in the case of a late death.

According to ICRP 26 [3] the probability of inducing a fatal cancer or leukaemia is $1{,}25 \cdot 10^{-4}$ per rem. Taking into account this value, the risks of late death derived from the yearly doses (500 mrem/a and 25 mrem/a) given for normal transport in chapter 1 are $6 \cdot 10^{-5}$/a for transport workers and $3 \cdot 10^{-6}$/a for members of the general public. We must remember, that these doses are upper limits used for planning purposes. Only some transport workers and very few members of the general public in reality get these doses over a longer period of their lives. According to a risk assessment made for the USA [4] it may be assumed that the mean radiological risk for

members of the general public is about 1 % or 1°/oo of this value.

The upper limit for the design basis accident is 3 rem. This corresponds to a risk of late death of $4 \cdot 10^{-4}$ per accident. To derive a yearly risk from this figure accident statistics are needed. Fortunately until now very few accidents with RAM did occur in the Federal Republic of Germany. Therefore only a very rough estimate is possible. A figure of 10 accidents per year may be derived from the available material as an upper limit for this country. For a conservative assessment it is further supposed, that in every accident not only one transport worker but also one member of the general public really receives 3 rem. Assuming that $2 \cdot 10^{4}$ transport workers of RAM and $6 \cdot 10^{7}$ other people are involved in the Federal Republic of Germany gives an upper risk limit of $2 \cdot 10^{-7}$/a for transport workers and $6 \cdot 10^{-11}$/a for members of the general public.

In an extreme accident not only the risk of late death but also the risk of early death must be considered. A comprehensive risk assessment including also these accidents is not yet available for the transport of RAM in the Federal Republic of Germany. The corresponding risk assessments made in USA [4, 5, 6] show, that for certain transport modes accidents with very high consequences are possible, but the occurence probability of these extreme accidents is so low, that they do not contribute significantly to the whole accidental radiological risk.

3. RISK ACCEPTANCE CRITERIA

3.1 General Remarks

The world wide nuclear energy debate of the last years has focused the interest in general risk acceptance criteria. It is not possible for me to refer here in detail to the different risk acceptance models developed e.g. by Starr and Rowe [7,8,9]. But it has to be mentioned, that all these ideas are far from being accepted internationaly on a broad basis.

Concerning controlled radiation exposures of workers and members of the general public the situation is much better. Here we have the system of dose limitation given in ICRP 26 that is - at least in principle - accepted world wide. But it has to be taken into account, that the ICRP system of justification and optimisation is derived only for controlled radiation sources. With regard to transport of RAM this means, that the ICRP system is certainly applicable in the field of normal transport, perhaps applicable so far the design basis accident is involved, but certainly not applicable for extreme accidents.

3.2 Acceptance Criteria for the Individual at Risk

The radiation induced risk of transport workers can be compared with occupational risks of different industries. Table 2 gives the fatal industrial accidents of the different german trade cooperative associations for which first indemnities were paid in 1978. Values are derived from [10], fatal accidents on the way to work are not included in the figures. Since about 20 million workers are associated in these cooperatives, the annual risk values given in the last column of table 2 are representative for the occupational fatal risk of the different industries in the Federal Republic of Germany. For transport workers the occupational mortality risk is $4{,}2 \cdot 10^{-4}$/a. Comparing this value with the maximum radiological risk of $6 \cdot 10^{-5}$/a given in section 2 results, that a yearly dose of 500 mrem may be acceptable for transport workers.

To derive acceptance criteria for the individual members of the general public is more difficult. According to ICRP 26 the risks of radiation exposures should be compared in this case with other risks, that an individual can modify only to a small degree and which are normaly accepted in everyday life. In table 3 the numbers of deaths for 1978 in the Federal Republic of Germany are put together as given in the official data of the "Statistische Bundesamt" [11] for the different causes that are listed in the first column. The corresponding mean death-probabilities for the inhabitants of this country

are given in the last column of table 3. It has to be noted, that these numbers are mean values. The individual mortality risk may vary to a great extent from these figures. As an example in figure 1 the age dependence of the mortality in the Federal Republic of Germany is shown for all natural death causes (mean value $1{,}1 \cdot 10^{-2}$ in table 3) and for cancer and leukaemia (mean value $2{,}5 \cdot 10^{-3}$ in table 3). Figure 1 is derived from a not yet published paper [12]. It is valid for men and the time period 1968-1977. If the yearly numbers of deaths are small, as e.g. for the death causes electric current and lightning in table 3, longer time periods than one year must be considered. The yearly fatal accidents due to electric current have decreased in Germany from about 320 in 1955 to about 240 in 1975 [13]. It is interesting to note, that the portion of deaths in private houses remained constant at about 80 per year in this time period. The value of 10 dead persons per year due to lightning is representative for the time between 1967 and 1979 [14]. During the time period from 1952 to 1966 about 30 persons were killed yearly by lightning. The maximum radiological risk derived in chapter 2 for members of the general public was $3 \cdot 10^{-6}$/a. Comparing this value with the figures given in table 3 results, that the risk involved with a yearly dose of 25 mrem seems to be acceptable for individual members of the general public. The same result is obtained if the 25 mrem are compared with the regional variations of the natural external radiation exposure in the Federal Republic of Germany. The corresponding annual dose equivalent values vary between 31 and 360 mrem.

As pointed out in section 2 of this paper the accidental radiological risks for members of the general public and for transport workers are some orders of magnitude lower than the risk values accepted for normal transport as discussed in the previous two paragraphs. Neglecting the difference in the acceptability of late and early death it may be stated therefore, that for individual members of the public all transport risks are acceptable.

3.3 Acceptance Criteria for the Society

The principal question to clarify here at first is:
Are additional acceptance criteria for the society needed, or is'nt it sufficient if the acceptable risk for the society is regarded as the sum of all individual risks accepted by it's members? The answer to this question has two aspects.

First aspect

Some dangerous activities may be so attractive to the members of the society, that the individual is prepared to accept a rather high risk for doing it. If enough members of the society are engaged in such a dangerous activity, the individual risks may sum up to such an amount, that the society - that means i.g. the government - does not accept it and will take some measures to reduce it. An example for such an activity is car driving. Despite all reglementations and efforts to reduce this figure we in 1978 had 14662 dead persons in the Federal Republic of Germany due to motor vehicle accidents. For the transport of RAM this first aspect is of no importance.

Second aspect

There may be activities that are normaly safe, but have the potential capability to cause an accident with very high consequences. If the occurence probability of such an accident is low enough, the risk for persons that may be involved in it is very low and therefore acceptable for the individual. But the question remains: Are the high consequences - e.g. 1000 or more dead persons - also acceptable for the society? The answer must be given by society itself. I can try here only to outline the consequences that the different answers may have for the transport of RAM.

When looking at low probability events with high consequences it is a general praxis to calculate risk spectra in the form of complementary cumulative probability curves f (x). If the unwanted event x is dead of persons, these curves start at x=1; they give the probability that an event with x or more

fatalities will occur per year. As an example the figure 2 gives the risk spectra for shipments of plutonium in the form of oxide powder by air and truck. Curves a and b are derived from [6] and are valid under certain assumptions for the entire USA at shipment levels expected (in 1977!) for the early 1980s. The risk spectrum for deaths by meteorites is given for comparison (curve c [6]).

If the answer to the question mentioned above is "yes", the collective societal risk K respresented by a certain curve f (x) is given by+)

$$(1)\quad K = \int_{1}^{N} f(x)\,dx + f(1) - N\,f(N)$$

N is the number of persons in the collective for which the risk assessment is made. The mean individual risk I_m for a single person of the collective is

$$(2)\quad I_m = \frac{K}{N}$$

Application of these formulae to the risk spectra a, b and c given in figure 2 leads to $K = 2 \cdot 10^{-2}$ fatalities per year for meteorites and plutonium shipments by air and $K = 3 \cdot 10^{-5}/a$ for plutonium shipments by truck. Assuming an US-population of $2{,}3 \cdot 10^{8}$ gives the corresponding mean individual risks of $1 \cdot 10^{-10}/a$ and $1 \cdot 10^{-13}/a$. It has to be noted, that only a part of the US-population is really at risk to be killed by plutonium shipment accidents. Therefore the risk for members of this critical group is probably one or two orders of magnitude higher then the I_m-values given before.

Whereas the risk spectra themselves show clearly whether events with very high consequences are possible, this information is completely lost, if only the corresponding K-values are given. This is demonstrated in figure 2 where the risk

+) Equation (1) is only approximative since in principal a summation instead of an integration process is needed to derive K, if x represents dead persons. The approximation is so much the better the greater the values of x are.

spectra a, c and d despite their completely different shapes have all the same K-value of $2 \cdot 10^{-2}/a$.

If the answer to the question mentioned above is "no", the problem will become more complicated. In this case the simple mathematical definition of the risk given in chapter 2 is not applicable to the extreme accident. As far as I can see there are two possible solutions for this problem.

a) Introduction of a weight - factor for the risks from accidents with high consequences. This procedure would be applicable not only for the transport of RAM but also for nuclear power plants.

b) Avoiding transports which have the potential capability to kill many persons in one accident. This could be done e.g. by setting an upper activity limit for certain shipments, or avoiding densely populated areas when transporting RAM with very high activities. It has to be noted that the risk is then shifted to accidents with lower consequences but perhaps higher frequencies. The collective societal risk K as defined before may be augmented by such procedures, especially of not only the radiological but also the conventional risks of the transports are taken into consideration.

I hope to have made it clear, that acceptance criteria are difficult to find for extreme accidents. Nevertheless, for the next revision of the IAEO transport regulations a way must be found, how to deal with these accidents. I personally believe it would be a good solution to require in the future regulations a multilateral approval for shipments with very high activities. It would then be in the responsibility of the national competent authority to allow a certain transport without any additional requirements or to reduce the risk of an accident with very high consequences in an appropriate way.

References

[1] IAEA, Regulations for the Safe Transport of Radioactive Materials, Safety Series No 6, Vienna (1979)

[2] IAEA, Advisory Material for the Application of the IAEA Transport Regulations, Safety Series No 37, Vienna (1980)

[3] ICRP Publication 26, Recommendations of the International Commission on Radiological Protection, Oxford Pergamon Press (1977)

[4] NUREG-0170 Final Environmental Statement on the Transport of RAM by Air and other Modes, USNRC Washington (1977)

[5] A.R. DuCharme et.al SAND-77-1927, Transport of Radionuclides in Urban Environs: Working Draft Assessment, Sandia Laboratories, Albuquerque (1978)

[6] T.I. McSweeney et J.F. Johnson, BNWL-2030 UC-71, An Assessment of the Risk of Transporting Plutonium Dioxide by cargo Aircraft, Battelle Pacific Nothwest Laboratories, Richland (1977)

[7] C. Starr, Social Benefit vs. Technological Risk, Science Vol. 165 page 1232 (1968)

[8] C. Starr etal, Annual Review of Energy, Palo Alto, Calif (1975)

[9] W.D. Rowe, An Anatomy of Risk, Johns Wiley and Sons, New Yourk (1977)

[10] Übersicht über die Geschäfts- und Rechnungsergebnisse der gewerblichen Berufsgenossenschaften im Jahr 1978, Bonn (1979)

[11] Sterbefälle nach Todesursachen (Einzelnachweis) 1978, Statistisches Bundesamt, Wiesbaden (1980)

[12] Private communication from J. Mehl, Bonn (1980)

[13] D. Kieback, Die zeitliche Entwicklung tödlicher Stromunfälle in der Bundesrepublik Deutschland, Zentralblatt für Arbeitsmedizin, Arbeitsschutz und Prophylaxe, Band 29 S. 197 (1979)

[14] Private communication from "Arbeitsgemeinschaft für Blitzschutz und Blitzableiterbau", Kiel (1980)

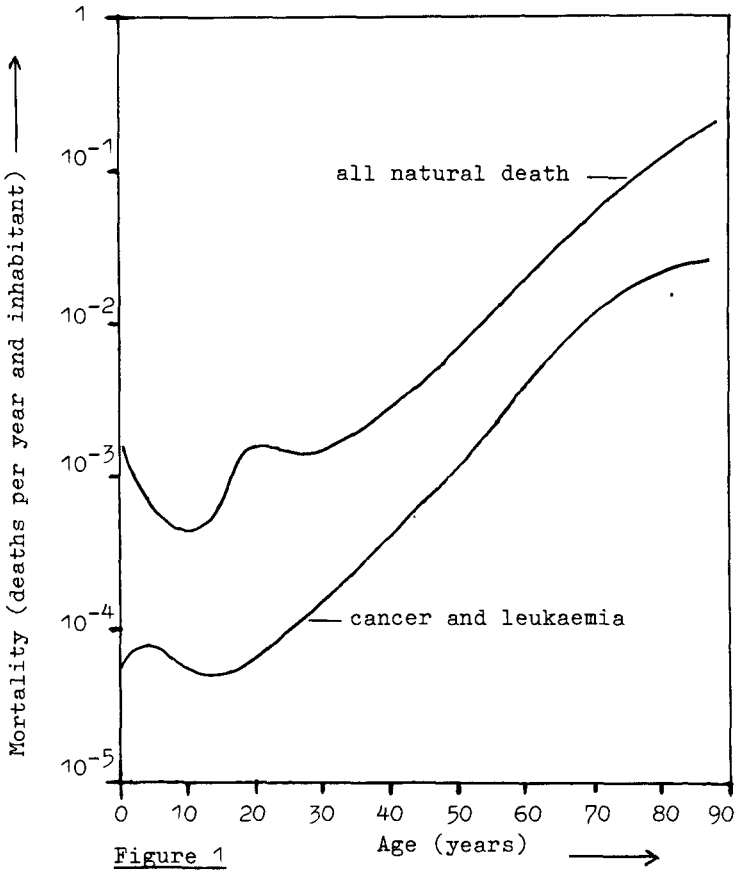

Figure 1
Age dependence of mortality

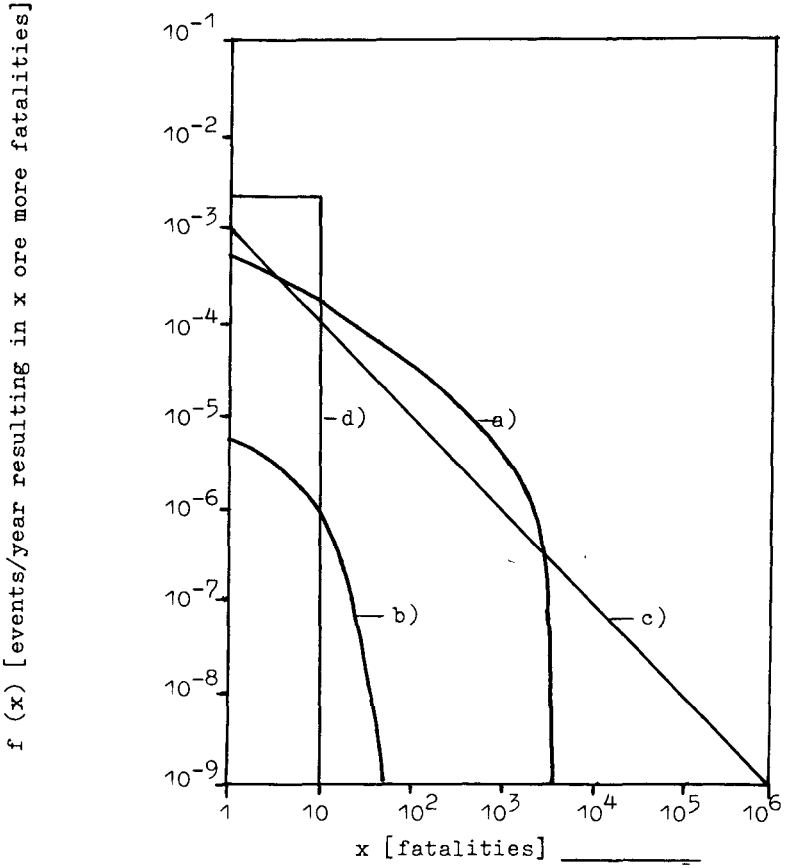

Figure 2

Risk Spectrum for Plutonium Shipments in the Early 1980's for the Entire U.S. (Single Container Analysis)

a) as oxide powder by air

b) as oxide powder by truck

For comparison the risk spectra for meteorites c) and a hypothetical spektrum d)with the same K-value are given

Table 1 Packaging types and activity limits (Criticality safety principles are not considered)

Type of packaging	required in schedule[+)]	Activity limits required for	
		packages	shipments
no packaging	5 (LSA I under full load)	not applicable	schedules 5 and 6[++)]; [+++)] 100 x A_2 for liquids and gases, for solids only for inland waterway crafts other schedules: none
normal	2 to 6	only for schedule 3 and 4	
strong industrial	7	none	none
Type A	8	A_1 or A_2	none
Type B	9 and 10	as given in the package approval	none

+) Schedule 2: Articles manufactured from natural or depleted uranium or from natural thorium
Schedule 3: Small quantities of radioactive materials
Schedule 4: Instruments and articles with radioactive materials
Schedule 5: Low specific activity materials (LSAI)
Schedule 6: Low specific activity materials (LSA II)
Schedule 7: Low lewel solid radioactive materials (LLS)
Schedule 8: Radioactive materials in Type A packages
Schedule 9: Radioactive materials in Type B (U) packages
Schedule 10: Radioactive materials in Type B (M) packages

++) additional requirements of ADR and RID are considered

+++) for tritiated water lower limits are valid

Table 2: Fatal accupational accidents in the various trade cooperative associations

Groups of trade cooperative associations	Deaths for which first indemnities were paid in 1978	Number of full-time workers in 1000	Deaths per full-time worker and year
Mining industry	223	206	$1,1 \cdot 10^{-3}$
Transport	295	703	$4,2 \cdot 10^{-4}$
Bulding trade	628	1882	$3,3 \cdot 10^{-4}$
Stones and earths	103	387	$2,7 \cdot 10^{-4}$
Iron and metal	579	3333	$1,7 \cdot 10^{-4}$
Paper and printing	77	524	$1,5 \cdot 10^{-4}$
Chemicals industry	132	855	$1,5 \cdot 10^{-4}$
Foodstuffs and Luxury comestibles	222	1494	$1,5 \cdot 10^{-4}$
Gas and water	12	84	$1,4 \cdot 10^{-4}$
Wood and carving materal	70	519	$1,3 \cdot 10^{-4}$
Commerce, finance and insurance, services	596	5151	$1,2 \cdot 10^{-4}$
Electrical engineering, precision mechanics and optics	203	1965	$1,0 \cdot 10^{-4}$
Textiles and leather	64	943	$6,8 \cdot 10^{-5}$
Health services	70	1355	$5,2 \cdot 10^{-5}$

Table 3: Deaths and their causes in the FRG

Causes	Numbers of deaths in 1978	Deaths per inhabitant
a) natural		
Total number of deaths	678000	$1,1 \cdot 10^{-2}$
Circulatory diseases	345000	$5,6 \cdot 10^{-3}$
Cancer and leukaemia	155000	$2,5 \cdot 10^{-3}$
b) other than natural		
Total number of deaths	45000	$7,3 \cdot 10^{-4}$
Total number of motor vehicle accidents	14662	$2,4 \cdot 10^{-4}$
- of these, as driver or passenger in private car	7082	$1,2 \cdot 10^{-4}$
- of these, as pedestrian	3788	$6,2 \cdot 10^{-5}$
Total number of fatal falls	9685	$1,6 \cdot 10^{-4}$
Total number of deaths caused by fire	530	$8,6 \cdot 10^{-6}$
- of these, fire in private dwellings	301	$4,9 \cdot 10^{-6}$
Electric current	171	$2,8 \cdot 10^{-6}$
Stroke of lightning	10	$1,6 \cdot 10^{-7}$

DISCUSSION

Question by H.-W. Levi, FR Germany: Does the extreme accident exist or can you always think of a worse accident?

Answer: According to the definition given in my paper, every accident resulting in a dose of more than 3 rem is an extreme accident. This means that - however strong the package may be - an extreme accident that also destroys this package is conceivable.

Comment by B. G. Pettersson, Sweden: I would like to refer to the preceding question. Recognising that one can always design an accident scenario more severe than the one previously established, would it then conceptually be possible to establish acceptance criteria for the low probability/high consequence accidents. The problem, of course, is that the consequence term appears to be impossible, or, at least, very difficult, to quantify, because any accident scenario could be argued to be "not severe enough."

Answer: I agree with this statement. One purpose of my paper was to draw attention to the fact that society must give an answer to the question, whether the pure mathematical definition of the risk is also applicable to the low probability/high consequence accidents. If this is the case, there is no problem with the acceptability of extreme accidents because they contribute only a very small amount to the overall risk of transportation. If this is not the case, then the proposal made in the last paragraph of my paper might be a compromise for future work on the IAEA Transport Regulations.

RECENT RADIOLOGICAL PROTECTION CONSIDERATIONS IN THE TRANSPORT OF RADIOACTIVE MATERIALS

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INTRODUCTION

All substances that are potentially harmful require appropriate procedures to be adopted so that they can be safely transported. Radioactive materials are subject to national and international transport regulations which come under periodic review.

The revision of existing transport requirements for radioactive materials should take account of recent advances in radiological protection. In radiological protection, there is now an emphasis on optimisation using techniques such as cost benefit analysis rather than on dose limits; such limits are considered more as boundary conditions (1). The idea of optimisation is not new, but its application in the transport context has been limited in the past.

The International Commission on Radiological Protection (ICRP) recommends a system of dose limitation the main purposes of which are to ensure that no source of exposure is unjustified in relation to its benefits or those of any available alternative; that any necessary exposures are kept as low as reasonably achievable; that the dose equivalents received do not exceed certain specified limits and that allowance is made for future development. The ICRP recommendations apply to planned or controllable situations which are expected to occur: for situations resulting from disruptive phenomena, acceptability criteria need to be defined.

In relation to transport, the source of exposure is under control in normal situations including minor mishaps. Following a severe incident, the decision to initiate remedial action will require that account be taken of the particular circumstances that prevail. In general, it will be appropriate to institute countermeasures only when their social cost and risk will be less than those resulting from further exposure.

APPLICATION OF OPTIMISATION

The basis in radiological protection of the present transport regulations is the control of external radiation and the limitation of possible intakes of radioactive materials (2). Limitations on the quantity of radioactive material in packages are given where appropriate as are limitations on external dose rates. There is no specific requirement to keep all exposures as low as reasonably achievable, with economic and social factors being taken into account. More emphasis should now be placed on this requirement.

Regulations based on dose limits make a considerable contribution to radiological protection and carry with them the advantages of standardisation. It would be inappropriate to attempt to carry out optimisation studies on a case by

case basis. Optimisation should, however, be employed in establishing some aspects of standards and regulations, and it should also be employed in their subsequent application.

Optimisation may be applied, for instance, with regard to a limit on the number of Type A packages that should be carried in one vehicle. At present, the only limit for such packages is based on the transport index, but this provides no limitation for packages containing radionuclides emitting non-penetrating radiations. In the event of a fire, the entire contents of Type A packages may be released. The resulting health effects can be calculated and so can the costs of reducing loads by the use of additional vehicles and drivers; account will need to be taken of the probabilities involved. The increase in occupational exposure and possibly public exposure will also be required as inputs to the optimisation study. This technique requires the harm from radiation exposure to be converted into monetary terms so that the costs of protection can be directly compared with the costs of the radiation exposure.

In the UK the Board has considered monetary values for the man-sievert in a Consultative Document (3) on the application of cost-benefit analysis to the radiological protection of the public. A "risk averse" model of human attitudes has been adopted in which the cost is not a linear function of the collective dose equivalent but depends upon the levels of individual dose equivalent. Following a consultative procedure, the Board expects to advise on values to be assigned to the man-sievert within the UK. International agreement on such values would be difficult to achieve.

Transportation usually involves a degree of handling of packages and hence a certain radiation exposure. Optimisation here can imply measures to reduce the amount of handling by using facilities such as lifting equipment and overpacks. Education and training of the workers involved can do much to reduce occupational exposure and help to keep the exposure of the public at a very low level. It is advisable that appropriate training, education and information be made available to transport workers, and that data and relevant radiological assessments on transportation be made widely available whenever possible.

It is often necessary to look at the best use of resources and to see where most benefit can be achieved. Changes to package specifications may be far less cost-effective than changes in operational procedures, an important consideration when resources are limited. It may therefore be profitable to compare the advantages of implementing changes in different areas of international regulations.

EMERGENCIES

The effects of transport emergencies can be minimised by appropriate package design and construction or by limitation on content. Measures can be taken to reduce the probability of an incident, and emergency plans can be drawn up to provide for remedial actions.

The potential consequences of accidents are taken into account in designing packages so that certain small quantities of radioactive materials may be transported and be able to withstand the rigours of normal transportation. Larger quantities of radioactive materials must be transported in containers designed to retain their integrity to a substantial degree in severe accidents.

Emergency control procedures are required for the protection of the population from significant unplanned releases of radioactive materials. Pre-planning is necessary, and various procedures have been established in the UK for

different types of materials. Specific emergency schemes for major movements of radioactive materials are established by the appropriate authorities and, in addition, the Board arranges a national scheme for minor incidents involving radioactivity (4).

Experience with the transport of radioactive materials extends over several decades and the record is good. However, it is necessary to consider potentially serious events of low probability and also the influence of changes in the pattern of transport, such as increased traffic density and speed. Radiological consequences following a release of radioactive material depend upon many circumstances including the nature of the release, the population distribution, the prevailing meteorology and any remedial actions taken. Realistic radiological protection assessments are required to provide information on potential radiation exposure from severe incidents during transportation. The cost of this potential detriment, account having been taken of the probabilities involved, can then be compared with the cost of measures to reduce the probability of an incident or lessen the potential effects.

Following an accident countermeasures may be needed. Emergency plans are required including dose levels at which various countermeasures have to be considered and dose levels below which no measures are required. On the basis of risk, Clarke (5) has proposed a system of reference doses for use in pre-planning for accidental releases of activity to the environment, and for short-term decision-making in the event of releases. The lower dose bound is set such that if doses are unlikely to exceed it, no action is necessary. If doses are likely to exceed it, however, the introduction of countermeasures should continually be evaluated on the basis of net risk avoidance. The upper dose bound is set at a level where countermeasures are certainly expected to be implemented. The recommended doses in this system are 5 mSv (500 mrem) for the lower bound and 500 mSv (50 rem) for the upper bound. The system is shown below in tabular form:

0 - 5 mSv (0 - 500 mrem)	No action.
5 - 500 mSv (0.5 - 50 rem)	Continual consideration of the introduction of countermeasures.
500 mSv ⟶ (50 rem ⟶)	Certain implementation of appropriate countermeasures.

This general system, if it is finally recommended for use, will be the background against which specific local plans can be made.

POPULATION EXPOSURE

In order to carry out optimisation studies, it is necessary to obtain comprehensive data. In the UK, for instance, work is in hand to provide detailed information on radiation exposure from the transport of radioactive materials.

An example of this work is the assessment (6) of the radiation dose to the general public from the movement by rail in the UK of used fuel elements from nuclear power stations of the Magnox type.

Rail routes from the power stations to Windscale have been chosen, for the purpose of calculation, from the British Rail system map: these routes are shown in Figure 1. Collective dose equivalents to the public along chosen routes have been calculated for the year 1977: values for the normal transportation of irradiated fuel elements are shown in Figure 2. The doses are very small, but

the contributions from the trains when on the move and when stationary have been calculated for the ten power stations. Most of the dose arises while trains are stationary at various marshalling yards. The highest value is about 10 man mSv y^{-1} (1 man rem y^{-1}) from a single power station. The total collective dose equivalent is less than 50 man mSv y^{-1} (5 man rem y^{-1}): the highest dose to an individual member of the public is estimated to be less than 0.1 mSv y^{-1} (10 mrem y^{-1}). For reference, the dose limit for an individual member of the public is 5 mSv y^{-1} (500 mrem y^{-1}).

In an accident one of the factors influencing the radiological consequences is the population density nearby. Information has been obtained on the population density within 1 km^2 of the track along rail routes between the power stations and the reprocessing plant. Figure 3 shows the population density along each of the chosen routes. Over most of their lengths, the routes are not heavily populated but marshalling yards may be in areas of high population density. A comparison can be made between routes in terms of the product of route length and population density within 1 km^2 of the track. Figure 4 shows that, for each of the ten Magnox power stations, the product of route length and population density is virtually constant for most of the stations.

Reductions in collective dose could be achieved if times in marshalling yards were reduced, and areas of high population density could be considered in relation to routeing, but the effects that this might have on other rail traffic would also need to be taken into account, together with the increased risk of collision and mishap from non-standard routeing. In the normal situation the collective doses are small and further routeing considerations would not be necessary.

CONCLUSIONS

Optimisation may be usefully applied in the setting of transport standards and in their practical application. Radiological assessments can provide information for such optimisation studies and can provide a firm basis for any alterations which may need to be made to existing requirements. There is a definite requirement for further detailed assessments and for the collection and examination of data on radiation exposure from transportation.

Emergency plans are required appropriate to specific movements of radioactive materials. Dose levels should be stated at which countermeasures have to be considered. Routeing considerations must take account of all potential risks both radiological and non-radiological.

References

1. International Commission on Radiological Protection, Recommendations of the ICRP. Oxford, Pergamon Press, ICRP Publication 26. Ann. ICRP, 1, no. 3 (1977).

2. International Atomic Energy Agency Bulletin, 21, no. 6. IAEA, Vienna, (December 1979).

3. The Application of Cost Benefit Analysis to the Radiological Protection of the Public. A Consultative Document. Harwell, National Radiological Protection Board (March 1980).

4. Bland, W F, National Arrangements for Incidents Involving Radioactivity (NAIR). Radiological Protection Bulletin No. 35, (July 1980).

5. Clarke, R H, Radiological Protection Criteria for Controlling Doses to the Public in the Event of Unplanned Releases of Radioactivity. IIASA Workshop, Laxenbourg, Vienna, (January 1980).

6. Beach, S A, Radiation Exposure of the Public from the Movement of Irradiated Fuel Elements by Rail from Magnox Power Stations. Harwell, National Radiological Protection Board, NRPB-R114 (to be published).

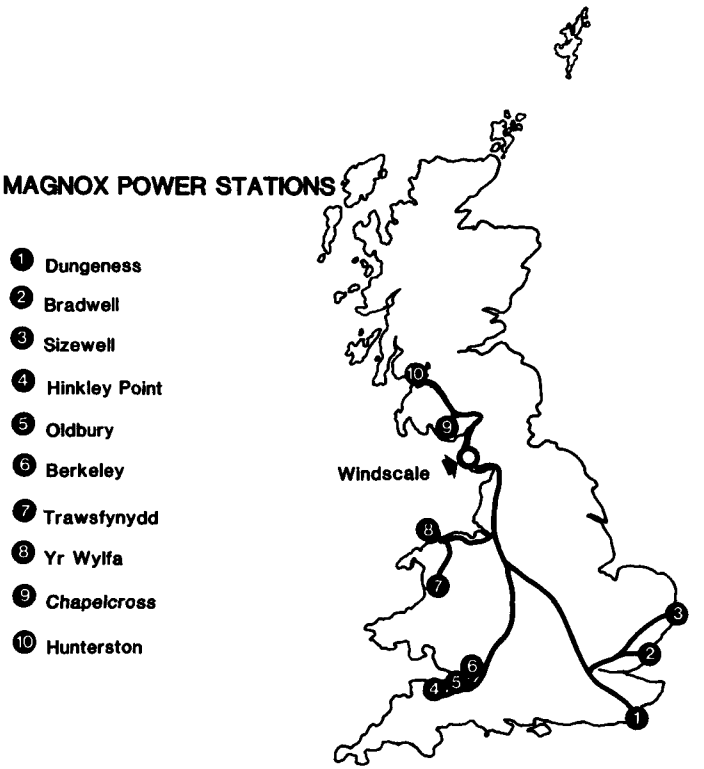

Figure 1: Chosen rail routes from Magnox Stations in Great Britain to Windscale.

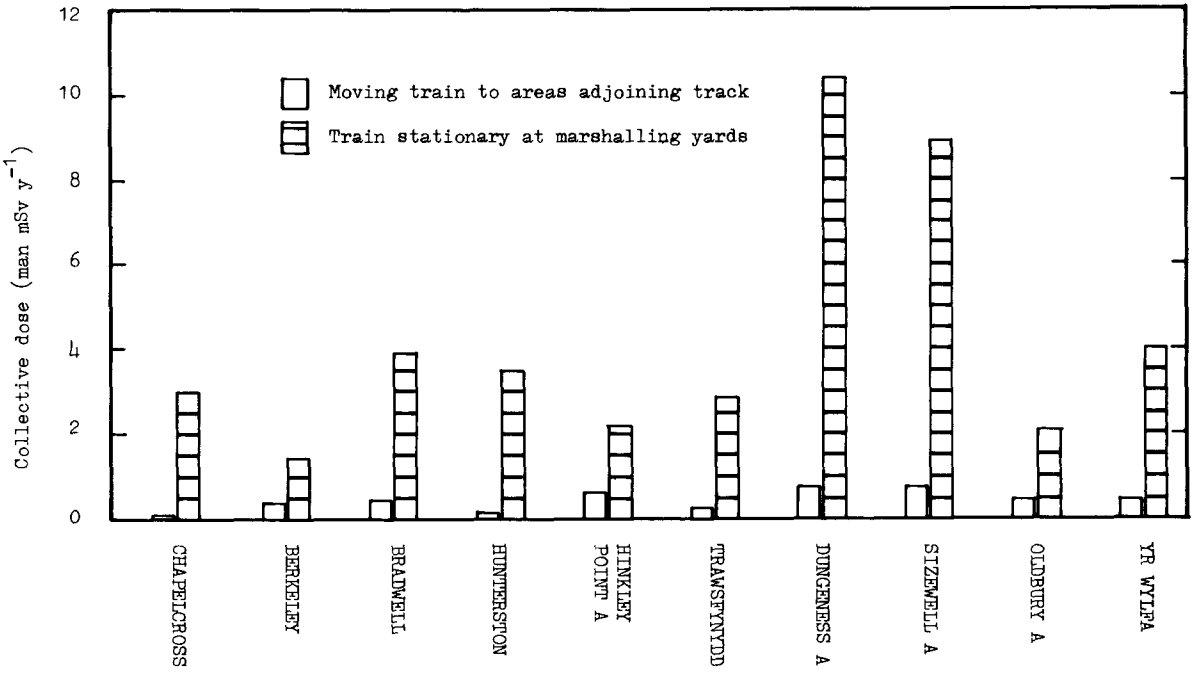

Figure 2: Collective dose to the public from the movement by rail of used fuel elements from Magnox power stations for the year 1977 (shipped in the years 1978-79)

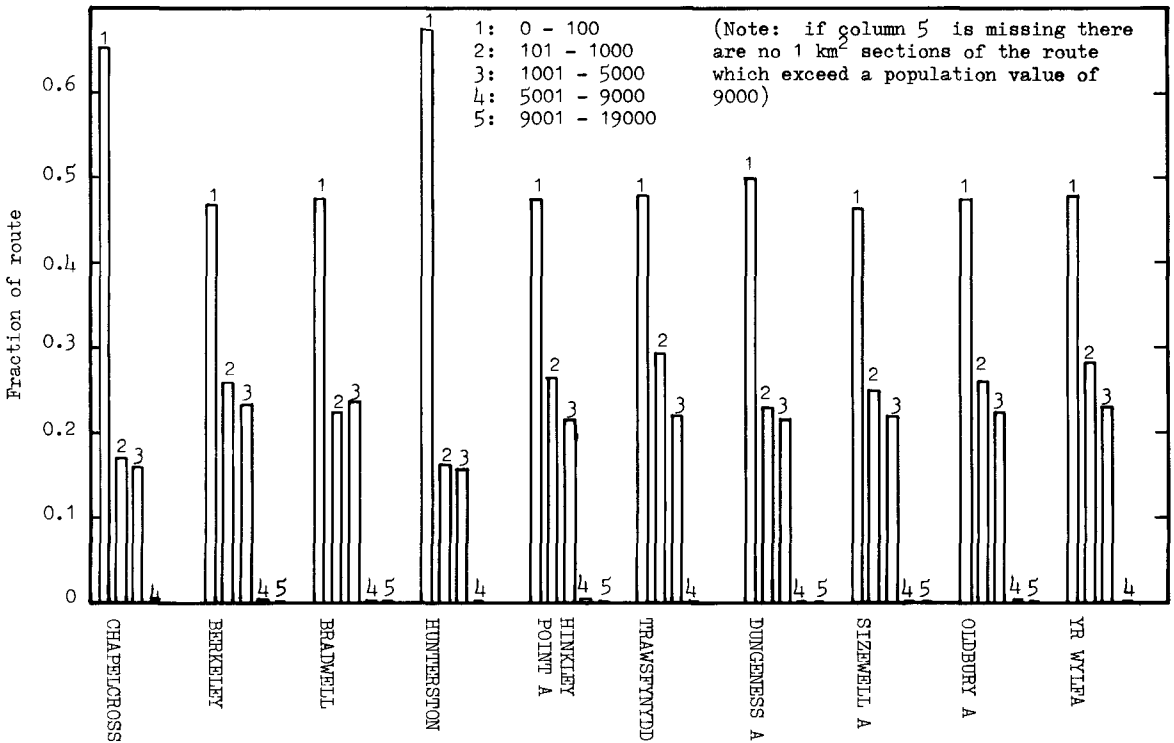

Figure 3: Fraction of route within 1 km^2 containing a given population density

Figure 4: The product of the route length (km) and the expected population living within a 1 km grid square traversed by a consignment of irradiated Magnox fuel

Question by G. Cohendy, France: In England is transport carried out in the whole train? If so, what about the collective doses in the marshalling yards?

Answer: The collective dose equivalents to the public from the rail transport of irradiated fuel are very small. However, as was shown in the paper, most of the dose arises while trains are stationary at various marshalling yards. In the normal situation, the collective doses are so small that further routing considerations would not be necessary.

Question by L. B. Shappert, USA: Is the high dose rate received in the marshalling yards due primarily to dose given to the population in general or to British railway workers or to a combination of the two? Which population group provides the greatest contribution to the dose shown in your diagram?

Answer: The information presented in this paper is concerned with the radiation exposure of the general public resulting from the movement of irradiated fuel elements by rail from Magnox power stations within the United Kingdom.

Question by A. Hoegl, FR Germany: Do you have any idea of the order of magnitude that will be assigned to the man-sievert by the authorities in the United Kingdom? For example, will one man-sievert be equal to pounds, 20 pounds etc?

Answer: The monetary value for the man-sievert is not a linear function of the collective dose equivalent but depends upon the levels of individual dose equivalents. It is important to reflect individual risk within the collective dose framework. Following a consultative procedure, the Board, in the very near future, will advise on values to be assigned to the man-sievert within the United Kingdom. This advice will be published as a Board ASP and will be more widely available.

TRANSPORTATION OF NUCLEAR MATERIALS: THE NUCLEAR FOCUS OF THE 80's

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I am pleased and honored to be here today to offer some thoughts to keynote this Sixth International Symposium on Packaging and Transportation of Radioactive Materials. It is appropriate that this symposium, the latest in the PATRAM series, is being held here in Berlin, one of the world's great cities with a long tradition of technical excellence.

Many of you may recall in the keynote address at the previous symposium, a little more than two years ago in Las Vegas, Nevada, Dr. George Cunningham of the US Department of Energy (DOE) characterized transportation as the "Vital Link". That transportation is a vital link is more than a keynote idea -- it is an immutable fact. It is also a fact that the transport of radioactive material has been carried out since the inception of the nuclear age (over 30 years) with an unparalleld safety record. In all of this time with millions of packages shipped, not one individual, so far as I know, has been injured in the transport of radioactive materials due to its radioactive nature. Despite these achievements there is no basis for complacency for it is paradoxically true that a large segment of the public -- at least in the United States -- view such transportation with great concern. As we move into the 80's with growing energy requirements and the nuclear option being counted upon in much of the industrial world to help meet our energy needs, increased transportation of nuclear materials will undoubtedly result. In the 1980's, we wll also be seeking to answer what I believe to be the single lmost important unanswered question involving nuclear power -- the safe disposal of radioactive waste. As we continue our efforts to solve this problem, we must not overlook the fact that safe efficient transportation systems will be an integral component of the management of radioactive waste.

The 1980's will also see advances in technology and medicine, many involving the use of radioisotopes for testing and diagnostic purposes, resulting in increased benefits for the health and safety of mankind. The greater benefits derived from these uses of radioisotopes will depend in part on the safe packaging and transportation of these materials.

Fortunately, through conscientious efforts, sound engineering, and effective regulation, we have an unequaled legacy of safe transportation to guide us through the 1980's. As enviable as this record is, we must not lose sight of the fact that one serious accident, no matter what the cause, resulting in the release of radioactivity can effectively negate this record in the eyes of the public. We must never cease to strive to improve our performance, to develop safer and more efficient transportation systems and moreover, to perform these tasks in a highly visable manner so that we can allay public concerns.

We now find ourselves at the crossroads, so to speak. We can either claim immunity from the fracas based upon our excellent record of safety, or we can intelligently and honestly interact with both the public and the technologists and pursue our efforts on the basis of technological fact and informed public

opinion. Certainly the past record is a part of that fact, but only a part of it. I believe that this international conference is testimony to the fact that the transportation segment of the nuclear industry is more than ever before committed to addressing the issues head-on. Here at PATRAM 80, for the first time in this series, there will be a session devoted to public perception which will attempt to involve the public in an active fashion.

But, in the same vein that the past record is not of itself sufficient, neither is public participation "the" solution to all the issues surrounding the transportation of radioactive materials. The solutions to the problems besetting the nuclear transport industry involve many disciplines, much of which must initially rest on a foundation of sound technology.

This conference is built around a core of papers on the developing technology of nuclear transportation. It is this continuing development of the base technology that transmits the message that where improvements can be made there is indeed an interest in such progress. There is, quite understandably, substantial body of papers in this conference on systems, their design and development, their manufacturing processes, their operation and the methodologies of quality assurance in each of these activities. I say understandably, because there is both a growing volume as well as a growing diversity of nuclear materials to be transported. Such growth and diversity demand both the expansion of existing systems and the creation of new systems for meeting the emerging needs. The recent increased use of testing has improved the industry's ability to develop improved transport systems through the advancement of analytical methodologies. Further, the test programs carried on worldwide have been effective in demonstrating the response of packagings to severe accidents in a manner much more understandable to the public and the regulators than the complex analyses which such test programs help validate.

It is in this activity that the role of the International Atomic Energy Agency (IAEA) is most important. But, the continued development of appropriate regulations requires broad based input such as that which is scheduled to be offered here at PATRAM 80. Existing regulations must, of necessity, be periodically evaluated. Such evaluation demands the availability of data which in some cases do not exist. This in turn exerts a pressure for developing the needed data. The IAEA is engaged in just such a data collection effort to compile information on the flow of radioactive materials, the mode of transport and the corresponding accident/incident experience. An emerging sector of the regulations addresses the need for providing additional protection for radioactive materials during shipment. Questions of what to protect, how to protect it and even the need to protect, face the world nuclear community. Coupled closely, although not entirely to the protection issue, is the emerging concern about how to deal with transportation emergencies, should they arise. The question of what to do in the case of a transport accident involving the possible release of radioactive contamination is often overwhelmed by questions about who is responsible for what, who should do the planning, who should be trained (and to what level) and how to pay for all this preparation. Again, PATRAM 80 will address parts of this problem.

When all of these efforts are put together, those of us involved in nuclear transportation, and especially those of us involved in managing these programs, must ask "What are we accomplishing?" What measures do you use to determine whether or not the effort is useful? The ultimate scale by which the effort must be weighed is the balance of risk and benefit. How much dos the effort cost and how much, if at all, does it reduce risk? Thus, risk assessment has become an important management tool in the nuclear transport community. While absolute numerical determinations of risk may not always be possible, comparative risk studies can serve to determine program direction. But, the present state

of worldwide risk assessment is quite inconsistent. Until lately, there has been little compatability between the risk studies done in one country with those done in another. In recognition of this discrepancy, the IAEA has initiated a program to develop a worldwide, uniform methodology to address the risks of transporting radioactive materials. Here again, the input data on flow rates, material types, mode of transport and other factors becomes important to the development of accurate risk assessments. Such efforts, to be reported at this conference, will contribute to the proper management of research programs. But, research management is not the only purpose of risk assessment. In the present climate, where political decisions often depend upon perceived fact, risk evaluations are important as input data to the public and to policymakers. The generation of publicly understandable risk assessments will contribute substantially to the formation of intelligent public opinion on the issues of nuclear transportation.

All of this leads to the discussion of the institutional issues impacting the transport of radioactive materials. The simple question of who decides such matters as "how safe is safe enough", or "which political entity has jurisdiction", or "how should the law be structured to provide a balance between public protection and public benefit," is as yet undecided. Just the first step, defining what the institutional problems are, is at best a difficult one. Once again, the papers at this conference reflect this emerging problem area.

In closing, let me say that the significance of this meeting is not so much the range or extent of the worldwide effort being expended on the continued upgrading of nuclear material transport, but rather the fact that you and your colleagues recognize the necessity of addressing both the real as well as the perceived problems of the industry in an objective and open manner.

REFERENCES

1. George W. Cunningham, "The Vital Line", Proceedings of the Fifth International Symposium on Packaging and Transportation of Radioactive Materials, Las Vegas, NV (May 7-12, 1978) pp 3-7.

SESSION I
REGULATIUONS, LICENSING AND STANDARDS

UNILATERAL AND MULTILATERAL APPROVALS

- Monsieur Yves SOUSSELIER C.E.A. Fontenay-aux-Roses

- Monsieur Michel GRENIER C.E.A. Fontenay-aux-Roses

The French writer of 18th Century, MONTESQUIEU, who, amongst other merits and honors, had got the one of being a member of the Royal Academy of Prussia for Science and Litterature, wrote in the first volume of his work : " De l'esprit des lois", about human laws :

" Elles doivent être tellement propres au peuple pour lequel elles sont faites que c'est un très grand hasard si celles d'une nation peuvent convenir à une autre".

If such is the case, when giving herself the task of proposing common laws to all peoples, at least in her field, the IAEA did not get an easy task, in spite of the universal character of the physical laws she founds upon.

But, still more difficult was to act in such a way, that the implementation of their rules in a country be valid in another, even without the agreement, sometimes, of the authorities of the latter. Nevertheless, the growing number of international shipments of radioactive materials had made it necessary, universality of physical laws and human nature made it possible.

The transport field presents indeed this feature that the rules of safety must be accompanied by a certain harmony between the different countries and even by some kind of uniformity.

After this harmony was reached, it has been found desirable, to allow the Competent Authority of an unique country to determine the conditions of safety of an international shipment.

Therefore were introduced in Safety Series n° 6 of the IAEA two types of type B approvals, the unilateral approval B(U) and the multilateral one B(M).

The first had to be signed only by the Competent Authority of the country of origin of the package, the second by the Competent Authorities of all the passed through or called at countries. That was established in order to make easier the setting up of approvals and to erase some difficulties.

We will begin by examining what were these difficulties and then to what extent they were lessened, and at last, whether it is needed to introduce any new development.

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I - The difficulties presented by Approvals

I.1 - The Approval File

The setting up and the justification of the safety report which must be supplied to support the application of an approval are not plain formalities.

Faced with the growing complexity of many transport casks, the files of approval of these casks become thicker and thicker, more and more elaborate. Some of the problems dealed with in these reports such as, for instance, the radiological protection or thermal calculations, etc ... need exchange of informations and explanations which may infer direct contacts, meeting and discussions, even rather numerous. That is true already at the only level of the approval of the design of a package.

I.2 - The actual packages

But that is not all. Safety, which lies theoretically on the design of the package, lies practically on the package which is actually circulating. That is obvious but this obviousness brings about other problems. The first is the compliance of the real packages with the approved design. That is at first under the responsibily of the applicant of the approval, but checking the carrying out of this responsibility is normal and may be necessary in order to verify that minor differences in the manufacture do not question the safety analysis.

I.3 - The evolutions of the packages

The time during which they remain safe is limited.

From this point of view too, the approval of the design only is not sufficient, as far as it does not give any mean of checking and sanctioning the actual state of each packaging, overall when many examplars of the same design are in use. It is nevertheless obvious that only these actual packagings are involved in the safety of transport.

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II - The present situation

The introduction of B(U) approval facilitated greatly the approval of the design, at the international level. It is obviously easier for the applicant to assure the good information of the Competent Authority of his own country. That does not lessen the difficulties rising when the country of origin of the package, which must be the applicant's one, is not the country where the packaging was designed and possibly manufactured. But these difficulties would exist in any case and even would be multiplied by the number of involved countries, in the case of a multilateral approval.

These close links, sufficient at the level of the design approval, make therefore greatly easier the international running of transport casks.

This is, besides, underlined by the difficulties experienced in front of packagings coming from countries which do not, or not yet, implement this unilateral approval concept.

On the contrary, the introduction of this unilateral approval is perhaps not as simplifying, when looking at the safety of actual packages, and not at the theoretical design.

It is, indeed, in the part of the Competent Authority, having approved the design, to verify, before use, the compliance and the quality of manufacture and during use, the general state of the packages which have to comply with this design. We will see, in the Quality Assurance Session, that this task is far from being easy, even at the sole national level.

With greater reason, how might this Competent Authority do this outside the geographical field of her sovereignty ?

On the other hand, on what basis the Competent Authority of a country where the package is actually circulating, could act, without having taken any part in the Approval of its design ?

This question of limits of sovereignty of Competent Authorities, in front of the necessity of following the packages and packagings during their use, is not less present in the case of B(U) packagings, sold to users, in another country that the one where they have been approved.

In other respects, the introduction of unilateral approval could, by backlash, make of the B(M) a somewhat devaluated one, giving not to the involved package the same "universal" freedom of movement as the one which is given by an unilateral approval.

We must therefore consider what can be done in order to further improve this situation.

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III - Some possible solutions

If, as we have seen, the introduction of B(U) and B(M) types has been not only effective but also beneficial, we have still to try to find a solution to the problems that we have just pointed out.

These problems come essentially from the necessity of following carefully the actual packages.

This need, which presents some psychological aspects too, in relation with public opinion, is a consequence of the situation developping nowadays. That is to say the spreading out of packagings and packages in very much more numerous and various hands, coming from the growing number of nuclear reactors, and the following operations of fuel cycle rendering in a manner radioactive transport common place.

More than ever, therefore, the safety of this transport is based upon packaging or rather upon "the packagings".

What can we propose ?

At first, to consider :

1/ the introduction of a compliance certificate to an approved design for any packaging, which will have to be marked and identified.

2/ the introduction of a checking of the maintenance and of the absence of modification for this same packaging, which could be sanctioned by a periodic marking.

That will obviously entail a spreading out of relations between Competent Authorities, which could go by an increase of the role of the IAEA, for instance about the understanding of foreign languages.

This could lead for instance to the issue of a glossary of equivalent terms in certain number of languages.

Moreover, this translation means will have to be accompanied by an harmonization of the procedures, and overall of the criteria of checking.

And it is there, that the part of IAEA should reveal itself essential.

These measures are far from being simple in their implementation which will demand a considerable effort.

Nevertheless, it is perhaps the only way to continue to carry out the international transport of radioactive materials, keeping the very high level of safety it has been given so far, in spite of the present increase, and overall the foreseeable increase of its volume.

Question by S. Williamson, United Kingdom: The United Kingdom accepts the principles of unilateral approval, simply by accepting the total principles of IAEA Regulations into national legislation. Experience has shown that other countries are not so accommodating in this matter and insist on their own complete safety assessment, quite outside the spirit of the IAEA Regulations, to which they are party by being member states. Do you think that the principles of unilateral approval will tend to disappear? If this is possible, do you consider that any measures could be taken so as to encourage the principles of unilateral approvals?

Answer: We certainly consider that it is necessary to take measures so as to encourage the principles of unilateral approval and to find solutions for keeping alive these principles. We are conscious of the difficulty of this task but we think that the latter would be at least as great, in the case of multilateral approval, even as far as real packages are concerned.

Question by B. W. Emmerson, United Kingdom: In the possible solution which you have proposed for facilitating B(U) and B(M) approvals, you included a check list for maintenance. Do you envisage this as a formal maintenance procedure to be carried out at specified intervals, and that this will be in addition to any tests already required under the Transport Regulations? If so, do you consider that routine maintenance procedures will need to be incorporated into future IAEA Transport Regulations?

Answer: We do not think that the details of the maintenance procedures and checkings have to be in the Regulations. We think that the principle of this check list must be in the Regulations, but that the check list itself should be found in the Approval Certificates, so that it can be adapted to each package type.

AN ALTERNATIVE APPROACH TO THE A_1/A_2 SYSTEM FOR DETERMINING PACKAGE CONTENTS LIMITS AND PERMITTED RELEASES OF RADIOACTIVITY FROM TRANSPORT PACKAGES

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1. INTRODUCTION

Currently the IAEA is undertaking a review of the Regulations for the Safe Transport of Radioactive Materials (IAEA, 1979), with a view to producing a further revision in 1983. In this paper an alternative to the A_1/A_2 system which forms the basis of the present Regulations is presented. The aim is to take account of recent recommendations of the ICRP (1977 and 1979), and to develop a system in which the radiological implications of the permitted activity releases are more directly considered. This work was undertaken under the terms of Research Agreement No. 2291/R1/CF, which forms part of an IAEA coordinated programme on the safe transport of radioactive materials.

2. BACKGROUND

In the 1973 Transport Regulations (IAEA, 1973 and 1979) the earlier group classification system was developed into the "A_1/A_2" system, in which each nuclide has a Type A package contents limit, A_1 curies, when in a special form and a limit, A_2 curies, when in a non-special form. The dosimetric bases of the A_1/A_2 system were not always fully elaborated (IAEA, 1978), although an emergency whole-body dose limit of 3 rem was assumed in the derivation of A_1. In addition, it was indicated that an intake of $10^{-6} \times A_2$ curies should correspond to half the maximum permissible annual intake for a radiation worker. In this paper an alternative "Q_o" system is developed, in which the Type A package limit, Q_o curies, is the least of several values Q_1, Q_2, Q_3, etc., determined from consideration of particular exposure pathways following damage to the package in an accident (or when subject to specified tests). The dosimetric criteria used are similar to those stated, or implied, in the 1973 Regulations since these have been found to provide an acceptable basis for the safe transport of a wide range of radioactive materials over a period of several years, while in situations involving the intake of radioactive material use is made of the new data and concepts recently recommended by the ICRP (1977 and 1979).

3. DETERMINATION OF THE TYPE A PACKAGE CONTENTS LIMIT Q_o

The Q_o value for a radionuclide is the least of several values of Q_1, Q_2, Q_3, etc., determined from consideration of the radiation dose via various pathways which might be incurred by a person exposed in the vicinity of a package following an accident. The distinction between special and non-special form material is

retained. Special form material is assumed to remain within its source container at all times and is thus treated as non-dispersible or fixed. The only source of dose from such material is then via radiation emitted through the container. Non-special form material is that which is transported in a form where its complete containment cannot be guaranteed at all times and it is thus treated as dispersible or non-fixed. This introduces additional potential exposure routes, such as the contamination of skin or the intake of material via inhalation or ingestion. Various arbitrary assumptions concerning the exposure pathways are made in the derivations of the Q_1, Q_2, Q_3, etc. values given below and it is recognised that these may not be valid in all circumstances. Thus in drafting new Regulations it is suggested that consideration should be given to permitting the package designer or consigner to make use of alternative parameters, provided it can be demonstrated that there is compliance with the basic dosimetric limits outlined below.

Independent of the form of the radioactive material being transported, the derivation of the Q_0 is based upon the following fundamental assumptions:

(a) the shielding of the package (as opposed to the container) is completely destroyed as a result of the accident;
(b) the emergency whole-body dose to a person exposed in the vicinity of the package following the accident should not exceed 5 rem;
(c) the emergency skin dose to a person involved in the accident should not exceed 50 rem;
(d) the exposure time at a distance of 1 m from the damaged package is unlikely to exceed 30 minutes.

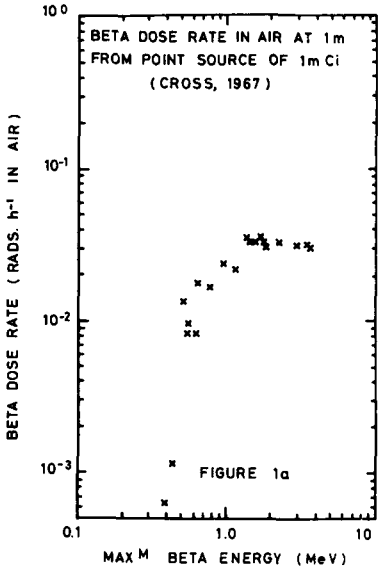
The emergency whole-body dose limit of 5 rem assumed above corresponds to the maximum permissible annual dose for a radiation worker recommended by the ICRP (1977). In absolute terms this represents a relaxation of the emergency whole-body dose limit of 3 rem assumed in the 1973 Regulations, but with the assumed exposure time leaves the radiation dose rate limit from the damaged package unchanged, i.e. 1 rem.h^{-1} at 3 m. The emergency skin dose limit of 50 rem is identical to the annual limit recommended for radiation workers by the ICRP.

4. EVALUATION OF Q_1, Q_2, Q_3, ETC.

The Q_1 value for a radionuclide is determined by consideration of the external radiation dose due to gamma or X-rays to the whole-body of a person exposed near a damaged Type A package following an accident. The source container is assumed to provide no residual shielding and the resultant whole-body dose rate at a distance of 1 m from the damaged package is taken as 10 rem.h^{-1}, which is approximately equivalent to a gamma exposure rate of 15 R.h^{-1}, giving a dose of 5 rem in the assumed 30 minute exposure time. Thus for a point source gamma emitter $Q_1 = 28.35/E_\gamma$ Ci, where $E\gamma$ is the mean photon energy per disintegration in MeV.

The Q_2 value is determined by consideration of the beta dose to the skin of a person exposed following an accident involving a Type A package containing special form material. The variation of beta dose rate with maximum beta-ray energy at a distance of 1 m from a 1 mCi source is shown in Figure 1(a); these results are taken from a tabulation for a range of radionuclides given by Cross (1967), and form the basis of the evaluation of the Q_2 values given below. In addition, the concept of a residual shielding or self-shielding factor (associated with the source container or the source itself) included in the current Regulations is retained here. The 1973 Regulations assumed a shielding factor of 3 for a 2 MeV emitter (equivalent to an absorber of $\sim$ 150 mg.cm^{-2} thickness), and in the present system the following factors are used:

Eβ max. (MeV)	< 0.3	0.3-0.5	0.5-1.0	>1.0
Shielding Factor	30	15	10	3



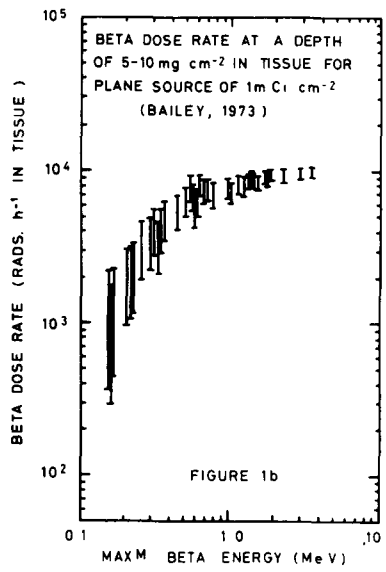


FIGURE 1. VARIATION OF DOSE RATES AS A FUNCTION OF MAXIMUM BETA-RAY ENERGY

Finally, the beta dose rate to the skin of a person at 1 m distance from the damaged package is taken as 100 rem.h^{-1}, which is approximately equivalent to a beta dose rate of 100 rads.h^{-1} in air, giving a dose of 50 rem in the assumed 30 minute exposure time. The Q_2 values evaluated on this basis are then as follows:

Eβ max. (MeV)	< 0.3	0.3-5.0	0.5-1.0	> 1.0
Q_2 (Ci)	1000	300	30	10

Again following the 1973 Regulations, arbitrary maximum Q_1 and Q_2 limit values of 1000 Ci are imposed.

The Q3 value for a radionuclide in non-special form is determined by consideration of the inhalation dose to a person exposed to activity released from a damaged Type A package following an accident. In order to estimate the intake of radioactive material which might occur the following assumptions are made, namely:

(a) The accident occurs in a storeroom or cargo-handling bay of arbitrary dimensions 3 x 10 x 10 m^3, under ventilation conditions corresponding to four air changes per hour, and

(b) the fraction of the package contents which is suspended as a respirable aerosol is 1%.

If the intake of activity by a person is limited to the Annual Limit on Intake (ALI) recommended by the ICRP for radiation workers, this should ensure that the emergency whole-body dose limit of 5 rem cited earlier is not exceeded. The ALI values are based on a "weighted" whole-body or effective dose equivalent of 5 rem, or a single organ dose of 50 rem, whichever is the more restrictive (ICRP, 1977). In those cases where the latter non-stochastic limit applies the effective dose

dose equivalent limit is only marginally less than 5 rem. Thus Q_3=ALI/2.22×10^5 Ci, where the ALI is expressed in becquerels (1 Bq = 27.03 pCi), and is taken from ICRP Publication 30 or an equivalent tabulation for radionuclides not considered by the ICRP (see for example Adams, Hunt and Reissland, 1978). The mixture of standard and SI units is used here for convenience in deriving Q_3 values and comparing them with the current A_2 values.

The assumptions used above combine to yield a nett intake of $6 \times 10^{-6} \times Q_3$ Ci for a person breathing at a rate of $2 \times 10^{-4} m^3 s^{-1}$, compared with an intake of $10^{-6} \times A_2$ Ci assumed in the 1973 Regulations. However, the use of the emergency whole-body dose limit of 5 rem, plus changes in the metabolic data contained in the latest ICRP recommendations, result in only marginal changes in the Type A package contents limits for the majority of radionuclides. Intakes via ingestion or a contaminated wound have not been included in the derivation of Q_3 values since it is considered that the inhalation pathway would pose the most important hazard in most accident situations.

The Q_4 value is determined by consideration of the beta dose to the skin of a person contaminated with non-special form material released from a Type A package following an accident. The variation of beta dose rate at a depth of 5-10 mg cm^{-2} as a function of maximum beta-ray energy for an extended plane source of 1 mCi.cm^{-2} is shown in Figure 1(b); these data are taken from the tabulation given by Bailey (1973), and the depth of 5-10 mg.cm^{-2} is that recommended by the ICRP for the evaluation of the dose to the sensitive basal layer of the skin. In order to estimate the level of skin contamination which might arise in a transport accident involving a Type A package the following assumptions are made:

(a) As a result of the accident 1% of the package contents are uniformly spread over an area of 1 m^2. Subsequent handling of the damaged package results in the pick-up of 10% of this activity (Fairburn, 1961), i.e. the resultant level of skin contamination is $Q_4 \times 10^{-7}$ Ci.cm^{-2}.

(b) The dose rate due to skin contamination should not exceed 10 rem.h^{-1}, this dose being determined at a depth of 5-10 mg cm^{-2} in skin.

The dose rate limit of 10 rem.h^{-1} implicitly assumes that any contamination will be removed from the skin within a period of 5 hours. The Q_4 values obtained on this basis are then as follows:

Eβ max. (MeV)	< 0.1	0.1-0.15	0.15-0.2	0.2-0 5	> 0.5
Q_4 (Ci)	1000	100	30	15	10

As with the Q_1 and Q_2 values above, and following the 1973 Regulations, arbitrary maximum Q_3 and Q_4 limit values of 1000 Ci have been employed here.

The Q_2 and Q_4 values determined here on the basis of skin dose considerations are compared with the A_1 values for beta-emitters contained in the 1973 Regulations (Table IX, IAEA, 1979) in Figure 2. The markedly different energy dependence shown by the Q_2 and Q_4 values arises from the variations in dose rate with maximum beta-ray energy given in Figures 1(a) and (b), coupled in the case of the Q_2 value also with the energy dependence of the residual shielding factor discussed earlier.

The assumptions made here in deriving Q_4 may also be used to estimate the uptake of activity via ingestion. If it is assumed that a person may ingest all the contamination from 10^{-3} m^2 of contaminated skin over a period of 24 hours (Dunster, 1962), the resultant intake of radioactive material is $10^{-6} \times Q_4$, compared with the intake via inhalation of $6 \times 10^{-6} \times Q_3$ derived earlier. Thus, noting that the dose per unit intake via inhalation is generally of the same order or greater than that via ingestion (see for example Adams et al., 1978), the inhalation route is likely to be limiting for internal contamination in the exposure situations postulated here.

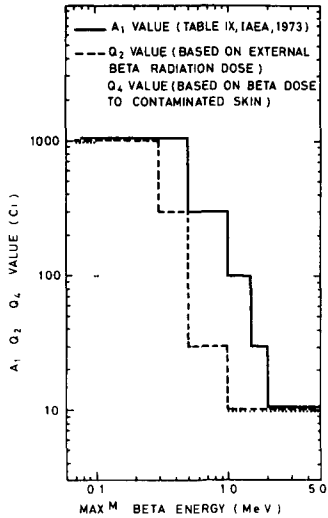


FIGURE 2. COMPARISON OF A_1, Q_2 AND Q_4 VALUES

The Q_1, Q_2, Q_3 and Q_4 values defined above are intended to cover the majority of materials transported in Type A packages. However, some radionuclides or forms of material require special consideration, e.g. gases, alpha and neutron emitters, highly volatile materials, etc. These are covered under the Q_o system by the Q_5, etc. values defined below.

The Q5 value for a gaseous radionuclide transported as a non-special form material is determined by consideration of the external radiation dose (beta or gamma) following a 100% release in an accident. The resultant airborne concentration of activity can be derived from assumption (a) of the derivation of Q_3 above, while the dose rate limits for the whole-body and skin irradiation by gamma and beta-rays are taken as 10 and 100 rem.h^{-1}. The Q_5 value is then the lesser of the two values, $100/E\gamma$ or $170/E_{\beta max}$ Ci. In practice these values are less restrictive than Q_1 and Q_2, thus Q_5 will only apply to gases transported in a shielded container whose integrity cannot be guaranteed under accident conditions.

Within the Q_o system alpha and neutron emitters transported in Type A packages as special form material may be covered in a similar manner to that in the current Regulations. Thus for alpha emitters an arbitrary limit, Q6 value, is defined by $Q_6 = 10^3 \times Q_3$ Ci, with a corresponding limit, Q7 value, for neutron emitters of 2 Ci (see p.82, IAEA, 1973). Finally, for highly volatile materials (or readily dispersible solids transported as fine particulates) transported in a non-special f assumption (b) of the derivation of Q_3 may be invalid since 100% of the package tents could be suspended as a respirable aerosol. The Type A package limit, Q8 value, for these materials is thus defined as $Q_8 = 10^{-2} \times Q_3$ Ci.

Some examples of the application of the Q_o system to a selection of radionuclides are given below for special form (SF) and non-special form (NSF) materials. In about half the cases cited the Type A package contents limit values are essentially unchanged from those set by the A_1/A_2 values in the present Regulations.

Radionuclide	Q_o(Ci) (SF)	A_1(Ci)	Q_o(Ci) (NSF)	A_2(Ci)
Na-22	10	8	10	88
Na-24	7	5	7	55
P-32	10	30	10	30
S-35	1000	1000	30	300
Co-57	230	90	90	90
Co-60	10	7	5	7
Kr-85	30	1000	30	1000
Sr-90	10	10	0.5	0.4
Y-90	10	10	10	10
Ru-106	10	10	2	7
I-131	30	40	9	10
Xe-133	300	1000	200	1000
Xe-135	30	70	30	70
Cs-137	10	30	10	20
Ce-144	10	10	2	7
Pm-147	1000	1000	15	80
Po-210	90	200	0.1	0.2
Ra-226	10	10	0.1	0.05
Pu-239	1	2	0.001	0.002
Pu-241	1000	1000	0.05	0.1
Am-241	1	8	0.001	0.008
Cm-242	45	200	0.05	0.2

This tabulation could be simplified somewhat by the use of a group classification for special and non-special form materials, as used in earlier versions of the IAEA Regulations; for example, a system based on three groups per decade, say in the ratio 1:2:5, would not involve undue loss of accuracy due to rounding off (see pp. 75-6, IAEA, 1973).

5. APPLICATION TO TYPE B PACKAGES

The 1967 Transport Regulations (IAEA, 1967) originally introduced the concept of specifying a containment standard for the package of a large radioactive source in terms of activity loss in relation to specified test conditions. The 1973 Regulations retained this concept in prescribing containment standards for Type B(U) and Type B(M) packages, requiring unilateral and multilateral approval respectively, in terms of the Type A package contents limit A_2 values. A difficulty arises with this approach because the A_2 values refer to individual radionuclides irradiating a single body organ, thus for mixed nuclide intakes they are non-additive. However, this problem is eliminated under the Q_o system by virtue of the new ICRP concept of a "weighted" whole-body or effective dose equivalent if the activity release limits for Type B packages are specified in terms of the Q_3 values derived earlier for Type A packages. This effectively assumes that the dominant exposure route is inhalation of material released from the package during normal

transport or under severe accident conditions. In addition, the external penetrating radiation dose rate limits contained in the 1973 Regulations would apply. Further, it has been suggested that the Type B(U) accident limit is unduly restrictive for severe accidents which are likely to occur only very infrequently (Macdonald and Mairs, 1978), and that under normal transport conditions separate consideration should be given to "small" Type B packages, which may be stored or handled in warehouses, and to "large" Type B packages such as irradiated fuel flasks, which are normally handled out-of-doors or in relatively large enclosures (Goldfinch, 1979). The need for this distinction in accident situations is less important since prompt evacuation is likely to minimise doses if an incident occurs indoors.

For normal transport design dose values of 25 mrem.a^{-1} for members of the public and 100 mrem.a^{-1} for workers handling transport packages have been proposed (IAEA, 1979a). Assuming continuous exposure of the public out-of-doors at a distance of 100 m under average weather conditions leads to a dose rate limit of $\sim$ 3 μrem.h^{-1}, corresponding to an activity release rate for large Type B packages of $\sim 1.7 \times 10^{-5} \times Q_3$ Ci.h^{-1}. For transport workers exposure indoors for 1000 hours per year under the conditions of assumption (a) of the derivation of Q_3 results in a dose rate limit of 0.1 mrem.h^{-1}, corresponding to an activity release rate for small type B packages of $\sim 2.0 \times 10^{-7} \times Q_3$ Ci.h^{-1}. Finally, for a severe accident occurring under the out-of-doors conditions above an emergency dose limit of 1 rem would lead to a release limit of $5 \times Q_3$ Ci, thus Type B package release limits are proposed as follows:

Normal Transport	(Large Type B:	$2 \times 10^{-5} \times Q_3$ Ci.h^{-1}.
	(Small Type B:	$2 \times 10^{-7} \times Q_3$ Ci.h^{-1}.
Severe Accident Conditions:		$6 \times Q_3$ Ci over 1 week.

The dose rate limits used here for normal transport are consistent with the 10 mrem.h^{-1} external radiation dose rate limit at 1 m from the surface of a Type B package contained in the current Regulations. At a distance of 100 m this is equivalent to a dose rate of 1 μrem.h^{-1}, while for exposure indoors with a few metres of a package dose rates would be in the range 0.1-1 mrem.h^{-1}. In addition, the emergency dose limit of 1 rem assumed above provides containment standards comparable with the Type B(M) package accident limits contained in the current Regulations (see for example Macdonald and Mairs, 1978).

6. CONCLUSIONS

The Q_o system described here provides an alternative to the A_1/A_2 system contained in the current IAEA Transport Regulations for the specification of package contents limits and permitted releases of radioactivity from transport packages. The Q_o system is based on the considerations of specific exposure pathways and the dosimetric considerations underlying the derivation of the various Q_1, Q_2, Q_3, etc. values have been clearly defined. Thus the level of radiation exposure to transport workers or members of the public arising from transport operations may be more readily assessed than under the current A_1/A_2 system. The resultant content limits for Type A packages are broadly similar to those obtained on the basis of the current Regulations, and could be simplified somewhat by the introduction of a group classification system similar to that employed in earlier versions of the Regulations.

For Type B packages the use of the ICRP concept of a "weighted" whole-body or effective dose equivalent gives the advantage of additivity of releases when applying the Q_o system to mixtures of radionuclides. In addition, it is suggested that under normal transport conditions separate activity release limits could be

applied to small and large Type B packages. Typically the former would apply to packages normally stored indoors, whereas the latter would be restricted to consignments such as irradiated fuel flasks which are generally handled out-of-doors.

ACKNOWLEDGMENT

This paper is published by permission of the Central Electricity Generating Board. The views expressed in this paper are the personal views of the authors and not necessarily those of the CEGB.

REFERENCES

Adams, N., Hunt, B.W., and Reissland, J.A., 1978, NRPB Report R82.

Bailey, M.R., 1973, CEGB Report RD/B/N2763.

Cross, W.G., 1967, AECL Report 2793.

Dunster, H.J., 1962, UKAEA Report AHSB(RP)R28.

Fairburn, A., 1961, Notes on Certain Aspects of the Regulations. IAEA Safety Series No. 7, IAEA, Vienna.

Goldfinch, E.P., 1979, Consultative Review Paper Prepared for the IAEA Advisory Groups on Basic Principles and Test Standards for the Transport Regulations, May 1979, Vienna.

IAEA, 1967, Revised Edition, Safety Series No. 6, IAEA, Vienna.

IAEA, 1973, Advisory Material for the Application of the IAEA Transport Regulations, Safety Series No. 37, IAEA, Vienna.

IAEA, 1978, Report of the First SAGSTRAM Meeting, October 1978, Vienna.

IAEA, 1979, Revised Edition (As Amended), Safety Series No. 6, IAEA, Vienna.

IAEA, 1979a, Advisory Group Meeting on Radiation Protection and Safety Principles, Paper 27, July 1979, Vienna.

ICRP, 1977, Publication 26, Pergamon Press, Oxford.

ICRP, 1979, Publication 30, Pergamon Press, Oxford.

Macdonald and Mairs, 1978, CEGB Report RD/B/N4440.

DISCUSSION

Question by B. G. Pettersson, Sweden: In determining the Q_3 and Q_4 values, the fraction of the contents released is assumed to be 1% (for non-gaseous and non-volatile substances). In your presentation you mentioned that this (and other) assumption is arbitrary. Recognising that this figure is 10 times higher than the release fraction which is at present one of the bases of the IAEA Transport Regulations, I am asking why the 1% figure was selected?

Answer: The 1% value is considered to be realistic for the respirable fraction of oxides formed when the package contents of non-special form materials are exposed to air or are dispersed in a fire environment. Lower releases may occur in particular situations, but it is intended to be a more representative value than the $0.1\%(10^{-3})$ value in the current Regulations. Indeed, in some cases a higher value may be appropriate and in such cases the Q_8 value would apply.

Question by J. T. Daniels, United Kingdom: I am not a health physicist and I have too much respect for the author of the paper to doubt its technical adequacy. However, I recall that in considering the current revision of the IAEA Regulations it was decided that the regulations should not be changed unless there was cause to believe that the current standards are resulting in an unacceptable risk to the public. Do you have any evidence that the current standards are inadequate and that change is required for that reason?

Answer: We are not claiming that changes are required because current safety standards are inadequate, but rather because in the 1973 Regulations the radiological protection criteria used are not clearly stated. It is therefore very difficult to establish what safety standards have in fact been used and to compare them with the standards applied in other facets of the nuclear fuel cycle. This is particularly true for beta emitters, where the energy dependence of the A_1 value is not correctly included (see Figure 2), and the possibility of skin contamination as an important pathway for non-special form materials is not recognised. The proposed Q_0 system addresses these problems and provides a self-consistent, defensible approach. In many respects it can be claimed to justify the numerical values in the 1973 Regulations, but not their method of derivation.

Question by F. Lange, FR Germany: You claim that your Q_0 system is more transparent from the point of view of a health physicist. I see a problem with the definition of a radionuclide as beta emitter or alpha emitter since it takes no account of the probability of the special decay mode. Could you comment on this point?

Answer: This is a problem with both the A_1/A_2 and the Q_0 systems in the case of radionuclides for which the maximum energy beta particle or alpha emission is not the only or the dominant decay mode. However, in practice this effect is important for relatively few radionuclides. For example, for most beta emitters the dominant emission has an end point energy close to $E_{\beta max}$, as is evident from Figure 1(a) where the data points form a fairly smooth curve with relatively little scatter. For alpha (and neutron) emitters the Q_6 and Q_7 values for special form materials are entirely arbitrary, as is the case with the current IAEA Regulations. This problem is recognised and will be discussed further in an extended version of our PATRAM paper to be published shortly.

EXPERIENCE IN THE NRC PROGRAM FOR INSPECTION AND ENFORCEMENT OF NUCLEAR WASTE TRANSPORTATION IN THE U.S.A.

by
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Background

The transport of commercial low-level nuclear wastes has been carried out in the USA for about 20 years, principally by the highway mode, utilizing exclusive use (full-load) common motor carriers. Although such transport had been generally well-established, it was not until mid-1979 that it actually attracted a great deal of public attention. The situation changed however, when on July 10, 1979, the Governors of the states of Nevada (NV), Washington (WA), and South Carolina (SC)* expressed concern in letters to the Secretary of Transportation (DOT) and the Chairman of the NRC; their letters calling attention to what the Governors considered "serious and repeated disregard for existing rules governing the shipments of commercially-generated low-level nuclear wastes." The Governors demanded that the two agencies develop a plan to upgrade inspection and enforcement of the rules for packaging and shipment of waste materials.

In responding to the Governors, the Chairman of the NRC agreed that improvements and actions were needed to upgrade the transport regulatory compliance program, and outlined a plan of action to bring about immediate improvements in the packaging, transport and disposal of low-level nuclear wastes.

This paper discusses that NRC plan of action and its implementation since July 1979, focusing particularly on the inspection and enforcement program.

Transport Regulatory Agencies In the USA

In the USA the transport of nuclear materials is regulated at the Federal level by two agencies-DOT and NRC. Due to this overlap in authority, the two agencies operate under a written Memorandum of Understanding which is intended to prevent duplication or overlap of effort. Generally under the Memorandum, DOT regulates the total spectrum of hazardous and radioactive materials transport, including both shippers and carriers. NRC has historically regulated only the shippers of the more significant quantities, i.e., Type B and Fissile materials, issuing specific approvals of the package designs therewith, as used by NRC licensees. Non-NRC licensees, i.e., those licensed by certain "Agreement" States or those using naturally-occurring or accelerator-produced materials such

*These states are the only ones in the USA having operating repositories for shallow land burial of low-level nuclear waastes.

as radium, Co-57, etc., are regulated totally by the DOT and the States. In general, therefore, prior to late 1979, the NRC program for inspection/enforcement of tranport had been limited essentially to activities involving Type B and Fissile packages and did not include inspection/enforcement of the overall shipper requirements of the regulations, even for shipments of Type A and LSA by its own licensees.

However, as the most significant part of its overall increased efforts on transportation, on December 3, 1979, the NRC amended its transport regulations in 10 CFR Part 71 to incorporate all of the requirements of DOT regulations in 49 CFR as direct NRC requirements, thus enabling NRC to carry out inspection and enforcement against those DOT requirements. What this means, is that all NRC licensees now are subject directly as NRC requirements, to all of the DOT regulations for packaging, marking, labelling etc. for all types of shipments, including Type A and LSA in addition to Type B and fissile material shipments.

NRC Actions

The NRC's plan of action to implement an augmented program to bring about improvements in the packaging, transport and disposal of low-level nuclear wastes directly addressed the problem areas raised by the three State governors. A comprehensive plan was initiated consisting of the following steps:

- Adoption of the aforementioned change to NRC regulations to permit the NRC inspection staff to insure compliance by NRC shippers with the applicable DOT, as well as NRC regulations and to take appropriate enforcement action as needed on violations.

- The issuance of instructional Bulletins to all generator/shippers of nuclear waste, informing them of their responsibility for more adequate compliance with the regulations for shipment of wastes.

- The issuance of Information Notices to provide assistance to shippers in learning the proper regulatory requirements for compliance.

- An enhanced inspection program at the waste generator/shipper facilities to detect deficiencies in waste packaging before transport commences.

- The dispatch of inspectors to the three waste burial sites to perform routine inspections of incoming waste shipments.

- The promulgation of guidelines for enforcement, including criteria for determining the severity level of violations and the inplementation of an agressive enforcement program for civil penalties against significant violations.

Further details of these steps are discussed below:

Bulletins to Licensees

On August 10, 1979 NRC issued two instructional bulletins to all its licensees (IE Bulletins 79-19 and 79-20, "Packaging, Transport and Burial of Radioactive Wastes"), calling for the immediate attention of NRC licensees to improve their programs for packaging and transport of nuclear wastes so as to eliminate discrepancies which had been observed in shipments to the three

waste burial sites. The specific corrective measures and improvements were to include:

- Maintenance of copies of all current regulations and requirements relating to packaging, transport, and transfer of nuclear wastes.
- Designation in writing of the persons responsible for the transport program.
- Establishment of management-approved instructions and procedures for periodic training of transport operating personnel in the regulatory requirements.
- Performance of an audit by management of the waste transport activities.
- A written response to NRC within 45 days, providing detailed information on the types and amounts of low-level waste shipped in the previous year. (response was required only from licensees believed to be generators of waste, e.g., those licensees who received Bulletin 79-19).

Information Notices

Several Information notices have been developed and issued to all NRC and State licensees to aid in understanding and complying with the requirements for transportation. These include:

a. IE Notice 79-21 "Transportation and Commercial Burial of Radioactive Material." This notice provided information regarding regulatory requirements for packaging, transportation and disposal of wastes, including a summary table of the requirements of each of the three burial sites, relative to acceptable material forms. It also included a copy of a training brochure on transport regulatory requirements.

b. IE Notice 80-24 "Low-Level Radioactive Waste Burial Criteria," provided an update on the changing material form requirements of the three burial sites.

c. IE Notice 80-25 "Transportation of Pyrophoric Uranium," provided guidance to generators and shippers of pyrophoric uranium on the proper methods of packaging and stablizing such materials.

d. IE Notice 80-32 "Clarification of Certain Requirements for Exclusive Use Shipments of Radioactive Materials," provided guidance on NRC application of the requirements for exclusive use highway shipments. (This Notice is discussed later in more detail)

Inspection Activities

The inspection activity of NRC can be categorized as routine or reactive. Routine inspections are those performed at specified intervals, based on considerations of the type of licensed activity and its scope of operation, with various categories and priorities established therewith. Reactive inspections are conducted whenever NRC is notified of a real or potentially significant safety hazard situation. All Inspections are performed by professional radiation safety specialists dispatched from either of 5 regional offices in Philadelphia, Dallas, Atlanta, San Francisco and Chicago. A total of about 100 Radiation Specialists (Health Physicists) are assigned to the inspection of

facilities licensed to use, possess and transfer licensed radioactive material. Since 1979, these inspectors have additionally been tasked to inspect against transport regulatory requirements as a part of their overall inspection activity. Specialized training in transport requirements has also been provided, and the inspectors' procedural manual has been modified accordingly to include these requirements. Inspection personnel from the San Francisco and Atlanta offices are also routinely dispatched to the Hanford, Washington; Beatty, Nevada and Barnwell, South Carolina waste burial facilities to inspect incoming shipments of waste for compliance.

Inspections of Waste Generators Facilities

As a means of verifying the adequacy of waste transport activities of its licensees subsequent to issuance of the IE Bulletin 79-19 in August 1979, a program of followup inspections was initiated by NRC in late 1979. A complete inspection of all operating power reactors was required by January 1, 1980 and all fuel facilities by April 1, 1980. Under this program, the total number of inspections performed through June 1980 included:

Power Reactors - 55 locations and 72 operating units
Fuel Facilities - 29 facilities
Materials Licensees - 655 licensed activities

This program has gradually transitioned to an on-going effort in which transport activities are now routinely inspected in addition to other radiological safety aspects of the licensee's operation. The inspections usually consist of an observation of the waste packaging operations and discussions with plant operating personnel, including those persons who prepare waste packages for transport. Documentation and records of previous shipments are examined for discrepancies. In some cases, completed packages awaiting transport are opened by the licensee in the presence of the inspector to verify that no free-standing liquid is present and that the packaging and its contents are in accord with the transport documents. During the July 1979-June 1980 period, about 200 such packages were opened.

It is not expected that these inspections at the waste generators' facility will derive a large number of non-compliance items, due to their before-the-fact (of transport) nature at the point-of-origin. The program is nonetheless considered to be extremely important for its educational value in achieving a higher level of overall safety and compliance by the shippers of nuclear materials.

Inspections at Burial Sites

On September 1, 1979, NRC Headquarters issued instructions to its Regional Offices establishing requirements for inspecting incoming shipments and packages of waste at the three waste disposal sites; Chem-Nuclear Services Inc. at Barnwell, SC and Nuclear Engineering Co. at Beatty, NV and Richland, WA. These inspections began in September 1979, and were directed at shipment activities of licensee-shippers, not the waste burial firms.

The instructions call for inspectors to spend an average of one day per week at each site, applied in 3-5 day increments at random intervals. These inspections are coordinated with the Agreement State licensing the burial site and with the local regional office of the Department of Transportation. The instructions specify that, where possible, and with the concurrence and presence of the State official(s), each day one shipment from an NRC-licensee is to be selected and one package from that shipment opened to verify that the contents

conform to the description on the shipping papers and conformance with other requirements, including the absence of free-standing liquids. Opening of packages at the burial site is, however, somewhat limited, by the absence of facilities for performing such operations.

Between November 1979 and July 1980, at the request of the State Governors, NRC provided two full-time resident inspectors at the Richland, WA and one inspector at the Beatty, NV site. At the present time, coverage at each of the three locations has reverted to the 3-5 days/month frequency. The overall inspection effort to date at these locations was as follows, through June 1980:

Site	Number of NRC Inspector-days on-site	Number of Shipments inspected
Beatty, Nevada	152	271
Barnwell, SC	41	418
Hanford, WA	298	494
Totals	491	1,183

These inspections have accounted for the bulk of the enforcement cases which have been processed by NRC involving violations of transport requirements by NRC licensees. A listing of the enforcement actions on transportation violations which have been processed by NRC through September 1980 is provided below.

Enforcement Activities

An important enforcement option available to the NRC is the imposition of a civil penalty. Such a penalty, along with a Notice of Violation is issued by NRC Headquarters for the most significant violations or for a large number of lesser violations. Other enforcement options available consist of orders to cease and desist further operation and suspension, modification, or revocation of the operating license of the violator. There are also less formal enforcement options, including enforcement conferences between management and NRC and immediate-action letters which confirm a licensee's committment to alleviate any potentially unsafe condition which has been identified.

In conjunction with the NRC's expanded program of inspection and enforcement of nuclear transport, it issued on December 3, 1979 a written notice to all its licensees, establishing specific criteria for enforcement actions for failures to comply with transport regulatory requirements. As published in the Federal Register, violations were to be categorized into one of four Severity Levels. Severity Levels I and II would normally result in civil penalties and included non-compliance wherein breach of packaging resulted; radiation/contamination above the limits or personnel exposures resulted; or non-compliance with other requirements caused an increase in potential safety hazard. Severity Levels III and IV, which usually do not result in a Civil Penalty involve noncompliance of lesser safety significance. such as labelling, placarding shipping paper, etc. deficiencies.

As a related matter, the NRC just recently on October 7, 1980 issued a new proposed General Statement of Policy and procedure for enforcement actions. It is intended that this policy, as finally adopted, will be formally codified into NRC regulations. This statement explains how NRC will utilize its various enforcement authorities and options. In this new policy statement six, rather than four, levels of Severity would be established. Also, because of the

considerable differences in the type of regulated activities, violations would be scaled according to their importance within each of seven areas: Reactor Operations, Facility Construction, Safeguards, Health Physics, Transportation, Fuel Cycle operations, and Materials Operations. Guidance is provided for determining the appropriate severity level for violations in each of these seven areas.

The new procedures would also provide for base levels of civil penalties in consideration of the class of licensee, with heaviest penalties to be imposed on licensees whose operations generally involve greater nuclear material inventories and greater potential consequences to the health and safety of the public or workers.

During 1979, as a result non-compliance items detected principally during inspections of incoming waste shipments to the burial sites, NRC has completed or nearly completed action on fifteen civil penalty cases. Each of these is described briefly, as follows:

1. A $4,000 civil penalty against a nuclear utility for excessive radiation levels (300 mrem/hr) on the surface of a package on a open transport vehicle (173.393(i), Severity Level II)

2. A $5,000 civil penalty against a nuclear utility for excessive radiation levels (17 mrem/hr) at 6 ft from the sides of an exclusive use vehicle (173.393(j)(3), Severity Level II)

3. A $1,000 civil penalty against a radioisotope supplier for failure to provide instructions on shipping papers for maintenance of exclusive use shipment controls (173.392(c)(9), Severity Level II)

4. Three civil penalties totalling $5,000 against an industrial laboratory for use of nonspecification Type A packaging, improper shipping paper description, and unauthorized transfer of liquids to a burial site (173.395(a)(1), 172.203(d)(1), and 10 CFR 30.41(b)(c), Severity Levels II and III)

5. Two civil penalties totalling $8,000 against a nuclear utility for excessive levels of radiation (300 mrem/hr) at the external surface of a closed vehicle and inadequate surveys of the vehicle (173.393(j)(2) and 173.393(n)(9), Severity Level II)

6. A $5,000 civil penalty against a nuclear utility for excessive radiation levels (600 mrem/hr) on the surface of an exclusive use vehicle (173.393(j)(2), Severity Level I)

7. A $4,000 civil penalty against a nuclear utility for excessive radiation levels (3.7 mrem/hr) in the truck cab of an exclusive use shipment (173.393(j)(4), Severity Level II)

8. A $4,000 civil penalty against a nuclear utility for excessive radiation levels (700 mrem/hr) at the bottom surface of a closed transport vehicle (173.393(j)(2), Severity Level I)

9. A $2,000 civil penalty an industrial radioisotope supplier for excessive radiation levels (300 mrem/hr) at the bottom surface of a closed transport vehicle (173.393(j)(2), Severity Level II)

10. A $3,000 civil penalty against an industrial radioisotope supplier for failure to provide strong, tight packaging for LSA waste (173.392(c)(1) Severity Level II)

11. A $2,000 civil penalty against an industrial radiographer for failure to properly secure a device in the vehicle so as to preclude its movement during normal transport (177.842(d) Severity Level II)

12. A $2,000 civil penalty against a fuel fabrication facility for excessive radiation levels (5 mrem/hr) in the truck cab of an exclusive use shipment (173.393(j)(4)

13. Two civil penalties totalling $2,000 against a waste collector for failure to properly block and brace packages of LSA within the vehicle and failure to use strong, tight packages (173.392(c)(1) and (6))

14. A $2,000 civil penalty against a radiopharmaceutical supplier for failure to properly label packages of non-exempt material (173.403)

15. Three civil penalties totalling $9,000 against a nuclear utility for use of a Type A packaging without a complete certificate and safety analysis of the design, use of non-leak resistant inner containment system therewith, and failure to use strong tight packages for LSA (173.395(a)(1), 173.393(g) and 173.392(c)(1))

Of these 15 enforcement actions (21 items) it was observed that the most frequent violation (8 out of 21 items) involved radiation levels in excess of regulatory limits, accounting for $30,000 of the $58,000 total civil penalties assessed. The next most frequent violations noted were failure to use strong, tight packages for LSA (3 items involving $4000 civil penalties) and failure to properly secure packages to prevent their movement during normal transport (2 items involving $3000). The remaining 8 items totalling $20,000 covered an assortment of matters.

As a means of increasing compliance by shippers, as well as to clarify the correct application of certain regulations, NRC issued in August, 1980, Information Notice 80-32, entitled "Clarification of Certain Requirements for Exclusive Use Shipments of Radioactive Materials." Copies were sent to some 20,000 NRC and State Licensees. The notice focused on two major areas, i.e., the application of the radiation level limits on exclusive use vehicles, and the other general requirements of exclusive use shipments. This latter area was deemed to be of major importance in clarifying the respective responsibilities of the waste "generator" as opposed to the waste collector or "broker." Frequently in the USA shipments of radioactive waste are made as "exclusive use" (full-load) under arrangements whereby the original generator of the waste utilizes the service of a waste collector (i.e., "broker,") who in turn usually engages a common or contract carrier to transport the shipment or transports the materials in his own vehicle as a private carrier. On occasion this "broker" may also be consignee, e.g., a waste burial site operator. Because of this complex arrangement, confusion often arises as to which party is responsible for performing the regulatory requirements of the "shipper" or "consignor."

Although the USA regulations do not specifically define a "shipper" or "consignor" they do specify regulatory requirements that apply to a "shipper" or person who "offers material for transportation" by a carrier. In the nuclear waste transportation business, more than one party (e.g., the original "generator"

and the "broker") can therefore be responsible for functions attributable to a "shipper." The Information Notice provides guidance on the respective responsibilities of each party, and is expected to help in achieving a higher overall level of compliance by both waste generators and brokers.

Observations and Conclusions

Questions frequently arise as to what trends of non-compliance are being experienced in the NRC inspection/enforcement program. Since the program is a relatively new one, and since there was no comparable NRC program prior to mid-1979, it has been difficult to compare the recent experience with trends from any prior period. It has become generally evident however, that overall compliance of nuclear waste shipments, has been increasing significantly. As evidence of this, the Director* of the regulatory program in one of the three states licensing burial sites, South Carolina, has noted that during April 1979, of some 300 incoming shipments inspected by their Agency, a rate of discrepancy of 7.8% was observed. By comparison, in January 1980, of some 410 inspected shipments, the discrepancy rate had droped to 2.7%. (Discrepancies in this case include all major and minor deficiencies, including carrier/vehicle items)

In addition to enforcement actions taken by NRC, the licensing authorities of the three states have, from time to time issued orders suspending the waste burial permits of shippers who are found in violation. These orders usually remain in effect until the shipper has submitted information outlining his corrective measures to prevent recurrence of violations.

The NRC Office of Inspection and Enforcement feels very strongly that during the last 18 months, it has developed and established a vigorous and effective program for inspection and enforcement of transport activities of its licensees. Whereas the initial emphasis has been on waste transport and inspections of shipments upon receipt at the disposal sites, inspection emphasis has more recently been shifting to include more before-the-fact inspections of packaging and shipment operations at the point-of-origin. Increased emphasis is also being placed on transport activities of shippers of non-waste products and other materials. The NRC has worked and will continue to work closely with the other regulatory agencies i.e., the DOT and the States, so as to continue its committment to an overall enhanced program of inspection and enforcement of nuclear transportation.

*Statements by Heyward G. Shealy, South Carolina Bureau of Radiological Health, February 20, 1980 during meeting on Low-Level Radioactive Waste, Incline Village, Nevada (NUREG/CP-0013)

Question by C. B. G. Taylor, United Kingdom: What classes of shippers were mostly concerned in these inspections? Government sites, hospitals, manufacturers of radioisotopes?

Answer: Government generators of waste are not usually involved in shipments to the three communally operated waste burial sites. The shippers generally fall into two categories: private utilities shipping low level waste accumulated from the operation of nuclear power plants; and waste from medical/academic institutions usually through waste collector (broker) organisations.

Question by C. B. G. Taylor, United Kingdom: Has the firmer application of the Regulations had any significant effect on the operations of these bodies? In particular, has nuclear medicine been hindered because hospitals could no longer dispose of their wastes?

Answer: Nuclear medicine operations, although threatened by short-term site closures in Washington and Nevada, were not seriously affected. However, the impact on medical shippers such as the utilities was affected causing a more than normal accumulation of wastes at their facilities.

Question by D. P. Ryskiewich, USA: What effect will the recent vote by the citizens of the state of Washington to ban radioactive waste from other states have? Please comment on this ban and its ramifications.

Answer: During the recent national elections in the USA on November 4, 1980, the state of Washington ballots included an initiative which was passed which will affect the operation of the low-level waste burial site in that state. The effect of this initiative which was adopted by the voters will cause the cessation of further acceptance of shipments to the Richland, WA, burial site after mid-1981, unless the wastes are originated from medical use applications or unless the waste is from a generator in a state which has become a party to a regional compact or plan for a low-level repository in the origin state or region. The initiative, although likely to cause initial hardship on non-medical generators of waste, may in the longer term have a more positive effect of nurturing and forcing the development of more regional sites for low-level waste burial.

Question by J. T. Daniels, United Kingdom: Is there any indication, from the violations so far detected, that non-compliance has stemmed from ignorance on the part of the consigners, who may have believed they were acting correctly, or has there been willful contravention in the belief that it would be cheaper than complying with regulatory standards?

Answer: My opinion is that there may have been some of both reasons, with a greater tendency towards non-compliance due to ignorance of the requirements as well as a knowledge that there was little inspection and enforcement being done against violations. I tend to think that willful non-compliance was generally not too common.

RADIATION LEVEL FOR LOW SPECIFIC ACTIVITY MATERIALS IN COMPACT STACKS

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In the "Regulations for the Safe Transport of Radioactive Materials 1973, Revised Edition" the transport index for low specific activity (LSA) materials means the number expressing the maximum radiation level (dose equivalent rate) at any point 1 m from the external surface of the load multiplied by the multiplication factors in Table II of the Regulations appropriate to the cross-sectional area of the load. These multiplication factors were originally devised by means of geometrical considerations depending on the radiation fields of point and plane sources. These factors take into account the different decreases of dose rates in the fields of these sources with distance /1/.

There are discrepancies if one compares transport indices estimated according to this rule with the values of the transport indices given, for example, in the segregation table of the "International Maritime Dangerous Goods Code". The following example using the rules of the code shows this. In lack of an actual radiation level measurement for chemical concentrates of thorium, 30 mrem/h (0.3 mSv/h) may be taken as the maximum radiation level. Assuming that the cross-sectional area of the load is greater than 20 m^2 the transport index in this case is 570. For the same size of the load the transport index for ores and physical concentrates of uranium and thorium is 760 according to the rules given in the code. In Paragraph 5.3 of the code, the maximum value of the transport index included in the segregation table is 400.

Determining transport indices for thorium nitrate from measured maximum radiation levels /2/ a value of 247 is obtained for the highest reported level, which is suitable for the application of the segregation table.

These discrepancies were the reason for the calculation of the dose rate on the axis of a cylindrical volume source containing thorium nitrate as a model for a stack of packages containing LSA material. These calculated dose rates were compared with those of a point source producing the same radiation level to find the relation between dose rates in the fields of volume and point sources.

As a lower limit of the photon energy E_γ contributing considerably to the dose rate, an energy of 0.5 MeV was chosen /3/7/. The nuclides together with the photon energies used for these calculations are given in Table I. Radium-228 and its daughter actinium-228 is not contained in fresh thorium nitrate. It is built up with a half-life of 5.75 years and reaches equilibrium with thorium-232 after about 30 years. Thorium-232 in equilibrium with its daughter products emits the most penetrating radiation of all LSA materials. The assumed content of the model source therefore covers the worst case of all LSA materials.

The composition of thorium nitrate is assumed to be

$$Th(NO_3)_4 \cdot 6\ H_2O$$

The measured density of a thorium nitrate sample is ρ = 1.61 g cm^{-3}

Table I

Nuclide	Energy MeV	Photon emission Probability %	Nuclide	Energy MeV	Photon emission Probability %
^{228}Ac	0.908	25	^{208}Tl	0.511	23
	0.960	20		0.583	86
				0.860	12
^{212}Bi	0.727	7		2.641	100
	0.785	1.1			
	1.625	1.8			

The dose rate J on the z-axis of the cylindrical volume source is given by /3/

$$\dot{J} = \sum_{j=1}^{n} S_V B_j C_j \int_0^D \left[E_1(\mu_j z) - E_1(\mu_j z \sqrt{1 + \frac{R^2}{(z+a)^2}}) \right] dz \qquad (1)$$

with

S_V = source strength per volume
B_j = energy-dependent dose build-up factor at energy E_j
C_j = factor transforming photon flux density to dose rate at energy E_j and taking into account the emission probability for photons with energy E_j
μ_j = linear attenuation coefficient for $Th(NO_3)_4 \cdot 6H_2O$ at energy E_j
a = distance between a point on the axis and the surface of the cylinder
R = radius of the cylinder
D = height of the cylinder

$$E_1(t) = \int_t^\infty \frac{e^{-x}dx}{x}$$

The contribution from photons scattered inside the volume of the source to the dose rate on the axis is approximately considered by the build-up factor B_j, which depends on the photon energy and the composition of the absorber. B_j is calculated by the approximation formula /4/

$$B_j = \begin{cases} \exp\left\{ \frac{\ln[y_j(y_j-1)+1]}{\ln 2} \ln(1+\frac{t_j}{y_j}) \right\} & y_j < 2.8 \\ \exp\left\{ \frac{y_j-1}{\ln 2} \ln(1+\frac{t_j}{y_j}) \right\} & y_j > 2.8 \end{cases}$$

with t_j = D relaxation length

$y_j = \frac{\mu_j}{\mu_{en,j}}$ attenuation coefficient / energy absorption coefficient

Because there are no tabulated values of the attenuation and of the energy absorption coefficients for $Th(NO_3)_4 \cdot 6\,H_2O$, these coefficients were evaluated using the tables issued by Hubbell /5/. For computing the factor C_j the emission proba-

bility given in /6/ and energy absorption coefficients for air tabulated in /5/ were used. The radii of the model source which were used correspond to the 4 areas F given in Table II of the Regulations. Height D of the cylindrical source should be so chosen that we calculated contribution to the dose rate from parts of the source lying at distance D behind the surface under consideration can be neglected. This influence of self-absorption was investigated by numerical integration of equation (1) for different values of D and fixed values for distance a and energy E_j. For distance a the value 1 m was inserted and for energy E_j the highest value given in Table I. These calculations show that for values of D greater than 0.8 m the dose rate on the axis at a fixed distance does not increase further. For the computation of the dose rate on the axis of the cylinder, a height D of 0.8 m was therefore chosen and distance a was varied from 0.5 m to 80 m. Absorption and scattering in air were neglected.

The results of this numerical integration are given in Fig. 1. All calculat dose rates were normalized at a distance a of 1 m.

In the form of representation in Fig. 1 the decrease of the dose rate in the field of a point source located at a = 1 m is represented by a straight line. Next to this is the curve representing the smallest volume source. With the increasing size of the volume, the decrease of the dose rate near the surface of the source varies more and more from that of a point source. Only for distances a greater than 15 m are the ratios between the dose rates almost constant, for all of the 4 ares F used for caluclation.

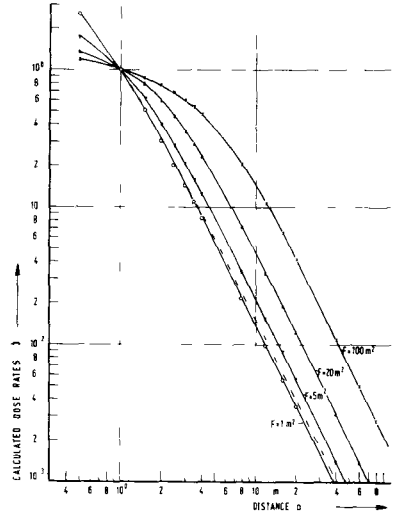

Fig. 1 Calculated dose rates on the axis of cylindrical volume sources and a point source located inside the load 0.288 m behind the surface (---)und consideration, normalized at a distance of 1 m from the load.

The calculated dose rates of the volume sources have now to be compared with those of a point source producing the same radiation level. It has been shown by Andronow /8/ that a point source producing a dose rate of 200 mrem/h (2 mSv/h)on the surface of the load and at a distance of 1 m, a dose rate of 10 mrem/h(0,1 mSv/h) is located inside the load 0.288 m behind the surface being considered. The dashed line in Fig. 1 represents the decrease of the dose rate in the field of such a point source.

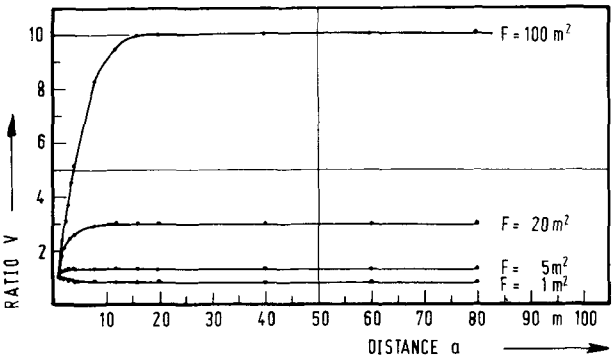

Fig. 2 Distance-dependence of ratios V between dose rates of volume sources and a point source located inside the load

Dividing the dose rates of the volume sources by those of this point source yields the curves given in Fig. 2. These curves clearly show that the ratios V between dose rates of volume sources of the size under consideration and those of a point source located inside the load are almost constant for distances a greater than 15 m. The constant values of these ratios V may have the same significance as the multiplication factors in the Regulations, because they are a measure for the differences in the radiation fields of the volume sources and the point source located inside the load.

A curve representing these ratios V is given in Fig. 3, which is derived from the curves in Fig. 2 at a distance a = 20 m.

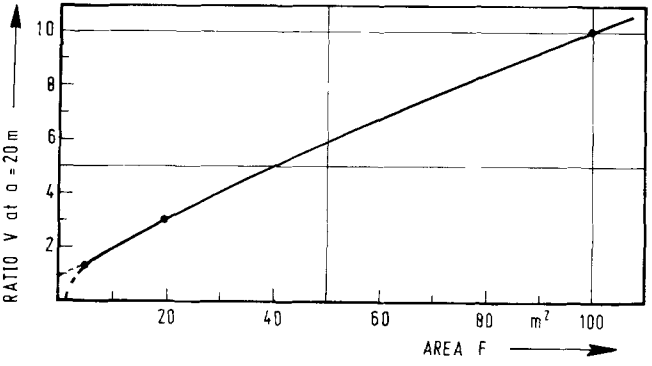

Fig. 3 Rations V between dose rates of volume sources and a point source located inside the load at a distance of 20 m from the load

In Table II the cross-sectional areas of the load with the multiplication factors from Table II of the Regulations and rounded-off values of the caluclated constant ratios are given for comparison. These newly calculated figures are considerably lower than the multiplication factors in the Regulations. Taking into account that thorium-232 in equilibrium with all its daughter products emits the most penetrating radiation of all LSA materials,and covers the worst case, the new figures should be considered for the 1983 revision of the IAEA Safety Series No.6.

Table II

Cross-sectional area of the load m^2	Multiplication factors from Table II of the S.S.,No. 6	Computed ratios for volume and point source for distance a >15 m
1	1	1
5	3	2
20	6	3
100	19	10

REFERENCES

/1/ K.J. Aspinall a. R. Gibson
The Control of External Radiation During the Transport of Radioactive Material. AHSB (RP)R31 February, 1963

/2/ Klinger,private communication

/3/ R.G. Jaeger et al.
Engineering Compendium on Radiation Shielding, Vol.I, Springer Verlag Berlin (1968)

/4/ O'Brien, K., W.M. Lowder a. R.L. Solon
A Semiempirical Method of Calculating the Energy Absorption Buildup Factor with an Application to a Uniformely Contaminated Space having Spherical Bounderies, Nuclear Science and Engineering 3, (1958)

/5/ Hubbell
Photon Cross Sections, Attenuation Coefficients and Energy Absorption Coefficients from 10 keV to 100 GeV NSRDS-NBS 29(1969)

/6/ Lederer et al.
Table of Isotopes Sixth Edition, John Wiley and Sons Inc., New York, London, Sydney (1968)

/7/ W. Kolb, private communication

/8/ IMCO CDG XXII/7(a)1 vom 25.1.1973

DISCUSSION

Question by A. H. Partridge, United Kingdom: As you are aware, the multiplication factors in the IAEA Regulations are used also to determine the transport index for large freight containers. Does your study indicate that transport indices calculated in a similar manner for large freight containers containing individual packages, such as type B packages, are also likely to be in error to the same extent?

Answer: The model calculations for large volume sources show that the distance dependence of the dose rate is mainly dependent on the size of the volume. The distribution of the source content, if it is distributed homogeneously or uniformly, has a minor influence on the dose rate. In the case of large freight containers containing individual packages, the difference between new figures calculated and the multiplication factors from Table II of the Regulations may therefore be of the same order as in Table II of the paper.

LOW SPECIFIC ACTIVITY MATERIALS--A NEW LOOK AT AN OLD CONCEPT

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I. Regulatory Approach for Radioactive Substances.

Embodied within the International Atomic Energy Agency (IAEA) "Regulations for the Safe Transport of Radioactive Materials"1/ are several different methods for ensuring appropriate control of the wide variety of hazards presented by radioactive substances in transportation. The approach that has been taken in transportation safety has been to require safety measures that are commensurate with the hazards of the substances being transported.

At the lower end of this hazard spectrum are those substances which contain radionuclides but are not ordinarily considered to be radioactive. In order to quantitatively identify these substances, a limit of 2.0 nanocuries per gram of material has been selected. Since substances with a radioactive content below this limit do not pose a radiological hazard in transportation, these substances are not subjected to any transportation safety requirements.

Requirements have been established for most radioactive substances that are based on the quantity of radioactivity which is contained in a package. For small quantities that present very little radiological hazard even if released from their packagings, only certain minimal requirements are placed on their transportation. Included in this category are certain instruments, devices and manufactured articles which also contain limited radioactive contents.

Another category which relies only partially on packaging integrity is that of Type A quantities of radionuclides. The Type A package content limits are based on consideration of a "median" accident in which 10^{-3} of the package contents are released and 10^{-3} of this amount is intaken by any individual. The resulting radiation dose from this accidental intake is limited to those values prescribed by the International Commission on Radiological Protection (ICRP). So Type A packages are required to withstand the normal conditions of transport and minor mishaps, are limited in content such that individual doses following a "median" accident are not unacceptable and are not expected to withstand severe accidents in light of their limited contents.

Quantities of radioactive substances which present a significant radiological hazard (Type B) or which present a criticality concern must be shipped in packages designed to withstand severe accidents. Such packages are expected to withstand a severe accident with little or no significant loss of effectiveness for containment of the contents, radiation shielding, decay heat dissipation and criticality control.

These categories of regulated substances have one characteristic in common--they are controlled according to total activity in a package. There has been a recognition of another class of materials which have such a low radioactive concentration that while they need to be regulated in transportation, they do not pose a significant radiological hazard regardless of the quantity of material or its packaging integrity.

II. Development of Concept.

In order to avoid imposing unreasonable packaging requirements on substances with low radioactive concentration, a category for these materials was developed along with commensurate safety requirements. These substances were designated "Low Specific Activity" (LSA) materials in reference to their low radioactivity per unit mass of material.

The early (1964 and 1967) IAEA regulations which dealt with LSA materials were designed to cope with certain recognized hazards and four basic subcategories of LSA materials.2/ These were:

a. Solids - The most significant recognized hazard related to solids involved internal radiation doses as a result of an accident in which the material was dispersed into the air and inhaled by humans. For modeling purposes it was assumed that the materials were airborne at a density of 10 mg/m^3 (very dusty) and that an individual at moderate exertion breathed at a rate of 2 m^3/h. For a one-half hour exposure an intake of 10 mg was considered conservative especially considering that no allowance was made for packaging containment. So where 10 mg of material would just equal the radiation dose accepted for a "median" accident involving a Type A package, a limit was established for LSA solid materials. Below this specific activity limit, intake of a higher internal dose was felt to be highly unlikely and so normal commercial packaging would provide more than adequate protection.

b. Tritiated Water - There was some disagreement as to whether or not liquids could be shipped under the provisions of the IAEA regulations apparently derived for solids. Provisions were made, however, for at least one liquid--tritiated water.

For modeling purposes, it was assumed that a release of tritiated water occurred in an enclosed room at a temperature of 20°C and that the resulting atmosphere would contain 17 g/m^3 of vapor. Given that the concentration of the material is 5.0 mCi/ml (a figure based on experience as being a transportation need), the resulting atmospheric activity concentration would be 10^{-1} uCi/ml. Hence a one-half hour stay in the room would result in an intake which produces a whole body dose of about 25 rem. Given that this was the ICRP limit for accidental exposures and that ventilation of the room or occurrence in an open area would reduce the expected exposure by several orders of magnitude, the 5.0 mCi/ml limit was accepted as reasonably conservative.

c. Surface Contaminated Materials - As in the case of LSA solids, the conditions established for allowing surface contaminated materials to be classified as LSA were that probable intake by any individual would be unlikely to exceed the accepted ICRP limits. Again, these conditions had to be met without regard to package containment or release fractions following an accident. The main difference between contaminated materials and other solids was that they were not homogeneous in nature so specific activity was not an acceptable characteristic to describe them.

Earlier work3/ had shown that a surface contamination level of 1 uCi/cm^2 resulted in a corresponding airborne concentration of $2x10^{-8}$ uCi/cm^3 under simulated very dusty conditions in open air. Given the conditions assumed for solids (one-half hour exposure, 2 m^3/h breathing rate) the resulting individual intake from airborne material would be $2x10^{-8}$ uCi/cm^3 x 10^6 cm^3/m^3 x 2 m^3/h x 0.5h or $2x10^{-2}$ uCi per uCi/cm^2 of surface contamination. Relating these values to the accepted maximum permissible intakes resulted in establishing limits of 0.1 uCi/cm^2 for the most radiotoxic nuclides and 1.0 uCi/cm^2 for other radionuclides2/. An additional conservatism was included by requiring that the surface contamination be in a "non-readily dispersible" form.

d. Some materials which were commonly shipped and composed of isotopes which in themselves were essentially LSA materials (natural thorium, Th-232, natural uranium, U-235, U-238) were specifically included. Consequently, ores, concentrates and other forms of these radionuclides were listed in order to preclude having to repetitively perform the calculation of the materials' specific activities.

Even though the development of the LSA material specifications assumed no retention of containment following an accident, packaging requirements were

established to avoid unnecessary and repeated exposure of workers to the materials. In order to provide flexibility in shipping three ways were prescribed:

a. Solid materials were authorized in strong industrial packagings which were labeled, handled and shipped the same as other packages of radioactive materials. These packages were limited to Type A quantities of LSA materials, the contamination limits and external radiation levels of other packages.

b. All LSA materials were authorized in strong industrial packages shipped under full load conditions. These packages were not required to be marked, labeled or meet contamination or package contents limits. Primary safety requirements were vehicle or compartment activity and external dose rate limits.

c. All LSA materials were authorized in bulk under full load conditions subject to vehicle or compartment activity and external dose rate limits.

Between the 1967 and 1973 revisions of the IAEA regulations, several areas of concern were addressed including tritiated water activity concentration limits and the ability of LSA materials to concentrate during transportation.

There was an expressed need to raise the limits for tritiated water above 5 mCi/ml and after the 1967 regulations were established some changes were made in the accepted radiological values assigned to tritiated water. Specifically, the quality factor (QF) was lowered from 1.7 to 1.0 and the biological half-life was accepted as 10 days instead of the earlier 12 days. Additionally, recalculation of the earlier values indicated an error in arriving at the 5 mCi/ml value.

A new model was developed which assumed a leaking drum in a confined area which resulted in an individual breathing the atmosphere and being wetted over one quarter of the body area for one hour. Account was made for three pathways of intake: inhalation, dry skin absorption of vapor and wetted skin absorption. Using the newer radiological values and the 25 rem whole body limit an allowable value of 15 mCi/ml was obtained. Later calculations were performed using the circumstances of wetting only from the splashing from a dropped drum along with the other assumptions and an accepted dose of 3 rem. This gave a finally agreed-upon limit of 10 mCi/ml which was adopted in the 1973 revision.

In preparing the 1973 revision of the IAEA regulations, it was agreed that the Transport Group system of classifying radionuclides would be replaced by the A_1/A_2 system which addressed each radionuclide on its own properties. As a result, the LSA limits had to be set on the basis of the A_1/A_2 system. Since the basis for the LSA system was an acceptability of the Type A "median" accident consequences (10^{-6} A_2), it was easily tied to the 10 mg material intake assumption. The resulting limit (10^{-6} A_2/10 mg or 10^{-4} A_2/g) gave LSA material values that directly reflected the various model parameters. In fact, this action represented a relaxation or raising of the specific activity limits for most radionuclides since they were no longer constrained by the most restrictive radionuclide in each Transport Group. For example, the iridium-192 limit was raised from 0.3 mCi/g to 2.0 mCi/g and americium-241 from 0.0001 mCi/g to 0.0008 mCi/g.

Considerable concern was expressed over the potential for LSA to concentrate under certain conditions of transportation with a resulting potential for consequences exceeding those originally modeled. In order to prevent this, materials were basically split into those which could increase in specific activity during transport and those that could not. For additional protection, those materials whose nature did not guarantee remaining as LSA materials had to be shipped under full load conditions. While all materials were required to maintain their LSA status during routine transport, only those that would remain as LSA following dissolution in water and recrystallization, precipitation, evaporation, combustion and abrasion could be shipped under non-full load conditions.1/ Considering these changes and the validity of the underlying assumptions, the previously imposed Type A contents limit was dropped for packages shipped under non-full load conditions.

A new category of materials was created to provide for the shipment of materials in which the activity was distributed throughout a solid object(s) or was in a solidified form and objects contaminated in excess of the LSA limits. This new category - Low Level Solid (LLS) material - had several conditions attached to it. Solids had to be of low enough solubility that even with a loss of packaging and total immersion in water the activity leached from the solid in one week would not exceed 0.1 x A_2 for the contained radionuclide(s). Such solids were restricted to a specific activity of 2×10^{-3} A_2/g or less. Non-radioactive materials contaminated with non-ready dispersible radioactive contamination could be shipped as LLS provided that the level did not exceed 20 times the LSA limit for similar materials. In order to provide adequate safety during the transport of LLS materials, they were required to be shipped under full load conditions and in packagings that were more stringent than those required for LSA.

The result of all of this development was a somewhat complicated system for classifying and transporting LSA and LLS materials. However, the technical basis for the system and the need to transport these materials in an economical yet safe manner provided the necessary rationale for continuing with the philosophy and regulatory provisions in spite of their complexity.

III. The Need For Improvement.

Member States of the IAEA have now had extensive experience in transporting a wide variety of LSA and LLS materials. This experience has led to some concerns about certain aspects of the classification, packaging and carrier requirements that pertain to these materials. As with all materials that are transported, there is a finite probability that they will be involved in accidents that range from minor to severe. In the case of LSA and LLS materials the probabilities have proven realities and the materials have been involved in severe accidents. This experience has indicated a need to review all aspects of these materials to ensure that the public's safety is maintained.

One of the foremost reasons for analyzing the LSA and LLS requirements involves products which are being shipped that are at the limits allowed. The quantities of these materials being transported is increasing. Materials such as spent ion exchange resins approaching the maximum allowable specific activities of a variety of radionuclides are frequently shipped in large volumes. The nature and transported quantities of materials such as these make it imperative that the LSA concept be reviewed for its adequacy in today's situation.

Several IAEA convened Advisory Groups and Technical Committees have indicated a need for refinement of the LSA category and a strengthening of some of the applicable packaging requirements. Recently, the following groups have met and made recommendations on LSA to the IAEA:

March 1977, AG-126 - The IAEA should expand the Advisory Materials in Safety Series No. 37 to more fully explain acceptable LSA procedures.

December 1977, AG-144 - The AG expressed concern over the possible external radiation levels from LSA and LLS packages following a loss of shielding. Such a concern can be construed as an implicit problem with possible radiation levels from the materials and a lack of appropriate packaging requirements for such materials.

July 1979, AG-225 - The AG recommended deleting the provision for shipping LSA materials that could concentrate as a result of an accident and that a requirement be added for either a retention of shielding following accidents or that the external radiation levels from the materials be limited to 1 rem/h at 3 meters (the same as the Type A limit).

August 1979, TC-272 - The TC reiterated the need to limit the allowable external dose rates following accidents and recommended establishing minimum package test requirements for LSA packages.

As a result of these and other related recommendations from IAEA groups, the subject was forwarded to the IAEA Advisory Group meeting in September 1980 on the Comprehensive Review and Revision of the Regulations.

These concerns have led to the investigation of some of the properties of LSA materials by regulatory agencies in the U.S. Work performed by SRI International and funded by the Nuclear Regulatory Commission has produced preliminary results concerning the properties of currently allowed LSA materials and the exposure pathways to humans following accidents4/5/. Basically, this work has quantified some of the concerns, such as the external dose rates from gamma emitting LSA materials. It has also shown that for other pathways such as ingestion, inhalation and skin doses from beta emitting LSA materials are relatively well controlled by the existing regulatory requirements. Specifically, one can conclude from the SRI International work4/5/ and a recognition of the background of LSA that:

a) The assumptions used in modeling inhalation (i.e. 10 mg of material) are reasonably conservative. Only for very finely divided powders such as silt and clay is it likely that more than 10 mg of material will be inhaled and retained. The likelihood of such finely divided material being involved in an accident under the assumed conditions of dispersion and personnel immersion in the dust with no respiratory protection is sufficiently remote.

b) The possibility of ingestion of a significantly hazardous amount of LSA is sufficiently remote. While there are some isotopes that could result in activity intakes exceeding quarterly occupational exposure limits when less than 10 mg is ingested, they are rare.

c) For some beta emitting radionuclides, the quarterly occupational skin dose limit can be attained in a relatively short time with conservative assumptions. For a deposition of 10 mg/cm^2 of skin, a radionuclide such as Ag-111 can result in a beta dose of 18.75 rad to the skin in 29 minutes5/. Consequently, under the assumptions of heavy deposition on the skin and no action taken to remove the material, the beta dose can be of concern following an accident.

d) For "hard" gamma emitting radionuclides the external dose rate from LSA materials can be high. Calculated dose rates for hypothetical materials with a specific activity at the LSA limit for a nuclide such as cobalt-60 give very high figures. Some of these values indicate that rescue workers could receive the maximum recommended accidental exposure (3 rem) in a short time following dispersal of the material or a loss of shielding.

IV. Proposed Changes.

One of the tasks undertaken by the September 1980 IAEA Advisory Group on the Comprehensive Review and Revision of the Regulations (AG-266) was to consider all of the information and recommendations related to LSA and LLS materials and to propose any necessary changes. After lengthy discussions and deliberations on the basic approach to LSA and LLS materials, the working group which was assigned the task recommended some fundamental changes in the application of the concept. The most substantial changes which were adopted by the Advisory Group involve the classification of materials, external radiation level limitations and packaging requirements.

The materials which were kept in this category were divided into two overall groups, Low Specific Activity (LSA) materials and Low Contamination Objects (LCO). While some people will undoubtedly have heartburn over this misapplication of the English language, the name LCO was the only one agreed to as reasonably

describing objects which are contaminated with relatively low levels of radioactive contamination.

In order to address the problem of external radiation levels from LSA materials and LCO, the Advisory Group accepted the fact that a somewhat arbitrary limit needed to be applied. The alternative was to not place a radiation level limitation on these materials and that was believed to be unacceptable. As had been suggested by Advisory Groups before, a limit of 1 rem/h at 3 meters from the unshielded material was adopted. It was stipulated that this limit would apply from "the surface of the unshielded and uncontained material" so that there is a limit on both the package (1 rem/h at the surface, paras 534 and 537 1/) and the LSA material or LCO within the package. Therefore, following an accident and dispersal of the material or a loss of shielding from the package the radiation dose rate would be limited. An example of what effect this might have on some substances can be seen by taking the case of cobalt-60 LSA in a 210 liter drum. Assuming a density of 2.25 g/cm^3 (comparable to sand) and 10^{-4} A_2/g of material (with A_2 for Co-60 = 7 Ci) gives $210\ L \times \frac{10^3\ cm^3}{L} \times 2.25\ \frac{g}{cm^3} \times 10^{-4}\ \frac{A_2}{g} \times \frac{7\ Ci}{A_2}$ or about 330 Ci per drum. This results in a calculated dose rate of about 58 R/h at 1 meter and 9 R/h at 3 meters. So the proposed limitation of 1 rem/h at 3 meters would force a substantial reduction in the specific activity of this hypothetical cobalt material.

The existing text in the IAEA regulations (para. 121(e) and (f)) is unclear as to whether routine transport conditions (i.e. "conditions likely to be encountered in transport") or accident conditions (i.e. "evaporation, combustion, abrasion, etc.") are to be considered in classifying LSA materials. The Advisory Group was of the opinion that only those conditions generally encountered in incident-free transport should be considered. While there could be some accident related circumstances which could concentrate the materials, they were believed to be extremely unlikely and not of major consequence. As a result, a general requirement was imposed on all LSA materials and LCO that "under conditions likely to be encountered in routine transport, (the material) remains essentially uniformly distributed or homogeneous."6/.

With these two conditions imposed on all LSA materials, they were subdivided into the following subcategories:

LSA-I; ores and concentrates, either solid or liquid, and solid compounds, mixtures or metallic forms of natural or depleted uranium and natural thorium.

LSA-II; materials in which the average estimated specific activity does not exceed 10^{-4} A_2/g for solids and gases nor 10^{-6} A_2/g for liquids.

LSA-III; solids which are essentially those presently categorized as LLS.

The reduction by a factor of 100 for the specific activity of LSA-II liquids was the result of considerable discussion. In related discussions on radiation protection it was decided that tritium should be treated as other radionuclides and not be subjected to the arbitrary limit of 1000 Ci placed on the A_2 value for beta emitters, since tritium has no Bremsstrahlung hazard. Consequently, tritium and tritiated water could now be treated the same as other liquids since their "true" hazard would be reflected in the A_2 values. However, recent analyses on tritiated water transport7/ when extrapolated indicated that 10^{-4} A_2/g was too high a value under accident conditions. To resolve the dilemma it was decided that the "true" A_2 value for tritium should be used and that all liquid allowances would be adjusted accordingly. Since 10^{-6} A_2/g was an appropriate value for tritiated water it was applied to other liquids as well since the assigned A_2 values would "normalize" the hazards.

The second basic category of materials was called LCO and was established to provide for the transportation of objects which are surface contaminated (as opposed to LSA-III which is activity within a solid). As with LSA materials, a limit of 1 rem/h at 3 meters from the unshielded material was imposed. Two levels of LCO were established which are:

LCO-I; when the non-fixed external contamination does not exceed 10 times the package contamination limits of Table XI, Section V 1/ and the fixed external contamination does not exceed 10^4 times that value.

LCO-II; when the total of fixed and non-fixed external contamination does not exceed 2×10^5 times the Table XI values.

While these classifications are somewhat arbitrary they do allow for the specification of appropriate packaging requirements for the hazard presented the materials. In keeping with the previous Advisory Groups' recommendations, minimum packaging requirements were established for each grouping and the requirements were generally increased. As before, packaging requirements were made more stringent for non-exclusive use shipments in recognition that the handling of packages would not be as controlled. A significant improvement in clarifying the packaging requirements was provided by the use of a matrix as follows7/:

Material		Packaging Level	
		Exclusive Use	Not in Exclusive Use
LSA-I:	Solid	1	2
	Liquid	2	3
LSA-II:	Solid	2	3
	Liquid and Gas	3	4
LSA-III:	Solid	3	4
LCO-I:		2	3
LCO-II:		3	4

Packaging levels:
1-Retains contents and shielding "under conditions likely to be encountered in routing transport."
2-Must meet paras 201-206, 208 and 219 1/.
3-UN Packing Group III plus paras 201-206, 208 and 219 1/.
4-Type A packaging, excepted from para 226 1/.

Provisions were made in the proposed regulations for transporting LSA-I, LSA-II and LCO-I under bulk or unpackaged conditions. In recognition of the greater potential consequences of bulk liquid shipments, however, the allowable specific activity was further reduced by a factor of 10 to 10^{-7} A_2/g. This was believed to be necessary since any failure of the package (vehicle) would likely result in complete loss of a large volume of liquid. Likewise the conveyance activity limits for bulk liquids were reduced to $10xA_2$ and for inland waterway craft were reduced to $10xA_2$ for packaged liquids and A_2 for bulk liquids.

V. What Now?

The changes proposed by IAEA AG-266 will be circulated to Member States for comment and it will be necessary to carefully review these proposals. Some changes have been proposed which address specific problems and it is necessary to ensure that they actually solve the problems. In some areas reductions have been proposed on radiological protection grounds and the cost of implementing these changes should be considered. There is considerable work that needs to be done and information collected and analyzed before the changes can be construed as

having solved the problems. Specifically, I believe the following areas must be addressed:

1) Is an external radiation dose rate limit of 1 rem/h at 3 meters adequate? Under some accident conditions this level could result in substantial exposures to personnel, especially when the material is dispersed.

2) Are the limits for packaged and bulk liquids reasonable? For radionuclides with an A_2 value of less than 2×10^{-2} Ci the permissible specific activity for bulk liquids is less than the theshold for unregulated materials (.002 uCi/g)! Additionally, is it reasonable to apply limits that are based on tritium to other liquids even considering that the A_2 values are measures of radiological hazard?

3) Should bulk (unpackaged) shipments of LSA-II materials be allowed with radiation levels up to 1 rem/h at 3 meters? Additionally, how much material has to be considered in determining if the radiation level limit is met?

So the LSA concept is once again in the process of being refined. Some of the early considerations are still valid but others need to be added. Further work is being performed in the U.S. and hopefully elsewhere. This additional information, in conjunction with the proposed changes to the IAEA regulations (modified as necessary), should enable the concept to be modernized so that it rests again on a firm public safety foundation.

References

1/ Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, International Atomic Energy Agency, Vienna, 1973 Revised Edition (As Amended).

2/ The Safe Transport of Radioactive Materials, edited by R. Gibson, Permagon Press Ltd.

3/ Fairbairn, A., The Derivation of Maximum Permissible Levels of Radioactive Surface Contamination of Transport Containers and Vehicles, IAEA Safety Series No. 7, Vienna, 1961.

4/ Krebs, J. S., Potential Radiation Hazards of LSA Materials, Preliminary Draft, Contract NRC-02-78-071, March 21, 1980.

5/ Krebs, J. S., Potential Radiation Hazards of LSA Materials, Draft Interim Report, Contract NRC-02-78-071, July 18, 1980.

6/ Report of Working Group A.5, Working Paper No. 78, IAEA AG-266, Vienna, September 1980.

7/ Dicke, G. J., Assessment of Radiation Risks in the Case of Accidents With Truck Transportation of Tritiated Water in Drums, RMEP-IR-790-8-R, Ontario Hydro., Toronto, May 1980.

DISCUSSION

Question by G. Cohendy, France: How would the materials generated by the decommissioning of nuclear power plants be shipped under this proposed system?

Answer: There are a variety of different types of waste material that would be generated by the decommissioning of nuclear power plants. These would range from highly radioactive internal structural components to very slightly contaminated building rubble.

The slightly contaminated materials could, if the materials m[redacted] the appropriate requirements (such as the proposed LSA-I category) be shipped in bulk or in practically any package. More highly radioactive materialsthat still qualify as LSA or LCO could be shipped in the prescribed package which may be as high as a type A requirement.

Materials that are so radioactive that they do not meet the requirements for LSA and LCO would have to be shipped in type A or type B packages depending on the total activity contained in the package.

Question by B. Wright, United Kingdom: In the case of the proposed change for 'LL solids', what is the justification for assuming loss of shielding and specifying a doserate limit? This is unreasonable and will have the biggest effect, if any, of the changes proposed; for 80% of reactor waste is shipped as LLS and this would all be affected. Consideration must be given to the probability of the accident, and the possibilities of retaining the shielding by design, and not solely the consequences of an arbitrary "loss of shielding". If necessary, the LLS could justify special consideration on its own, because of the large amount of transport that will come under this category. I would ask the IAEA review committee to look at this again. Perhaps for LLS there should be a category between type A and type B for the package design.

Answer: The fact that wastes with high radiation levels are being shipped with increasing frequency is part of the reason that the concept of LSA and LLS needed a review and revision. It is not reasonable to allow materials with very high radiation hazards to be shipped in less stringent packaging than that required for type A quantities of materials which are limited in their radiation hazard (1 rem/h at 3 meters maximum). Additionally, the Advisory Group decided that in developing the requirements the commercial value of the material would not be considered since this has no bearing on the potential public hazard. These considerations were factored into the AG's deliberations and recommendations.

I agree that the proposed changes will have quite a pronounced effect on LLS materials that exhibit high radiation levels, but this is not necessarily bad. Since there is no assurance that any degree of shielding will be retained following an accident, the radiation hazard must be controlled in another manner. An external radiation level limit such as the one proposed is a direct and effective method which does not penalise lower hazard LSA (and LLS) materials.

I do not agree that in the absence of an assurance that the package will maintain its shielding integrity we can hide behind the low probability of its occurrence. The hazard of these materials must be more directly and surely controlled.

Your comments should be channelled through your competent authority for consideration by the next IAEA AG meeting on this.

Comment by W. Kolb, FR Germany: I fully agree with Mr. Wright that the changes proposed for the LSA section are the most significant ones in the new draft of the 1983 revised regulations.

For certain radionuclides specific activities as low as 10^{-6} A_2 r liquids in packages or even 10^{-7} A_2 for liquids in bulk are already below the definition level of 2 nCi/g. But these are the consequences of the application of the ICRP 26 concept as has already been decided.

Answer: I agree.

Comment by W. A. Brobst, USA: Mr. Wright is correct, and also Dr. Kolb. Many materials which can now be shipped under the LLS category will not be able to be shipped under the 1983 IAEA rules for LSA/LCO if adopted. Those materials are the ones which would have high radiation levels if the shielding around the materials is lost in an accident. This change was deliberately recommended by the IAEA AG in September to eliminate those high radiation materials from the LLS category. Interested people should read very closely the new IAEA proposals to determine the effect of this change and to prepare for it. Comments should be sent to the IAEA.

Answer: I agree except that comments to the IAEA should be directed through the appropriate competent authority.

DESIGN RULES FOR CONTAINMENT SYSTEMS FOR NUCLEAR SPENT FUEL AND HIGH-LEVEL WASTE TRANSPORT PACKAGINGS

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1. ORGANIZATION OF MAIN COMMITTEE

Within the past year an ASME (American Society of Mechanical Engineers) Committee on Containment Systems for Nuclear Spent Fuel and High-Level Waste Transport Packagingings has been formed for the purpose of developing, maintaining, and coordinating codes and standards for the construction* and inservice inspection requirements for these packagings. The formation of the Committee was precipitated by the nonexistence of a consistent set of rules for packaging containment systems. Such rules are required not only for guidance to the designer and fabricator but also for evaluation by the regulatory agencies.

The chronology for establishing this Committee was as follows:

- ASME Pressure Vessel & Piping Panel (PVP) Session in San Francisco, June 1979
- Two-Day Workshop, San Jose, September 1979 (ASME-PVP Ad Hoc Task Force)
- Presentation to ASME Nuclear Codes & Standards Committee, New York, November 1979
- Second Presentation to ASME Nuclear Codes & Standards Committee, New York,
- Creation of Main Committee, January-February 1980
- First Meeting, Main Committee, Dallas, March 1980

*Construction, as used herein, is an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection and certification required in the manufacture and installation of containment systems.

At the present time, the Committee reports to the oversight ASME Nuclear Codes and Standards Committee. It interfaces with Section III, Nuclear Power Plant Components and Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components of the ASME Boiler and Pressure Vessel (B&PV) Code, the Nuclear Quality Assurance Committee, and the Operations and Maintenance Committee.

The new committee, called NUPACK, has established five Task Groups charged with responsibility for:

1. General Requirements
2. Materials
3. Design
4. Fabrication, Examination and Testing, and
5. Inservice Requirements

These five Task Groups and the parent NUPACK Committee have balanced representation from the six interest groups -- owners/operators, designers/constructors, manufacturers, regulators, insurers/inspectors, and general interest.

One of the first actions of the NUPACK Committee was to establish that the rules of construction would be patterened after Section III of the B&PV Code. One of the distinguishing features of Section III are the requirements for a certified Design Specification and a certified Design Report. Subsection NA, General Requirements defines the contents of these documents. It also assigns the responsibilit for providing the Design Specification to the Owner and the Design Report to the N Certificate Holder. It is fully intended to retain this concept of separation of responsibilities for NUPACK, although the definitions of Owner will have to be revised for transport packagings.

2. TASK GROUP ON DESIGN

The Task Group on Design has a leading role and the remainder of this paper is restricted to design aspects. The Task Group has developed interfaces with two existing counterparts within the Section III Committee structure -- the Subgroup on Design and the Subgroup on Materials -- and is attempting to write rules that can be accommodated in some way by the present ASME B&PV Code format. A number of options have been discussed, including one extreme that would involve modifications to one of the existing Subsections in Section III and another extreme that might entail a completely separate document. This document could be a stand-alone Subsection representing a distinct safety classification, such as those covering nuclear power plant metal containments (Subsection NE) and supports (Section NF), or an intact Code Case that would be non-mandatory for a probationary period.

Several different Subsections are potential candidates for direct incorporation into the existing Code. In order to determine which of the top candidates should be pursued, the deliberations to date have concentrated on design philosophy -- that is, the balance between design by formula (such as that contained in NC-3300), design by analysis (such as that contained in NB and NC-3200), and design by experiment.

Design philosophy is an important consideration because extreme opinions tend to divide proponents of each of these three approaches. However, design by formula can be virtually eliminated as a candi[redacted] Those sections of the ASME B&PV Code that rely on design by formu[redacted] are dominated by concerns about pressure stress, which are not of primary importance in shipping container design. Containment systems for transport packagings are typically sized for shell or head thicknesses by a combination of shielding requirements and faulted events, such as the 9-meter drop or puncture. Therefore, design by analysis is the more probable choice, with consideration given to design by experiment for extreme events. The uncertainties and the expense associated with dynamic, elastic/plastic, large deformation finite element analysis -- especially for three-dimensional geometric models -- tend to reinforce the design by experiment position.

3. DESIGN CRITERIA

In order to provide some guidance on the acceptability of design criteria by the U.S. Nuclear Regulatory Commission (NRC), it has issued Regulatory Guide 7.6, Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels. This document relies substantially on the terminology of the ASME B&PV Code. It invokes the design-by-analysis approach, but is not recognized by ASME and falls well short of being comprehensive enough to cover all construction aspects of containment systems for transport packagings.

The ASME Code classifies extreme events similar to those that govern the shipping container design as Level D loadings; i.e., the 9-meter drop is a loading event of such severity that "gross general deformations, loss of dimensional stability, and damage requiring repair" may be observed. However, such an event must not compromise the containment function. If the NC-3200 rules in existence today were adopted, a provision for an experimental stress analysis is made in NC-3211.1(c), as an alternative to a complete stress analysis. This provision is reinforced by allowances for experimental stress analysis in a non-mandatory Appendix F (Rules for Evaluation of Service Loadings with Level D Service Limits), paragraph F1321.3, in spite of the footnote in Table NC-3217.1 that says "When a comp[redacted]e analysis is performed in accordance with NC-3211.1(c), the stress[redacted] limits of Appendix F may be applied". This footnote should be interpreted to mean <u>either</u> a complete experimental or numerical stress analysis.

Appendix II (Experimental Stress Analysis) provides the rules for conducting both cyclic and non-cyclic tests, but does not cover the principal concern in the Level D event -- loss of containment. Basically, the non-cyclic tests are designed to measure collapse load, plastic instability load, or stress/strain limits. Allowable instrumentation (e.g., strain gauges, photoelastic models) does not directly address leak measurements during or following the Level D event. Therefore, the Task Group on Design of NUPACK will consider a modification to Appendix II to permit direct measurement of a functional (as opposed to a structural) failure mode.

4. MATERIALS

Major area of concern is the fracture toughness and material acceptance rules for ferritic steels. The existing rules for Class 2 equipment (NC-2300) permit exceptions from impact testing and flaw evaluation by presumably guaranteeing upper shelf (ductile) behavior. A thickness-dependent temperature "margin of safety" is permitted for the ferritic steels listed in Table NC-2311(a)-1, and no impact testing is required for section thicknesses less than 5/8 in. (16 mm) or for lowest service temperature (LST) greater than 150F (66 C). When the exemption is not justified, two material acceptance paths are available, depending upon whether the section size exceeds 2-1/2 in. (64 mm), drop weight testing is required in order to find a reference transition temperature, T_{NDT}. A thickness-dependent upper shelf temperature margin is then prescribed between the lowest service temperature and T_{NDT}. Linear elastic fracture mechanics (LEFM), as contained in Appendix G, are offered as an alternative. Unfortunately, the lowest service temperature for shipping containers must take into account extreme climatic conditions during transport, with a value of -40F (-40C) a potential lower bound. Few ferritic steels are available which would permit an upper shelf temperature margin of even the 30 F (17 C) required for a 2-1/2 in. (64 mm) section. Also, the Appendix G rules explicitly avoid addressing Level C and D events, which means that an LEFM flaw assessment for a 9-meter drop would have to be justified on a case-by-case basis.

The Task Group on Design is examining this issue with care, and is considering the possibility of an elastic-plastic fracture mechanics (EPFM) approach that blends transition region evaluation into upper shelf response.

Other areas that are currently being studied are the S_m material allowables in Table I-1 of Appendix I, especially for existing cask materials and for operation at temperatures below 100 F (38 C); the strain rate effects that are permitted in Appendix F; and possible revisions to the buckling rules.

5. CONCLUSION

The need for establishing consistent rules for the construction of containment systems for irradiated fuel and high level waste transport packagings was recognized and discussed at a Panel and at a Work Shop approximately one year ago. The ASME Codes and Standard bodies were then approached with a request to develop such rules under their auspices. Upon acceptance the NUPACK Committee was formed earlier this year reporting to the ASME Nuclear Codes and Standards Committee.

The NUPACK Committee commands a high priority within ASME and is well staffed with enthusiastic, knowledgeable individuals representing a balance of different interest groups. They understand the need for developing these rules and recognize the problems they face. Although it is too early to predict a completion date, a goal has been set of completing a first draft in approximately two years.

DISCUSSION

Question by G. Malkoske, Canada: The rules being developed by 'NUPACK' will apply to containers that will be used for international transportation. Do you see any difficulty in harmonisation between the 'NUPACK' rules and the US NRC regulations, which are based on international regulations?

Answer: I see no difficulty in harmonisation. The codes and standards which will be developed by'NUPACK'shall be in full compliance with the US Federal Code of Regulations 10 CFR Part 71 which, in turn, incorporates the international IAEA Regulations for the safe transport of nuclear materials. Representatives of the US Nuclear Regulatory Commission are members of the 'NUPACK' Main Committee and Task Groups.

TRANSPORT OF RADIOACTIVE MATERIALS BY POST. A NEW APPROACH FOR RULE-MAKING.

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Background.

Earier, the transport of radioactive materials by post has not been allowed in Denmark, Finland and Norway. In Sweden transport by post has been allowed and the IAEA Regulations have been used in practice. The demand for this mode of transport of radioactive materials has increased, mainly due to the increased use of these materials in consumer products and for in-vitro use in hospitals. Therefore, in 1978 the competent authorities in the Nordic countries formed a workinggroup with the task of identifying problems relating to this mode of transport and to work out a proposal for regulations for the transport of radioactive materials by post. This proposal was intended to form the starting point for negotiations with postal and other pertinent transport authorities and organizations aiming at the preparation of national and possibly common regulations within the Nordic countries. In 1979, regulations based on the proposal was implemented as national regulations in Norway.

Consumer products containing radioactive materials approved by the authorities can normally be sold and used freely in the individual countries. A substantial number of smoke detectors and watches have been distributed. Restrictions as to the transport of devices that can be bought and handled freely by members of the public, does not seem reasonably. The large amount of RIA kits containing small amounts of radioactive materials used today in clinic laboratories is another materialgroup where transport by post would be appropiate. It is felt better to have reasonably rules which users will comply with rather than having illegitimate transports of radioactive materials by post.

Allowing the transport of these items by post permits a more effective and rational way of distributing these materials. The handling by the postal services of these radioactive consignements should not be different from the handling of other post items. In order to comply with this requirement the element of risk should be demonstrated to be so low that the radioactive materials are non-dangerous even in non-packed conditions and even if the package

is severely damaged. In other words the level of risk should be demonstrated to be so very low that no special precautions such as labelling of the outside of parcels, controls for transport and storage in transit, education and information of postworkers etc. is deemed necessary. Therefore, acceptance from the postworkers to participate in the carriage of these radioactive materials should be possible.

The activity limits for transport by post specified in the IAEA "Regulations for Safe Transport of Radioactive Materials, 1973 Edition" appears to be quite high regarding this "absolute" safety demand, e.g. the activity limit for iodine-125 in liquid form is 700 microcuries i.e. 25 times the ALI value given by ICRP.

By lowering the activity limit for the transport of radioactive materials by post to the ALI value the group considers that one can demonstrate that the element of risk is sufficiently low. For instance a post parcel with this limited activity can not - whatever happens- result in a radiation dose to anyone which exceeds the annual dose limit for a radiation worker. Higher activities could be allowed for sources in consumer products approved by the authorities.

Based on these consideration the group has formulated a proposal for the transport of radioactive materials by post. The principal contents of the proposal can be summarized as follows:

Proposal on regulations for the transport of radioactive materials by post.

1. Permitted radioactive content.

a. no restrictions applies for:

- naturally occuring radioactive materials
- radioactive material having a specific activity of less than 0,002 microcuries per gram
- approved consumer products containing radioactive material.

b. the total activity per package shall not exceed the most restrictive ALI value for a single radionuclide.

in case of a mixture of different radionuclides the following conditions shall be fulfilled

$$\sum_{i=1}^{N} A_i/ALI_i \leq 1$$

where N = number of radionuclides

A_1 = activity of the ith nuclide

ALI_1 = the ALI value of the ith nuclide

2. External radiation.

The radiation level originating from the package shall not at any time during normal transport exceed 0,1 mrem/h at any point on the external surface of the package.

3. Packaging design requirements.

The packages shall be able to withstand the normal conditions of transport. Packages for liquids shall be provided with sufficient absorbent material to safely absorb at least twice the volume of the liquid contents.

4. Labelling of packages.

Packages shall not be labelled on the outside but a warning of the presence of radioactive material shall be visible on opening the package. The radioactive content should possibly be declared in the transport documents.

The implementation of the described proposal can be done either in form of general regulations specifying activity limits, packaging requirements etc. or by licensing each distributor of radioactive materials. In the latter case applications should be sent to the postal authority, which also issues the licences after consulting the competent transport authority.

DISCUSSION

Question by R. Neider, FR Germany: You gave two kinds of limits for the content of radioactive materials in packages sent by post. Do these limits apply alternatively or do both conditions apply together in all cases?

Answer: The two kinds of limits apply alternatively, e.g. an approved consumer product can be sent by post even if the activity content exceeds the appropriate ALI value. For all materials, which are not mentioned under 1.a, the activity is limited by the ALI value.

A MODAL STUDY OF TRANSPORTATION SAFETY

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U.S. Nuclear Regulatory Commission

The U.S. Nuclear Regulatory Commission is conducting a research program to identify and evaluate alternatives to current transport regulations. The principal objective of the program is to determine whether a set of mode-dependent transportation regulations can be developed which both represent, and are perceived to represent, an improved and more comprehensive technical basis for assuring continued protection of public health and safety. Specifically, the focus of this study is directed at those shipments of radioactive material in which potentially high consequences could result from failure of packages under extremely severe, but credible, transportation accidents. This program has been given the title of the "Transportation Modal Study."

BACKGROUND

Four and one-half years ago, the U.S. NRC decided to reevaluate its regulations governing air transport of radioactive materials. A course was embarked upon which, in December of 1977, led to issuance of an environmental appraisal of the impacts of essentially all radioactive transport activities within the United States (Ref. 1). The appraisal included an assessment of the risks presented to the public by the transport of all radioactive materials. The results were not too surprising. During normal transport activities, the statistically expected health effects resulting from the extremely small exposures of transport workers and the general public were calculated from the transportation of radioactive material in the United States during 1975, as 1.7 genetic effects and 1.2 latent cancer fatalities. From exposures due to potential transportation accidents, expected effects were several hundred times smaller than these calculated normal values. On this basis the NRC concluded that "the environmental impacts of normal transportation and the risks attendant to accidents are sufficiently small to allow continued shipment by all transportation modes and no immediate changes to the regulations are needed." Yet, even with this study, the public and, for that matter, many in the nuclear community, do not appear content with the current state-of-affairs. The reason for this uneasiness is related to three apparent deficiencies in our current regulations as shown in figure 1. First, the current performance tests and package acceptance criteria have not been shown to have a strong technical connection to transport accidents. Second, the current performance tests are not perceived as being representative of real-world accident conditions. Finally, the current regulations for type B packages do not give special consideration to potentially high consequence events which could be postulated to result from low probability but severe transport accidents. To be useful, the Modal Study must attempt to address each of these deficiencies.

MODAL STUDY - PERFORMANCE TEST DEFINITION

The initial task in the Modal Study is to establish a set of performance tests which reflect credible accident environments appropriate to each transport mode. To begin work on this task the NRC has initiated a commercial contract to accomplish this objective for the road and rail transport modes.

FIGURE 1

THREE DEFICIENCIES OF CURRENT REGULATORY PERFORMANCE TESTS

- No strong technical connection to actual accident conditions
- Not perceptually representative of extremely severe real world accidents
- Potential high consequence—low probability transportation accident events not given any special (formalized) regulatory consideration

The first step in this contract is to document a representative history of severe accident environments in the United States. All types of accident forces and loadings--mechanical, thermal, chemical, etc.--will be included. These historical accident environments, which are generally described in only a qualitative fashion, would then be quantified in terms of forces or loadings which could be transmitted to a RAM package. This process will consider reasonable extensions of historical data to develop accident scenarios which take into account the package interfaces with combinations of real-world conditions appropriate to a particular transport mode (e.g., bridge heights, impact surfaces). Finally, these forces or loadings will be translated into a representative set of package performance tests. Certainly if NRC is successful in accomplishing this task for road and rail transport, and future efforts are able to do the same for the other transport modes, the resulting performance tests will have a strong technical connection to severe accident conditions. To assure that the relationship of the performance tests to real-world accidents is perceptually appreciated, the outputs anticipated from these initial tasks will include a fully documented report describing how the process of selecting representative accident environments was carried out. Pictorial and illustrative examples of historical accident environments, such as the example shown (Figure 2), will depict how certain classes of forces associated with severe accidents were bounded and quantified. The major difficulty in these initial studies, however, will be to define a set of performance tests which will properly recreate the mechanical and thermal loading enviornments experienced by a RAM package if it was involved in an extremely severe transportation accident. Some desired attributes of any selected test set will include: (1) the practicality of subjecting RAM packages and contents to such tests, (2) the capability to analytically predict the significant parameters of package response, and (3) as has been emphasized previously, the perception of the tests as being representative of extremely severe accident environments.

PACKAGE ACCEPTANCE STANDARDS

This discussion of performance tests represents only one of two equally important tasks within the Modal Study. The second task involves first estimating the effects of these performance tests on packages; that is, what if any, release of RAM can be expected and second, defining what constitutes package acceptance criteria that are reasonable and practical.

To address this second task, the NRC intends to define, construct, and test several prototype or scale models of existing packages with simulated contents. The severity of the tests could cause extensive structural damage to the package and its contents. However, the principal measure of package performance will be the magnitude of any release of RAM. At this point, a range for lower bound acceptance values for RAM release would be specified and the possibility of making minor package design changes to meet this acceptance value or to further minimize the release would be examined. This process is illustrated in figure 3 for a hypothetical package labeled A.

If however, package testing indicates that the release of radioactive material exceeds this specified lower bound acceptance value (even after considering minor package design modifications), other possible alternatives to minimize material release would be evaluated. These alternatives would include: (1) modification of content (RAM) form, (2) consideration of more extensive package design modifications, and (3) consideration of operational or administrative controls on transport to minimize the severity of the environments to which the package could conceivably be exposed. This process is illustrated in figure 4.

FIGURE 2 **THE EXTREMELY SEVERE PUNCTURE/CRUSH ENVIRONMENT?**

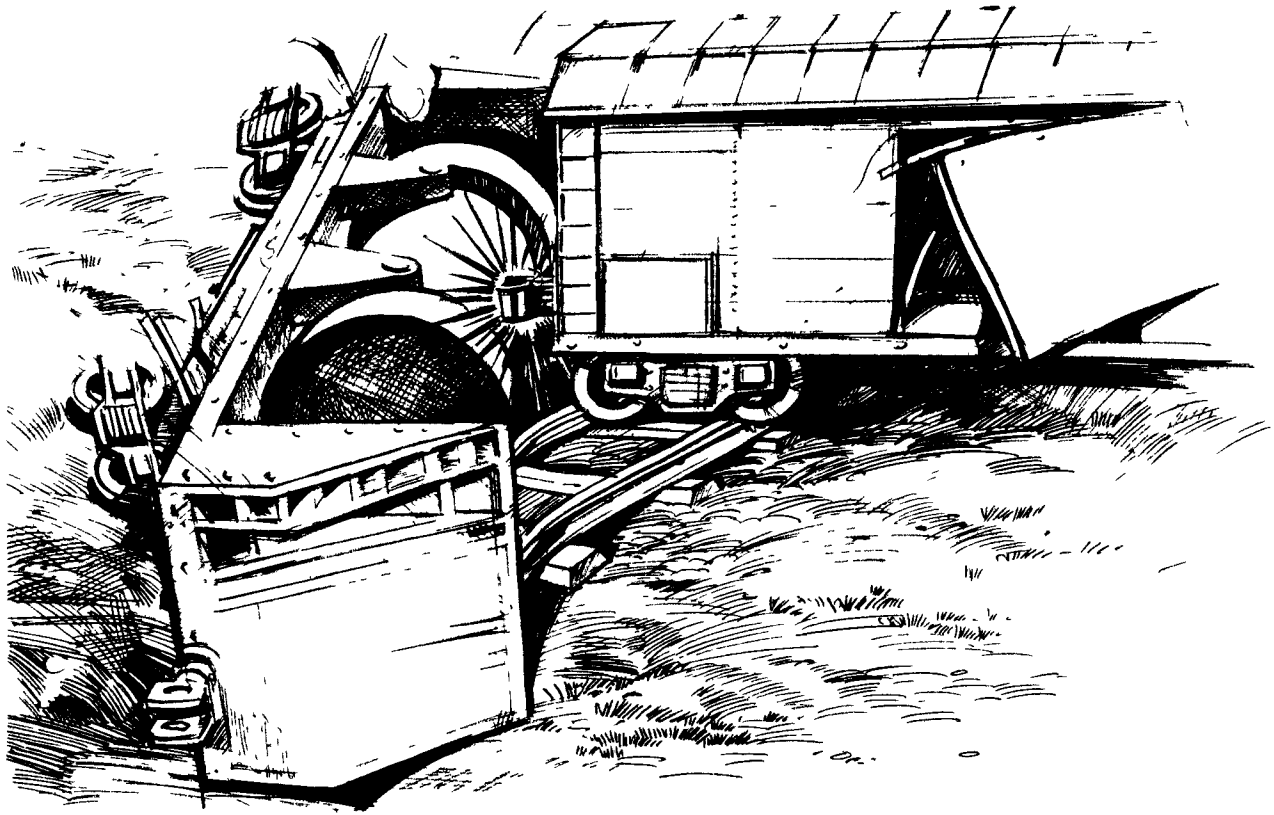


FIGURE 3

EVALUATION LOGIC FOR TRANSPORT REQUIREMENTS

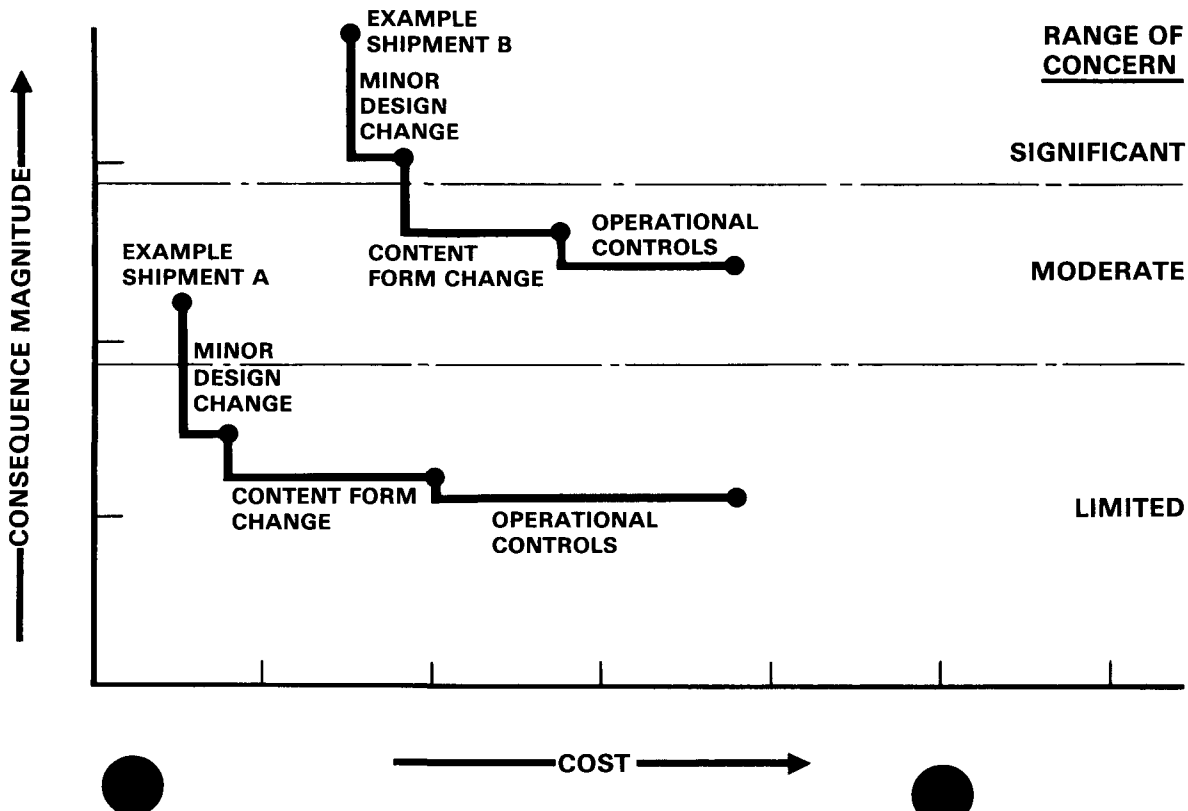
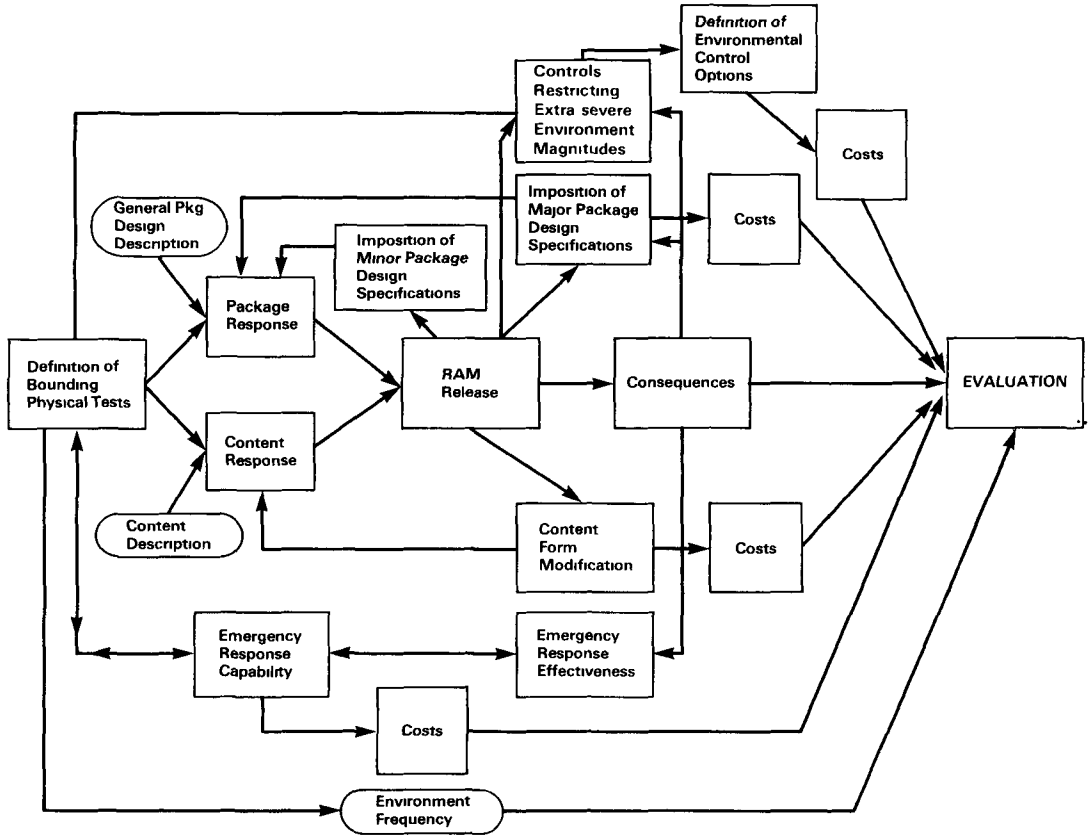


FIGURE 4

MODAL STUDY PROGRAM PLAN



To assess the effectiveness of any of these alternatives, the costs of such actions must be weighed against the benefits provided to public health and safety. Health effects and economic impacts are obviously expected to be the major measures of effectiveness. Since the magnitude of potential consequences from severe accident events is expected to influence regulatory judgments, their evaluation requires that specific populations and land areas must be assumed. In addition, the significance of certain extreme environments postulated for this study may also be related to population density or specific geographic location. These interrelationships give rise to the indicated feedback loop in the value-impact assessment process. Yet, the most difficult task in this process will involve the development of the algorithm relating consequence magnitude, risk, or a combination of these factors to the costs of implementing any of the alternatives which could reduce both consequence magnitude and risk. Our current thinking as shown in figure 3, pictures a lower bound on the consequence magnitude scale below which a linear risk-cost tradeoff would represent the proper regulatory posture. As the potential consequence levels increase from this bound, the value of any fixed increment of consequence reduction would increase in a non-linear fashion. It should be noted that the coupling of stringent package performance tests with more realistic and practical acceptance standards may not have a significant impact on package design practice as it currently exists.

SCHEDULE AND PROGRESS

The initial Modal Study task, the definition of the bounding physical tests for the rail and truck transport modes, is underway, and should be completed in 1981. Similar investigations regarding other modes will be begun late this year or in early 1981. The procedural processes for soliciting a contractor to conduct the testing tasks required in the Study have just begun. Initial test activities could occur in late 1981. The schedules beyond this point are extremely tentative. Best estimates are that the first value-impact appraisals will be initiated in the latter part of 1982. Other supporting studies would closely follow with completion of the Modal Study anticipated in 1984-85.

REFERENCES

1. Final Environmental Statement On the Transportation of Radioactive Material by Air and Other Modes, NUREG-0170, December 1977.

DISCUSSION

Question by E. P. Goldfinch, United Kingdom: Will the studies include an examination of accidents of severity between the so-called median accident and very severe accidents, to type A packages, in other words accidents such as small fires which can disperse 100% of the contents and lead to very high doses. My concern is that as well as examining low probability/high consequence accidents, high possibility/high consequence accidents should be included.

Answer: This study will concentrate principally on examination and documentation of a representative history of severe accidents and "high hazard" situations, i.e. those of low probability but high consequence. In general, type A package situations will not be considered in the study.

Answer by P. E. Eggers of Ridihalgh, Eggers and Associates, USA: The scope of the study entitled "Definition of Bounding Physical Tests Representative of Transport Accidents - Rail and Truck" being performed by Ridihalgh, Eggers and Associates for the US Nuclear Regulatory Commission is directed at the study of severe highway and rail accidents involving type B and fissile material packages and does not currently include type A packages. However, the results of this study - the forces and thermal loads associated with severe accidents - can certainly be applied subsequently to type A or other packages.

SESSION II
LMFBR SYSTEMS CONCEPTS

DESIGN OF A TRANSPORT PACKAGING FOR PLUTONIUM CONTAINING UNIRRADIATED FBR FUEL ELEMENTS

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INTRODUCTION

This paper refers to the study design and first states of testing of a packaging for the transport of the unirradiated plutonium containing fuel elements to be used in the PEC reactor.

This plant, a FBR experimental facility a-building since several years in Italy, will have a maximum power output of 120 MWt and was designed specifically for testing fuel elements of unitary power up to 3 MWt.

The packaging in consideration has been studied jointly by CNEN-RAD Division of Casaccia and Ist. Impianti Nucleari of Pisa University in order to meet the IAEA safety standards as well as to assure the geometrical and functional integrity of the fuel elements during the transport and handling normal operations.

MAIN FEATURES OF THE PACKAGING

The PEC fuel element has a lenght of 3000 mm and an exagonal section with opposite sides distance of about 86 mm. The packaging which contains only one element, is intended to be transported in a 9 unity rack assembled on a commercial truck specifically fitted up for the particular use.

The packaging (Fig. 1) is formed by two separate parts
1) an internal containement bottle,
2) an overpack with a central cavity for the bottle.

The fuel element is inserted and mantained in position inside the containement bottle by an elastic support system (actually a pneumatic one) This bottle is a cilindrical stainless steel vessel, with double ORing seal ard a key closure on each base.

The bottle itself is designed in order to assure the containement of the fissile material in normal and accidental transport conditions as well as the protection of the fuel element during several handling operations in the plant.

The overpack, with a cilindrical external surface of 530 mm in

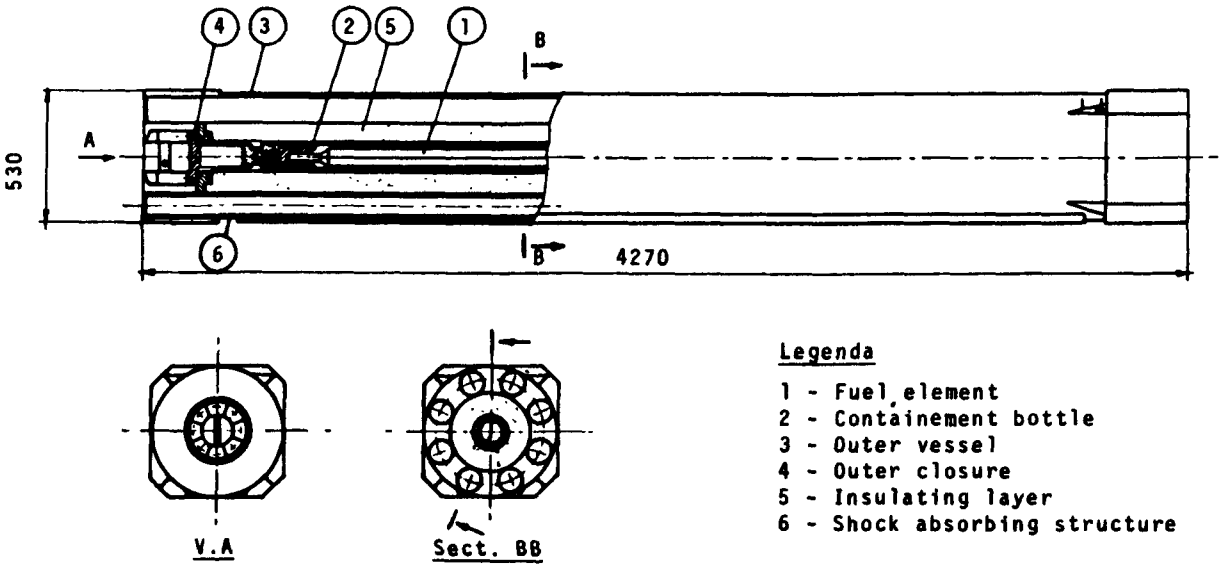

Fig. 1 - PEC unirradiated fuel element transport packaging prototype.

diameter and about 4300 mm of lenght, contains two concentric anular cavities delimited by three steel cilinders. The inner cilinder has shock absorbers equipped closures on the bases for the protection of the containment bottle.

Nevertheless the main shock absorbing device is constitued by 8 tubes with a diameter of 100 mm inserted in the outer anular cavity. These tubes estend over the closures on the bases and the whole lenght of the packaging, with the aim to reduce the mechanical effects of accidental drops of the packaging itself in axial and lateral directions.

The inner anular cavity of the overpack contains a light vermiculite conglomerate layer which, beside a possible further smoothing of the dynamic loads, assure the thermal insulation of the bottle and its seal systems in accident conditions and is suitable for possible shielding use.

The package, made out of commercially available raw materials requiring relatively simple manufacturing operations (mainly weldings), has an overall weight (fuel element included) under 1000 kg.

DESIGN ORIENTED STUDY PROGRAM

The packaging study included the design of the structure in the relevant mechanical and thermal IAEA accident conditions, and the calculation of the shielding and criticallity for the stack of 9 unities in the most dangerous envisaged configuration.

As regards to the tube shock absorbing devices used to protect the container against the effects of axial and lateral dynamic loads, a preliminary design was carried out on the basis of the available bibliographical data, e.g. /1,2,3,4/.

These data were considered fulfilling for the design of the axial tube shock absorber, as they included also tests results pertaining to the same material type and dimension range preliminarily envisaged for the container.

On the contrary in order to define more precisely the behaviour of the relevant components of the shock absorber under lateral dynamic loads, a test program was performed by means of the drop facility of the Impianti Nucleari Institute of Pisa University.

This program included drop tests with a guided hammer on tube sections during which the dynamic loads on the sections bearing plate as well as the accelleration of the hammer were recorded by means of a suitable instrumentation.

The experimental data attained resulted to be in a good agreement (in terms of load for length unity versus the radial deformations (Fig. 2)) with theoretical calculations specifically developed in the program taking in account of the strain-rate and strain-hardening effects on the material behaviour, for a four plastic hinges de-

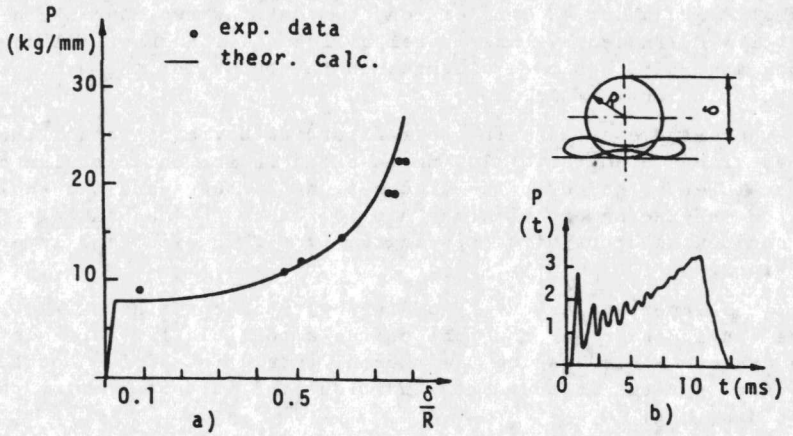

Fig. 2 a) Load versus deformation for tube under impact load normal to the axis.
b) Load versus time curve registered in an actual shock test on a tube section of 100x3 mm.

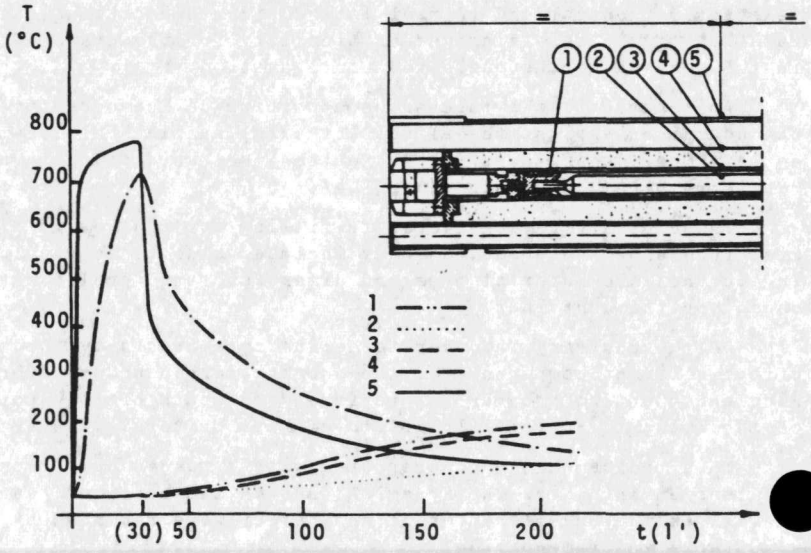

Fig. 3 - Thermal behaviour of the package in the thermal test (calculated).

formation model of the tube sections.

Another test serie of the same type was carried out in order to define and characterize from a mechanical point of view the insulating conglomerate.

The thermal behaviour of the package was calculated by means of finite difference and finite elements methods based codes /5,6/.

In Fig. 3 the temperature versus time transients are reported, as calculated with the HEATING 3 code, in several internal points of the packaging supposed in the conditions resulting after a 9 m drop on the base. The low levels of the temperature in the bottle closure [redacted]ams to confirm the possibility of the use of silicone elastomer OR [redacted]ls.

Good results from a safety point of view were also attained (as was possible to foresee) in criticallity calculations performed for the stack of nine damaged containers with the KENO 4 code /7/.

CONCLUSIONS

The attained results seem to justify up to now the reliability of the packaging general design.

Anyway these results have to be confirmed by the envisaged experimental program to be carried out on full scale prototypes.

The first prototype was already manufactured and is attually under test (loaded with an inert moke up of the PEC fuel element) in order to acquire further data for a possible optimization of the design as well as for the eventual approval of the Competent Authority.

REFERENCES

/1/ J.A. De Runts & P.G. Hodge
CRUSHING OF A TUBE BETWEEN RIGID PLATES
Journ. Appl. Mech. 30-391-395 (1963)

/2/ W. Johnson S.R. Reid T.Y. Reddy
THE COMPRESSION OF CROSS-LAYERS OF THIN TUBES
Int. Journ. Mech. Sci. 19 423 (1977)

/3/ R.J. Burian & E.C. Lusk
COMPACT METALLIC IMPACT LIMITERS FOR SHIPPING CONTAINERS
Third International Symposium Packaging and Transportation of Radioactive materials - Conf-710801 (Vol. 2)
Richland - USA August 16-20/1971

/4/ S. Barsali & G. Forasassi
STRUTTURE DI PROTEZIONE E SICUREZZA PER UN CONTENITORE PER IL TRASPORTO DI ELEMENTI DI COMBUSTIBILE IRRAGGIATO
Atti Istituto Impianti Nucleari Università degli Studi di Pisa
RL 326(70)

/5/ W.D. Turner & M. Simantov
HEATING 3 AN IBM 360 HEAT CONDUCTION PROGRAM
ORNL TM 3208 - 2/1971

/6/ Collier
THREAT A 3 D HEAT TRANSFER PROGRAM
UKAEA TRG 2684(R)

/7/ L.M. Petrie & N.F. Cross
KENO IV AN IMPROVED MONTE CARLO CRITICALITY PROGRAM ORNL 4938
UC-32 Math. & Computers 11/1975.

DISCUSSION

Question by R. Portal, France: Don't you have any problem of radiation shielding? This problem is related to the nature (isotopic composition) and age of the employed plutonium.

Answer: The calculations performed up to now for the intended fuel have shown a radiation dose outside the container within the IAEA limits.

Moreover, I would like to stress that the container is intended for transportation of one fuel element and that it is possible to improve its shielding capability, if needed, both by adding suitable materials to the inner conglomerate and increasing, within certain limits, the thickness of its various steel layers.

DESIGN OF A CONTAINER FOR FRESH LMFBR FUEL ASSEMBLIES TYPE SNR 300

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Introduction

For the shipment of non irradiated plutonium fuel assemblies type SNR 300 Transnuklear has designed a special container with a capacity of 6 - 7 fuel assemblies. SNR 300 is a 300 MW prototype LMFBR under construction at Kalkar, Germany.

2. Fuel assembly data

The fuel elements are typical for LMFBR's and comprise an array of stainless steel clad fuel pins surrounded by a hexagonal shroud tube. Main data are given in table 1.

Table 1: SNR 300 fuel assembly data
(MKIa version for first core)

number of pins	169
fuel zone length	950 mm
total length	3 700 mm
cross section	110 mm across flats
type of fuel	UO_2/PuO_2
weight of Pu	9.5 kg
Pu isotopic composition	
Pu-238	0.5 w/o
Pu-239	70.0 w/o
Pu-240	24.5 w/o
Pu-241	4.5 w/o
Am-241	0.5 w/o
activity	1.3×10^6 Ci
decay heat	63 W
neutron source strength	2×10^7 1/sec

It can be seen that special attention has to be given to heat transfer and neutron shielding.

3. Design requirements

In view of the container handling facilities and the transfer schedule at the power station the container should have the maximum possible capacity.

- loading has to be performed in horizontal position
- unloading has to be performed in vertical position. All handling operations have to be performed from the top of the container.
- a maximum bending tolerance of 1.3 mm for the assembly during transport is accepted.
- assembly accelerations from normal transport conditions should not exceed 3 g in lateral and axial direction.

4. Package design description

Taking into account all of a.m. requirements and additionally the IAEA regulations of 1973 the following detailed type B (U) design can fulfill these conditions.

The container has a maximum capacity of 7 assemblies. The loading and unloading can both be performed as well in horizontal as in vertical position. In consideration of dimensional accuracy the assemblies will be fixed in longitudinal axis by spring tension and in lateral axis by special formed packing rings. The packing rings can be constructed adjustable, too. Then the capacity of the container amounts to only 6 assemblies.

Build up of container body:

- hollow cylindric double walled closure by bolted flange
- neutron shielding as well as heat insulating between the double wall
- fixed or removable shock absorber at cylindrical sphere of container
- removable shock absorber at the lid and bottom
- support plugs integrated at the shock absorber as cylindrical sphere.

Build up of container basket:

- fixed construction of up to 7 tubes for holding the assemblies
- the basket is fixed by springs which can absorb an acceleration of 3 g (vertical and horizontal)
- the basket will only be removed from container body in case of repairs or maintenance
- the tubes of basket have only one combined lid
- each position should be fixed by adjustable springs.

Weight of container:

container body with basket	6 000 kg
shockabsorber	760 kg
grossweight loaded with 7 assemblies	7 530 kg

5. Results of criticality safety analysis, shielding and heat transfer calculations

According to criticality calculations the envisaged number of 7 assemblies per container is allowed. Furthermore the number

of containers per shipment is not limited.

The results of shielding calculations are showing that the dose rate due to gamma-radiation is negligible. The expected neutron dose rate during transport is shown in table 2.

Table 2: Maximum expected dose rate during transport (dose rate mrem/h)

surface of the cask	80
1 m distance from the cask	12
2 m distance from the cask	4

The cask is designed to dissipate 500 Watts to the ambient taking into account the IAEA-regulations and the a.m. conditions. The maximum expected temperature of the outer surface will be about 65 °C. The result of the safety analysis shows that the dose rates and temperatures at the cask are well within the limits given by the regulations.

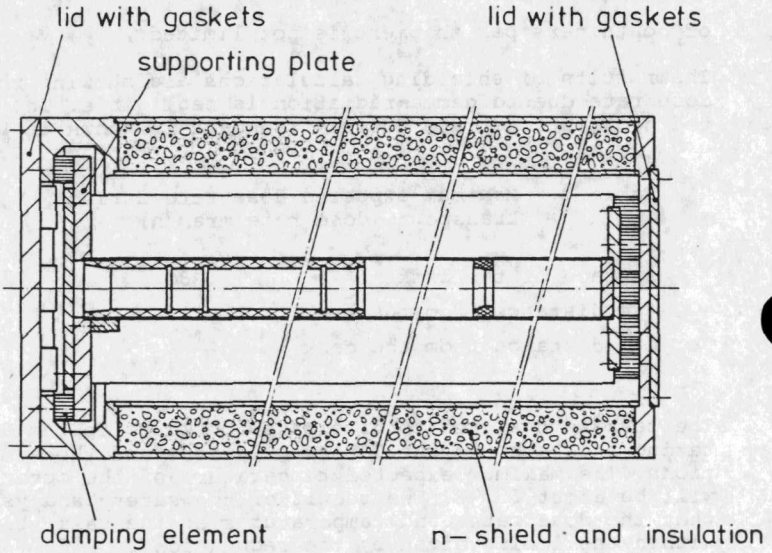
6. Future aspects

As it has been shown a high capacity transport cask for fresh Pu-fuel has in principle the same characteristics as the well known containers for irradiated spent fuel i.e.:

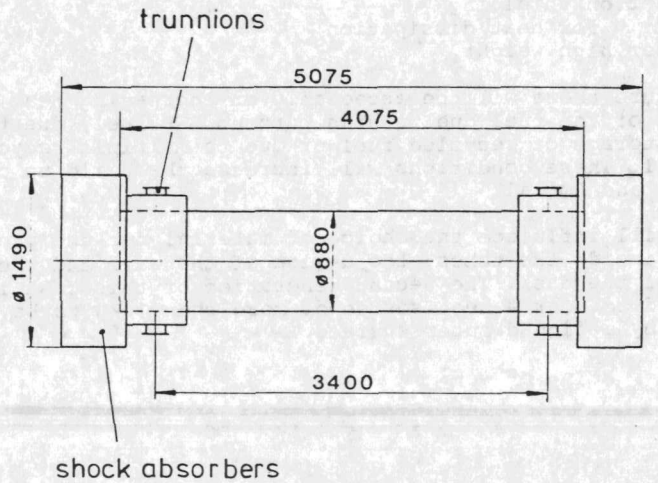
- type B
- radiation shielding
- means for heat dissipation
- rather high weight.

In future there will be especially an increasing heat generation of the fuel due to high burn-up LWR fuel, due to long term storage of recycled fuel or due to multiple recycling of fuel. These conditions will increase the build up of Pu 238 and Am 241.

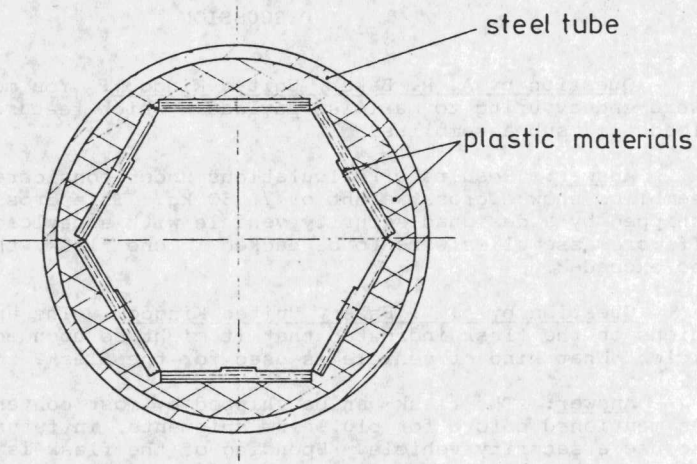
This will influence the choice of material for the packing rings inside the tubes, the design of the wall and the surface of the cask. The second generation of this cask is envisaged to bear a heat source of more than 1 kW to be dissipated by a finned outer surface.



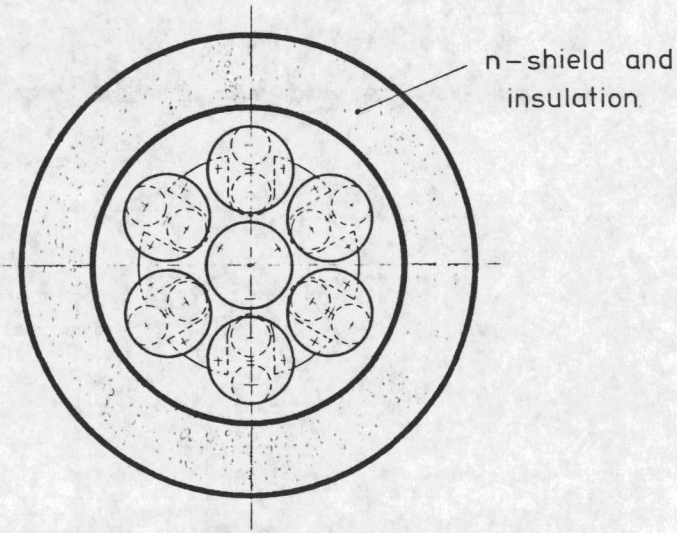
container section



overall dimensions



support of the assemblies



container cross section

DISCUSSION

Question by A. R. Baker, United Kingdom: You mentioned that you were endeavouring to maximise payload. Which feature limited the payload to 7 sub-assemblies?

Answer: Results of calculations under consideration for 7 assemblies show a gross weight of 7,530 kg. This gross weight can be shipped by a designed security vehicle with a payload limit of 8 t. If more assemblies were to be packed in one flask, this limit would be exceeded.

Question by A. R. Baker, United Kingdom: The appearance of trunions on the flask indicated that it might be upended on an open vehicle. What kind of vehicle is used for the flask?

Answer: The flask can be shipped by most conventional trucks. As mentioned before for plutonium shipments, in future we will have to use a security vehicle. Upending of the flask is performed outside of the vehicle.

Question by G. Arendasy, Austria: Why is neutron shielding in the lid and base regions very limited?

Answer: The length of the fuel zone of the assemblies is only 950 mm. Furthermore the assembly has a breeder zone with a length of 400 mm on both sides of the fuel zone.

T-3, A PACKAGING AND TRANSPORTATION SYSTEM FOR IRRADIATED US BREEDER REACTOR EXPERIMENTS

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Department of Energy

EXPERIENCE

The first major irradiation testing of fast reactor fuels and materials in the United States (US) started in the late 60's and early 70's at the Experimental Breeder Reactor II (EBR-II) and Transient Reactor Test (TREAT) Facility in southeastern Idaho. The irradiation testing program utilized hot cell facilities at the Hanford Engineering Development Laboratory (HEDL) in eastern Washington, the Los Alamos Scientific Laboratory (LASL) in northern New Mexico as well as the Hot Fuel Examination Facility (HFEF) adjacent to EBR-II and TREAT at Argonne National Laboratory West (ANL-W) (Figure 1). The substantial distances between these hot cells required

Fig.1 MAP OF WESTERN UNITED STATES

a packaging and transportation system for the experimental payloads ensuring experimenters that changes in the specimens were irradiation induced and not the result of packaging and transportation. The majority of the early irradiated breeder reactor experiments were transported to hot cells in one of two T-2 casks.

T-2 CASK AND INTERNAL PACKAGE

The first of two T-2 casks was fabricated by General Electric Company in 1968 to support shipment of transient test experiments to and from the TREAT Facility in Idaho. The second T-2 was fabricated under the cognizance of E. I. duPont de Nemours and Company, Incorporated in 1972. At that time both casks were assigned to ANL to support the US Breeder Reactor Program being conducted primarily in the TREAT and EBR-II facilities. The casks were licensed by the US Nuclear Regulatory Commission (NRC) as part of their first cask licensing activity. The first license was somewhat restrictive allowing only 800 grams of fissionable material and 40 watts of decay heat.

A relicensing effort for the T-2 design was initiated to increase the cask payload capability and to license the casks to the new 10-CFR-71, Appendix E. This effort was successfully completed in January 1980 when the Department of Energy (DOE), Chicago Operations and Regional Office (CORO) issued a certification of compliance allowing shipment of 1.71 Kg fissile material and 57,000 curies of fission products producing up to 208 watts of decay heat.

The T-2 cask upgrade to 10-CFR-71, Appendix E was done by providing a shipping case (overpack) for the cask and a cask liner to meet the requirements for payload containment. Quarter scale models of the cask and shipping case were fabricated and used in drop tests to show compliance with 10-CFR-71, Appendix E. Testing was performed on a full-scale cask liner to show compliance with "Special Form" requirements.

Figure 2 shows a T-2 cask supported vertically for a load/unload operation at HFEF. Shown in the cask liner are 19 fuel pins in protective tubing sleeves. The fuel pin/tube assembly is in a carrier equipped with a wire bail for handling and insertion into a 2R container. The T-2 liner has an internal shield plug to allow hands-on bagging (sealing) of the liner to the cask loading/unloading port.

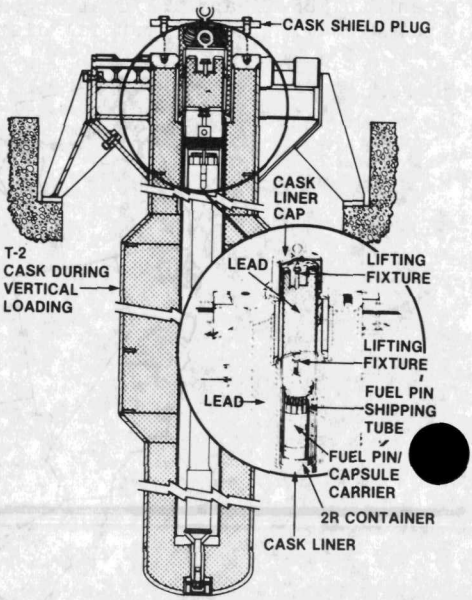

Fig.2 T-2 Cask

TRANSITION FROM T-2 TO T-3

The Fast Flux Test Facility (FFTF) is the second generation US fuels and materials fast test reactor. The longer (3.62-m) FFTF core components required development of a longer shipping cask. The new cask interfaces were also at HEDL, ANL-W and LASL again requiring a system for interfacing and protection of the experiments during transport. Studies were made of the T-2 cask design criteria and the payloads from the testing program in EBR-II. Design criteria developed for the new shipping cask, designated T-3, included:

- Individual fuel pin packaging,
- Fuel pin container sealed to prevent contamination spread.
- Remote insertion/withdrawal of payloads.

The payloads are limited to the following:

- Fissionable material 5 kg
- Decay heat 600 W
- Cask surface radiation dose 100 mRem/hr
- Fuel-pin surface temp. 427°C

Shipping cask requirements were:

- Capability for both vertical and horizontal loading.
- Deliverable with NRC certificate in two years.
- Weight under 18,000 kg.
- Outside diameter less than 840 mm.
- Compatible with existing internal container designs.

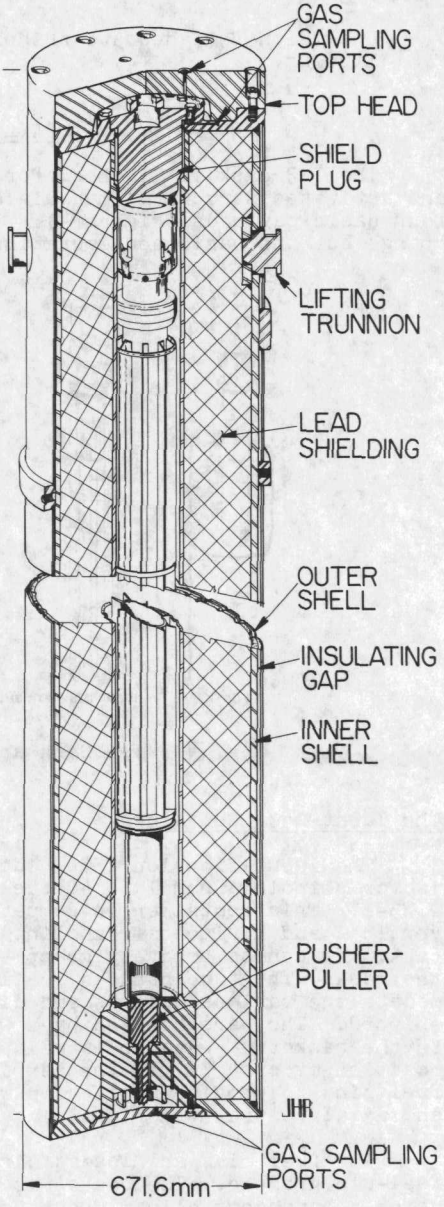

Fig. 3 T-3 Cask

- Design compatible with US-Department of Transportation and NRC regulations,
- Provision for adequate shipping-cradle support, tie-downs and lifting.

PAYLOADS

The T-3 cask is licensed for 200 000 Ci of mixed fission products and activated structural materials. The cask was designed to ship four basic payloads (Figure 4). The contents of the four payloads change but the basic configurations remain the same.

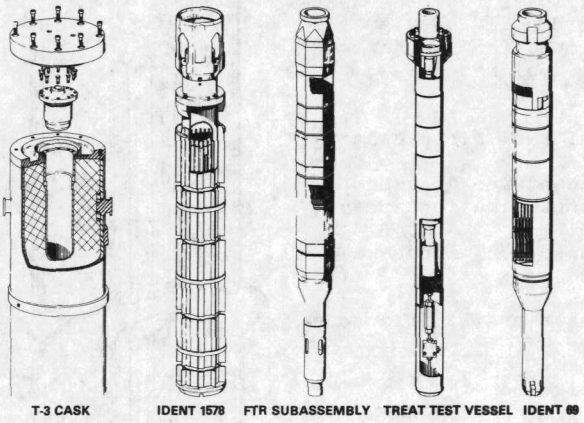

Fig. 4 T-3 Cask and Payloads

The Ident 1578 Container

The identification 1578 (Ident 1578) container shown in Figure 4 is for shipping from 1 to 40 fuel pins in lengths from 1.02-m to 2.35-m. This container is compatible with all facilities (both vertical and horizontal handling) identified for receipt of the T-3 cask. It can be remotely grappled for removal from or insertion into the cask. The tube bundles, called "baskets," vary in size to accommodate the various lengths and diameters of the fuel pins being shipped. They are handled by a wire-rope bail which attaches directly to the basket. A spacer is welded to the bottom of each basket to position it properly in the carrier. A protective cap covers the fuel pins to provide protection while handling the basket assembly and axial restraint for the pins and for the basket assembly.

The fuel pins are loaded directly into the basket tubes by a fuel-pin grapple. The basket is suspended on top of the carrier by using a horseshoe clip. When the basket is full, the protective cap is installed, the basket is picked up by the cable bail, the horseshoe clip is removed, and the basket is lowered into the 1578 container and then the 1578 container into the cask completing the loading operation.

As stated in the system requirements, the maximum allowable decay-heat load for the cask is 600 W and the limiting fuel pin cladding temperature is 427°C. Under certain conditions, fewer than 40 pins would be shipped in the Ident 1578 container to stay below the cladding-temperature or decay-heat limitations.

The FTR Assembly

The second payload shown in Figure 4 is the FTR* assembly. The T-3 cask is designed to ship both fuel and absorber-material subassemblies in much the same manner as the Ident 1578. For fuel, a relatively long decay period will be required to reduce the decay heat to less than 600 W. For a medium-power FTR driver assembly, the decay period will be about one year. Because absorber assemblies contain no fissionable material, the decay-heat problem will not restrict their shipment.

The TREAT Test Vessel

The third payload shown in Figure 4 is the TREAT static capsule for transient testing of FTR-length (2.35-m) fuel pins. This payload is not critical in terms of curie content or decay heat since only one to three fuel pins are used. However, the 3.62-m length of the static capsule makes it the longest payload shipped in the T-3 cask.

The Ident 69 Container

The fourth payload shown in Figure 4 is the Ident 69 spent-fuel storage shipping container. This container is similar to a subassembly in configuration. It can accommodate up to 109 spent fuel pins for return to the Fuel Storage Facility at the FFTF site in Eastern Washington. The pins are loaded into separate tubes (similar to the Ident 1578) wherein they are restrained during shipment and storage.

PAYLOAD RESTRAINT

The differences between payload outside dimensions and the cask cavity dimensions required a payload restraint system. The following criteria were applied to the restraint system design:

- Weight less than 67 kg,
- Easily installed and removed,
- Does not impair payload heat transfer,
- Cost effective for replacement vs. decontamination, and
- Provides positive restraint insofar as remote insertion/withdrawal handling tolerances will allow.

* The core of the FFTF reactor is known in the physics community as the "Fast Test Reactor" (FTR).

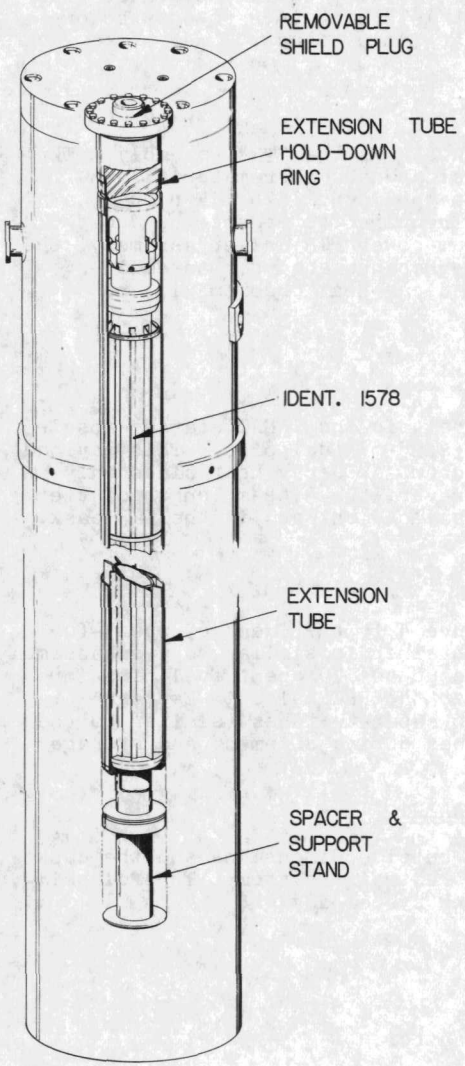

Fig. 5 T-3 Cask with Ident 1578 Pin Container

The payload restraint system (Figure 5) underwent several design iterations and prototype tests before reaching final configuration.

Longitudinal positioning was accomplished with a spacer support stand at the cask cavity bottom. The flat top-surface of the stand provides a resting point for the Ident 1578 and a tapered internal seat for the S/A and Ident 69 payloads. An extension tube rests on the stand and extends to near the cavity top providing radial restraint for payloads. An extension hold-down ring is attached to the upper cask cavity with three set screws. This ring prevents the extension tube and spacer support stand from moving when payloads are pushed or pulled from the cask.

The payload restraint system was fabricated from aluminum and black anodized to assist heat transfer.

T-3 CASK DESIGN, LICENSING AND FABRICATION

The task of incorporating the above noted cask requirements, payload restraints and design criteria into a licensable package was given to Nuclear Packaging, Incorporated (NuPac) of Tacoma, Washington.

Since it was necessary that the payload transfer be conducted in a totally sealed and controlled atmosphere, each end of the cask containment was equipped with a sealing or bagging ring. This would allow the cask to be sealed by way of a flexible bag. Once sealed, the primary shield plug could be remotely removed and drawn through the bag into the controlled cell area. The bottom end of the cask was equipped with a shielded push plug assembly that allows the payload to be pushed into the cell area during horizontal loading. Vertical loading was accomplished by way of a remote grapple lowered into the cask cavity.

Once the functional and operating features were incorporated into the design, the package was then reviewed in light of regulatory requirements. Recent regulatory emphasis on avoiding nil ductility and brittle fracture characteristics for structural materials combined with the tight program time schedule, dictated that the cask be fabricated entirely from 300 series stainless steel.

A detailed Safety Analysis Report was completed, using a variety of computerized analysis techniques. Standard thermal analysis routines were used to handle the effect of the hypothetical accident condition on the rigid foam impact limiters and wire spaced thermal shields. Normal conditions of transport represented a more difficult thermal restraint since it was important that the payload heat generating capacity be adequately dissipated to keep the fuel internal temperature below 427°C. Temperatures above this limit would affect the fuel pin test results, but would have little affect on the package.

Impact conditions required the development of special programs designed to evaluate the package for a complete range of impact orientations. NuPac generated a dynamic program (OBLIQUE) that parametrically examines the affect of all possible impact orientation angles from vertical to horizontal. This approach is necessary on relatively long, slender packages that may experience significant body bending moments during impact. Results from the (OBLIQUE) program were fed directly into finite element programs that evaluated not only the casks outer shell but also the containment vessel. Containment vessel stresses resulting from bending and impact loads were then judged relative to the acceptance criteria.

To minimize licensing review time, relatively high margins of safety were maintained throughout the design. The review process required approximately 16 months from the initial submittal to receipt of a Certificate of Compliance. The T-3 was the first spent fuel cask to be licensed in the USA since 1976.

The final configuration for the T-3 provided a payload cavity of 3.7 meters long by 203 millimeters in diameter. Gross weight, including two removable impact limiters, is 17,326 killograms. Shielding is provided by 203 millimeters of lead, combined with 38.1 millimeters of stainless steel. Access to the containment vessel is by way of a 314 millimeter thick stainless steel closure plug. A double O-ring system ensures sealing integrity (Figure 3).

Three packages as shown in Figure 6 were fabricated to a quality system meeting the requirements of RDT 2.2 and 10-CFR-71, Appendix E. This system ensures adherence to all design and construction requirements for the T-3 casks. Fabrication, lead pour and delivery of three units were completed in the summer of 1980.

ACKNOWLEDGEMENT

The authors recognize the expert assistance of P. R. Clements and L. A. Pember of Westinghouse Hanford Company and John D. Simchuk, Charles A. Skorupa and Dick Haelsig of Nuclear Packaging, Inc. for their contributions toward the successful completion of the T-3 cask program.

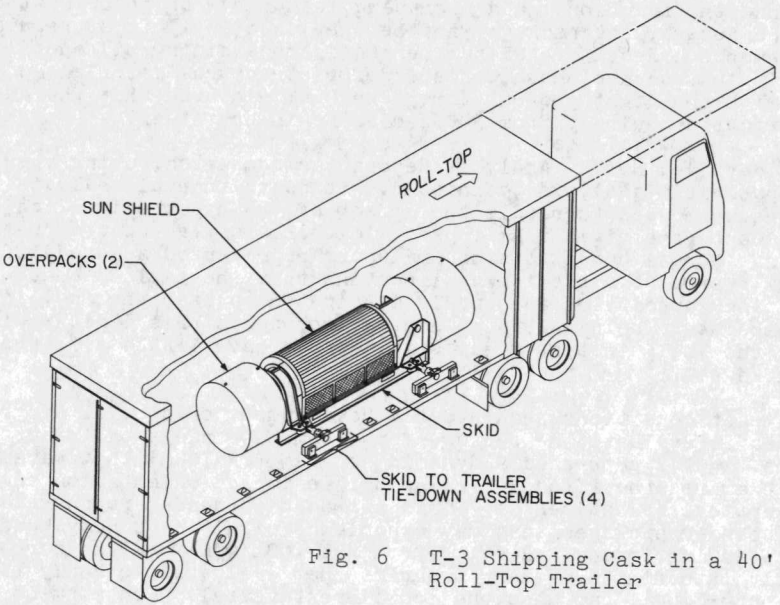

Fig. 6 T-3 Shipping Cask in a 40' Roll-Top Trailer

DISCUSSION

Question by K. Schneider, FR Germany: 1. Why was the dose rate limited to 100 mrem/h? 2. Why was the surface temperature limited to 60°C? 3. Which limit did you take for the dose rate at the 1m and 2m distance from the surface of the cask? 4. Will you use a special trailer?

Answer: 1. A maximum dose rate of 100 mrem/h was required for facility interface with the cask. 2. 60°C was considered a safe cask surface temperature for handling. 3. 1m and 2m dose rate requirements were consistant with applicable regulations. 4. We plan to use a closed trailer but the primary requirement is weather protection.

Question by W. Gibson, Canada: 1. I understand that you did not perform either drop tests or fire tests on the T-3 cask, only analysis? 2. Do you think that the 16 months required to receive licensing approval could have been reduced if you had performed actual tests on a prototype? 3. Did the licensing authorities accept the thermal and impact analyses easily?

Answer: 1. The T-3 cask design was licensed strictly by analytical technique. 2. A difficult question, but in this case I don't think so. 3. Four iterations were required to satisfy the NRC questions asked about analyses. In some cases, reanalysis was required.

Question by M. Brown, United Kingdom: The thermal barrier in your cask design appears to be based upon an air gap. Does this make it vulnerable to an impact from the side followed by a fire and have you analysed this?

Answer: We are currently reviewing the cask design to 1973 IAEA Regulations and will have this answer soon. This is a potential problem area.

CASK DESIGN FOR SPENT LMFBR FUEL ELEMENTS TYPE SNR 300

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H. Diem
Transnuklear GmbH

1. Introduction

For the spent fuel elements of the German 300 MW prototype LMFBR named SNR 300 a shipping cask concept has been developed by Transnuklear GmbH. The plant is now under construction and start up is envisaged for about 1984.

The following design requirements had to be taken into account:

- large capacity without exceeding a 80 t cask weight and dimensional limits given by the power station handling facilities
- dry loading at the reactor site with a given loading technique
- dry or wet unloading at a reprocessing or storage facility
- encapsulation of the elements prior to loading in view of water pond storage requirements.

2. Fuel element data

The fuel elements are typical for LMFBR's and comprise an array of stainless steel clad fuel pins surrounded by a hexagonal shroud tube. Main assembly data are given in table 1.

Tabele 1

Data of SNR-300 fuel assemblies

number of pins	169 (MKIa-version)
active length	950 mm
total length	3 700 mm
shroud tube dimension	110 mm across flats
operating power	$\simeq$ 4 MW
decay heat at the date of loading	$\leq$ 1.5 kW (corresponding to a cooling time of about 200 d)

The decay heat is limited by the encapsulation facility.

3. Cask design (see fig. 1, 2)

The cask concept is based on the TN technology for LWR spent fuel casks and comprises the following features:

- a forged steel body with stainless steel cladding serving as gamma shield and containment
- copper fins for external heat transfer
- a resin-type neutron shield embedded at the root of the fins
- a removable aluminium basket with 12 borated stainless steel clad lodgements
- fixed lateral and removable front shock absorbers
- bolted trunnions
- lateral orifices
- a shielding lid with two penetrations for insertion of the encapsulated fuel assemblies. The penetrations are closed by plugs.

Cask data are given in table 2. Aspects of handling and safety are discussed below.

Table 2

SNR-300 spent fuel cask data

overall diameter	2 240 mm
overall length with/without shock absorbers	6 490/5 730 mm
cavity diameter	849 mm
cavity length	4 250 mm
weight	75 t
capacity	12 assemblies
design heat load	12 x 1.5 = 18 kW

3.1 Loading/Unloading

At the SNR-300 power station the loading of the cask is accomplished by using a transfer container with a single element and a rotating device which is put on top of the cask lid. Loading is performed through two ports in the shielding lid, the ports being in line with the lodgements which are arranged on two concentric rings. During the loading procedure the cask lid is connected to the handling system and can be rotated.

Dry unloading affords a similar system or a hot cell. In the case of wet unloading, the cask is cooled down and the shielding lid is removed under water in the same way as for LWR fuel casks.

3.2 Containment system

All the fuel assemblies are enclosed in stainless steel capsules which are closed by welding. The capsules are however not considered as a part of the containment system which is completely based on the integrity of the cask body and the closures. All closures con be leak tested independently.

3.3 Heat transfer

The decay heat source is concentrated an active length of 950 mm. Heat transfer to the cavity wall is primarily by radiation inside the fuel assembly and conduction through the high conductivity basket. To keep temperatures as low as possible, the capsules and the cavity are filled with helium. Loss of helium during normal or accidental conditions will however not result in unacceptable temperatures for both fuel pins and the basket. The temperature distribution over the cask cross section is given in fig. 3. Temperature data are given for low grade helium and air (data in brackets).

3.4 Criticality safety

A parametric study has been performed for the given basket geometry. Results are given in fig. 4. In the dry condition keff is only 0.55. All other cases were calculated for water penetration into the cask cavity and the capsules. The influence of basket material is not very pronounced. keff is between 0.79 - 0.74 for the materials aluminium, steel and copper. These rather low values of keff which are obtained without poison addition are due to the fact that the assemblies are substantially undermoderated. If an extreme situation is assumed where the rods are rearranged on a larger cross section, moderation and consequently keff increases.

In fig. 4 the larger cross section is described by a larger assembly diameter. Even a complete rearrangement of the fuel pins inside the capsule diameter of 15 cm would however be acceptable if some poison is added to the stainless steel tubes of the basket.

It is more than questionable if the rod cluster could really rearrange in this way in view of the presence of spacers and the massive shroude tube. In a realistic addicent situation the assemblies would at most tend to flatten, which results in reduced pin spacing and therefore in a reduction of keff.

3.5 Shielding

Whereas γ-shielding requirements are comparable to LWR fuel casks, the high neutron source concentration of spent LMFBR fuel necessitates a substantially stronger neutron shield of about 150 mm thickness. The major contribution to the total surface dose rate of about 65 mrem/h is from neutrons. An uncertainty factor for the prediction of neutron source strength has been included.

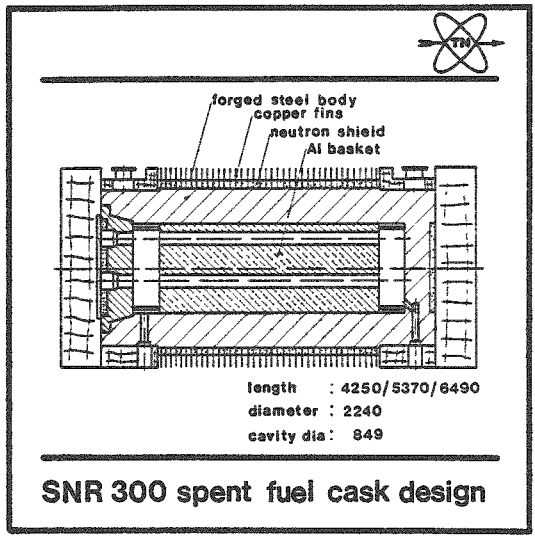

forged steel body
copper fins
neutron shield
Al basket
length : 4250/5370/6490
diameter : 2240
cavity dia : 849
SNR 300 spent fuel cask design

Fig. 1

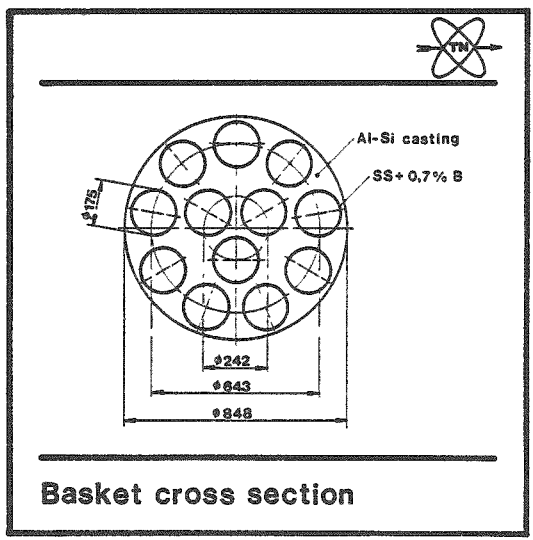
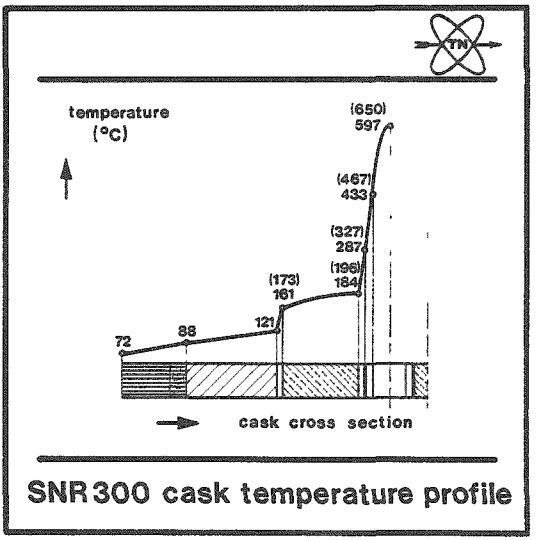

Al-Si casting
SS+0,7% B
ø125
ø242
ø643
ø848
Basket cross section

Fig. 2



SNR 300 cask temperature profile

Fig. 3

condition	assembly diameter (cm)	basket material	k_{eff}
dry	10,78	Al	0,55
wet	10,78	Al	0,79
wet	10,78	St	0,76
wet	10,78	Cu	0,74
wet	12,00	Al	0,88
wet	13,12	Al	0,95
wet	15,00	Al + borated St	0,94

Results of criticality calculations

Fig. 4

A UK FEASIBILITY STUDY FOR AN LMFBR SPENT-FUEL TRANSPORT FLASK

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Risley, Warrington, Cheshire*, and
Winfrith, Dorchester, Dorset**

ABSTRACT

A programme of LMFBRs in the UK in the next century will require the transport of spent fuel at a high level of decay power in order to obtain a total out-of-reactor time for plutonium of 9 months. Further work has been done to update the previous UK study of a conceptual design of transport flask in order to show its feasibility in the long term and to identify the features that might be demonstrated for the Commercial Demonstration Fast Reactor. The substantial margins in meeting the IAEA regulations are explored. The proposed design carries 3 sub-assemblies doubly contained in separate internal canisters and in the body of the flask itself. The canisters contain sodium as a heat transfer medium. The steel main body which provides the gamma shield also gives good resistance to fire. There is an annulus of water-extended polyester surrounding the steel body to provide neutron shielding. The neutron and gamma outputs of fuel in a short out-of-reactor fuel cycle are estimated in order to show that the external radiation dose levels would be sufficiently low. Heat transfer calculations are presented to show that the fuel pins would not reach failure conditions in normal service or in accidents. The further development work required to complete the study is discussed.

1. INTRODUCTION

As part of the demonstration of the feasibility of building and operating a large number of Liquid-Metal Fast Breeder Reactors (LMFBRs), it is necessary to establish that spent fuel may be routinely transported, safely and economically, from the reactors to the reprocessing plants. This report summarises a design study that has been undertaken to demonstrate the feasibility of providing a transport flask and to identify those features needing development work before the preparation of a detailed design.

It has been shown in many assessment studies that LMFBRs can make a substantial contribution to the World's power needs while conserving the sto of natural uranium if they are operatd with a short out-of-reactor time recycled plutonium. The objective set for the design study is the transport of sub-assemblies which have been cooled after irradiation for an average of 140 d. This is consistent with objectives set for the various stages of the timetable for the out-of-reactor fuel cycle which leads to a total out-of-reactor time of 9 months (1).

Most estimates of future reactor programmes indicate a requirement for the transport of short-cooled LMFBR fuel early in the 21st century and the flask feasibility study has this date in mind. No decision has yet been made on the extent of the demonstration of the transport of short-cooled fuel for the planned Commercial Demonstration Fast Reactor (CDFR).

A conceptual design for a flask for 5 CDFR sub-assemblies was produced some years ago (2). This design has been extensively reworked, the pay-load reduced to 3 sub-assemblies, and greater attention has been given to the requirements of security. The substantial margins in meeting the IAEA regulations (3) have been explored.

2. GENERAL DESCRIPTION

In Fig 1 there is a general assembly drawing of a flask for 3 sub-assemblies, each with a maximum heat output of up to 8 kW. The body is a single massive steel forging which gives a simple robust design of flask with good impact resistance as well as resistance to malevolent attack. Fins are provided on the outside of the flask to dissipate the heat to the air and to act as impact limiters in the event of a side impact. Impact limiters are placed at the ends of the flask to decelerate the flask in the event of an end-on impact. The mass of the flask is about 95 t which is suitable for transport on a purpose-designed railway wagon. Transport by road may be necessary for short distances to and from rail heads.

The canisters (shown in Fig 2) containing the sub-assemblies are placed in 3 holes bored along the length of the body on a pitch circle diameter of 450 mm. The canisters have a nominal outside diameter of 220 mm and the diametral clearance of 12 mm between the canisters and the holes is provided to take account of the permissible variations in the diameter and out-of-straightness of the canisters. This argon-filled annular space has been minimised in size to reduce the temperature drop across the gap.

The decision was taken to adopt sodium as the medium within the canisters for a number of reasons. Sodium is a good heat-transfer fluid which will remain in single phase at the high predicted temperatures of the fuel pins. The choice eliminates any interchange of the sub-assembly between different fluids before despatch and sodium-cleaning operations of sub-assemblies of high decay power are not required at reactor sites. The long-term storage life of sub-assemblies in sodium is already established.

The use of sodium presents one feasibility isue for the canisters - the production of satisfactory seal welds at the reactor for potentially sodium-contaminated surfaces and the reopening of the canisters at the reprocessing plant. Development trials with a 5 kW CO_2 laser for both purposes are giving promising results and improvements on the details shown in Fig 2 are being considered to facilitate laser welding and cutting.

One end of each hole in the body is permanently welded closed and the other is a removable plug. Containment is ensured by a 'Helicoflex' flexible metal face seal in the plug. A metal seal is adopted to withstand leakage of sodium from the canister. This type of seal, developed originally for the CEA in France, is available commercially. The seal in the plug can be leak tested by using an interspace leak-test facility, the interspace being provided by an elastomer 'O' ring concentric with the metal seal.

Double containment of each sub-assembly (which may contain some fuel pins failed in service) is provided by its welded canister and by the plug in the hole. In addition, there is a further large removable plug covering all 3 smaller plugs which it may be sufficient to regard as a weather seal and security cover, so allowing less rigorous operational procedures.

The external diameter of the forging is covered over its central part by a moderator jacket which is required to reduce the external neutron dose rate. The moderating material is water-extended polyester (WEP). The proposed method of construction is to cast the fins which pass through the moderator jacket in short lengths each with 4 fins. The external diameter of the forging and the internal diameter of the short length with fins are such that the fins can be shrink-fitted on to the forging. In the same way, the steel outer jacket and the extension of the fins radially outwards from the moderator are shrink-fitted to the moderator jacket fins.

It is expected that, in a fire, the fins will expand and produce an air gap at the shrink-fit interfaces which will reduce the heat input into the flask and also reduce the thermal stresses. Good thermal contact is expected in normal conditions across the shrink-fit interfaces as controllable stresses can be readily achieved using tolerances that can be attained in normal workshops. A temperature difference of about 50°C is all that is required between the shrink-fitted components for an air gap to start to appear. The air gap reduces the heat input into the flask by about 50% in a fire. At the end of the fire, the fins will rapidly cool and again produce a shrink fit which will aid the cooling of the flask.

To provide for end-on impacts, limiters are mounted on the ends of the flask. They are of all-welded construction using aluminium plates and the energy is absorbed by buckling the plates. The preliminary design of the impact limiters is based on the model tests performed for the earlier design. The cooling fins will act as impact limiters in the event of a sideways impact on the flask. Further drop tests will be required when the design is finalised. The earlier tests also demonstrated that a canister of similar dimensions and material to the one now proposed was able to withstand the drop from 9 m required by the IAEA regulations (3) with a considerable margin. It is expected that the present design of canister will behave similarly.

Trunnions are provided at each end of the flask for lifting and tilting the flask and also for securing the flask on the railway wagon. The flask will be tilted at an angle of 5° to the horizontal during transport to permit any ullage and gas bubbles in the molten sodium to float away from the fuel pins.

A vented cover, placed over the flask during transport, has the main purpose of increasing security by adding an additional barrier which has to be circumvented before access to the flask can be obtained. It also provides a barrier to solar radiation and, if it is adequately ventilated, it will not increase the temperature within the flask.

3. TECHNICAL ASSESSMENT

The shielding provided by the body of the flask is designed for a source consisting of the irradiated pins whose fuel occupies only a length of 1 m positioned near the centre of the flask. The shielding over the central length of 2.6 m of the flask consists of a minimum thickness of steel of 470 mm followed

by 100 mm of moderator (WEP) and a further 25 mm of steel. The neutrons produced by the irradiated fuel are reduced by the steel to epithermal energies before they reach the moderator which further slows them down and permits them to be captured. The final steel layer absorbs the gamma rays produced when the neutrons are captured.

The various contributions to the dose rates at the mid-plane of the flask where they are highest are shown in Table I. The gamma ray dose rate calculations refer to a particular choice of fuel management procedure that may be used once plutonium is being recycled with an out-of-reactor time of 9 months. The calculations refer to a maximum power sub-assembly after 152 d cooling (or to a minimum power sub-assembly after 83 d cooling, which gives essentially the same gamma source and decay power of 8 kW) for a CDFR sub-assembly as defined in a previous paper (1). Despite the normal short out-of-reactor time, there is a possibility that the flask may have to carry an occasional batch of fuel that has been stored for a number of years before irradiation, so allowing to be built up the nuclide ^{242}Cm which is largely responsible for the neutron source. The neutron dose-rate calculations are based on fuel which has been stored for 10 y before irradiation and which has been cooled 271 d after irradiation. The calculations are, therefore, for separate worst conditions for gamma rays and neutrons, which is pessimistic as they would not be expected to occur simultaneously.

The maximum dose rates allowed by the IAEA regulations (3) for a flask such as this which can be regarded as a 'Full Load' are given for comparison in Table I. There is a margin of more than a factor of 10 on the IAEA dose limits for the general public (3). Such a factor will probably be necessary, however; to protect the operatives at the reactor or the plant (within the limits of say an annual dose of 1 rem) who can be expected to be in closer proximity to flasks and to be concerned with every consignment.

TABLE I

Calculated dose rates at the mid-plane of the flask

Source	Dose rate (mrem/h)	
	At surface	At 2 m
Fission-product gamma rays	0.4	0.02
Fast neutrons	9.3	0.69
Epithermal neutrons	1.2	0.02
Capture gamma rays	2.3	0.14
Calculated total	13.2	0.87
Allowed by IAEA regulations	200	20

The flask is designed for 3 sub-assemblies each producing up to 8 kW of heat. (In addition, absorption of sunlight could lead to an additional heat input to the flask of 4 kW.) The internal heat source mainly occupies the central length of 1 m in each sub-assembly. The heat is conducted from the fuel pins through the sodium to the canister. From the canister it must traverse a gas gap to reach the steel body of the flask through which it is conducted to the fins and hence to the atmosphere by natural convection and radiation.

The largest resistance to heat flow is across the gas gap between the canister and the body of the flask. This gap is assumed filled with argon as will be ensured by the proposed method of flask handling. This is a pessimistic assumption since any in-leakage of air will replace the argon by a gas of higher thermal conductivity. An appreciable proportion of the heat flow across the gap takes place by radiation and this depends on the emissivity of the stainless steel surface of the canister and of the carbon steel surface of the hole. After a literature search, it was decided that pessimistic values (giving the highest temperatures) were 0.2 and 0.3, respectively.

The effect of natural convection in the canister is evaluated by extrapolation of data reported in the literature (4,5) from experiments using electrically heated sub-assemblies in sodium-filled canisters. In the horizontal orientation, there was very little temperature variation along the length of the canister. In the vertical orientation, there was very little temperature variation from the bottom of the core upwards but from the bottom downwards there was an abrupt reduction in temperature. For inclinations up to 10°, the behaviour closely resembled that for the horizontal orientation and above 30° it resembled that for the vertical orientation.

In the design calculations for the canister, for the flask in the horizontal orientation, it is assumed that the heat production is effectively along its complete length, whereas in the vertical arrangement the heat production (except that from the lower axial breeder) is spread only along the length from the core upwards. By taking account of conduction and radiative heat transfer outside the canister it is possible to calculate the temperature of the canister wall. A separate calculation shows that the maximum pin temperature is 40°C higher. The maximum pin temperatures obtained in this way are 430°C and 490°C, respectively. These temperatures are much too low to lead to failure of fuel pins.

When the ambient temperature is at the limit specified in the IAEA regulations (3), 38°C, for a horizontal flask, the maximum temperature at a fin tip is 78°C , just within the allowed value, 82°C. (This calculation neglects the presence of the outer cover.)

It is expected that fuel pins can withstand a temperature of at least 700°C without appreciably increasing the number of failures. Calculations have been performed for the fire conditions representing a transport accident in the IAEA regulations (3). The rise in pin temperature is only 30°C on taking into account the shrink-fit arrangement or 65°C without it. The flask would be satisfactory in a much more severe fire and, in that case, the benefit of the shrink-fitted arrangement would be correspondingly greater.

4. DISCUSSION

The study has shown the feasibility of the conceptual design for a transport flask for 3 sub-assemblies of spent fuel each of output 8 kW. The design incorporates double containment of the fuel, sodium-filled internal canisters and a steel gamma shield. The dimensions of the flask are within the rather small

loading gauge required on UK railways. The steel gamma shield gives substantial resistance to malevolent attack and provides a large margin for fire resistance.

Work is continuing to improve the understanding of the difficult features of the design so that eventually an optimised design can be produced, possibly carrying a larger number of sub-assemblies. Consideration is also being given to the nature of the demonstration of the eventual requirements for short-cooled spent fuel in connection with the CDFR.

Development work is planned to explore the natural circulation of sodium in a canister, horizontally or vertically. This will enable an improvement to be made in the rather empirical design methods used so far and will also allow a study of bubble behaviour in inclined flasks. An angle greater than 5° cannot be accommodated due to the height limitation on UK railways. The tests will investigate whether gas bubbles can collect within the fuelled region of sub-assemblies, so leading to additional pin failures.

The theoretical work indicates that the shrink-fitting of the external fins will further increase the already substantial fire resistance but confirmation by experiment and manufacturing trials are necessary. The arrangements for metal seals will also be included in manufacturing trials.

There is a requirement for more detailed information on the performance of impact limiters and for model-scale drop testing of the flask itself. As improved information on impact limiters is a general requirement for all types of flask, the work on an LMFBR flask will benefit from other programmes.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the contributions made by Mr A F Avery who performed the shielding calculations and Mr D F Collins who performed the heat transfer calculations.

REFERENCES

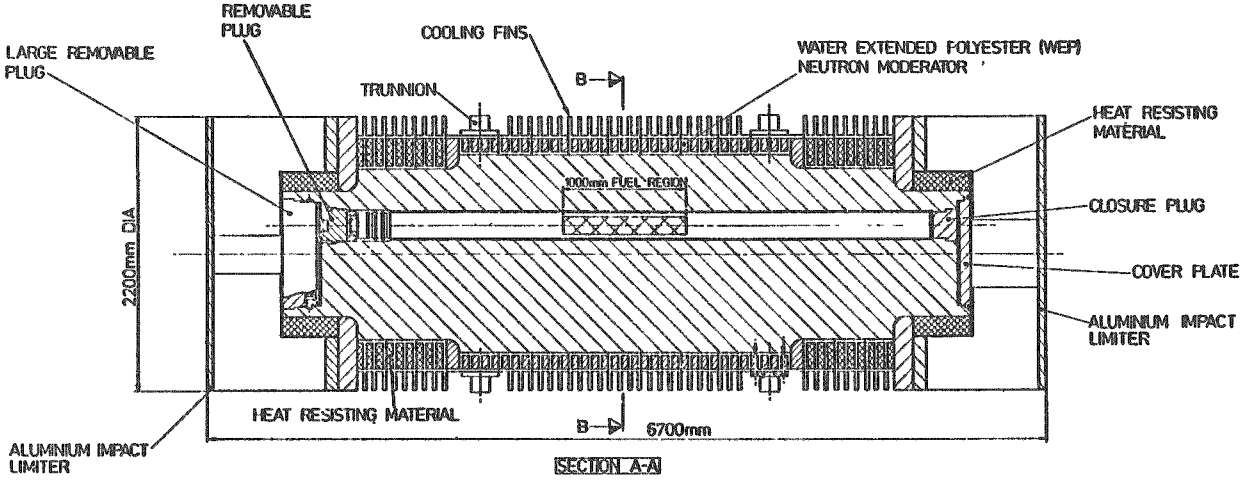
1. BAKER, A R, BURSTALL, R F, and PHILLIPS, A W. Decay power and radioactive inventories of fuel in the out-of-reactor fuel cycle. Paper IAEA-SM-244/31 for the International Symposium on Fast Reactor Physics, Aix-en-Provence, September 1979.

2. HENRY, K J, et al, UKAEA, Risley, unpublished work, 1972.

3. Regulations for the safe transport of radioactive materials. IAEA Safety Standards. Safety Series No. 6, 1973 revised edition.

4. PRUSSMAN, M, RAPPLE, B, and WEINERT, E. Heat removal experiments on an SNR fuel element using a sodium-filled transport container. KfK 2174 (1975) - translation EURFN-1301.

5. KLIMA, B B. LMFBR spent fuel transport single assembly heat transport test. ORNL-TM-4936.



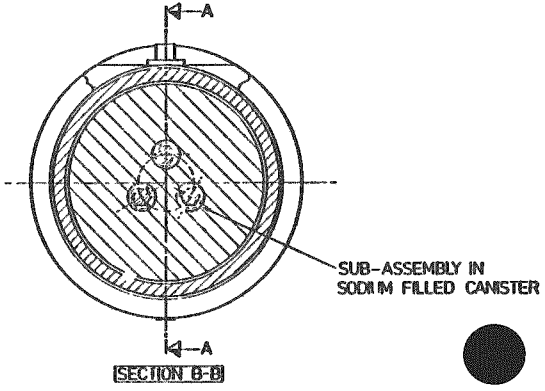


Fig.1 General assembly of transport flask

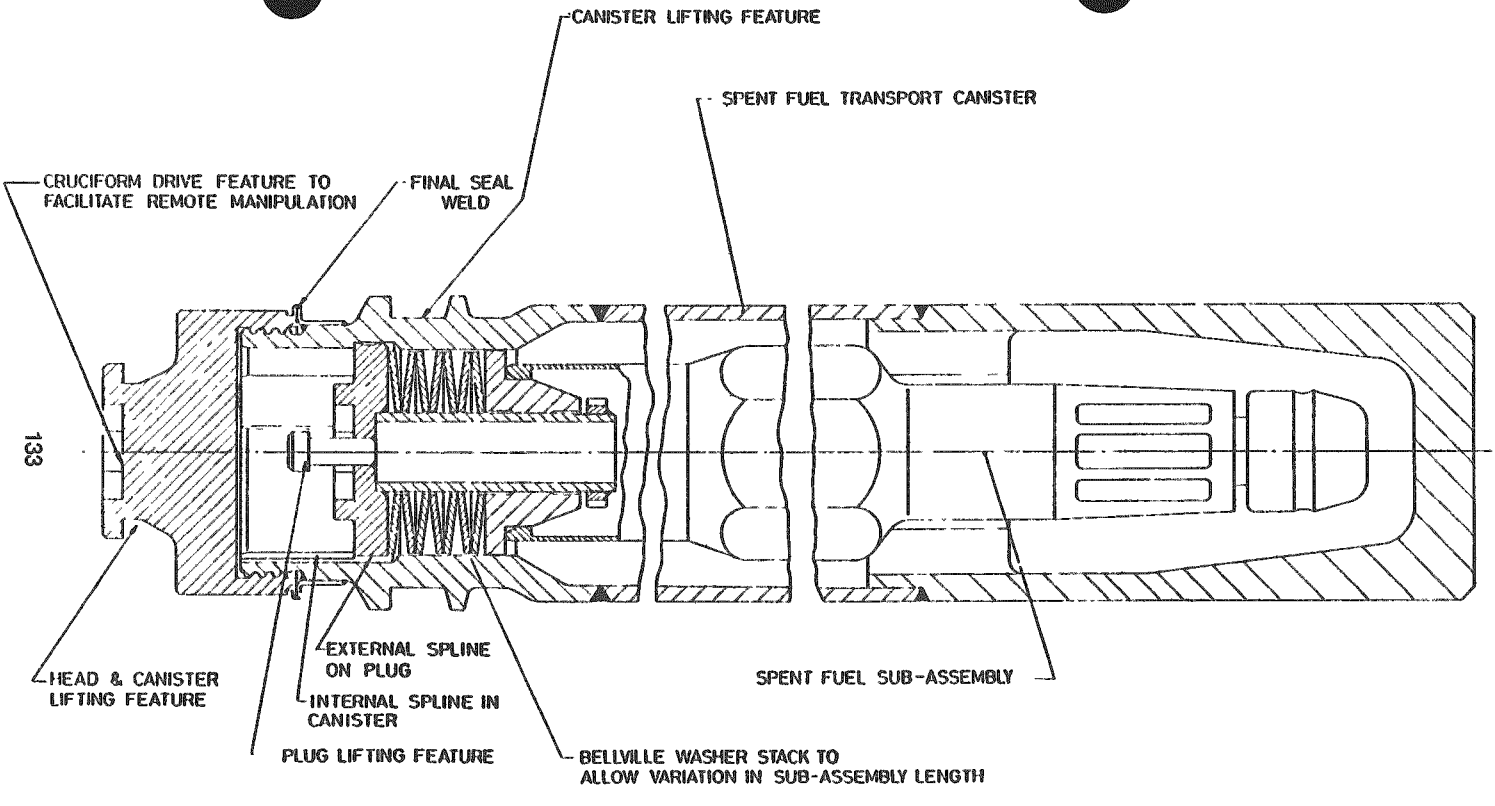


Fig.2 Sub-assembly and canister

DISCUSSION

Question by L. B. Shappert, USA: How many times do you expect to be able to reuse the fuel element canisters considering the welded seal on the canister?

Answer: It is important, as you realise, to ensure the recycling of canisters to reduce the amount of active waste for disposal. The number of times of reuse depends on the detailed final design. Our objective is to obtain reuse about 10 times.

Question by R. F. Halliburton, United Kingdom: 1. Since this flask is only for use in the United Kingdom, would you please comment on the design criteria being adopted compared to the IAEA Regulations, particularly with reference to accident conditions? 2. What is the current estimate of the overall weight of the flask?

Answer: 1. The flask is being designed to conform with the IAEA Regulations but, if necessary, a narrower range of external temperatures could be adopted for a flask only used in the United Kingdom. 2. The overall weight of the flask is estimated at 100 t.

STATUS OF LMFBR SPENT FUEL CASK DEVELOPMENT IN JAPAN

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INTRODUCTION

The Japanese fast breeder reactor program was commenced in October, 1967 when the Power Reactor and Nuclear Fuel Development Corporation (PNC) was founded as the sole organization responsible for researching and developing such advanced types of reactors as the experimental fast reactor "JOYO" and the prototype fast breeder reactor "MONJU".

The spent fuel cask development program was established based on the fast breeder reactor program. Therefore, PNC's effort has been made mainly for the development of analysis, design and fabrication technology for "JOYO" and "MONJU" spent fuel casks.

This paper describes the status of LMFBR spent fuel cask development in Japan, including cask specifications and future plans.

BACKGROUND

Construction and successful operation of experimental reactor "JOYO" followed by prototype reactor "MONJU" are major milestones in the LMFBR development program in Japan. Table 1 shows the schedule of construction and operation of the reactors, fuel fabrication facilities and spent fuel reprocessing plants.

"JOYO", the initial excavation of which had started in early 1970 was completed in late 1974 and reached its first criticality in April 1977. After many tests early in 1978, the first full power 50 MWt of MK-I core was achieved in July of that year. Since that time, two operational cycles at 50 MWt and three operational cycles at 75 MWt have been completed. One operational cycle of "JOYO" consists of 45 days of continuous operation and 15 days of shut down for refueling. After completing MK-I core operation, we will proceed to MK-II core for the irradiation tests and the operation at 100 MWt.

The environmental impact of "MONJU" is now being assessed by both the Japanese Government and the Fukui Prefectural Government. Its construction will be started in 1981.

Regarding reprocessing, which is a key to the fuel cycle, an adequate technology must be developed to cope with the increased burnup and changes in the consituents of spent fuel from LMFBRs. The Chemical Processing Facility (CPF) under construction is one of our basic research facilities, consisting mainly of hot cells in which fundamental study of reprocessing of highly burnt fuels and waste treatment will be conducted by using "JOYO" spent fuels. Conceptual design of the FBR Fuel Reprocessing Test Facility for "MONJU" and "JOYO" spent fuels is being carried out.

Table 1 Schedule of construction and operation of the PNC's facilities

(Oct. 1980)

Items	Name of Facility	Site	Year: 1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989
Reactor	JOYO (Experimental Fast Reactor)	O-arai, Ibaraki Prefecture	Critica	Tests	MK-I Operation		Conversion	MK-II Operation							
				50MWt		75MWt		100MWt							
	MONJU (Prototype FBR)	Tsuruga, Fukui Prefecture		Detailed Design				Construction					Critical Commissioning	Operation	
				Environmental Assessment		Licensing									
Fuel Fabrication	Plutonium Fuel Development Facility	Tokai, Ibaraki Prefecture	Reconstruction				JOYO MK-II Fuel Fabrication								
	Plutonium Fuel Fabrication Facility	Tokai, Ibaraki Prefecture	Detailed Design				Construction						MONJU Fuel Fabrication		
						Licensing									
Spent Fuel Reprocessing	Chemical Processing Facility	Tokai, Ibaraki Prefecture	Detailed Design & Construction						Operation						
			Licensing												
	FBR Fuel Reprocessing Test Facility	Undecided	Preliminary Design		Conceptual Design			Detailed Design					Construction		
										Licensing					

Table 2 Schedule and plan of development of LMFBR spent fuel casks

(Oct. 1980)

Items	Year: 1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989
JOYO Spent Fuel Cask -Outer dimensions total length : 3,350mm envelope diameter : 1,640mm -Mass of load packaging : 26,000kg	Conceptual Design		Design		Safety Tests		Licensing		Fabrication		Transports (Overseas)					
Irradiated Fuel Specimens Cask (I) -Outer dimensions total length : 3,264mm envelope diameter : 1,400mm -Mass of load packaging : 11,000kg				Conceptual Definition		Detailed Design		Model Tests	Licensing	Fabrication	Transports					
Irradiated Fuel Specimens Cask (II) -Outer dimensions total length : ~2,500mm envelope diameter : 1,400mm -Mass of load packaging : ~6,000kg							Detailed Design		Licensing	Fabrication	Transports					
MONJU Spent Fuel Cask -Outer dimensions total length : ~6,425mm envelope diameter : ~2,430mm -Mass of load packaging : ~76,000kg					Conceptual Definition			Preliminary Design	Detailed Design	Safety Analysis	Licensing	Fabrication				
									Plug Seal Tests	Thermal Tests	Shielding Tests	Collapse Tests	Drop Tests			
MONJU Irradiated Subassembly Cask -Mass of load packaging : <30,000kg							Conceptual Definition	Preliminary Design	Detailed Design	Safety Tests	Safety Analysis	Licensing	Fabrication			

CURRENT STATUS

Up to now, two different types of casks have been developed. Table 2 shows the development schedule of those casks.

I. "JOYO" Spent Fuel Cask

The shipping cask for "JOYO" spent fuel is designed for transport of fuel pins irradiated in the core of the "JOYO" reactor to the Marcoule reprocessing plant. Its contents consist of fuel pins in tight stainless steel capsules. The packaging can contain 12 of these capsules, each of them receiving up to 91 pins of a fissile fuel assembly. The cask is designed as a type B(U) package. The main design was completed in 1976 and safety tests for licensing were carried out in 1977-8. The licensing of the cask and fabrication have been postponed for a few years because an agreement between PNC and COGEMA concerning reprocessing for "JOYO" spent fuels has not yet been reached.

Cask Specification

Package general view is shown in Figure 1.

- Outer dimensions:
  - total length: 3350 mm
  - envelope diameter (without impact limiters): 1640 mm
  - width with impact limiters: 1974 mm
- internal cavity dimensions:
  - length: 2015 mm
  - diameter: 634 mm
- approx. mass of empty packaging: 25300 kg
- approx. mass of load packaging: 26000 kg
- package heat output: 7800 W
- package activity: 2.1×10^6 Ci
- load capacity: 12 capsules

(Each of the capsules contains 91 irradiated fuel pins or tubes for damaged pins.)

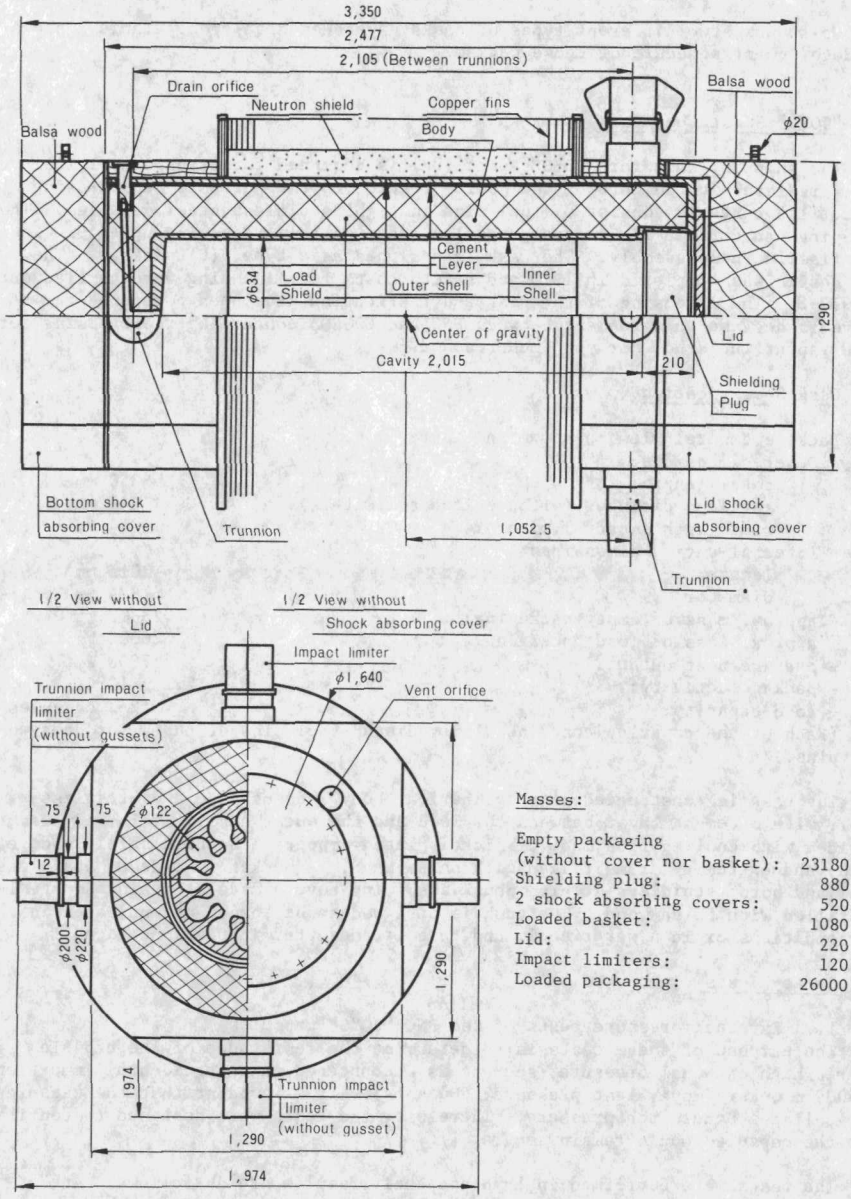
The cask is constructed of a cylindrical body made of lead and stainless steel, with a cement layer between the lead and the outer shell. This body is provided with cooling fins made of nickel plated copper, a neutron shield made of resin, and extremity drums filled with balsa wood. A basket is made of stainless steel and boron stainless steel. Shock absorbing covers made of stainless steel and filled with balsa wood. The cask is designed to be loaded and unloaded in dry conditions or in a water pool, and to be transported in dry conditions.

Safety Tests

1. External pressure tests of the PNC "JOYO" capsules

The purpose of these tests is to determine the resistance of the capsule under a high external pressure, such as is encountered in an accidental immersion in 5000 m water (equivalent pressure: 500 bars). It is evident that the capsule will collapse under such pressure. Therefore, the tests were designed to confirm that the capsules would remain secure.

The tests were performed in 1978 and their results are as follows. At 35 bars, the extremity caps were slightly bent, but the body was not crushed. From 100 bars to 500 bars, we detected a very significant crushing of the body. Between 500 bars and 700 bars, the capsules dimensional change was observed to be small. The capsule remained tight, when exposed to an external pressure of 700 bars.



Masses:

Empty packaging (without cover nor basket):	23180 kg
Shielding plug:	880 kg
2 shock absorbing covers:	520 kg
Loaded basket:	1080 kg
Lid:	220 kg
Impact limiters:	120 kg
Loaded packaging:	26000 kg

Figure 1 Package general view of JOYO Spent Fuel Cask

2. Drop tests of the PNC "JOYO" capsules and the 2/5 scale model cask

The drop tests were performed in May and September, 1977. The purpose of the drop tests of the PNC "JOYO" capsules was to determine the leaktightness. The purpose of the drop tests of the 2/5 scale model cask was to introduce the documents concerning these tests and to compare the tests and impact calculation results. The analyses to use the tests results were completed in 1978.

3. Experimental verification of the thermal characteristics of the cask

Heat dissipation tests and regulatory fire test using a full-scale slice model were performed in 1977. The main purpose was to introduce the documents concerning these tests and to compare the tests and heat transfer calculation results.

The results of the first open fire test were not judge satisfactory as resin did not stop burning even after the flame was extinguished. Therefore the second fire test was performed after changing the resin composition. The average temperature of thermocouples in fire ambiance reached 800°C. After 30 minutes, the fuel was drained from the pan but no attempt was made to extinguish the flames of the resin combustion. Most of the flames were localized along the resin partition plates. This appeared to be due to thermal decomposition of resin by the heat radiated by these plates. This combustion decreased quickly. After 30 minutes, it was completely over.

II. Irradiated Fuel Specimens Cask (I)

The cask is for shipment of irradiated fuels and materials from the reactor site hot laboratories at DFR, Rapsodie and Phenix to the hot laboratories of O-arai Engineering Center. This is also utilized for the transportation of "JOYO" spent fuel from FMF to CPF for reprocessing. Conceptual definition for the cask was completed in 1978. The detailed design and safety model tests for licensing are presently in progress. It is scheduled to obtain the approval for the packaging design as a type B(U) from the competent authorities of Japan, France and the U.K. in 1980 or 1981. Work on the design of a Irradiated Fuel Specimens Cask (II), which is very similar to the Cask (I) but a little shorter, is scheduled to begin in September, 1980.

Cask Specification

General view of the Irradiated Fuel Specimens Cask (I) is shown in Figure 2.

- Outer dimensions:
  - total length: 3264 mm
  - maximum external diameter with shock absorbing covers: 1400 mm
  - width with trunnions: 930 mm
- Internal cavity dimensions:
  - length: 2000 mm
  - diameter: 142 mm
- approx. mass of empty packaging: 10900 kg
- approx. mass of load packaging: 11000 kg
- package heat output: 260 W
- package activity: 5.5×10^4 Ci
- typical contents: irradiated fuel pins with following characteristics
  - external diamter: 5.5 mm to 6.5 mm
  - total length: maximum 1880 mm
  - active length: maximum 720 mm
  - mass of fissile material: maximum 2160 g
  - mass of fertile material: maximum 990 g
  - numbers of fuel pins: maximum 15

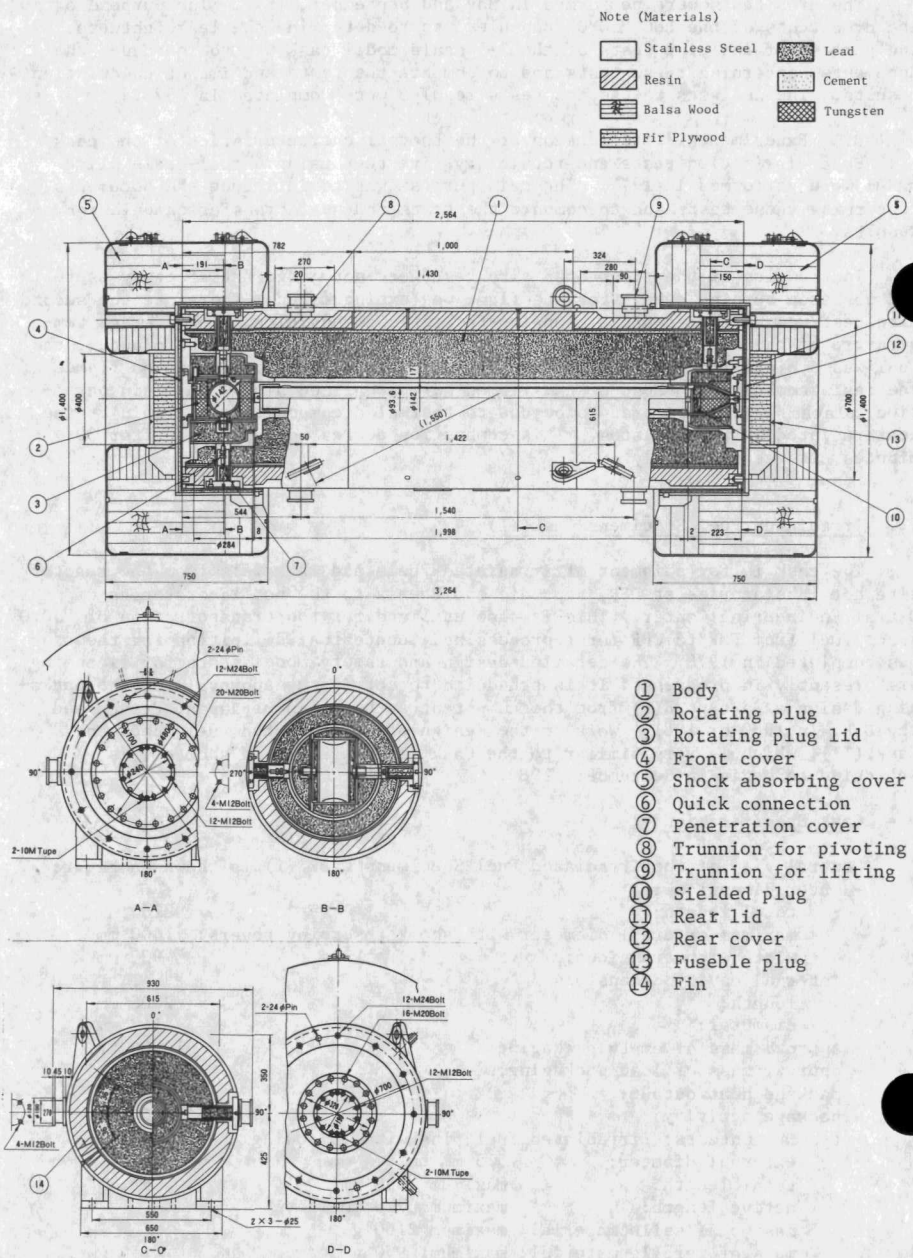


Figure 2 General view of Irradiated Fuel Specimens Cask (I)

The cask is designed in accordance with safety requirements of hot laboratories in France, the U.K. and Japan. The cask is composed of a cylindrical body made of lead and stainless steel with a cement layer between the lead and the outer shell. A rotating plug lid facilitates connection to a hot cell and resists to regulatory accident conditions. A rear shield plug connected to the capsule allows transfers to a hot cell in horizontal and vertical positions by use of a pushing bar (horizontal transfer) or a cable connected to a winch (vertical transfer). A resin neutron shielding with copper radial plates provides neutron shielding without excessive thermal barrier. Balsa wood shock absorbing covers provide shock and thermal isolation of packaging extremities.

Safety tests

Drop tests of a 1/2 scale model cask were performed in May 1980. The main purpose was to compare the tests and structural calculation results and to demonstrate the leak tightness of the cask in accident conditions. The model cask was made to simulate the rotating plug lid.

The results satisfied our design criteria, but when punch tests were performed, shock absorbing covers were penetrated by a solid mild-steel cylindrical bar 75 mm in diameter and 250 mm long.

The above results meant that balsa wood was bared, making the thermal test necessary. The test model which was 150 mm in diameter and 200 mm in depth on balsa wood covered steel was exposed to a furnace test meeting the conditions of the thermal test of the IAEA Regulations. The results of the first furnace test were not satisfactory because the balsa wood did not stop burning within the required three hours. Therefore the second furnace test was performed after adding flame retardant material to the balsa wood. After 30 minutes, the second test model was withdrawn from the furnace. The flames decreased quickly and were extinguished in a few minutes. The results of the second test satisfied our design criteria.

FUTURE PLAN

The development program of "MONJU" casks was begun recently. The program includes the development of two type casks: one is a high capacity cask for sea transport, the other is a truck-mounted cask.

1. "MONJU" Spent Fuel Cask (high capacity cask)

The concept of the "MONJU" Spent Fuel Cask was defined in 1979. The works were carried out in two stages. In the first stage, handling of spent fuels at "MONJU" and reprocessing sites and transport conditions were studied. The basic specifications were determined as follows:

(1) helium cooled dry cask weighting not more than 100 tons
(2) loading capacity of not less than 6 fuel assemblies
(3) limited seal structure with plug

In the second stage, conceptual safety analysis was carried out concerning structure, thermal analysis, containment, shielding and criticality. Basic design for two types of casks was proposed. The preliminary design and plug seal tests will be initiated in 1982 and drop tests of the cask are in planning.

II. "MONJU" Irradiated Fuel Subassembly Cask

The conceptual definition of "MONJU" Irradiated Fuel Subassembly Cask is in progress. The cask is for shipment of irradiated fuel subassembly from "MONJU" to the Fuel Monitoring Expansion Facility for the purpose of post irradiated examinations. Mass limitation for the cask is 30 tons because the cask will be transported by truck. Maximum decay heat per subassembly is expected to be between 2 and 5 kW.

ACKNOWLEDGEMENTS

We greatly appreciate the engineering contributions of Transnucleaire S.A., Kimura Chemical Plants Co., Ltd., and Hitachi Shipbuilding and Engineering Co., Ltd. in development of these casks.

DISCUSSION

Question by R. Brissier, France: During furnace tests you experienced difficulties with balsa wood. You overcame this problem by adding flame retardant material. What material did you use?

Answer: We added a saturated water solution of boric acid and borax to the balsa wood and then dried it to reach the same contents of water as in untreated balsa wood.

SPECIAL SPENT FUEL ASSEMBLIES PACKAGING METHOD FOR THE CREYS-MALVILLE POWER PLANT

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1 - INTRODUCTION

Certain spent fuel sub.assemblies from the Creys-Malville fast neutron reactor will be conveyed to examination laboratories. They may have a high residual power (about 10 KW). They must be conveyed whole in order to be dimensionally examined in special hot cells, before their disassembling. These two constraints - high residual power and whole assembly - led to the conception of a special package allowing the shipment of a spent fuel assembly in liquide sodium. The assembly is first placed in a sealed sodium - filled canister which is then placed for transport in a suitable package according to the regulation for radioactive materials transport.

2 - CANISTER AND SPECIAL SHIPPING CASK

2 - 1 - Design criteria of the canister

The canister is designed to hold the amount of sodium needed for proper cooling of the sub.assembly fissile zone. Its diameter, length and sodium content are therefore such that this fissile zone is always immersed, even in the horizontal position and after fuel cladding ruptures during a transport accident.

The fastening is sodium and gas tight and the wall thickness is calculated to resist the stress caused by a transport accident.

The volume of the cover gas is calculated in such a way :

- the upper part of the sub.assembly remains out of sodium in a vertical position in order to make the handling easier,
- the pressure does not excede 15 bars after the cladding ruptures caused by a transport accident.

2 - 2 - Main caracteristics of the canister

The canister (fig.1) is made up of two distinct zones :

- the body itself and the head.

The head fulfils several functions :

- handling of the canister (empty and loaded)
- primary seal, especially for cell handling with sodium.

This is provided by means of a mechanical plug which presses a metallic seal.

- Definitive tightness for shippment is then ensured by a welded lid.

The canister is made of stainless steel weighing 650 Kg empty and about 1.500 Kg loaded. It may be used again after a "clean cutting" of the welded lid with a laser beam.

2 - 3 - Design criteria for the shipping cask (fig.2)

The shipping cask contains only one canister for weight restriction reasons considering the examination laboratories capacities. This is not problem because there are few shipments.

In addition to the sodium and gas tightness provided by the canister, the shipping cask is sealed itself by two fastenings in series :

- a cell plug ensures a sodium and gas tightness around the canister. For this a metallic seal is used.
- a lid, covering the cell plug, ensures a water tightness with an elastomer seal.

These two fastenings are protected by a shock - absorbing hood which can also be a fire-guard.

Inside the shipping cask, the canister is hold by its head in such a way that thermal expansions take place freely and the welded lid is not subjected to mechanical stress caused by shocks for example.

2 - 4 - Main caracteristics of the shipping cask

The gamma shielding, a steel cylinder about 0,4 m thick, provides a very mechanically resistant shell which meets transport regulations. The neutron shielding is located outside the steel cylinder. It is a compound protected by a steel shell and radialy crossed by metallic fins for heat transfert.

The maximum working temperature is 425°C on the canister. The package weigh is about 60 Tons.

The shipment is to be done in a slight sloped position. It will allow to keep the fissile zone of the sub.assembly constantly immersed in the liquid sodium despite the lappings.

3 - THE REACTOR BUILDING HOT CELLS

3 - 1 - General description

The spent fuel shipment hot cells are located beside the intermediate storage tank adjoining the reactor vessel (fig.3). There are two hot cells (fig.4) :

- The upper cell, wich partially extends over the storage deck; the fuel sub.assembly is removed from the storage by means of the cell crane and prepared for insertion in a special canister which is closed before being lowered to the lower cell.
- The lower cell, where the canister is weld-sealed and inserted in the shipping cask.

3 - 2 - Operations and equipment in the upper cell.

This cell communicates with the storage vessel by means of two redundant penetrations, which are shut-off by special full-flow safe-seal valves. When a valve is open, the storage argon cover gas (at the temperature of 180°C) flows into the cell, which is filled with impure Nitrogen. A spent fuel assembly is lifted by the cell crane using an automatic gripper. This crane is fitted with a safety hoist and provision is made for remote-controlled servicing. It is also equipped with a fan to blow the ambient Nitrogen into the assembly fuel bundle.

The fuel sub.assembly is weighed and its residual power measured by gamma counting and spectrometry; at the same time, the residual sodium film flows back into the storage vessel. The purpose of this proceeding is to ensure that the residual power is low enough to permit safe spent fuel handling.

After closure of the storage vessel valve, the sub.assembly can, if necessary, be further cooled by means of an N_2. CO_2 mixture in drilling well. A special device drills six holes in the upper part of the hexagonal tube so as to ensure proper circulation of the sodium in the canister, in the event of fuel pin leakage associated with an abnormal canister configuration inside the shipping cask.

The sub.assembly residual power can sometimes be accurately measured in the same well, by a gaseous flow calorimetric method.

Should the sub.assembly not be properly cooled by the crane fan or by the drilling well blowing circuit, it can be placed in another well fitted with an additional cooling system.

The sub.assembly is then hoisted and lowered into the canister filled with sodium raised to a temperature of 250°C. The canister is positionned with its top end facing the upper cell roof, from which it is separated by a communicating well.

Before heating, its provisory plug is removed by a special "press an turn" device. This plug is replaced by the primary plug after heating and insertion of the sub.assembly. The primary plug is described in part 2 - 2. The "press and turn" device thus fulfills three functions :

- opening the provisory plug,
- fitting the cell plug, and removing it in the event of an abnormal occurence,
- ensuring leaktightness between upper and lower cells to prevent dispersion of radiative products,

The upper cell is obviously provided with all the usual hot cell equipment - periscopic lenses, remotely-operated handling apparatus etc... Its 1,2 m thick heavy reinforced concrete walls are protected by a stainless steel liner.

3 - 3 - Lower cell operations and equipment.

This cell is designed for canister handling; its permanent separation from the upper cell ensures its cleanliness as well as that of the canister itself.

In the reactor building, the spent fuel shipping cask is placed in contact with the bottom of the lower cell, whose main well is then opened by means of an auxiliary crane.

The second lid of the cask is removed (the first lid has been previously removed in the special workshop described later on) by means of an unscrewing device attached to the auxiliary crane.

The frozen sodium filled canister and its (still unwelded) lid are moved into the lower cell. The canister (without its lid) is placed on a hoist and raised so that its top is positionned against the communication well leading to the roof of the upper cell. Then its provisory plug is opened by the "press and turn" device, as described above. The sodium is melted in about one hour and a half by an induction furnace located around the lifted canister in the lower cell. After plug closure, the hot canister now containing the spent fuel sub.assembly is lowered into a well where its second lid is welded by the plasma welding. This is made with argon backing shielding gas, without addition of filler metal. This process requires the use of a stem located on the upper side of the lid. This stem is itself finally welded.

The canister is hoisted and placed in the spent fuel shipping cask, whose second lid is then installed by the screwing device.

The lower cell is also provided with various remotely operated machines, etc... Its 1,2 m thick heavy reinforced concrete walls are protected by a stainless steel liner.

3 - 4 - Auxiliary workshop.

The spent fuel shipping cask containing the canister full of frozen sodium is delivered to a special workshop, where its shock-absorbing hood and first lid are removed. Similary, after insertion of the hot canister, the cask first lid and hood are reinstalled.

Shipping cask leaktightness tests and non-contamination inspections are also performed in the workshop.

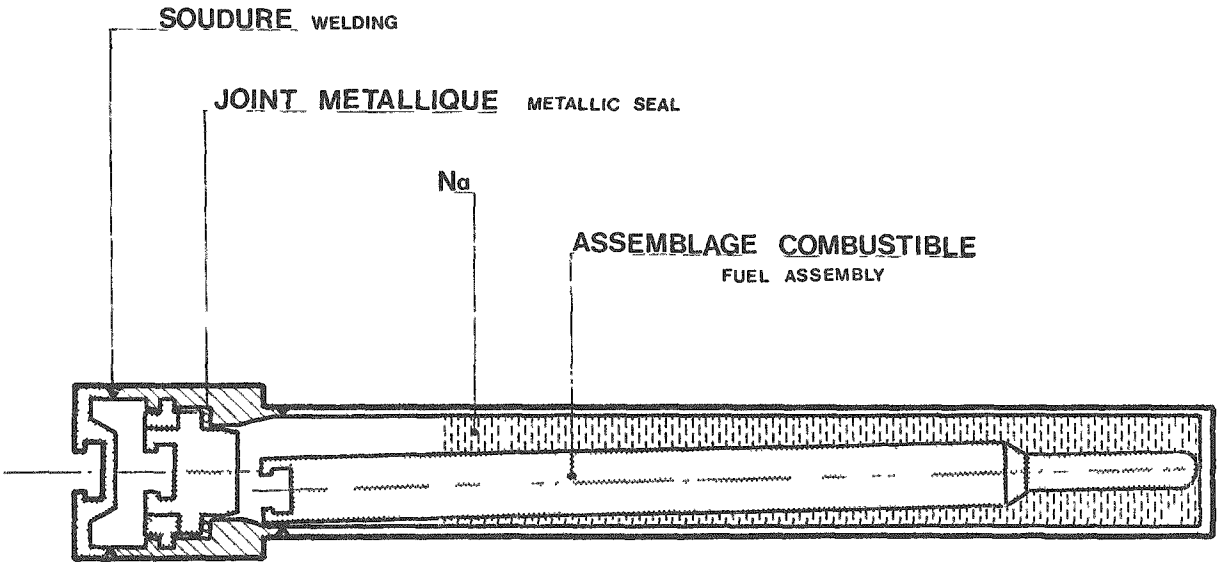

Figure 1

SPX 1	ETUI DE TRANSPORT FUEL SHIPPING CANISTER

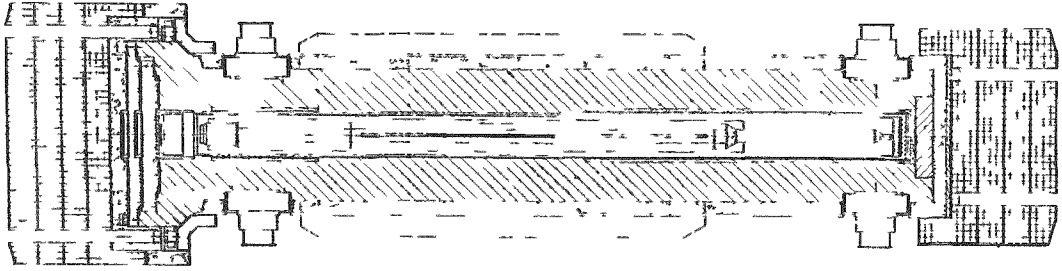

Figure 2

SPX1 **EMBALLAGE**
CASK

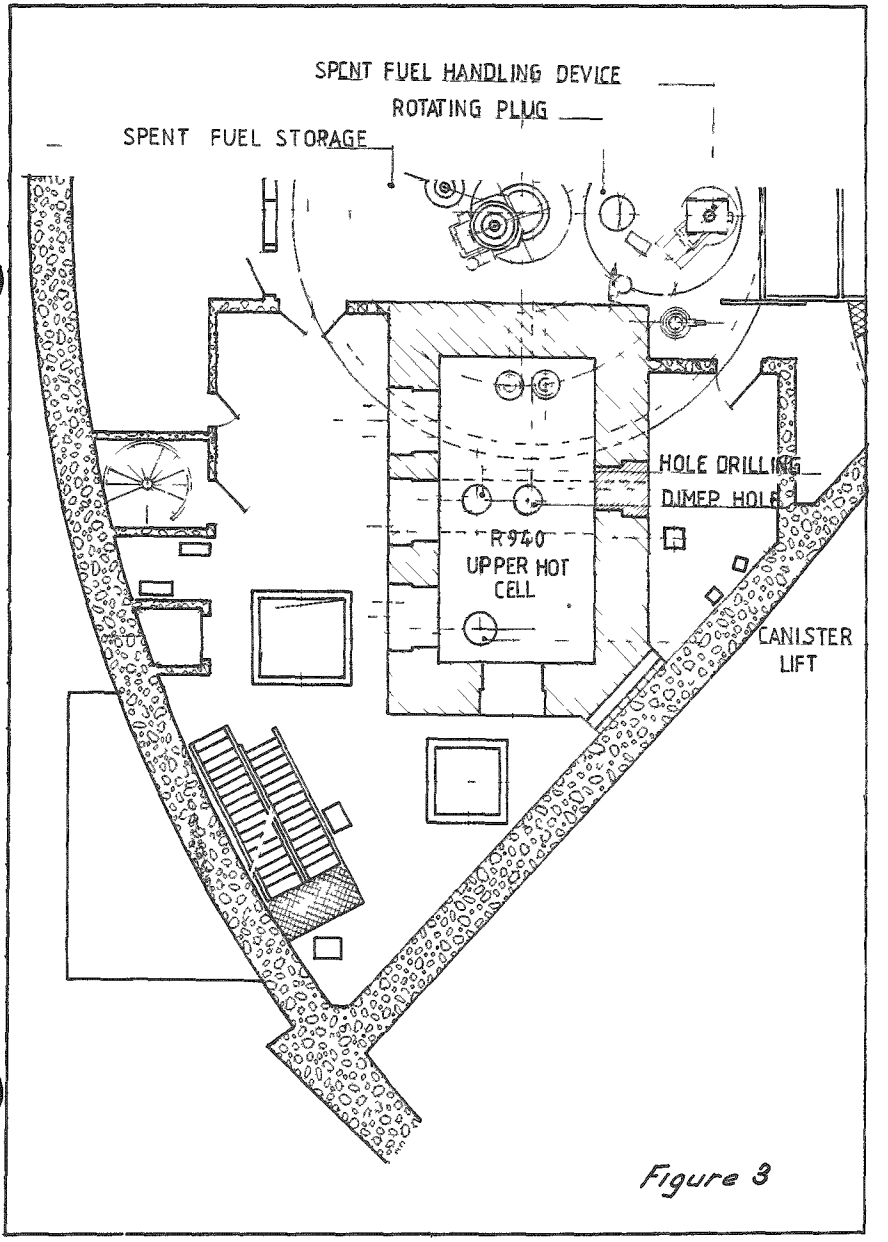
SPENT FUEL HANDLING DEVICE
ROTATING PLUG
SPENT FUEL STORAGE
HOLE DRILLING
DIMER HOLE
R940
UPPER HOT
CELL
CANISTER
LIFT

Figure 3

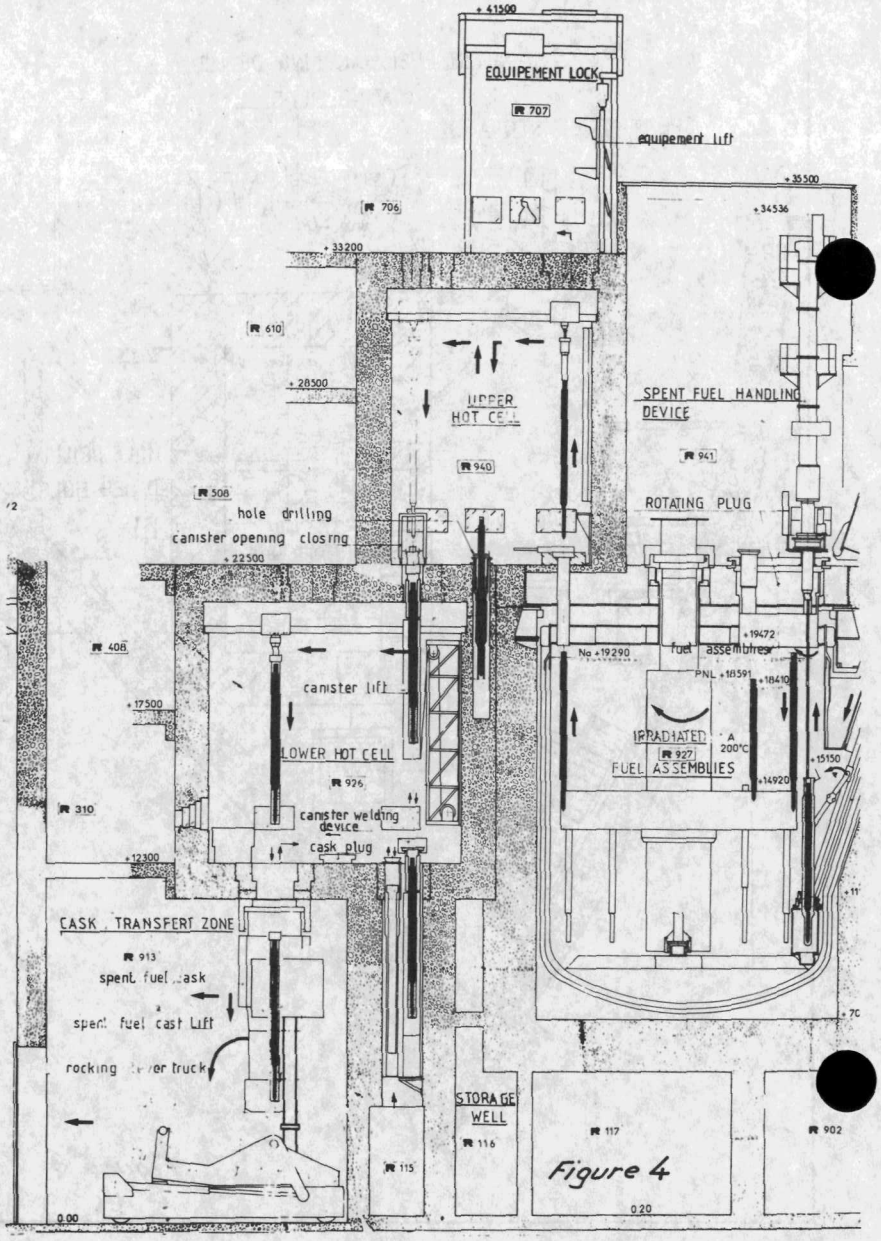
+ 41500
EQUIPEMENT LOCK
R 707
equipement lift
+ 35500
+ 34536
R 706
+ 33200
R 610
+ 28500
UPPER HOT CELL
SPENT FUEL HANDLING DEVICE
R 940
R 941
R 508
hole drilling
canister opening closing
ROTATING PLUG
+ 22500
+ 19472
No + 19290
fuel assemblies
PNL + 18591
+ 18410
R 408
canister lift
+ 17500
IRRADIATED FUEL ASSEMBLIES
A 200°C
R 927
+ 15150
+ 14920
LOWER HOT CELL
R 926
R 310
canister welding device
cask plug
+ 12300
CASK TRANSFERT ZONE
R 913
spent fuel cask
spent fuel cast Lift
rocking lever truck
STORAGE WELL
R 116
R 117
R 115
R 902
0.00
0.20

Figure 4

DISCUSSION

Question by R. Christ: You have described packaging and transport of special assemblies from the Super Phénix plant. What is the concept for transporting all spent Super Phénix assemblies?

Answer: Shipping sub-assemblies in a canister full of liquid sodium is especially convenient for the sub-assemblies which are to be examined at an early state in a laboratory, and thus shipped under the double constraint of a high residual power and a low peak cladding temperature. For the other assemblies, other methods - such as shipping after dismantling, or further decay storage - may be preferred due to their lower cost.

Question by A. R. Baker, United Kingdom: The promotion of circulation of the sodium within a canister is obviously very important and I should like to enquire about further details of the holes piercing the hexagon wrappers. Where are the holes placed, how big are they and how did you decide what to do?

Answer: Assume the canister is in a vertical position, the sodium is frozen at the bottom and assume there are several cladding ruptures: the sodium will flow inside the ruptured pins, and the sodium level inside the sub-assembly will decrease in such a manner that the fuel bundle can be uncovered.

Thus it is necessary to pierce holes (Ø 5 mm) at the upper part of the hexagonal tube, between the top of the fuel bundle and the bottom of the massive head of the sub-assembly.

ASSESSMENT OF LMFBR SPENT FUEL SHIPPING CASK CONCEPTS FOR THE CRBRP AND THE U.S. CONCEPTUAL DESIGN STUDY*

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INTRODUCTION

Two major liquid metal fast breeder reactor (LMFBR) power programs are in the conceptual study design and/or initial construction phases in the United States: the Clinch River Breeder Reactor Plant (CRBRP), designed to produce 380 MWe, and the Conceptual Design Study (CDS), estimated to result in a 1000 MWe plant. Shipment of spent fuel assemblies producing 4 to 7 kW decay heat each is required for the CRBRP. The design of the CDS plant requires shipment of spent fuel assemblies producing 8 to 11 kW decay heat. This paper describes the results of continuing studies to define shipping casks for transporting spent fuel from these plants. Loading and unloading problems are assessed and cask/plant interface criteria are discussed.

DISCUSSION

Various constraints have been defined for both plants and for the Hot Experimental Facility (HEF) which will receive and reprocess the spent fuel. For example, the handling surfaces of the cask cannot exceed 60°C (140°F) during loading and unloading operations, and the fuel clad temperature cannot exceed 650°C (1200°F) for normal operations. In addition, LMFBR spent fuel assemblies, unlike LWR assemblies, consist of an array of pins encased in a hexagonal stainless steel shell, and are further characterized by a short fuel zone from which a major portion of both decay heat and radiation are emitted. This design poses problems during loading and shipment of providing adequate shielding in the cask body near the fuel zone and efficiently distributing the heat away from the fuel zone. Previous studies (References 1-3) have shown that, to adequately distribute decay heats of 4 to 11 kW from a spent LMFBR fuel assembly through conduction, thermal radiation and natural convection, liquid coolants, such as sodium, will be required.

The basic conceptual design developed at Sandia National Laboratories (SNL) to support the CRBRP and the CDS reactors consists of a spent fuel shipping cask having a monolithic steel structure with seven basket cavities. Each cavity would contain a fuel assembly in a sealed, sodium-filled stainless steel canister. The cask cavity would be backfilled during loading, shipping and unloading with helium,

*Work done for the U.S. Department of Energy under Contract DE-AC04-76DP00789.

or possibly nitrogen. As currently envisioned, triple containment of the sodium would be provided by (1) the canister, (2) an inner cask lid and (3) an outer cask lid, as illustrated in Figures 1 and 2. If the fuel pin cladding is considered to be a containment boundary, this constitutes four levels of containment for the fuel and fission products. Surrounding the steel cask body, longitudinal steel fins on the inner circumference extend through a solid neutron shield of Borosilicone rubber. This neutron shield concept is described in detail in Reference 4. External heat rejection is accomplished with spaced finlets welded to an outer steel shell encasing the neutron shield.

The specific cask designs for CRBRP and CDS are similar but of different sizes. The CRBRP fuel, shorter and smaller in cross section, requires a cask of 1.7 m in diameter and 5.1 m in length with a loaded shipping mass of approximately 55 tonne. The canisters (0.178 m OD, 4.53 m long) would fit into 0.19 m ID cavities arranged in a circle of six on a 0.44 m diameter, with one in the center. For the CDS spent fuel, the cask would be 2.1 m in diameter and 6.6 m in length with a loaded shipping mass of approximately 88 tonne. The canisters for CDS fuel (0.229 m OD, 5.284 m long) would fit into 0.24 m ID cavities, one in the center and six on a 0.54 m diameter.

Three alternate methods for designing the outer lid, the inner lid (or lids) and an alternate shield plug are illustrated in Figure 2. Crane lift capacities and other cask/plant interfacing criteria will eventually determine the concept chosen, although the cask having two lids with the shield incorporated into the inner lid is preferred. The other options add complexity to the loading/unloading procedure, the cask fabrication, or both.

Two canisters for CRBRP fuel are currently under consideration. Both are of stainless steel pipe construction with a welded bottom of tapered stainless steel. The two designs differ in the sealing mechanisms at the head of the canisters. In one (Reference 5) a mechanical seal is used, while the other canister is seal-welded. The mechanically sealed canister would be indefinitely reusable, but the sealing mechanism reliability may be in question. The welded lid design is configured to allow the welded area to be mechanically removed and rewelded approximately ten different times. This design should provide an extremely reliable sealing mechanism.

Drop testing of the mechanically sealed canisters with surrogate fuel and coolant has been initiated. In the first test, which was a 9 m, end-on drop onto an unyielding target, the canister was filled with an 80-percent glycerine/20-percent alcohol mixture serving as the sodium surrogate. This mixture provides the same acoustic impedance as liquid sodium at 450°C. The test was conducted with the canister and contents at an ambient condition of approximately 24°C. Following impact, the canister appeared to be structurally sound with no buckling or cracking. Drop tests from 9 m have also been performed with the canister impacting side-on, and in a "slap-down" orientation with the bottom end of the canister impacting first. In these tests significant canister deformation was observed, but no cracks were found nor was there any leakage of the liquid from the canister body. However, the mechanical seals did fail, causing leakage of the surrogate coolant. Further tests at lower drop heights will be performed to define the failure threshold of the seals, and the thresholds can then be analytically related to the performance of the canister in the cask during impact. Based upon the drop tests to date, it is postulated that the seal-welded canister should survive a 9 m drop independent of the cask and of impact orientation.

Heat transfer analyses of the CRBRP and CDS conceptual designs were used to examine operating temperatures under normal solar input at 55°C ambient. The assembly heat output is assumed to be 7 kW for CRBRP with a backfill gas of helium between the canisters and the cavities after the cask is sealed. The CDS cask temperatures were examined with helium and nitrogen as backfill gases. The heat load used for

CDS is a ring of six 11 kW assemblies with one 4 kW core component in the center cavity to prevent overheating at that location. Resulting cask temperature profiles are shown in Table I. No analysis has been attempted to accurately estimate pin temperatures inside the assembly. However, it is felt that, based on data from References 1 and 2, the additional rise from the canister temperature to the maximum pin temperature is on the order of 30 to 50°C for the decay heat outputs considered.

An emissivity of 0.3 was generally assumed for the canister and cavity surfaces. Higher canister temperatures were calculated for the nitrogen-backfilled cask than for the helium-backfilled cask. This can be offset by increasing the emissivity of both canister and cavity as shown in Table I, which could be accomplished with various surface treatments.

One of the problems addressed has been the method of canister loading. Because of sodium expansion and cover gas pressure buildup in the canister, the simple method of loading a hot assembly into liquid sodium at or just above sodium melting temperature was rejected. Instead, heating the canisters and sodium to the approximate normal operating temperature before loading is proposed. The operating temperature in this case is defined by the maximum cask load of 49 kW for CRBRP and 70 kW for CDS (see Table I). The amount of sodium contained in each canister is governed by accident (off-normal) conditions. The cask analysis in a fire environment has not been completed. However, calculations using a bounding, i.e. worst case, condition of 676°C (1250°F) for the fuel cladding (and therefore the sodium) and a 10 atm pressure in the canister cover gas, indicate a sodium level at loading should be 0.254 m below the top of the CRBRP assembly. Further analysis of the cask response to an 800°C thermal source, which is discussed later, suggests that the actual temperature response may be less severe and that the sodium level might only need to be 0.038 m below the top of the assembly. Data from Reference 1 show that the heat rejection in the vertical position is not significantly affected if the sodium is below the top of the assembly. An alternate loading method would be to heat the sodium to a temperature greater than that expected during normal operation. This would result in a gas pressure lower than 10 atm in the canister at off-normal (i.e., accident) temperatures and subatmospheric pressure during normal shipping.

Table I. Axial Center Cask Temperatures
During Steady State Operation

CASK CONFIGURATION	CRBRP	CDS	CDS	
Spent Fuel Assemblies	Seven, 7kW	Six, 11kW One, 4kW	Six, 11kW One, 4kW	
Cask Cavity Cover Gas	Helium	Helium	Nitrogen	
Canister/Cavity Emittance	0.3	0.3	0.3	0
Cask Surface Temp (°C)	170	190	190	1
N-Shield/Steel Body Interface Temp (°C)	290	330	330	330
Outer Cavities Temp (°C)	330	385	385	385
Outer Canisters Temp (°C)*	405	465	535	459
Inner Cavity Temp (°C)	375	440	450	439
Inner Canister Temp (°C)*	430	460	495	453

*Maximum fuel pin temperature would be 30 to 50°C above the canister temperatures shown (see References 1 and 2). Maximum allowed pin temperature is 650°C.

The requirement of maintaining cask-handling surface temperatures below 60°C (140°F) presents difficulties in both loading and unloading modes. The loading sequence assumed was that one canister would be loaded per hour. The large thermal mass of the cask significantly delays surface temperature rise during loading. However, external natural convection alone is insufficient to prevent overheating while the cask is in plant. Surface temperatures were calculated to exceed 60°C (140°F) in 9 to 12 h after loading is initiated. This time is insufficient to allow for cask loading and sealing operations and removal from plant.

To provide a feasible method of cooling during loading and unloading, internal channels can be provided in the cask body. The assumed coolant flow pattern is one pass up and down, then out. Twelve pairs of channels, each with a cross-sectional area of 7.39×10^{-4} m^2 for the CRBRP cask and 1.098×10^{-3} m^2 for the CDS cask are placed between the steel structure and neutron shield. Thus the coolant channels are outside of the containment boundary and are near the cask surface, allowing the surface temperatures to be reduced rapidly by the pumping of coolant. The coolant inlet and outlet are at the bottom of the cask, with the channels extending upward to just below the lid.

For the CRBRP cask, calculations showed that a flow rate of 288 litre/min (76 gal/min) of 20°C (67°F) water in the channels maintains the surface temperatures of the cask below the 60°C requirement during loading. For the CDS cask, calculations showed that a flow rate of 490 litre/min (130 gal/min) of 20°C water maintained the surface temperatures below 60°C for approximately 42 h; and a flow rate of 980 litre/min (260 gal/min) maintains the surface temperatures below 60°C indefinitely. Further effort is required here to assess methods for enhancing the heat transfer to the water channels and reducing the required flow rates. The results of these calculations are summarized in Table II. Because it was assumed that the cooling channels did not extend into the lid, this portion of the cask could heat up during bolting and sealing operations. This problem is discussed later.

As indicated by the temperature profiles in Table I, the cask on arrival at the HEF will have a surface temperature far above the allowable 60°C. External air cooling at reasonable flow rates is ineffective in reducing surface temperatures following shipment. External water cooling improves this somewhat, although access to the cask surface is then limited. Furthermore, because large quantities of heat are stored in the cask body, external water cooling cannot be suspended during the unloading operation without rapid surface temperature rise to unacceptable levels.

Table II. CRBRP and CDS Cask Heat-Up During Loading

CASK TYPE	COOLING METHOD	CAVITY BACKFILL GAS	TIME TO REACH 60°C AT SURFACE
CRBRP	Natural Convection	He or Ar	9 h
CRBRP	288 litre/min of 20°C water	He or Ar	Steady state surface temperature less than 60°C
CDS	Natural Convection	He or N_2	12 h
CDS	490 litre/min of 20°C water	He or N_2	42 h
CDS	980 litre/min of 20°C water	He or N_2	Steady state surface temperature less than 60°C

Several modes of cask cooldown using internal water flow have been examined. Only the most promising method is reported here. The water flow would initially enter at 10 atm, 179°C (354°F), and would utilize the heat of vaporization in extracting heat from the cask. When the coolant exit temperature falls below 180°C (355°F), the inlet water temperature would be lowered to 95°C (202°F), while still maintaining the pressure. Once the exit temperature falls below 100°C (212°F) the pressure may be reduced and water at 20°C (67°F) introduced. In addition to the internal water cooling, external 20°C forced air cooling over the lid only was required for reasonable cooldown times.

The CRBRP cask cooldown results show that 25 to 53 h are needed to attain surface temperatures under 60°C. The calculated base case was 326 litre/min (86 gal/min) with no external air cooling on the lid, which corresponds to 53 cooldown. For the base case, the lid was the slowest cooling portion. Both t air flow rate over the lid and the water flow rate in the cask were varied to assess those flow rates which lead to shorter cooldown times. The greatest assumed velocity of air over the lid for CRBRP was 12.2 m/s, and the greatest flow rate of water was 974 litre/min (257 gal/min), which gave a 25 h cooldown. The results are shown in Table III.

The CDS cask cooldown analysis indicates times of 27 h to greater than 72 h. Once again, the longest time to cool resulted from the calculated base case of 480 litre/min (128 gal/min) internal water cooling with no forced external cooling. To obtain 27 h cooldown, internal water cooling of 960 litre/min with air velocity on the lid of 15.4 m/s was required. These results are also presented in Table III.

With the internal cooling of the cask body only, the lid is the slowest portion of the cask to cool. The method considered here to improve cooling times was external air cooling of the lid. Another option, separate cooling channels in the lid, could improve the cooling times even further. From examination of the cooldown of the base sections, which, like the lid, have no heat source nearby, it may be safely assumed that lid cooldown could be eliminated as the controlling factor in cooldown time if such internal cooling channels are used.

TABLE III. Cask Cooldown at HEF

CASK CONFIGURATION	AIR VELOCITY OVER LID (m/s)	WATER FLOW RATE IN CASK CHANNELS (litre/min)	TIME TO COOL HANDLING SURFACES TO 60°C (h)
CRBRP	0	326	53
	6.1	326	42
	12.2	326	35
	6.1	652	33
	12.2	652	27
	6.1	974	31
	12.2	974	25
CDS	0	480	>72
	7.6	480	69
	12.2	480	40
	15.4	480	35
	7.6	960	68
	12.2	960	33
	15.4	960	27

Initial calculations of the response of the cask to the regulatory thermal test exposures have been completed. Maximum canister temperatures were found to occur 9 to 10 h following initial exposure to the 800°C (1475°F) thermal source. Table IV compares the maximum canister temperatures during normal operation and following exposure to the thermal test environment. From these data it can be concluded that the thermal test does not control the design.

TABLE IV. Comparison of Maximum Canister Temperatures for CRBRP and CDS Casks

	NORMAL OPERATION (°C)	THERMAL TEST (°C)
CRBRP Cask	430	455
CDS Cask, Helium Backfill	465	480
CDS Cask, Nitrogen Backfill	535	545

Beginning in October 1980, a detailed conceptual design study was initiated for a CRBRP spent fuel shipping cask. The study is based upon the results of the efforts described above. The detailed concept for the cask should be completed in late 1981.

CONCLUSION

In conclusion, the study of conceptual shipping systems for CRBRP and CDS spent fuel has shown that systems significantly different from those used for LWR spent fuel will be required. In the conceptual design, liquid sodium was assumed to be the coolant in canisters containing the spent fuel assemblies, and multiple levels of containment were provided by canisters, an inner cask lid and an outer cask lid. Cask cooling at the reactor site during loading, and cooldown at the receiving site prior to unloading are signficant but tractable problems.

REFERENCES

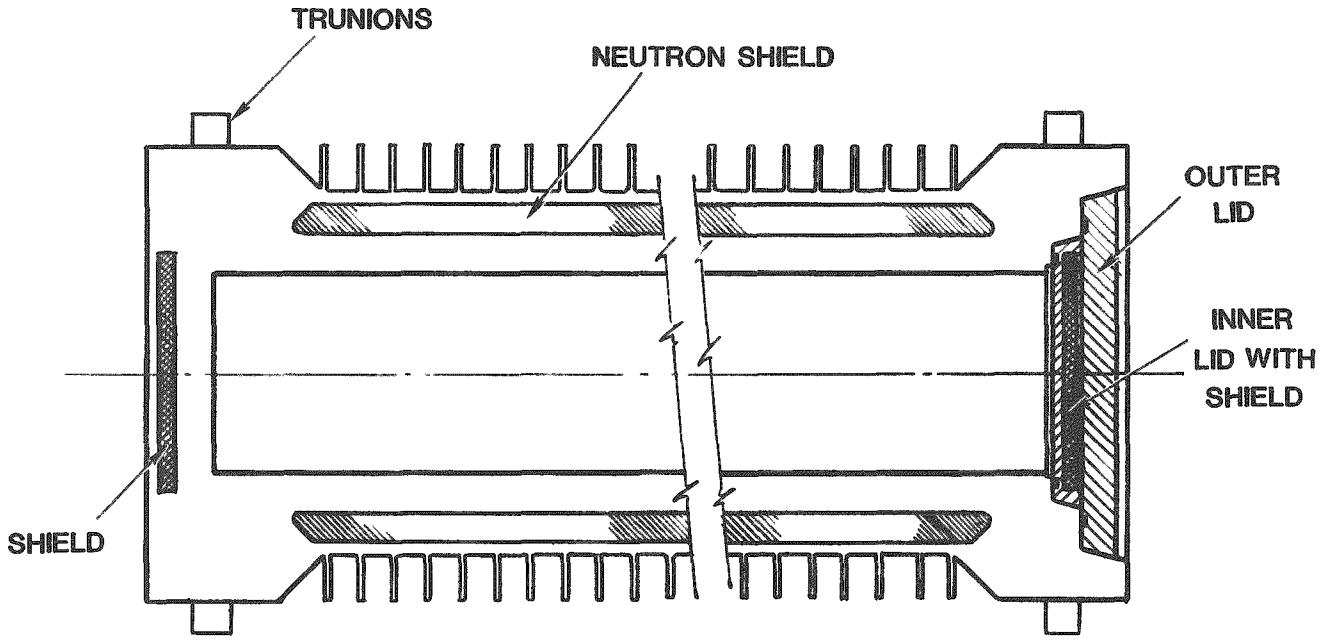
1. M. Prussman, B. Rapple, and E. Weinert, "Verusuche zur Warmeabfuhr von einem SNR-Brennelement in einer mit Natrium gefullten Transportfuchse," Gesellschaft fur Kernforschung mbH, KFK-2174, August, 1975.

2. B. B. Klima, "LMFBR Spent Fuel Transport: Single Assembly Heat Transport Test," Oak Ridge National Laboratory, ORNL-TM-4936, October 1975.

3. R. B. Pope, "Evaluation of Heat Transfer in Conceptual Designs of Spent Fuel Shipping Casks for the United States Breeder Technology Program," SAND77-1781, May 1978.

4. R. B. Pope, G. C. Allen, H. J. Rack, B. J. Joseph, and S. A. Dupree, "Developmental Testing of Partially Volatile Neutron Shields for High-Performance Shipping Casks," Presented at PATRAM '80, November 10-14, 1980, Berlin, FRG.

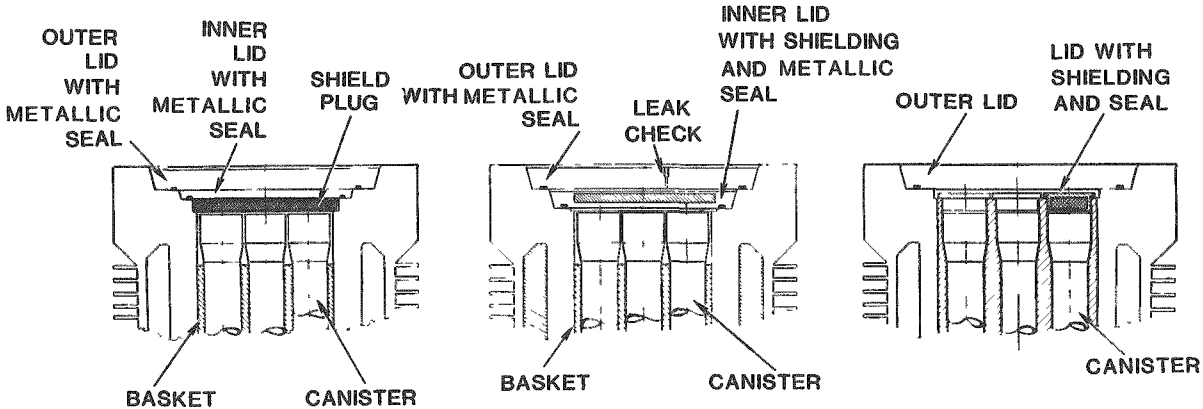
5. G. C. Allen, Jr., et al., "Conceptual Designs for LMFBR Shipping Casks," SAND77-1483, Sandia Laboratories, Albuquerque, NM, March 1978.



CASK CONFIGURATION

	CRBRP	CDS
SIZE	1.7mDX 5.1mL	2.1mDX 6.6mL
WT.	55 TONNES	88 TONNES

Figure 1



SEVEN ELEMENT CASK WITH SEPARATE SHIELD PLUG

SEVEN ELEMENT CASK DESIGN WITH TWO LIDS

SEVEN ELEMENT CASK WITH INDIVIDUAL INNER LIDS AND SHIELD PLUGS

Figure 2

DISCUSSION

Question by C. R. Davis, USA: 1. Why was an 11 kW decay heat level pin fuel assembly for CDS chosen for design purposes? 2. Will future design work use CDS fuel handling criteria for decay heat levels?

Answer: 1. This was the guidance provided at the time the study was initiated. 2. No future work supporting the CDS project is planned due to lack of funding.

Question by A. Onodera, Japan: What is the material used for gamma shielding? 2. Why was the canister drop tested while it is not contained in the cask cavity? 3. Do you intend to use depleted uranium?

Answer: 1. Steel, as shown in the paper. 2. The failure threshold of the canisters is currently being defined without the cask. The response of the canister to impact environments defined in this manner will then be coupled analytically with the cask to define its response while contained in the cask.

Question by H. Keese, FR Germany: You report in your paper about a solid neutron borosilicone rubber shield surrounding the cask body. How do you fix this borosilicone rubber in the cask body or between the cask body and the fins?

Answer: The neutron shielding material is injected into cavities in the external structure of the cask body, solidified in place, and then the holes through which it was injected are sealed with fusible plugs. For further information I refer you to the PATRAM '80 paper number 121 in Session IV.

Question by J. D. Hart, United Kingdom: 1. Is it necessary to have each of the three containment features independently surviving accident conditions? 2. Are you satisfied with the performance of metallic seals under accident conditions?

Answer: It is assumed that three separate levels of containment will be useful in the licensing process since sodium is proposed as the primary coolant. It will also help in the public acceptance of the cask. 2. Sandia National Laboratories currently has a test program underway to define the performance of metallic seals. I cannot answer your question at this time. The test program which is planned is described in PATRAM '80 paper number 151 in Poster Session A.

Question by R. Christ, FR Germany: Please comment on the rather high surface temperature of the cask.

Answer: The surface temperatures could be lowered by adding additional fins to the cask external surface. With the design concepts described here, the high surface temperature did not constrain the predicted operation in terms of excessive pin temperatures, although it does add to the in-plant heat-up and cool-down problems. Additional fins will be considered in future cask designs.

SESSION III
RISK/SAFETY ASSESSMENT I

RADTRAN II - A COMPUTERIZED MODEL FOR RISK ANALYSIS OF TRANSPORTATION OF RADIOACTIVE MATERIAL

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The RADTRAN computer code [1], which formed the basis for the 1977 U.S. generic transportation risk assessment, has been extensively updated. The updated version of the code [2], denoted RADTRAN II [3], includes changes based on findings from other transportation risk studies as well as changes based on reevaluation of earlier assumptions, analyses, and computerization techniques.

The environmental impact of the transportation of radioactive material can be envisioned as consisting of five components as shown in Figure 1. RADTRAN II is designed to evaluate both the incident-free and the accident contribution directly and can be used to evaluate the contributions of breaches of security and quality assurances deviation if some alterations in coding are made. Non-radiological impacts are not addressed.

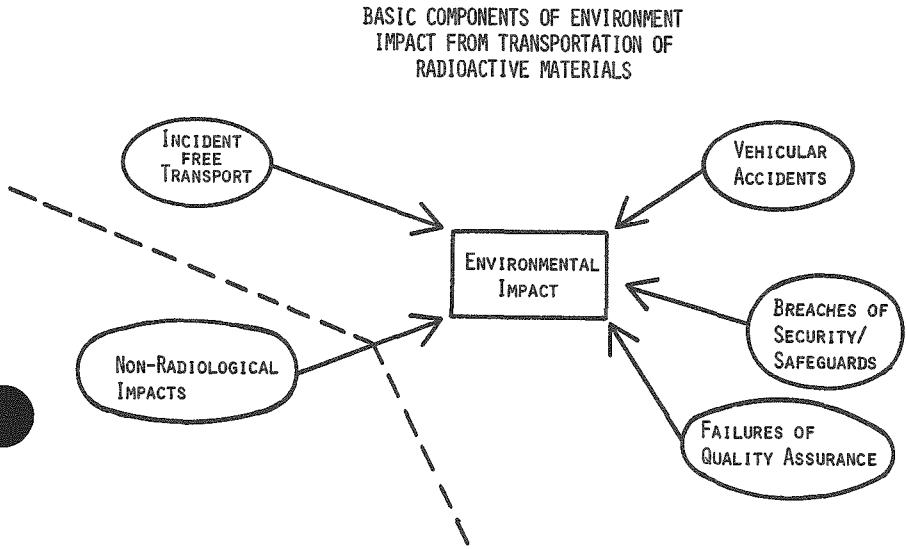


Figure 1

The contribution to risk from incident-free shipments is apportioned among seven population subgroups and is separately calculated for several transport modes. In addition, occupational exposure to crew and handlers of exclusive-use shipments is distinguished from non-occupational exposure. Example of the population subgroup and mode combinations are shown in Figure 2.

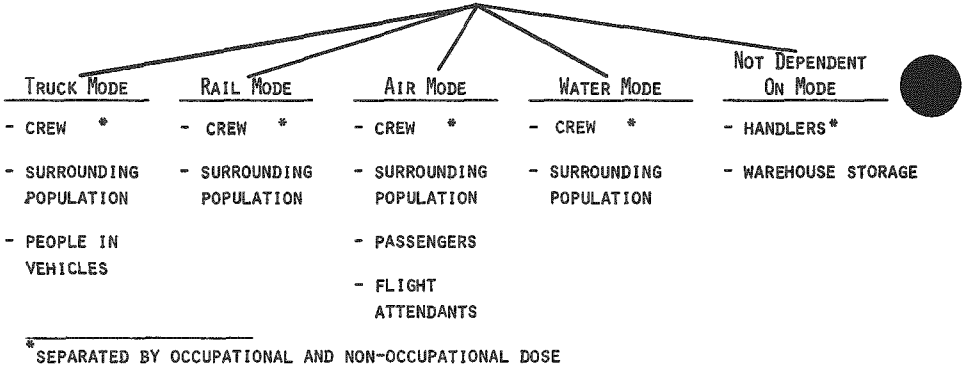


Figure 2

The calculational scheme for incident-free dose is based on a point-source geometry in most cases (line-source geometry is used for close-in exposures to large casks). The source term uses the measured value for transport index (TI) provided by the shipper. Four basic exposure geometries, shown in Figure 3, form the basis for the integrated close calculations. The figure-of-merit used for comparison and analysis of results is total integrated dose in person-rem.

The radiological hazard from accidents can occur via several dose pathways as shown in Figure 4. RADTRAN II only evaluates those pathways shown in blocks. The remaining dose pathways can be shown to be insignificant in transportation accident scenarios [2,4,5]. The calculations are based on the curie content of each shipment and the figures-of-merit are the expected numbers of various acute and chronic human health effects.

The calculations of radiological impact from accidental dispersals use atmospheric dispersion data generated by Gaussian diffusion models to predict downwind airborne concentrations and ground deposition. Standard dosimetric conversion factors are applied to convert airborne concentrations to expected organ doses from inhalation. External exposure from ground contamination is analyzed using an infinite plane source model and assuming that authorities either interdict a contaminated area or decontaminate to agreed-upon contamination levels.

Basic Exposure Situations

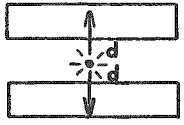
Situation	Description	Geometry	Exposed Subgroups	Basic Integrated Dose Relationship
1	Fixed distance		Crewmen, handlers of large packages	$\propto TI \cdot \frac{DIST}{VEL} \cdot \frac{1}{d^2}$
2	Storage, stops		Warehouse storage, in-transit stops	$\propto TI \cdot LN\left(\frac{r_2}{r_1}\right) \cdot \Delta T$
3	In-Transit		Surrounding Population, persons in vehicles	$\propto TI \cdot \frac{2\pi}{VEL} \cdot \frac{1}{d}$
4	Measured Dose Rates	Various	Airline passengers, flight attendants, handlers of small packages	$\propto K \cdot TI$

Figure 5

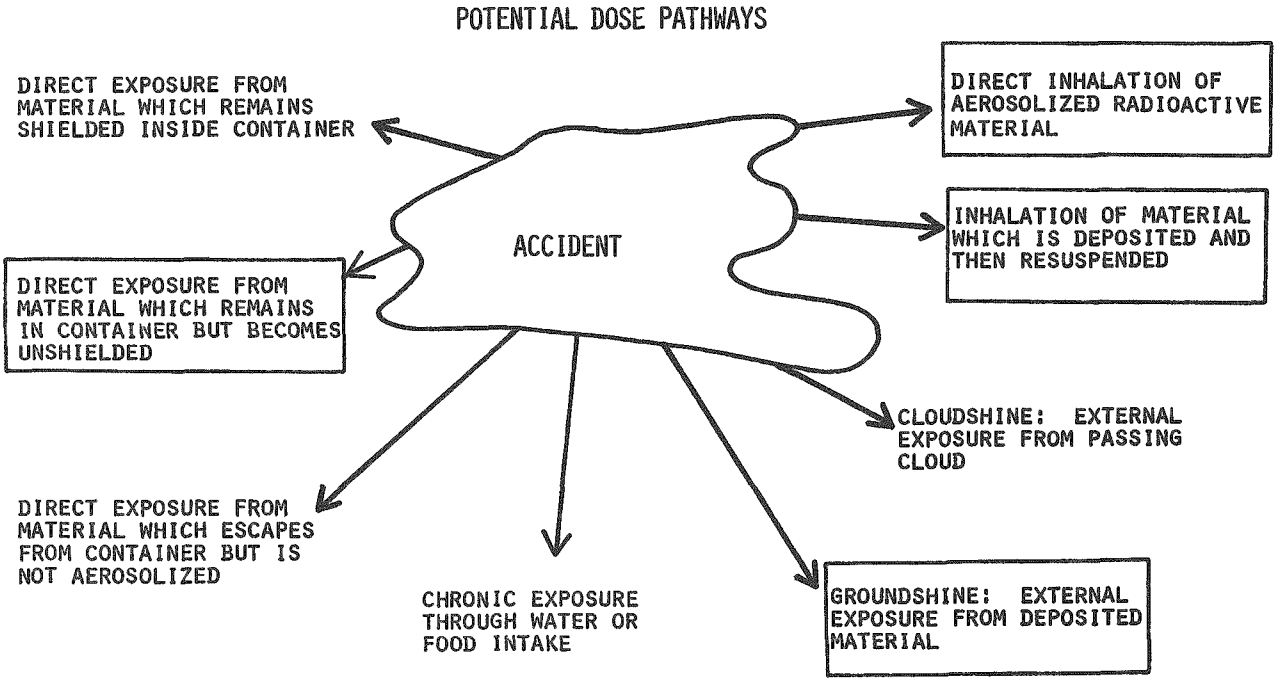

POTENTIAL DOSE PATHWAYS
DIRECT EXPOSURE FROM MATERIAL WHICH REMAINS SHIELDED INSIDE CONTAINER
DIRECT INHALATION OF AEROSOLIZED RADIOACTIVE MATERIAL
INHALATION OF MATERIAL WHICH IS DEPOSITED AND THEN RESUSPENDED
ACCIDENT
DIRECT EXPOSURE FROM MATERIAL WHICH REMAINS IN CONTAINER BUT BECOMES UNSHIELDED
CLOUDSHINE: EXTERNAL EXPOSURE FROM PASSING CLOUD
DIRECT EXPOSURE FROM MATERIAL WHICH ESCAPES FROM CONTAINER BUT IS NOT AEROSOLIZED
CHRONIC EXPOSURE THROUGH WATER OR FOOD INTAKE
GROUNDSHINE: EXTERNAL EXPOSURE FROM DEPOSITED MATERIAL

Figure 4

Accidents involving non-dispersible materials which may become unshielded in an accident are analyzed in an analogous manner to that used for the incident free dose calculations. In this case, however, the TI value assigned to a package is replaced by the curie content of the package.

A key factor in the analysis of accidents involving either dispersible or non-dispersible materials is the degree to which packaging integrity and/or shielding integrity is degraded. Also of importance in the analysis of accidents involving dispersible materials is the amount of the material which is released in aerosol form. In RADTRAN II, both parameters are specified as a function of the severity of the transportation accident environment encountered. This severity classification scheme, which is similar to that used in earlier reports [2,4], assigns occurrence probabilities to each of eight accident severity categories as a function of the transport mode employed.

The overall accident calculation can be visualized as shown in Figure 5. Note that both risk and consequence calculations are performed by RADTRAN II. This allows specific scenarios to be examined in some detail while, at the same time, allowing a large-scale analysis of overall risk which includes the results from many additional scenarios.

A sensitivity analysis is being incorporated into the RADTRAN II model to determine the impact of each model parameter on the results. The technique uses weighted values of partial derivatives to provide a measure of the importance of each parameter to the results. This information can be employed by users of the code in optimizing their specific data collection endeavors.

RADTRAN II has been designed as a flexible analytical tool. The code can be applied to specific scenario evaluations or model comparisons as well as to generic problems such as the ongoing IAEA worldwide transportation risk assessment. Initial evaluations of the results of RADTRAN II analysis on the shipment data set used in the preparation of the U.S. generic risk assessment [2] show good agreement. In general, RADTRAN II values are slightly higher for incident-free dose and slightly lower for accident doses. These variations are attributable to modeling improvements in RADTRAN II over the earlier RADTRAN code.

REFERENCES

1. "RADTRAN: A Computer Code to Analyze Transportation of Radioactive Material," Taylor, John M. and Daniel, Sharon L., Sandia National Laboratories, Albuquerque, New Mexico, April 1977, SAND76-0243.

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

3. "RADTRAN II: A Revised Computer Code for Analysis of Transportation of Radioactive Materials," Taylor, John M., Daniel, Sharon L., Luna, Robert E., Biringer, Betty E., Sandia National Laboratories, Albuquerque, New Mexico, SAND80-1943, To be published.

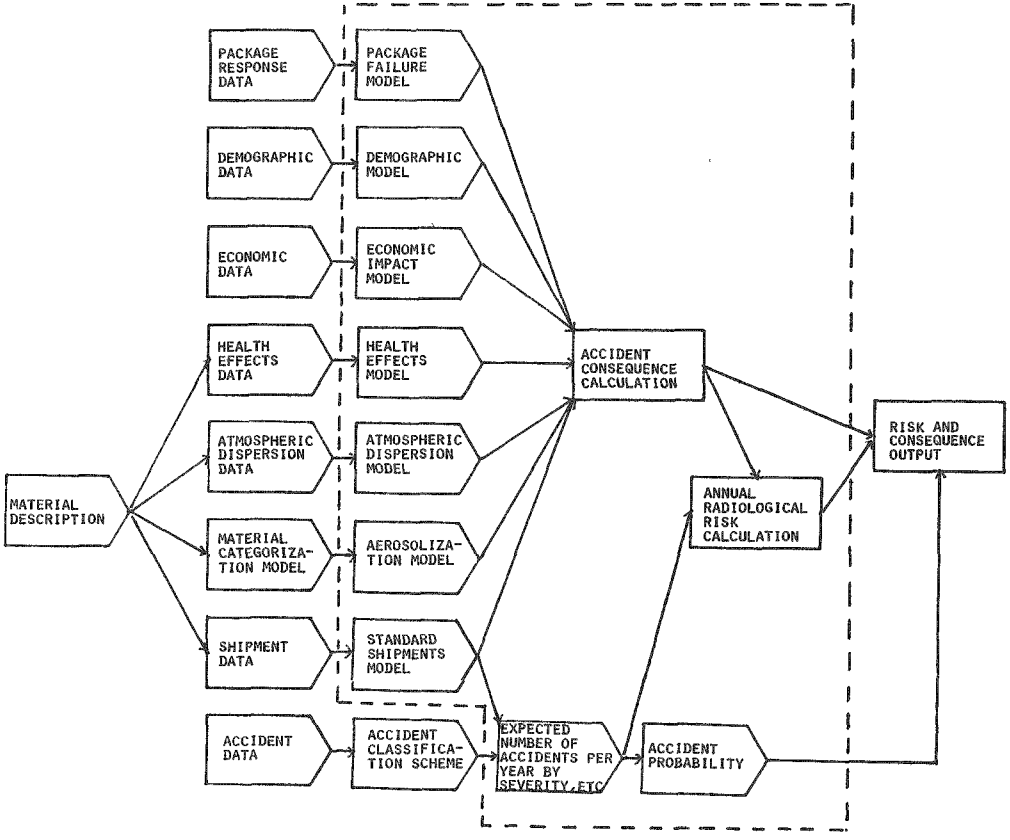


Figure 5

FLOW DIAGRAM FOR
ACCIDENT CALCULATIONS

4. "Transportation of Radionuclides in Urban Environs: Draft Environmental Assessment," Finley, N. C., et al., Sandia National Laboratories, Albuquerque, New Mexico, July 1980, NUREG/CR-0743 (SAND79-0369).

5. "Calculation of Reactor Accident Consequences (Appendix VI to Reactor Safety Study)," U.S. Nuclear Regulatory Commission, Washington, DC, October 1975, WASH-1400.

RADIOACTIVE WASTE TRANSPORTATION RISK

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The work presented in this paper is performed as part of a Nordic project funded by the Nordic Council of Ministers.

In a joint Nordic project a reference waste handling system is analyzed in order to show the interrelation between the various steps: temporary storage, transportation, and disposal, to correlate them with the necessary waste product specifications, and to show how possible radiation doses to humans can be assessed.

Calculations of risk are used as a tool for planning overall waste handling systems or judging acceptability of proposed solutions. Parts of the risk assessments have been carried out in Finland, Norway and Sweden. A Finnish study (Ref.1 and 2) evaluated the probability of and release resulting from road accidents. Swedish studies present the risk from boat transportation (Ref.3) and the risk related to permanent storage (Ref.4). A Norwegian study is concerned with the risk from road transportation of radioactive waste, and the risk related to a fire in the temporary storage facility.

The present paper is mainly concerned with the Norwegian study. The waste will be transported in drums with resin incorporated in concrete or bitumen. In other parts of the present Nordic project it has been assessed that the fragmentation of bitumen will be insignificant compared to that of concrete. Likewise it has been assessed that fire damage to concrete will be insignificant compared to that of bitumen. It is also found that the atmospheric release following fragmentation of concrete will be small compared to the release accompanying a bitumen fire. Accordingly the largest doses and highest conditional risk are related to the bitumen fire case. Whether this is true also for the absolute risk, is not yet known, since the probability of a fire following a collision is not yet determined. The atmospheric release following a fire in the temporary storage is comparable to the release accompanying the fire case.

Due to the close similarity of the cases, fire in the temporary storage has also been analyzed, and some of the results from these calculations will be presented in this paper.

The methodology used in the calculations has been adopted from the field of nuclear reactor accident analysis, using the most advanced methods available at the present time, but some modifications have been introduced to adapt the methods to Scandinavian conditions. This concerns meteorology and population, but also some agricultural data needed for calculation of the doses via nutrition.

RISK FROM TRANSPORTATION

The risk from transportation of radioactive materials in Sweden has been calculated in a report (Ref.3) for the Swedish Nuclear Power Inspectorate.

The report is designed to calculate the total risk due to shipping of radioactive materials in Sweden. As Sweden is planning to ship most of the nuclear fuel cycle materials by ship, the basic shipment model contains mostly sea transportation of this kind of materials. Alternative shipment models have also been studied.

The results from the calculations show that, for the basic shipment model, the additional population dose in Sweden due to both normal and accident conditions in the transportation of radioactive materials will be 30-50 manrem per year.

ROAD COLLISION, FIRE

In order to evaluate the consequences of road collisions, drop tests were performed using small blocks and full scale drums with resins incorporated in concrete and bitumen (Ref.1). The probalibity of a traffic accident has also been evaluated.

A fire either in the temporary storage or during transportation needs to be evaluated with respect to its possible consequences. In order to assess the amount of activity that can be released, fullsize tests were performed (Ref.2). The probability of a fire has however not been evaluated.

FIRE ACCIDENT SCENARIO

The waste considered consists of granular ion exchange resin from the reactor water clean-up of a BWR, incorporated in bitumen and packed in standard 200 litre steel drums. The assumed nuclide content per drum after 5 years decay in the temporary storage is as follows: 1.4 Ci of Co-60, 0.81 Ci of Cs-137, 0.18 Ci of Cs-134 and 0.04 Ci of Sr-90.

For transportation by truck, the drums are placed in transport containers made from reinforced concrete. The containers take 8 drums each and provide sufficient radiation shielding for road transportation. The accident is assumed to breach severely <u>one</u> container and to involve an oil and gasoline fire lasting long enough to ignite the resin/bitumen content of the 8 drums in the container (experiments in Finland have indicated that drums filled with bitumen do not start burning unless exposed to an oilfire through 15-20 minutes).

It is assumed that the fire will result in release of 100% of the cesium content in gaseous form. Of the remaining nuclides, 15% will remain in the ashes, 60% will be released as gases or as particles less than 10μ. The fire temperature is about 1000°C. The fire will last 1 hour and the energy release will be 3 GJ.

For fire in the temporary storage almost identical conditions are assumed. The waste drums are stored in wells under the concrete floor of the storage in such a manner that it is very unlikely that a fire could involve more than 8 drums. This scenario differs from that of the transport accident since the heat released during the fire is given off to the structural materials of the storage building, so that the temperature of the release will not significantly exceed

the ambient temperature. Also the fire in this case may involve "fresh" waste. The actual calculations are performed assuming "fresh" waste, as this will give the highest doses and risk.

CALCULATION METHOD

Doses and risk have been calculated using the computer program CRAC, which was originally developed for use in the American Reactor Safety Study (Ref.5). The following exposure pathways are included in the calculation:

- External irradiation from passing cloud.
- Inhalation of radioactive materials from passing cloud.
- External irradiation from radioactive materials deposited on ground.
- Inhalation of resuspended radioactive materials.
- Intake of milk, vegetables and meat contaminated due to deposition of radioactive materials on grass and other plants.
- Intake of milk, vegetables and meat contaminated due to root uptake of radioactive materials.

All the calculated doses are effective whole-body dose commitments. The effective whole-body dose is calculated from the doses calculated by the program to the bone marrow, lung, breast, bone, thyroid, gonads, digestive tract, and other tissue. This is explained, and the numerical values used are given in (Ref.6).

Some of the calculations give individual dose commitment as function of distance as result; some of the calculations give the risk, expressed as collective dose commitment. All dose commitments are calculated over all future, but since the half-life of the most long-lived isotope included in these calculations is about 30 years, the doses are in effect integrated over roughly a life time.

The collective doses are in principle integrated over all space (over the global population). This is however done in a rather rough and somewhat conservative manner. Both these aspects are described at further lenght in (Ref.5 and 7).

AGRICULTURE AND NUTRITION DOSES

For the first four of the exposure pathways listed in the preceding chapter, the collective dose in a certain land area will be proportional to the number of persons living in that area.

This is however not true for the nutrition doses. The contaminated milk, vegetables and meat will not necessarily be consumed by the persons living in the area. The individual doses received and the number of persons involved will depend upon the way in which the milk etc. is distributed. This will be different from one area to another, and will vary with the seasons. But while it is difficult to estimate the individual doses, it is relatively simple to calculate the collective doses.

The reasoning behind the calculations is simply that a certain amount of food has the possibility of causing a certain collective dose. This collective dose may be distributed in different ways; a low individual dose to a large number of persons or a high individual dose to a few persons, but the collective dose will be the same, regardless how it is distributed. In the present calculations the collective dose is calculated in the following way:

- calculate an individual dose, assuming that a person covers all his need of a certain type of food with food from the contaminated area.
- determine the number of persons who could be fed from the contaminated area, using the above assumption.
- the collective dose is the product of the calculated individual dose and the number of persons.

In the present calculations it is assumed that 570 m^2 pasture land supplies milk to one person, and that 6000 m^2 is needed to supply all other agricultural products to one person.

Further information about these calculations is found in (Ref.7).

DOSES AND RISK

Two quite different types of results are presented in this paper. Individual doses as function of distance from the release point - and the risk associated with a certain release.

When individual doses as function of distance shall be calculated, the conditions must be "frozen". The calculation must be performed for one specific release, and one specific weather condition. In the present case we have performed several calculations where these conditions are "frozen", but the conditions differ from one calculation to another. Calculations have been performed for two different releases; release from a fire in the temporary storage, and fire during transportation. Calculations have been performed for two weather conditions; Pasquill C and 3 m/s, and Pasquill F and 1 m/s. The calculations could have been performed for other weather conditions as well, but it is felt that these two weather conditions give an idea of the way the doses may vary from one set of weather conditions to another.

When the risk is calculated it is necessary to take the population distribution into account. In the present calculations the consequence of a release is expressed as the resulting collective dose. The consequence might be expressed in other ways, e.g. the number of cases of cancer or genetic damage expected to result from the release (using certain assumptions regarding the relationship between cancer/genetic damage and the dose). The collective dose is the sum of all individual doses resulting from the release. The individual dose will depend upon the actual geographical position of the person relative to the release point, and the population distribution must therefore be known if the collective dose is to be determined.

The individual dose in a certain geographical position will also depend upon the weather conditions. For a certain location there is a certain weather pattern. Even though the actual weather conditions will vary, certain types of weather are more probable than others. A weather statistic for the location will tell the probability of certain types of weather and also of the wind directions. When the risk is calculated, these probabilities should be taken into account. Consider the case in which there is a large population concentration in a certain direction from the release point. If it very seldom blows in that direction, the risk will be less than if such a wind direction is frequent.

The weather statistic may be taken into account with a varying degree of refinement. In the present calculations a whole year of meteorological data on an hour-by-hour basis is stored in a data file. For each hour throughout the year is stored the atmospheric stability, the wind speed and information on whether there is precipitation or not. Wind direction information is included in a simpler manner, as four frequency distributions, one for each of the seasons.

It would be possible to calculate the consequence of the release (the collective dose) for each one of the 8760 hours in the year, and construct a probability distribution from the results. The computer time required is however in many cases prohibitive, and in the present case calculations have been performed for 91 different hours, distributed over the year in a semi-random manner, and each result has a probability of (1/91). Such a probability distribution is often called a risk curve.

CALCULATION CONDITIONS

The present calculations are performed using the data specified in this chapter. Additional information on data and methods are found in (Ref. 5 and 7).

Release

From temporary storage:

18.0 Ci Co-60
0.31 Ci Sr-90
7.2 Ci Cs-134
7.2 Ci Cs-137

From transportation:

9.5 Ci Co-60
0.27 Ci Sr-90
1.3 Ci Cs-134
6.4 Ci Cs-137

Further information in chapter Fire Accident Scenario.

Duration of release

1 hour.

Release heat content

None when the release takes place from temporary storage.
$3.1 \cdot 10^6$ cal/sec when release takes place during transportation.

Size of building and truck

The building has height 10 m and width 30 m.
The truck has height 3 m and length 10 m.
These dimensions will determine the initial spread of the plume.

Agricultural area

A distribution of inhabited area around the release point is given in 16 sectors and 34 distances from release point out to a distance of 500 km.

The distribution is identical to the one used in the calculations for the *Norwegian Government Commision on Nuclear Power* (Ref 7 and 8), and is typical of the Oslofjord area.

In addition it is assumed that 76% of inhabited area is agricultural area, and that 8,6% of agricultural area is pasture land.

A constructed distribution is used in some additional calculations performed in order to examine the case where a transport accident takes place in the middle of a large city. This distribution is assumed to contain no agricultural area within a radius of 9.5 km. The remainder of the distribution is identical to the previous one.

Population

A population distribution around the release point is given in the sectors and distances mentioned above.

The population distribution is also identical to the one used in the calculations for the Norwegian Government Commission on Nuclear Power (Ref. 7 and 8), and is typical of the Oslofjord area.

Like for the agricultural distribution, a population distribution is constructed for use in calculations performed in order to examine the case where a transportation accident takes place in a large city. In this case, the population density within a radius of 9.5 km is assumed to be 1763 persons/km^2, corresponding to a city of $\frac{1}{2}$ million inhabitants and a radius of ca. 10 km. The remainder of the population distribution is kept identical to the above.

Deposition

Dry deposition is calculated using a deposition velocity of 1 cm/sec.

Wet deposition is calculated using a deposition coefficient of 0.0001 sec^{-1}.

Meteorological data

All meteorological data are identical to the ones used in the calculations for the Norwegian Government Commission on Nuclear Power (Ref. 7 and 8), and are typical of the Oslofjord area.

RESULTS

Waste in bitumen

The results from the dose calculations performed for specific weather conditions are shown in figures 1 - 4. Figures 1 and 2 show the effective wholebody dose commitment as function of distance from the release point for two types of weather conditions. Figure 1 is for release accompanying a fire following a collision during transportation. Figure 2 is for release accompanying a fire in the temporary storage.

The individual effective wholebody dose commitments are found from the calculations performed for specific weather conditions to have maximum values of about 1 rem when the release takes place during transportation and about 10 rem when the release takes place from the storage building. There are two reasons for this difference. It is assumed that transportation takes place not earlier than after five years storage in drums, which gives a certain decay. And the release from the building will have not thermal buoyancy.

Figures 3 and 4 show details of the results, namely the dose due to each one of the four isotopes, compared to the total release. This break-up of the results has been performed only for one of the weather conditions, and only for release from the temporary storage. Notice that although one isotope, Sr-90 gives an insignificant contribution to the total dose, none of the other three are dominant.

Another break-up of the results has also been performed. It has been determined for one of the weather conditions, and release from the temporary storage, how the different exposure pathways contribute to the total dose. The results are shown in table 1, where the percentage of the dose received via each exposure pathway is shown.

TABLE 1. Percentage of dose received via each exposure pathway

External radiation from passing cloud	0.005 %
Inhalation from passing cloud	0.01 "
Inhalation of resuspended activity	0.2 "
External radiation from activity deposited on ground	44.0 "
Exposure via nutrition	56.0 "

The dose is received almost exclusively via two longterm exposure pathways. The relative contributions may be somewhat different for other weather conditions and release scenarios, though the difference is not expected to be large.

Figures 5, 6 and 7 show the results of the risk calculations. The probabilities are conditional probabilities [1]: the probability of a certain consequence provided that the release has taken place. The absolute probability can not be determined unless the probability of a fire (during transportation or in temporary storage) is known. The distributions are furthermore complimentary cumulative; that is they show the probability that the consequence will be of the corresponding magnitude or larger.

REFERENCES

(1) Rämö, E. et al.:
Damage of reactor waste packages in transport accidents.
NKA Report AO(79)47.

(2) Rämö, E.:
Effect of fire on solidified reactor waste packaged into steel drums.
NKA Report AO(79)16.

(3) Ericsson, Ann-Margret, Kemakta Konsult AB:
Transportation of radioactive materials in Sweden - A risk study.
Avail. from the Swedish Nuclear Power Inspectorate.

(4) Stenquist, Curt:
Final Storage - Safety Study.
NKA Report (to be published).

(5) Reactor Safety Study - An Assessment of Accident Risks in U.S.
Commercial Nuclear Power Plants.
WASH-1400 (NUREG 75/014).

(6) Iversen, J.:
Reflections on calculation of individual and collective doses.
NKA Report AO(80)7. (In Norwegian).

(7) Tveten, U. et al.:
Accident Analysis Performed for the Norwegian Government Commission on Nuclear Power.
Progress in Nuclear Energy, Vol. 4, 1979, pp. 215 - 254.

(8) Nuclear Power and Safety.
Report from the Norwegian Government Commission on Nuclear Power.
NOU 1978: 35 C.

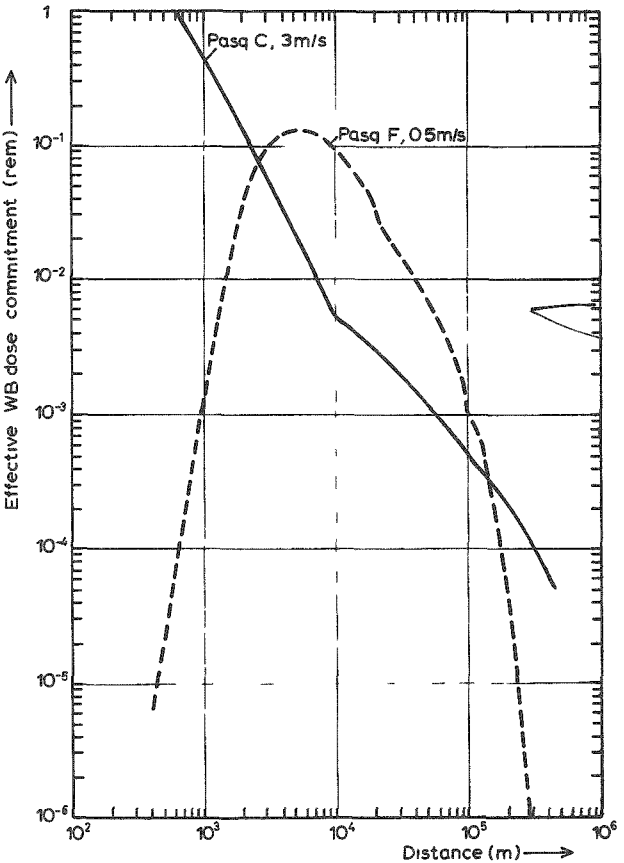

Fig 1 Release during transportation

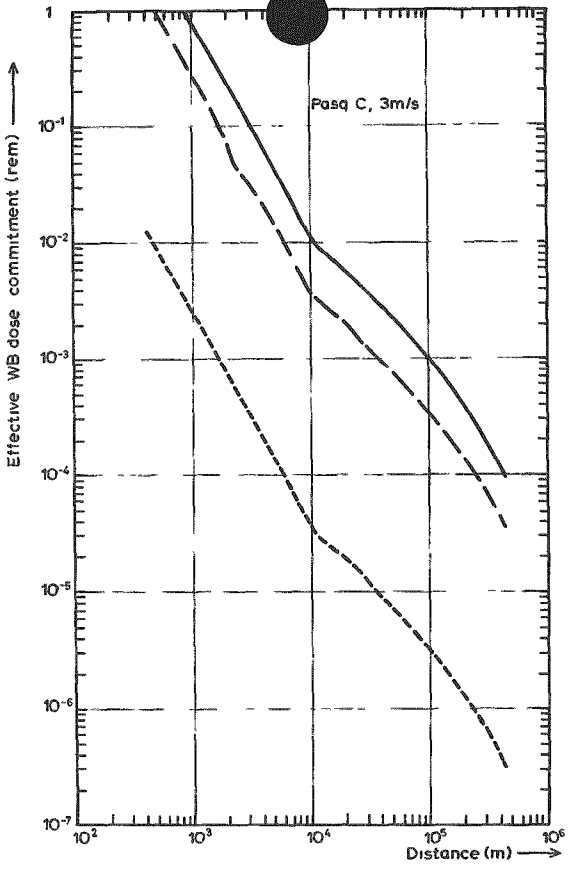

Fig. 3 Release from storage.
——— All isotopes
– – – – Sr-90 only
— — Cs-137 only

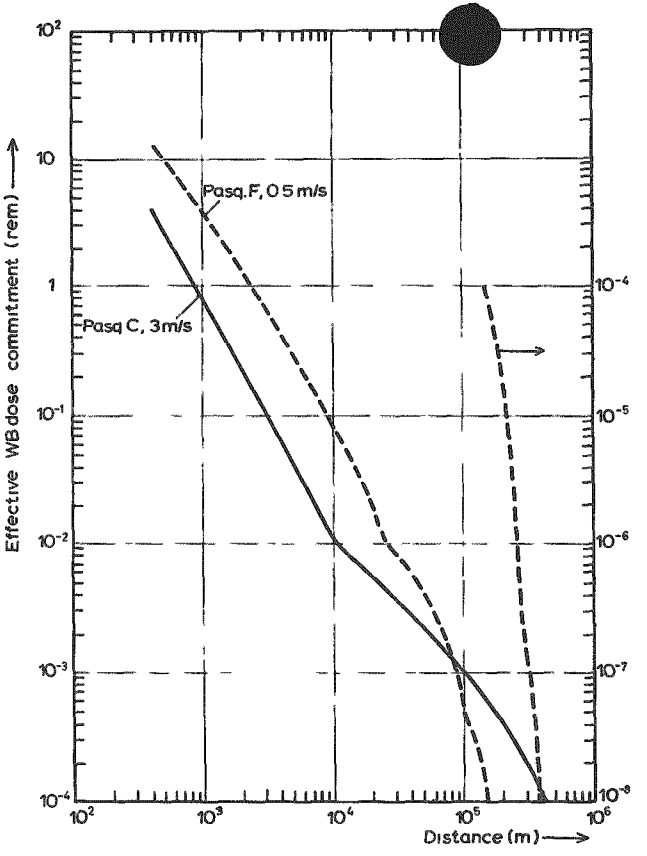

Fig. 2 Release from storage.

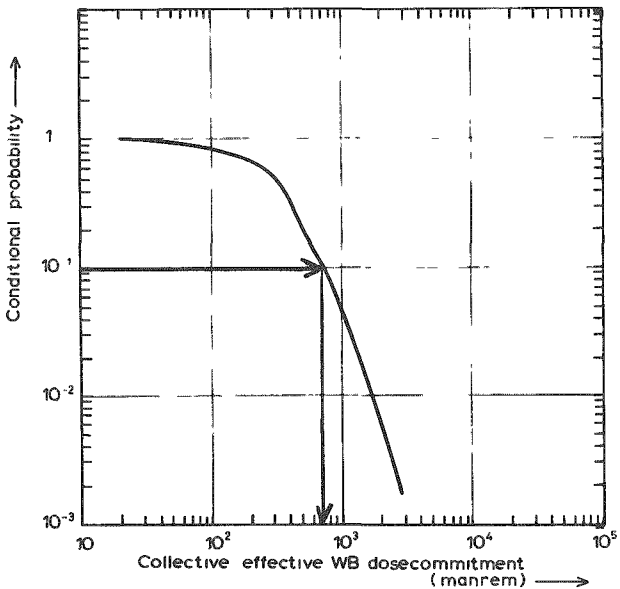

Fig. 5 Risk curve Release during transportation.

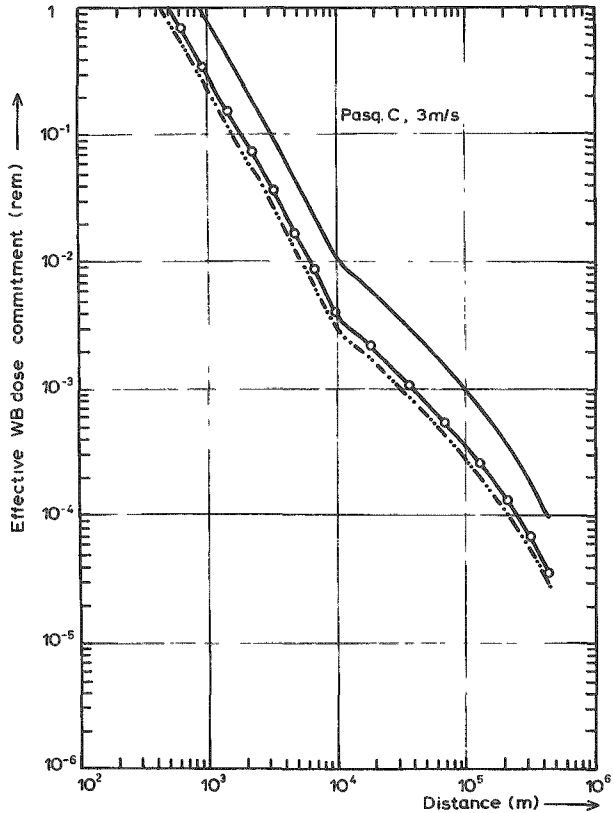

Fig 4 Release from storage.

All isotopes
Co-60 only
Cs-134 only

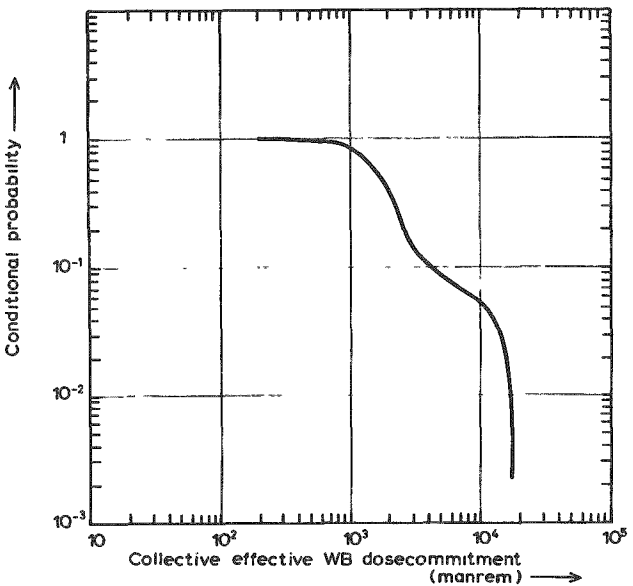

Fig. 7 Risk curve. Release during transportation. Accident in city.

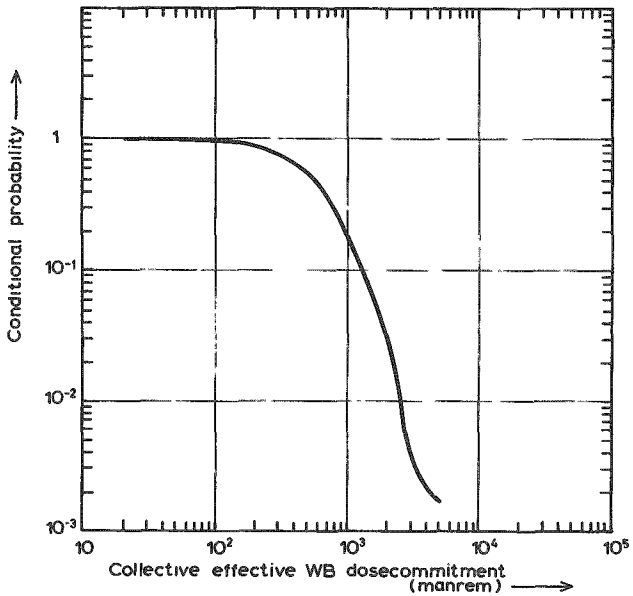

Fig. 6 Risk curve. Release from storage.

DISCUSSION

Question by J. Mishima, USA: Where was the fractional release used obtained and what were assumptions as to the characteristics of the released materials?

Answer: The fractional release and release assumptions were specified by another group in the project, and I am afraid I am not familiar with the details. The data was obtained partially from experiments conducted in Finland. In the calculations it is assumed that particle size is small enough to allow the release to be treated as gaseous.

Question by J. Hamard, France: I am surprised by the big difference (several orders of magnitude) between the doses obtained by directed pathways and indirect pathways (resuspension for example). Maybe that is due to the choice of resuspension factors. Relative to the doses per ingestion the doses are evidently too high and have to be reduced by hypothesis relative to the duration of consumation of contaminatd products (intervention). Have you made any hypothesis relative to this duration?

Answer: You may be right in supposing that the resuspension factor may be too high. It is the one used in WASH-1400, and is really valid for a desert area. Resuspension factors for other types of area are to my knowledge not available. Whether the agricultural products should be interdicted has not been specifically discussed in the project. Relative to the interdiction doses concerned in relation to reactor accidents, the presently calculated doses do not seem to require interdiction. This is however more a political than a technical question.

Question by B. W. Emmerson, United Kingdom: The transport and storage fire scenarios that you have described, and the conversion of the waste materials into their gaseous or particulate release forms, indicated high source temperatures. These in turn could generate significant bouyancy. Was the effect of bouyancy taken into account in both the transport and storage fire scenarios when assessing their impact on the environment?

Answer: Bouyancy is taken into account in the transport accident scenario, but not in the storage building scenario as the fire then takes place indoors.

Question by G. Wehner, FR Germany: What are the assumptions you have made concerning the incorporation via the food chain?

Answer: These assumptions are really too complicated and extensive to be explained here. One main assumption is that the area is self-providing regarding agricultural products. Some assumptions are incorporated into the paper and another of my reports is also referred to where a full explanation is given.

TRANSPORTATION RISKS IN THE U.S. NUCLEAR FUEL CYCLE

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U.S. Department of Energy
by
Battelle Memorial Institute

INTRODUCTION

For the past five years, Pacific Northwest Laboratory has been conducting a program to study the safety of transporting energy materials.* The overall objectives of this program are to develop information on the safety of transporting hazardous materials required to support the major energy cycles in the U.S. This information was developed for use in making energy policy decisions; in designing and developing new or improved transportation systems for these materials; to help establish research priorities; and as an aid in developing effective transportation safety regulations. Risk analysis was selected as the approach for performing these studies. A detailed methodology has been developed for analyzing the risks associated with the transportation of hazardous materials. This methodology has been applied to rail and highway shipments of nuclear fuel cycle materials and liquid and gaseous fossil fuels. Studies have been completed that analyze the risks of transporting uranium hexafluoride by truck and train (Geffen 1978); spent nuclear fuel by truck (Elder 1978); plutonium by truck (McSweeney 1976), train (Hall 1977), and cargo aircraft (McSweeney 1976); gasoline by truck (Rhoads 1978); propane by truck and train (Geffen 1980); and chlorine by train (Andrews 1980). Studies of the risks of transporting spent nuclear fuel by train and uranium ore concentrates (yellowcake) by truck are expected to be issued early in 1981. Analyses of the risks of transporting reactor waste and transuranic wastes are in progress. The work completed to date for nuclear material transportation makes it possible to estimate the transportation risks for the entire fuel cycle in the U.S. Results of this assessment are presented in this paper. Because the risk analysis studies for the transportation of gasoline, propane, and chlorine have been performed using a methodology, basic assumptions and data that are consistent with the studies that have been performed for nuclear materials, comparisons among the risks for nuclear materials and these materials can also be made. Results of these comparisons are also presented. It should be noted that it is not the intention of these comparisons to judge the safety of one industry in comparison to another. In fact, the risks from transporting all of these materials in the U.S. are low when compared to the risks from other activities. These comparisons can, however, provide some insights into the regulatory philosophy for hazardous materials transportation. The remaining sections of the paper briefly review the risk analysis methodology used in these studies, provide an overview of the systems analyzed, summarize the results and present conclusions.

* Work conducted for the U.S. Department of Energy through the Transportation Technology Center at Sandia Laboratories under Contract DE-AC06-76RLO 1830.

RISK ANALYSIS METHODOLOGY

The risk analysis methodology used in the studies described in this paper evolved from a number of risk analysis models originally developed for use in analyzing safety aspects of fixed nuclear facilities. These facilities have a well-defined population distribution and the population in the immediate vicinity of the plant (the exclusion area) is controlled by the facility operator. The population distribution in the vicinity of a transportation accident, however, is highly variable, and a variety of geographic and meteorological conditions can be encountered. This adds a degree of complexity not found in risk analysis of fixed sites.

The sophisticated transportation risk analysis tools currently available were developed over a number of years to meet the growing needs to understand and improve the safety of transporting hazardous materials. The first transportation risk analysis approaches developed were based on statistical analysis of accident case histories. Much useful information can be obtained with these statistical approaches, but their utility is limited. In many cases, there may be insufficient accident experience or the accident data may not have been collected in a way that permits accurate risk analysis. Historical data also tends to identify the most probable kinds of accidental releases, while accidents with lower probabilities but potentially larger consequences may not be identified. Analysis of hazardous materials transportation systems that do not have a good historical data base required the development of "predictive" risk assessment techniques. These methods use general information on the frequency and severity of transportation accidents to predict the probability that accidents with various consequence levels will occur during transport of hazardous materials. These methods permit analyses to be made of the risk of transporting materials for which there is little or no actual accident experience. Predictive risk analysis techniques were first used in two Holmes and Narver studies to determine tne risk of transporting bioweapons (Garrick 1969) and radioactive material (Transportation 1974). These techniques have been further refined in the work being performed at PNL and applied to a variety of transportation systems for hazardous energy materials.

Four basic steps are followed in the transportation risk analysis methodology used in the PNL studies. These four basic steps include:

- System Description. The basic information on the shipping system to be analyzed is collected in this step. The industry being studied is characterized by gathering data on facility locations, industry shipping requirements, and shipping destinations. Shipping packages and vehicles used to transport the materials being considered in the study are described in detail. Information is collected on the physical and chemical properties of the materials transported. Population distribution and weather characteristics along the shipping routes are characterized.

- Event Sequence Identification. Possible sequences of events are identified that could result in a release of hazardous materials during transportation. Fault tree analysis has been used to identify potential release sequences in all of the studies that have been performed to date using the PNL transportation risk analysis methodology.

- Event Sequence Evaluation. This step includes estimation of event sequence probabilities and evaluation of the potential consequences of each event sequence. This requires information on the response of the shipping system to normal and transport accident forces, together with knowledge of the forces present in transportation accidents. Mathematical analysis or data available from testing programs is generally used to estimate system failure thresholds.

This can be combined with accident environment data such as that developed at Sandia Laboratories (Clarke 1976a, Clarke 1976b, Dennis 1978) or statistical analysis of other accident data to calculate failure probabilities.

The consequences of a release of hazardous materials during transportation depend on the system characteristics, the nature of the failure that produced the release, the location of the release along the shipping route, and the weather and population conditions at the time of the release. These consequences may be determined from historical accident data, from information on tests that have been conducted with the material being shipped, or by engineering analysis. In general, consequences must be evaluated for each type of failure that can occur and for each combination of weather conditions and population distribution that can be encountered along the shipping routes that are used. The probabilities of encountering the various population distributions and weather conditions along the route are used in estimating the frequency of the various consequence levels that can result from a given release.

- Risk Calculation. The information developed in the previous steps is used to calculate the risk for the shipping system being studied. The results are then analyzed to determine the primary contributors to the risk and to identify alternatives that could reduce the system risk. Since the information to perform the risk analysis has been developed in discrete data blocks, sensitivity studies can also be carried out to test the effect on the system risk of assumptions and approximations that were made to develop key pieces of information. This may identify areas where further analysis is required or delineate the limitations of the analysis. Sensitivity studies can also determine the effectiveness of design or regulatory changes in reducing the system risk.

SYSTEMS ANALYZED

Transportation systems discussed in this paper include materials shipped in the U.S. nuclear fuel cycle, the propane distribution system, the gasoline distribution system, and the chlorine distribution system. An overview of the characteristics of each of these systems is presented in this section. A summary of the key systems description information used in the risk assessment studies is presented in Table 1. All systems are based on industry projections for the mid-1980's in the U.S.A.

- Nuclear Fuel Cycle. Current plans in the U.S. are to indefinitely defer reprocessing of spent fuel. This policy results in a "once-through" fuel cycle; with spent fuel being stored for a number of years and eventually disposed of as waste. Table 1 contains estimates of the materials required to be transported to support the operation for one year of a once-through fuel cycle with 100 GWe of installed generating capacity. Steady-state operation of a complete fuel cycle is assumed. The materials shipped include uranium ore concentrates (yellowcake) from uranium mills to UF_6 conversion facilities; natural UF_6 from the conversion facilities to enrichment plants; enriched UF_6 from enrichment plants to fuel fabrication plants; fresh nuclear fuel to reactors; spent fuel to interim storage facilities and from the storage facilities to a geologic repository; and low-level wastes from all the facilities to shallow-land burial grounds. Shipping containers and transport modes presented in the table reflect current U.S. practices or most likely future practices. Shipping distances are actual distances for existing facilities and conservative estimates for facilities that are not currently in existence.

LE 1. Summary of Transportation Systems Descriptions for Selected [illegible]gy Materials in the U.S.A[a]

Material Shipped	Transport Mode	Shipping Container	Container Capacity	Containers/ Shipment	Quantity Shipped (MT/yr)	No. of Shipments/Yr	Avg. Shipment Distance (km)	Total Distance (km/yr)	MT-km/yr
Nuclear Fuel Cycle									
Yellowcake (U_3O_8)	Truck	Steel Drums	360 kg	50	2.4×10^4	1,350	2,600	3.5×10^6	6.3×10^7
Natural UF_6[a]	Truck	Steel Cylinder	9.5 MT or 12.5 MT	2 or 1	2.5×10^4	1,700	650	1.1×10^6	1.7×10^7
	Rail	Steel Cylinder	12.5 MT	4	3,700	75	540	4.0×10^4	2.0×10^5
Enriched UF_6[b]	Truck	Steel Cylinder with Overpack	9.5 MT or 2.3 MT	1 or 5	5,300	490	860	4.2×10^5	4.6×10^6
Fresh Fuel	Truck	Overpack	1 PWR or 2 BWR Assemblies	6 PWR or 16 BWR	3,890	840	1,000	8.4×10^5	4.7×10^6
Spent Fuel[c]	Truck	Cask	1 PWR or 2 BWR Assemblies	1	1,170	1,830	930	1.7×10^6	1.1×10^6
	Rail	Cask	7 PWR or 13 BWR Assemblies	1	2,720	630	930	5.8×10^5	2.5×10^6
Reactor Wastes	Truck	Drums Overpacks	210 ℓ or 1.4 m^3 or 2.8 m^3	40 or 150 1	3.4×10^4 5.3×10^4	1,800 8,400	1,200 1,200	4.4×10^5 1.0×10^7	7.3×10^6 4.9×10^7
Spent Fuel[d]	Rail	Cask	15 PWR or 41 BWR Assemblies	1	3,890	375	4,600	1.7×10^6	1.8×10^7
Other Wastes	Truck	Drums	210 ℓ	150	6,800	450	1,600	4.5×10^5	6.7×10^6
Totals					1.6×10^5	1.8×10^4	1,150	2.1×10^7	1.7×10^8
Chlorine	Rail	Steel Tank	82 MT	1	4.5×10^6	5.5×10^4	450	2.5×10^7	2.0×10^9
Propane	Truck	Steel Tank	9 m^3 or 39 m^3	1	3.2×10^7	3.0×10^6	120	3.6×10^8	5.3×10^9
	Rail	Steel Tank	130 m^3	1	1.5×10^6	2.5×10^4	400	1.0×10^8	6.4×10^9
Gasoline	Truck	Aluminum Tank	32 m^3	1	3.2×10^8	1.4×10^7	80	1.1×10^9	2.5×10^{10}

(a) Expected about 1985.
(b) Includes shipments between enrichment plants.
(c) Reactors to interim storage.
(d) Interim storage to disposal site.

- Chlorine Distribution. Chlorine is used as a chemical feedstock, as a bleaching agent, and in purification of domestic water supplies. In the U.S., about 70% of the chlorine shipped from the manufacturing plants to the user in railroad tank cars. This is the portion of the chlorine distribution system considered in the risk analysis. The railroad tank cars used for chlorine transport have a capacity of 82 MT of chlorine. The tanks are insulated and equipped with pressure relief valves.

- Propane Distribution. In the U.S., propane is used as a domestic fuel, primarily in sparsely populated rural areas where the cost of providing piped natural gas service would be prohibitive. Propane is separated from natural gas at processing plants near the gas fields and also produced as a refinery by-product. It is transported from these facilities to retail distributors by pipeline, barge, or railroad tank car and distributed to customers primarily in large- or medium-sized tank trucks. The rail and truck portions of the distributions systems were analyzed in this study. They account for most of the transportation risks that result from operation of the system. The rail tank cars used for propane transportation are large steel pressure cyclinders with a liquid volume capacity of 130 m^3. They are equipped with thermal insulation, pressure relief valves, and head shields and shelf couplers to prevent coupler puncture of the tank ends in accidents. The tank trucks used in the system are also large steel pressure cylinders. Two typical trucks have capacities of 9 m^3 and 39 m^3 of propane. The tanks are equipped with pressure relief valves but are not insulated.

- Gasoline Distribution. In the mid-1980's, gasoline is expected to provide energy for about 10^{17} vehicle miles of transportation in the U.S. Gasoline is shipped in bulk quantities from refineries to wholesale distribution terminals by pipeline or barge. Tank trucks are used to distribute the gasoline to the retail outlets. Truck transportation of gasoline was analyzed in this study. It represents most of the transportation risk in the gasoline distribution system. The truck tanks used in this system have aluminum walls and a capacity of 32 m^3. Both truck-trailer and tractor-semitrailer vehicles are used.

RESULTS AND CONCLUSIONS

Estimated risks associated with accidental releases of materials transported for each step of the nuclear fuel cycle are presented in Table 2. These estimates are based on the work presented in References 1 through 5 and preliminary results from studies that are nearing completion. The risk numbers were calculated for the reference fuel cycle described in Table 1. The risk estimates include both immediate and latent fatalities caused by releases of these materials in transportation accidents. Studies of the risk of transporting yellowcake, fresh nuclear fuel and low level wastes from the front end of the fuel cycle have not been completed. Existing information does permit estimates of the risks to be made. The estimates presented in the table result from the very low hazards associated with release of these materials. These estimates are consistent with the result of other studies.

The results presented in Table 2 show that risks from all the fuel cycle transportation steps are low. The results also indicate that the total transportation risks associated with the nuclear fuel cycle are distributed about evenly between the front end (fuel supply) and the back end (waste management) of the cycle. Risks in the front end of the cycle result primarily from the

TABLE 2. Estimated Transportation Accident Risks From the Once-Through Nuclear Fuel Cycle

Fuel Cycle Step	Material Shipped	Transportation Accident Risks (Fatalities/yr)
Mining/Milling	U_3O_8 (Yellowcake)	$<10^{-6}$
Conversion/Enrichment	UF_6	6×10^{-5}
Fuel Fabrication	Fresh Fuel	$<10^{-6}$
Power Production	Spent Fuel	10^{-5}
	Reactor Wastes	2×10^{-6}
Fuel Storage/Disposal	Spent Fuel	2×10^{-5}
Waste Management	Other Fuel Cycle Wastes	$<10^{-6}$
	Total	10^{-4}

chemical toxicity of the materials transported (even though transportation of these materials is regulated on the basis of their radioactivity). For example, public health risks from transportation accidents that release UF_6 are due almost to the hydrogen fluoride (HF) gas that is formed when UF_6 reacts with water vapor in the air. This is the only mechanism in amy of the fuel cycle steps with potential to cause immediate fatalities from accidental releases during transportation. All other fatalities associated with accidental releases of fuel cycle materials during transportation are latent effects associated with the toxicity of heavy metals or the increase in cancer rates in populations exposed to increased levels of radioactivity.

The results of the risk analysis studies for transportation of nuclear fuel cycle materials are compared with the results for the three studies that have been completed for non-nuclear systems in Table 3. A number of risk measures are presented to facilitate comparisons among the various systems. These risk measures include the usual risk number (expected fatalities/yr) and the expected number of fatalities per shipment, per km and per MT-km. Conversion factors for the latter quantities are taken from the system description information in Table 1. An estimate of the risk normalized to the energy content of the material transported is also presented. These numbers are based on the electricity generated by a 100-GWe installed nuclear fuel cycle and the estimated amount of electricity that would be required to replace the energy supplied by the propane and gasoline. The risk analysis methodology used in these studies identifies the complete spectrum of potential accident consequences and estimates the probability of events producing that level of consequence. The maximum number of fatalities predicted for each material is present in Table 3. A variety of risk measures have been used because of the inherent difficulties in making risk comparisons. Examination of a number of risk measures can provide additional insights and help guard against conclusions that are dependent on the [illegible]he risk information has been developed and displayed.

The results summarized in Table 3 indicate that the risks from transporting these materials are all relatively low in comparison to other risks in society. For example, in the U.S., highway fatalities are about 55,000/yr; about 6500 people die in building fires each year; 160 are killed by lightning; and 90 die in tornadoes. The results also show that for all the risk measures, gasoline, propane, and chlorine transportation risks are very similar. The risks from shipments in the nuclear fuel cycle are, in general, several orders

TABLE 3. Comparison of Risk Analysis Results for Transportation of Selected Energy Materials in the U.S.A

Transportation System	Risk Measure: Fatalities/ Year	Fatalities/ Shipment	Fatalities/ km	Fatalities/ MT-km	Fatalities/ GW-hr	Maximum Fatalities
Nuclear Fuel Cycle Shipments	1×10^{-4}	6×10^{-9}	5×10^{-12}	6×10^{-13}	2×10^{-11}	100
Chlorine	9	2×10^{-4}	4×10^{-7}	4×10^{-9}	--	600
Propane	15	5×10^{-6}	3×10^{-8}	1×10^{-9}	4×10^{-5}	800
Gasoline	29	2×10^{-6}	3×10^{-8}	1×10^{-9}	2×10^{-5}	40

of magnitude less than the risks for the other hazardous materials. This is especially true when comparing risks normalized by the energy content of the material transported. This results primarily because nuclear materials have a high energy density, but this energy cannot be released in a transportation accident. Effects of release of fuel cycle materials are secondary ones caused by the radioactivity or chemical toxicity of the material transported. The concern in releases of gasoline or propane is the uncontrolled release of the stored chemical energy in these materials, resulting in a fire or explosion.

The relatively low risks associated with nuclear fuel cycle shipments in comparison to the other materials can be attributed to a number of factors. Foremost is probably the traditional conservatism built into standards and regulations in the nuclear industry. The transportation regulations for radioactive materials require increased levels of protection as the potential hazard from the material shipped is increased. Significant quantities of radioactive materials are provided protective packaging designed to prevent releases in severe transportation accidents. Packagings for other hazardous materials are generally designed to provide containment during normal transportation with limited protection during accidents. Package resistance to accident stresses is reflected most directly in the fatalities/km risk measure. The value of this risk measure for fuel cycle materials differs from the value for the other hazardous materials by about four orders of magnitude. This is approximately the difference in risk that would result from transporting a hazardous material in a package designed to withstand severe accident conditions instead of a package designed to withstand only normal tranport forces.

These calculated risk numbers provide a basis for examining the current hazardous material transportation regulatory philosophy. There seems to be consistency in the level of protection required for hazardous materials such as chlorine, propane, and gasoline. However, there seems to be an inconsistency when these are compared to nuclear shipments. For all the risk measures except maximum credible consequences, nuclear materials transportation risks are much lower than those for other materials. Even for maximum consequences, risks from propane and chlorine are 6 to 8 times higher. This would indicate that risks from nuclear material shipments are perceived to be more hazardous than they really are. There also seems to be a different risk/benefit philosophy applied to nuclear materials than to other hazardous materials. The risk/GW-hr risk measure indicates that nuclear material transportation has by far the lowest risk/benefit ratio of the materials analyzed. Nevertheless, significant efforts are still under way in the U.S. to provide even more stringent controls over these shipments. Based on the figures presented in Table 3, it is the opinion of the authors that there appears to be little real need to attempt to substantially increase the level of protection provided nuclear materials in transport.

REFERENCES

Andrews, W. B. et al., An Assessment of the Risk of Transporting Liquid Chlorine by Rail. PNL-3376 (TTC-0114), Pacific Northwest Laboratory, Richland, Washington, March 1980.

Clarke, R. K. et al., Accident Environments Expected in Air Force C-5, C-141, and C-130 Aircraft Accidents. AFWL-TR-75-247, Sandia Laboratories, Albuquerque, New Mexico, January 1976a.

Clarke, R. K. et al., Severities of Transport Accidents. SLA-74-0001, Sandia Laboratories, Albuquerque, New Mexico, July 1976b.

[illegible]is, A. W., Severities of Transportation Accidents Involving Large Packages. SAND-77-0001, Sandia Laboratories, Albuquerque, New Mexico, 1978.

Elder, H. K. et al., An Assessment of the Risk of Transporting Spent Nuclear Fuel by Truck. PNL-2588, Pacific Northwest Laboratory, Richland, Washington, November 1978.

Garrick, B. J., Gekler, W. C., Baldonado, O. C., Elder, H. K., and Shapley, J. E., A Risk Model for the Transport of Hazardous Materials. HN-204, Holmes and Narver, Inc., Los Angeles, California, August 1969.

Geffen, C. A. et al., An Assessment of the Risk of Transporting Propane by Truck and Train. PNL-3308 (TTC-0115) Pacific Northwest Laboratory, Richland, Washington, March 1980.

Geffen, C. A. et al., An Assessment of the Risk of Transporting Uranium Hexafluoride by Truck and Train. PNL-2211, Pacific Northwest Laboratory, Richland, Washington, August 1978.

Hall, R. J. et al., An Assessment of the Risk of Transporting Plutonium Dioxide and Liquid Plutonium Nitrate by Train. BNWL-1996, Pacific Northwest Laboratory, Richland, Washington, February 1977.

McSweeney, T. I., Hall, R. J. et al., An Assessment of the Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck. BNWL-1846, Pacific Northwest Laboratory, Richland, Washington, August 1975.

McSweeney, T. I. and Johnson, J. F. An Assessment of the Risk of Transporting Plutonium Dioxide by Cargo Aircraft. BNWL-2030, Pacific Northwest Laboratory, Richland, Washington, June 1976.

Rhoads, R. E. et al., An Assessment of the Risk of Transporting Gasoline By Truck. BNWL-2133, Pacific Northwest Laboratory, Richland, Washington, November 1978.

[illegible]sportation Accident Risks in the Nuclear Power Industry 1975-2020. [illegible]S 8191.1, Nuclear and Systems Sciences Group, Holmes and Narver, Inc., Los Angeles, California, November 1974.

DISCUSSION

Question by R. Neider, FR Germany: 1. When calculating the different risks did you take into account both types of risk, the specific risk e.g. from radioactive materials or chlorine on the one hand and the conventional risks from an accident on the other? 2. How would the figures look if you had also taken into account the conventional risks?

Answer: 1. The risk figures presented in the paper include only the risks resulting from the hazardous nature of the cargo being shipped. 2. In all the studies we have performed, conventional accident risks have been found to be much higher than risks due to the nature of the cargo.

Question by K. B. Shaw, United Kingdom: How did you arrive at the number of maximum consequences for nuclear fuel shipments? Does this number include probability factors and were various release scenarios taken into account?

Answer: The methodology used in our studies identifies the full spectrum of potential accident consequences. The figures I presented were the largest consequence accidents identified in our studies. Probabilities for these high-consequence accidents are all very low.

INDIVIDUAL AND COLLECTIVE DOSES ARISING IN THE TRANSPORT OF IRRADIATED NUCLEAR FUELS

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1. INTRODUCTION

Consignments of irradiated fuel from CEGB and SSEB nuclear power stations in transit by road and rail within the UK to the BNFL works at Windscale, Cumbria for reprocessing are controlled under the terms of Approvals or Special Arrangements granted by the Department of Transport. This ensures that such transport operations are carried out within the conditions set out in the IAEA Regulations for the Safe Transport of Radioactive Materials (IAEA, 1979). These Regulations are intended to establish an acceptable level of control of radiation hazards to persons, property and the environment during transport of radioactive materials. In addition, recent recommendations of the ICRP have implied that cost-benefit analyses should be undertaken as a part of the process leading to the establishment of acceptable levels of radiation dose (ICRP, 1977), which requires the evaluation of the radiological impact of all stages of the nuclear fuel cycle.

In this paper a method is described for predicting the individual and collective doses arising during the transport of irradiated fuel from a system of reactors to a central reprocessing plant. External radiation and potential exposure to airborne radioactivity are included in order to evaluate the doses to both transport workers and members of the public living along the transport routes. Measurements of the dose rate near a Magnox fuel transport flask are used to determine the doses resulting from Magnox transport operations within the UK. These doses are associated with an installed capacity of 4.2 GW(e) and are compared with other sources of dose arising at other stages of the nuclear fuel cycle. Finally, the individual and collective doses implied by the Type B(U) and B(M) activity release limits contained in the IAEA Transport Regulations are discussed and compared with those used to determine discharge limits from reactor sites within the UK.

2. METHODOLOGY

In general, consignments of irradiated fuel in transit may lead to low level exposure of transport workers and the public via several pathways:

a) External penetrating radiation direct from the fuel flask.
b) External or internal radiation due to exposure to a release of airborne radioactive material.
c) External radiation from ground contaminated by dry deposition or rainout of airborne radioactivity.
d) Internal radiation due to consumption of foodstuffs contaminated by dry deposition or rainout of airborne radioactivity.

However, in the case of Magnox fuel movements within the UK there are two barriers to the release of airborne activity, namely the flask lid seal and the fuel element cladding for intact fuel or the lid seal and transit bottle for damaged fuel. Decontamination of the external surfaces of the flask prior to dispatch ensures that exposure due to residual levels of surface contamination ($< 10^{-4}$ μCi.cm^{-2} β/γ) is minimal, thus the dominant contribution to doses arises from penetrating radiation emerging direct from the flask and the treatment given below will concentrate on this pathway.

2.1 Individual Doses due to Stationary or Moving Transport Flasks

For direct external radiation or exposure to airborne radioactivity the variation of dose rate with distance from a stationary fuel flask may be represented by a function of the form:

$$D'(r) = \frac{A}{r^n}, \qquad (1)$$

where n = 2 for direct external gamma exposure from a point source (see Figure 1 and Section 3), and n ≈ 1.2-1.9 for inhalation doses or cloud gamma exposure (Macdonald and Mairs, 1978). In general, A is also a function of distance, r, and for external gamma radiation is proportional to the product of $e^{-\mu r}$ and B(r), where μ is the linear attenuation coefficient and B(r) the dose build-up factor (Jaeger, 1968). However, for gamma radiation of ~ 1 MeV energy, typical of that emerging from the surface of a transport flask, this product is approximately constant for values of r in the range over which the major contributions to doses are likely to arise.

Within the point source approximation the integrated dose during the time period t_1 to t_2 due to the passage of a transport flask is given by:

$$D(d) = \int_{t_1}^{t_2} D'(r)\, dt, \qquad (2)$$

where d is the perpendicular distance of the dose point from the transport route. Assuming that the flask is travelling at a constant speed, v, and that the distances over which significant contributions to dose arise are much less than the length of the transport route, the above integral for n = 2 reduces to:

$$D(d) = \frac{\pi A}{vd}. \qquad (3)$$

2.2 Collective Doses due to Fuel Movements within the UK

In order to estimate the population doses resulting from irradiated fuel movements associated with CEGB and SSEB reactor operations, the transport routes from the power stations to Windscale were divided into a series of route elements. Since precise representation of the transport routes would require a prohibitively large number of route elements, those used in this study were selected in order to reproduce the major variations in population density along the actual routes and their locations relative to the main centres of population as represented in the collective dose computer code POPDOS2 (Merlin and Henning, 1979).

The collective dose contributions arising from moving fuel flasks were evaluated by summing over the route elements, using dose surfaces generated by assuming a constant dose at a given distance from the route element along its entire length,

with the fall off of dose with distance from the route given by equation (3) above. The contributions from stationary flasks at marshalling yards en route were estimated assuming a uniform angular variation of dose, with the dose as a function of radial distance taken from equation (1) above.

In all cases lower limits on the range of numerical integrations were applied since restriction of access to railway property, which comprises the overwhelming majority of the routes traversed by fuel flasks, makes it unlikely that the public will closely approach flasks in transit. For moving flasks a lower limit of 30 m was set, while for stationary flasks at marshalling yards limits in the range 30-200 m were used, depending on the geographical location and extent of public access to the sites involved. The special cases of rail passengers and transport workers are discussed separately below.

2.3 Doses to Rail Passengers and Transport Personnel

Detailed estimates of the doses to rail passengers and workers would require data on transport and work patterns which are not readily available, thus approximate methods have been used to evaluate the potential importance of these contributions to the overall transport doses. For rail passengers it is assumed that each consignment passes within a few metres of 500 passengers during each hour of its journey, excluding the periods spent at marshalling yards.

The doses to transport workers may be divided into four principal contributions; those to drivers of the road vehicles used to transfer the fuel flasks from the power stations to the nearby railheads, to rail staff loading the flasks at each railhead, to the crews of the goods trains carrying the flasks and to rail staff working in the vicinity of stationary flasks at the marshalling yards en route to Windscale. Detailed assumptions applied to each of these groups have been discussed elsewhere (Macdonald and Mairs, 1978).

3. INDIVIDUAL AND COLLECTIVE DOSE ESTIMATES

The results of measurements of the external gamma dose rate near a Magnox fuel transport flask are presented in Figure 1. These results apply to fuel irradiated to 4.5 $GWd.t^{-1}$ and cooled for a period of 100 days, which is the normal minimum storage time at the power station sites. However, for operational reasons the current practice is to transport fuel after at least one year's storage, while in many cases fuel has been stored for two years. This reduces the gamma dose rates shown in Figure 1 by factors in the range 5-10, indeed the data shown were derived from measurements of flasks containing fuel having a range of irradiation and cooling histories with corrections applied for the variation in gamma emissions as a function of irradiation and cooling time. Early experience in the transport of irradiated AGR (advanced gas-cooled reactor) fuel indicates that the dose rates near these flasks will be very similar to those shown in Figure 1 for the Magnox fuel flasks.

3.1 Individual Doses

The annual doses to individual members of the public and transport workers resulting from Magnox and AGR fuel movements within the UK were estimated using the data shown in Figure 1 and the methods outlined in Section 2. For doses from a moving flask a mean transit speed of 30 $km.h^{-1}$ was assumed; this is based on typical flask movement timetables which indicate average speeds between major stops en route in the range 26-38 $km.h^{-1}$ (Heap, 1978). The doses to rail

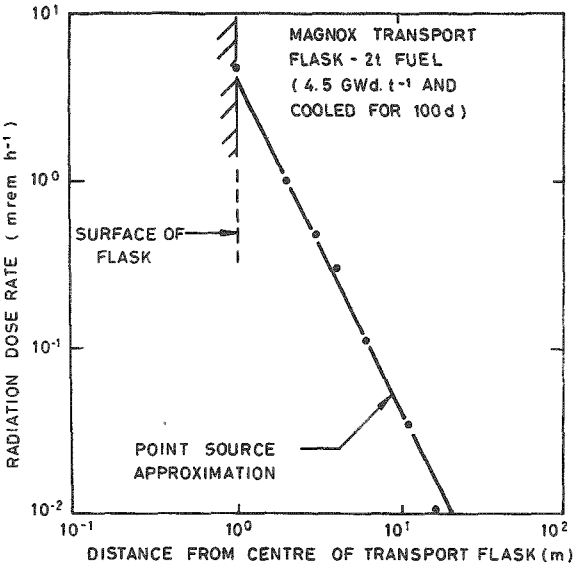

FIGURE 1.. External Gamma Dose Rate near an Irradiated Magnox Fuel Transport Flask.

passengers were evaluated for a relative passing speed of 100 $km.h^{-1}$, assuming exposure at a mean distance of 3-4 m; this yields an individual dose per pass of approximately 0.05 μrem. The maximum individual annual doses to the various groups considered here are summarised as follows:

a)	public living near transport routes	0.01 mrem
b)	public near marshalling yards	10 mrem
c)	road vehicle drivers	4 mrem
d)	rail staff loading flasks	29 mrem
e)	train crews	8 mrem
f)	rail staff at marshalling yards	20 mrem

In general, the levels of individual dose due to Magnox and AGR transport are well below the average natural background radiation level in the UK of ~ 100 $mrem.a^{-1}$. The two highest contributions are to rail staff loading the flasks at the railheads or working at marshalling yards where the flasks stop en route. Loading operations require two men to be in close proximity to each flask for ~ 5 minutes in an assumed radiation field of 2.5 $mrem.h^{-1}$, while doses to staff at marshalling yards were based on an assumption of continuous exposure of 10 men at each yard at a mean distance of 30 m from a stationary flask in a field of 5 $\mu rem.h^{-1}$.

3.2 Collective Doses during Normal Transport

The annual collective doses to members of the public and transport workers

due to Magnox fuel traffic from CEGB and SSEB nuclear power stations are summarised as follows:

a)	public living near transport routes	0.36 man rem
b)	public near marshalling yards	1.8 man rem
c)	road vehicle drivers	0.06 man rem
d)	rail staff loading flasks	0.25 man rem
e)	train crews	0.01 man rem
f)	rail staff at marshalling yards	0.55 man rem
g)	rail passengers	0.19 man rem

In estimating the contributions to members of the public living along the transport routes or near marshalling yards no allowance has been made for the shielding effects of buildings or local topographical features, such as railway embankments or tunnels; these effects are likely to reduce the doses cited above by at least 50% (Aldrich, Ericson and Johnson, 1978). Thus the overall collective dose estimate of $\sim$ 3 man rem.a^{-1}, associated with an installed Magnox reactor capacity of 4.2 GW(e), should be regarded as an upper limit value. The corresponding figure for the current programme of five AGR power stations, representing an installed capacity of $\sim$ 6 GW(e), is estimated to be $\sim$ 3.5 man rem.a^{-1}, although at present only two of these stations have been fully commissioned.

Finally, it is of interest to compare the collective dose estimates for fuel transport given here with those to radiation workers and the public arising in other phases of the nuclear fuel cycle (see for example Macdonald and Mairs, 1978; Macdonald, Pepper and Gammon, 1979). The transport contribution represents $\sim$ 0.1% of the overall collective dose due to the Magnox fuel cycle, which in turn is about 0.3% of the collective dose to the UK population due to natural background radiation.

4. ACCIDENT DOSE LIMITS

Within the terms of the IAEA Regulations an irradiated fuel transport flask constitutes a Type B package for which the accidental activity release limits are specified in terms of the Type A package contents limit A_2 values. The permitted releases for Type B(U) and Type B(M) packages are respectively $A_2 \times 10^{-3}$ and A_2 Ci in one week. For an accident occurring out-of-doors under average weather conditions and exposure within 50-100 m of the flask, it can be shown that these release limits are equivalent to integrated effective whole-body doses of approximately 1 mrem and 1 rem for Type B(U) and Type B(M) packages respectively (Macdonald and Mairs, 1978). These values compare with the Emergency Reference Level (ERL) of dose to the whole body of 10 rem recommended by the UK Medical Research Council (MRC, 1975), and used as the basis for the derivation of emergency action levels at CEGB and SSEB nuclear power station sites within the UK (Macdonald, Ballard, Thompson, Goldfinch and Orchard, 1977).

Thus the Type B(U) and B(M) activity release limits are respectively four and one orders of magnitude more restrictive than the ERL limit used as the criterion for emergency action at power station sites. These differences may be rationalised by noting that transport accidents can occur anywhere along the route from the power station to the reprocessing plant, including relatively densely populated urban areas. In this context it is of interest to compare the collective doses associated with accidental releases from reactor sites and in urban areas, and this is done for four representative locations in the UK in Figure 2. The azimuthal variation in collective dose for twelve wind directions uniformly spaced at 30° intervals is shown, normalised to an individual dose of 1 rem at 100 m downwind under average category C/D weather conditions.

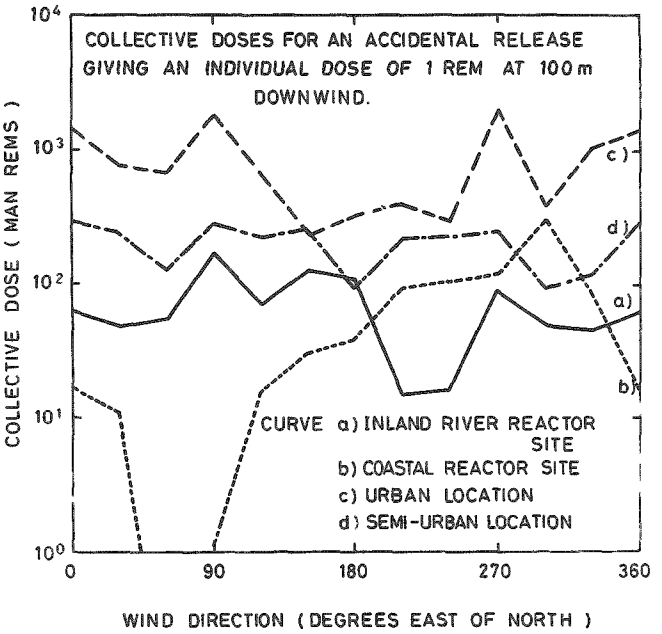

FIGURE 2. Directional Dependence of Accident Collective Doses for Selected UK Sites.

For the most unfavourable wind direction and an accident occurring in a densely populated urban area, the collective dose is approximately an order of magnitude greater than for the equivalent accident occurring at a more remote reactor site. Alternatively, noting the factor of 10 difference between the ERL dose level and that associated with the Type B(M) limit, the collective dose for a reactor accident at the ERL level would be approximately equivalent to that for an urban transport accident at the Type B(M) limit. Thus if one associates similar probabilities with a transport accident of severity sufficient to lead to the maximum permitted releases within the IAEA Regulations and a reactor accident leading to doses at the ERL level, the current Type B(M) limits result in similar safety standards for fuel transport and reactor operations. Applying the same logic to the Type B(U) activity release limit following a severe accident suggests that this limit is about three orders of magnitude too restrictive when compared with safety standards applied at power reactor sites.

5. CONCLUSIONS

A method for evaluating the individual and collective doses associated with the transport of irradiated fuel from a system of nuclear power stations to a central reprocessing plant has been described and used to estimate the doses arising during normal transport of reactor fuels within the UK. In general, individual doses are well below natural background radiation levels, while irradiated Magnox fuel movements result in an annual collective dose of $\sim$ 3 man rem, with AGR fuel movements likely to make a similar contribution when stations currently under

construction come on-line. These collective doses represent ~ 0.1% of the overall collective dose attributable to the nuclear fuel cycle.

The accidental activity release limits for Type B(U) and B(M) packages contained in the IAEA Regulations are respectively factors of 10^4 and 10 times more restrictive than the emergency action levels currently employed at power station sites in the UK. Assuming parity between transport accidents leading to a release equivalent to the Type B(M) limit and a reactor accident giving doses at the ERL level, the Type B(M) limit is justifiable in terms of the possible collective dose arising from a transport accident occurring in a densely populated urban area. However, the Type B(U) limit appears unduly restrictive when compared with safety standards applied at power reactor sites in the UK.

ACKNOWLEDGMENT

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REFERENCES

Aldrich, D.C., Ericson, D.M. and Johnson, J.D., 1978, Sandia Laboratories Report SAND 77-1725.

Heap, G.F., 1978, private communication.

IAEA, 1979, Regulations for the Safe Transport of Radioactive Materials 1973, Revised Edition, (As Amended), Safety Series No. 6, IAEA, Vienna.

ICRP, 1977, Publication 26, Recommendations of the International Commission on Radiological Protection, Pergamon Press, Oxford.

Jaeger, T.G., (Ed.), 1968, Engineering Compendium on Radiation Shielding, Vol. I Shielding Fundamentals and Methods, Springer-Verlag, New Yoark.

Macdonald, H.F., Ballard, P.J., Thompson, I.M.G., Goldfinch, E.P. and Orchard, H.C., 1977, J.Brit.Nucl.Energy Soc., 16, 177-86.

Macdonald, H.F. and Mairs, J.H., 1978, CEGB Report RD/B/N4440.

Macdonald, H.F., Cammon, K.M., and Pepper, R.B., 1979, Proc. of VGB Conf. on Power Stations and the Environment, 30-31 May, Essen.

NRC, 1975, Criteria for Controlling Radiation Doses to the Public after Accidental Escape of Radioactive Material, HMSO, London.

Merlin, J.H., and Henning, M.J., 1979, CEGB Report RD/B/N4686.

DISCUSSION

Question by Y. Sousselier, France: The figures you have given in your paper for the dose commitment are extremely low, nevertheless the major part is the dose received by the public near the marshalling yards. Have you envisaged, as we do in France, the use of special trains without stopping in marshalling yards?

Answer: The use of "special trains" would not be justified on radiological protection grounds, given that the low levels of individual and collective dose cited are upper limit values. The operati procedures on the British Rail system are different from those in many countries and further contribute to the safety of irradiated fuel movements. For example, strict segregation rules apply to fuel flasks, and the majority of the routes over which they pass are maintained to the high standards required for passenger traffic. Also, the location of all flasks is monitored at all stages of their journey using British Rail's TOPS computer system.

Question by K. Schneider, FR Germany: Is it right that you took the IAEA requirements as a basis for calculating the accidental risk?

Answer: Not exactly. The results of Figure 2 were used to illustrate the relative risks, expressed as man rems, of a given accidental release occuring from remote and urban sites. It was then suggested that under United Kingdom conditions and given certain assumptions concerning the probabilities of transport and reactor accidents, the type B(M) limits result in similar safety standards for fuel transport and reactor operations, while the type B(U) limit is unduly restrictive.

RELATIVE CONSEQUENCES OF TRANSPORTING HAZARDOUS MATERIALS*

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INTRODUCTION

The transportation of hazardous materials has become a matter of increasing public interest and concern which has been voiced in requests for improvements in shipping systems and increased protection from the hazards resulting from such shipments. Included within the term hazardous materials are basic categories of materials which exhibit anoxic, corrosive, explosive, flammable, radioactive or toxic effects or characteristics. The media has expanded its coverage of accidents and incidents involving hazardous materials and there has been an increase in the number of laws and regulations governing such shipments. These rules have been promulgated because of the accidents and incidents which have occurred, and because of the increased numbers, volumes and types of materials involved. These additional rules have resulted in greater complexities of transporting, storing and disposing of hazardous materials. Industry has responded with modifications to (or acquisition of new) equipment which has significantly improved performance of some transport systems in accidents. In addition, companies have been founded to clean up spills of hazardous materials and to return the routes and transport systems to operational status.

One aspect of the awakened interest in hazardous materials transport is reflected in the numbers of analyses describing the potential harm and risks involved. Many analyses and techniques have been developed to show the quantitative risk that accrues from transporting these materials. A number of these studies have been used to show that transportation is "safe" but some of the same studies have also been used to show how many problems and potential disasters are waiting to occur. This paradox is especially interesting to those involved in radioactive materials transportation. They weigh the objective record of no deaths or injuries from 30 years of transporting radioactive materials against that of any other category or class of hazardous material and then observe which industry "enjoys" the principal attention of the public, media, regulators and lawmakers at all levels of government. The problem here is the public's perception of the relative hazards posed by a material and their perception of the relative benefits they receive from it. The understanding of this process is part of the program at the Transportation Technology Center (TTC), and this paper deals with the relative hazards portion of the problem.

The objective of this paper is to discuss methods under study at TTC to develop a perspective on how technical measures of hazard and risk relate to perception of hazards, harm and risks associated with transporting hazardous materials.

This paper is concerned with two major aspects of the relative hazards problem and is based on a study performed by Science Applications, Inc. The first

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aspect is the analyses of the possible effects associated with exposure to hazardous materials as contained in the following two parts: (1) outlines of possible problems and controversies that could be encountered in the evaluation and comparisons of hazards and risks and (2) description of the various measures of harm (hazards or dangers) and subsequent comparisons thereof. The second aspect of this paper leads into a presentation of the results of a study which had the following purposes: (1) to develop analytical techniques for a consistent treatment of the phenomenology of the consequences of a release of hazardous materials, (2) to reduce the number of variables in the consequence analyses by development of transportation accident scenarios which have the same meteorological conditions, demography, traffic and population densities, geographical features and other appropriate conditions and (3) to develop consistent methods for presenting the results of studies and analyses that describe the phenomenology and compare hazards. The results of the study are intended to provide a bridge between analytical certainty and perception of the hazards involved.

ANTICIPATED PROBLEMS

There are many possible problems associated with performing relative comparisons of hazards. The most important problem is how the results of a particular comparative study will be received by the public. Typically a statement comparing hazards becomes an advocacy position whether explicitly stated or implicitly indicated. For example, comparisons of natural disasters to hazards of transportation may be informative, but are of little value, because there is a benefit associated with one and not the other. Development of econometric measures of possible effects also may be informative, but will certainly create similar disagreements because of difficulties in setting prices on injuries or fatalities. Analyses of the harmful effects associated with the manufacture, processing, use or transportation of one hazardous material are often stated to be superior or safer than the same activities associated with another material. As a result, those advocating the maligned material or activity quickly assemble their resources and outline deficiencies in the original analysis. In concluding this cycle, it can be stated that both "sides" in this controversy end up with reduced credibility and the public and the decision makers have increased skepticism about both materials and activities.

An example of a problem and resulting controversy created by comparisons of the risks of nuclear power with other energy systems is described by a study performed by Inhaber.[1] The concept of risk to human health divided by the net energy produced by the energy system considering the total energy cycle was used as the comparative resource in this study. The goals of the study were to (1) assess the risks of nuclear power, and (2) place the results of the study into perspective. The results of this study showed that risks (total deaths per megawatt year) for solar power (space heating, thermal, and photovoltaic) were eight to forty times greater than nuclear power when the entire energy cycle was considered (acquisition of raw materials, fabrication of building materials, construction and operation). However, the problems of storage and disposal of nuclear wastes were not considered in this analysis. Regardless of whether the storage and disposal of nuclear wastes pose an actual risk to public health, the exclusion of these activities in the risk assessment violated one of the ground rules of the study by not considering an entire energy cycle. This exclusion is significant because these storage and disposal activities have been the focus of interest groups in criticism of the plans, procedures, and policies developed for nuclear wastes. An evaluation of this study indicates that its results were informative but that the relative safety of nuclear power to solar power or other energy forms was not "proven" conclusively and that the benefits of this analysis to the safety of both solar and nuclear power were questionable. Inhaber's study is

typical of many that have been performed which result in adverse or construed advocacy positions for a system or material. It is sufficient to state that so many analyses have been performed to prove whatever particular result was desired that it has become difficult to perform objective analyses and develop methods to compare hazards. These analyses should be performed only under carefully controlled conditions, with stated objectives, and for selected audiences.

MEASURES OF HAZARDS AND HARM

A variety of methods have evolved or been developed to estimate harm and hazards to people and the environment. Risk assessment activities have resulted in the strongest technical analyses of these hazards. With continued interest in protecting man and the environment from possible adverse effects of hazardous materials, the perception of harm, hazards and risks has led to many positions and regulations which have little basis on the technical facts. In order to place the harm, hazards and risks in perspective, some methods have been developed which define risks in terms of cartesian coordinates. These range, in descriptive terms, from involuntary and certain fatal consequences to voluntary and certain non-fatal effects. Using such descriptions it is possible to plot activities, processes or industries on these scales.[2] Another definition set indicates whether an activity is acceptable by technical analyses and accepted by the public or is unacceptable by technical analyses and not accepted by the public.[3] The relationship of these and other imprecise measures of the perception of risk, hazards and harms to technical analyses is the goal of the study reported in this paper. The problem is extremely complicated because of the many effects that must be considered as possible results of exposure to a release of hazardous materials (e.g., death, injury, illness, birth defects, latent health effects, acute effects, genetic effects, monetary losses for cleanup and reconstruction, contamination, facility denial, evacuation, property damage, species extinction, loss of recreation, reduced aesthetic effects, social impacts in terms of disruption of social processes and reduction in the quality of life). Development of comparative criteria which places appropriate weighting factors on each of these effects has not been universally successful, but is frequently achieved in small groups by decision analyses.

Indexes of harm do provide for comparative analysis and decision procedures. These indexes can be developed with different criteria (scales) describing and linking any of the seventeen considerations noted above or other appropriate effects. An example of using such index of harm and subsequent comparison is outlined later in this paper. Until some acceptance of some quantitative methods emerges, the methodology contained in this paper may be used to provide insight (from a technologist's point of view) into this problem of relative hazards and comparisons of harm in transporting hazardous materials.

CONSEQUENCE ANALYSES

The consequences calculated in this paper are for the effects created by a spill of hazardous materials. The emphasis is on fatalities only, but the methodology developed is applicable to and can account for various other measures of hazard (i.e., injuries, property damage and social effects) previously outlined. The effects of a release of hazardous materials and the consequences associated with the spill are summarized in a flow chart shown in Figure 1.

A general relationship was developed from the various equations describing the possible effects of various hazardous material on human health as follows: (all

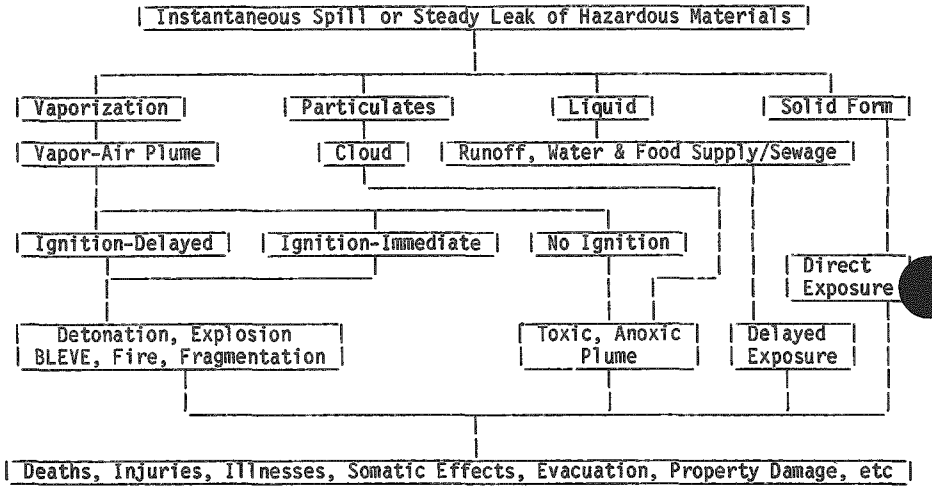


Figure 1. Flow Chart of Possible Effects Resulting from a Spill of Hazardous Materials

categories of materials previously indicated are described by this equation with the exception of corrosives).

$$C = ABF(m)^f/d^a \qquad (1)$$

where

C = Consequence (fatalities, injuries, property damage, etc.)
A = Fraction of population, property, etc., within an area affected by the released material ($0.0 < A < 1.0$, $1.0 - A$ is fraction protected by natural or man-made features)
B = Population density, property value or other appropriate measure of people or geography
F = Constant (threshold factor for fatalities, injuries, damage, etc.)
m = Mass of hazardous material released
f = Factor depending on category of hazardous material
d = Distance from release point
a = Relationship for dilution

Equation (1) is applicable for immediate effects caused by exposure to a fireball, blast overpressure and toxic materials where the cloud produced by instantaneous release is basically spherical in form. With the population density (B) and an estimate of the people shielded from the effects (A) known for an affected area, the consequences resulting from an instantaneous release of materials can be calculated. (For example, for flammable materials, the consequence C is equal to the heat loading $q(Btu/ft^2)$ to cause fatalities within distance x from a source, K is equal to 7785, m is equal to the weight of the material in kilograms and f is equal to 5/6. This product, multiplied by the population

density of the affected area and the fraction of the population affected by the released material, established analytical estimates of the consequences.)

As the released materials begin to be affected by the meteorological conditions and terrain, Equation (1) requires redefinition of and additional terms to describe these more complicated conditions. Among the factors which will help define these conditions that must be considered are the following: (1) dispersion of materials which are usually in vapor-air or particulate cloud forms and including the effect of buoyancy, (2) mitigating or enhancement effects offered by natural or man-made features which aid dilution or concentration of the materials, (3) the nature of the release (instantaneous or continuous) and (4) the type of effects created by the released materials (immediate, latent, genetic, etc.)

Restricting the discussion to health effects (with the continued implied application to other effects), the major contribution to these effects is the dispersion over a wide area of vapor-air-particulate cloud of sufficient concentrations to cause hazards to the public. This dispersion and dilution depends upon the meteorological conditions and whether the material is instantaneously or continuously released. The relationship describing consequences as a function of the materials released is shown in Figure 2. Relationships between the consequences and time are shown in Figure 3.

Figure 2. Relationship between consequences of release of hazardous material and mass and duration of materials released.

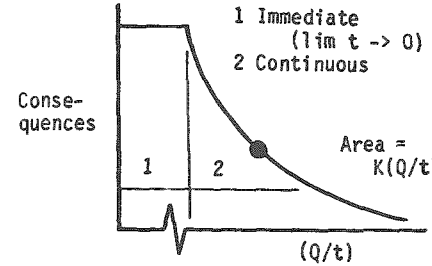
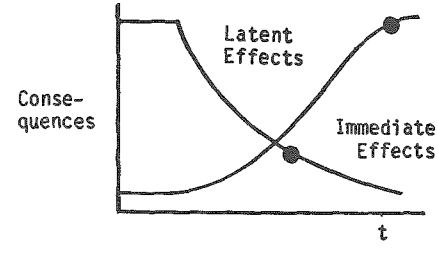


Figure 3. Possible relationships between time for health effects to occur and consequences.



To account for the various factors and relationships for dispersion and dilution, Equation (1) requires modification and redefinition as follows to describe continuous releases:

$$C = ABKQ^{m}(t)^{n} \qquad (2)$$

where

C = Consequences of release of hazardous materials
A,B = Previously defined
Q = Volume of material released

t = Time
m,n = Exponents describing meteorological conditions and material released
K = f (material spilled and area involved) with the Area under the curve in Figure 2 = K (Q/t) because at a given time, Area, Q, n are known as shown in Figure 2 and K is the intercept which can be calculated.

SIMULATED ACCIDENT CONDITIONS

Two accident scenarios (urban and rural) were postulated to develop quantitative comparisons of health effects produced by releases of various hazardous materials. In addition, the two cases would provide a uniform set of conditions to evaluate the methodology and eliminate as much subjectivity in the analyses as possible. The scenarios offer the available pathways (direct exposure, airborne, aquatic and food chain) for possible health effects to occur. The rural location consisted of a freeway overcrossing a state highway adjacent to a river as shown in Figure 4. Also in the immediate area were a gasoline station and a food market. The traffic densities were 20 and 100 vehicles per hour on the state highway and freeway, respectively. The prevailing wind of 2 m/sec was directed toward a town 3 km away which had a population density of 2400 people/km^2. The urban scenario was also a freeway overcrossing a city street (Figure 5). The traffic density was 8000 vehicles per hour on the freeway and 1000 per hour on the city street. The population densities ranged from 35,000 people/km^2 near the source to 20 people/km^2 at 80 km with an approximate geometric decay between. A wind speed of 2 m/sec existed. Tall buildings and other features (e.g., pedestrians) were included.

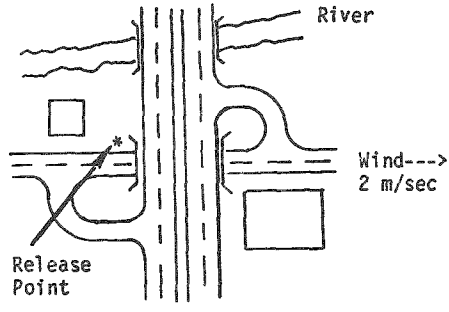


Figure 4. Rural accident location.

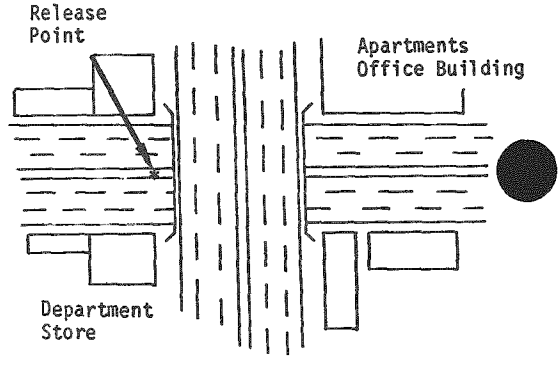


Figure 5. Urban accident location.

RELATIVE CONSEQUENCES

Detailed calculations were made of health effects produced from releases in these two situations using specialized and detailed techniques. Equations (1) and (2) were used as models to relate the results of the more detailed calculations to the determination of the consequences of spills of the following materials: anoxic (chlorine, which is also toxic), high explosives, flammable (LNG and LPG), radioactive (spent fuel) and toxic (parathion and toxaphene). These individual materials were selected as representative of their particular category and because basic characteristics and data describing the materials were available.

Analyses and subsequent comparisons of the results were accomplished through a variety of methods. One of the approaches is outlined in the following schematic shown in Figure 6. Here, the object is to develop a relationship between the measures of harm for materials A and B when shipments are subjected to similar accident environments in similar settings. The central plot of Figure 6 contains measures of harm for materials A and B and the other diagrams indicate the conceptual flow of the calculation. Beginning in the lower left with material A, accident severities (categories I through VIII from NUREG-0170) are related to package response as indicated. The release of materials (if it occurs) is related to the consequences as calculated using the appropriate form of Equations (1) and (2) and either of the two scenarios described. These consequences (in this case health effects only) form the measure of harm. The numbers on the scales merely indicate increasing values. Beginning in the upper right with material B, a similar analytical procedure is outlined in Figure 6. With sufficient calculations for analysis of the effects of various accident severities, a curve comparing the measures of harm of two materials is described as a "figure of merit". Analysis of this "figure of merit" plot can indicate (through deviation from a diagonal line) significant differences in the regulatory climate or risk perceptions of the two materials.

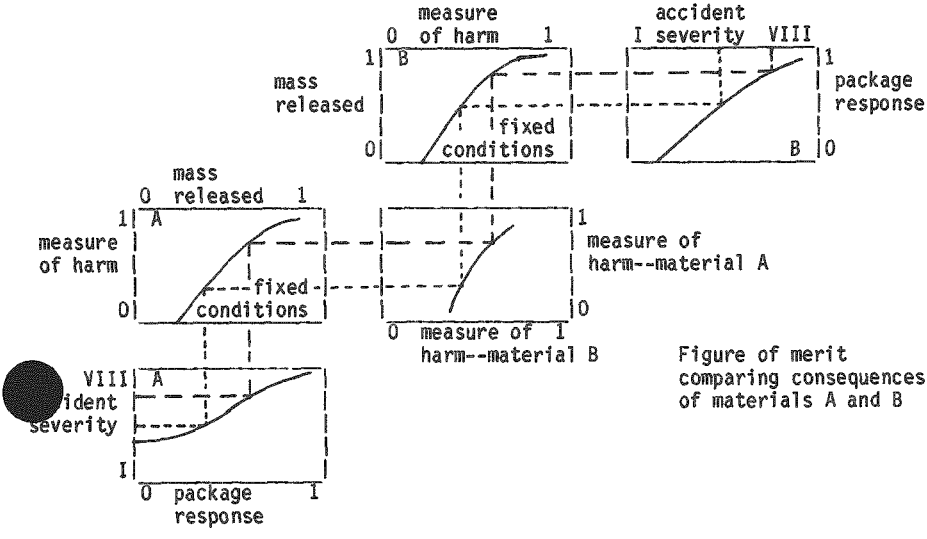


Figure 6. Schematic for Comparisons of Hazards from Two Materials

CONCLUSIONS

Understanding the differences in perception of hazards resulting from transport of various hazardous materials is fraught with difficulties in isolating the qualitative and quantitative features of the problem. By relating the quantitative impacts of material hazards under identical conditions, it is hoped that the perceived differences in material hazards can be delineated and evaluated.

REFERENCES

1. H. Inhaber, "Is Solar Power More Dangerous Than Nuclear Power?," IAEA Bulleti[redacted] 21(1), pp. 11-17, 1977.

2. B. Fischoff, P. Slovic, S. Lichtenstein, "Weighing the Risks," Environment 21(4) pp. 17-20 and 32-38, May 1979.

3. European Atomic Energy Community, Nuclear and Non-Nuclear Risk - An Exercise in Comparability, EVR 6417, Pollution Prevention Consultants, Ltd., Crawley, England, 1980.

DISCUSSION

Question by B. W. Emmerson, United Kingdom: 1. Does your release model for urban areas take into account persons who may be inside buildings and therefore at comparatively lower risk? 2. Have you extended the model to assess the consequence of a release into a conurbation area of high population density, many of whom will be located in multistory buildings? Experience indicates that such an area is difficult to model and such models must be very stylised. 3. Finally, have you had any indication of the degree of acceptance of such presentations by the general public?

Answer: 1. Yes, the release model accounts for protection offered to the public by both natural and man-made barriers through the parameter A, through sensitivity analyses, and the modeling of the dispersion of the vapor-air-particulate cloud. 2. Yes, the model of the dispersion of the cloud of vapor-air-particulate mixture accounts for cloud height and concentrations as functions of height and distance. 3. No, this work is part of an effort supporting analyses and assessments of the perception of hazards of transporting hazardous materials.

Question by W. Heudorfer, FR Germany: In calculating the consequence term of densely populated areas such as towns, do you also have to take effects such as air circulation around the building into consideration? Could you please explain how you do this with parameters n and m of your modified equation?

Answer: The dispersion of the vapor-air-particulate cloud depends upon the density of the cloud, the rate of release, the meteorological conditions and natural and man-made constraints present. Part of this modeling is handled in the term A which indicates a fraction of the people protected. The next part is the Q term which includes the area involved and if geographical constraints are present, then the dispersion is altered. m and n are basically the Pasquill dispersion coefficients and these together with much more complicated but consistent modeling procedures for the various categories of hazardous materials will lead to an adequate assessment of the flow around buildings and other obstacles.

APPLICATION OF ALARA PRINCIPLES TO SHIPMENT OF SPENT NUCLEAR FUEL

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Current U.S. policy is to indefinitely defer reprocessing of spent fuel. This policy has resulted in the requirement that the fuel be stored in a safe manner for extended periods of time until disposal facilities become available. Due to storage limitations at nuclear power plants, away-from-reactor (AFR) fuel storage facilities are planned. Use of these facilities will increase the frequency of spent fuel shipments in the next few years. Shipments to AFRs and subsequent reshipment of fuel to repositories for disposal will result in the exposure of transportation workers and the general public to low levels of gamma and neutron radiation emitted from spent fuel shipping casks during transport. The maximum external dose rates from these shipments is regulated by the U.S. Department of Transportation and the Nuclear Regulatory Commission. Recently, the possibilities of reducing maximum allowable dose rates and requiring that shipments maintain radiation doses "As Low As Reasonably Achievable" (ALARA) has been discussed. This report presents the results of a study (Greenborg et al. 1980) performed by the Pacific Northwest Laboratory to identify and evaluate alternatives in policy, operating practices and equipment design with potential to reduce dose levels within an ALARA philosophy.*

An accurate description and model of applicable portions of the nuclear transportation industry is required to estimate radiation dose from shipping spent fuel. This includes spent fuel shipping schedules, transportation routes, shipping casks, cask loading/unloading procedures, population distributions, shipping crew activities, and shipping procedures. The model that was constructed for this study is summarized below. Details can be found in references to this paper.

Adequate spent fuel storage capacity is required to facilitate the operation of the nuclear power plants. The quantities of fuel being shipped to offsite storage reflect the actual onsite nuclear power plant storage capacities and spent fuel discharge rates. These quantities are also strongly influenced by policies, regulations and facility characteristics for offsite spent fuel storage and disposal. Indefinite deferral of reprocessing and a lack of adequate offsite stora has resulted in many nuclear power plants seeking licenses for added storage capacity through the use of high-density fuel storage racks in existing storage pools and/or construction of larger pools. Construction schedules, capacities and

* Work conducted for the U.S. Department of Energy through the Transportation Technology Center at Sandia Laboratories under Contract DE-AC06-76RLO 1830.

location for spent fuel storage and disposal facilities are not known at this time. These unknowns produce considerable uncertainty in the logistics parameters for spent fuel transportation. Because of these uncertainties, three spent fuel storage logistics cases were examined in the study to represent a range of possible scenarios that could be expected:

1. No reracking of storage pools at nuclear power plants allowed and a full core reserve (FCR) must be maintained so that an entire reactor core could be placed in storage to facilitate repairing of the reactor.

2. No reracking is allowed and a FCR is not maintained (i.e., the space reserved for the fuel from the reactor core is used for spent fuel storage instead).

3. All nuclear power plants rerack to maximum capacity (typically 10 or more years storage) and FCR is maintained.

Shipping schedules were determined using the Variable Site Spent Fuel Transportation Logistics Model (Andrews et al. 1979) developed at PNL. This computerized model calculates minimum-cost shipment schedules between nuclear power plants, away-from-reactor storage facilities (AFRs) and spent fuel disposal sites. The data base used by the model contains the location, storage capacities, and other pertinent information on U.S. nuclear power plants in operation or under construction and planned for startup by 1990. The model assumes spent fuel is shipped from power plants (when necessitated by insufficient onsite storage capacity) to AFR storage sites located in the Midwest, Northwest and Southeast and a spent fuel repository (a disposal site) located in the Southwest. The model determines the age and amount of fuel to be shipped by rail and by truck depending upon the receiving capacity of the AFR and the most economical mode of transportation.

Existing spent fuel casks were used as a reference for this study. Truck shipments in the NAC-1 cask consists of 1 PWR or 2 BWR spent fuel elements. Shipments by rail were assumed to be transported in NLI 10/24 casks. The NLI 10/24 cask can carry 10 PWR or 24 BWR fuel elements. Shipment routes were selected based on interviews with the major transporters of radioactive materials. Actual routes were used when possible. Truck shipments typically move via interstate highway; rail shipments were assumed to be handled as commercial freight. Special trains were not used.

A mathematical model was constructed to calculate the population dose for the transport of spent nuclear fuel. The following groups were considered in the analysis:

- Workers at the power plants and storage facilities who operate fuel handling equipment during loading and unloading of shipping containers,
- Truck drivers or train crews who are exposed during transport of the fuel,
- Bystanders and inspectors at truck stops and train sidings,
- Travelers enroute moving in the same and in the opposite direction of the shipments, and
- Residents living along the path of the shipment within 0.8 km of the right of way.

The population densities enroute and traffic densities along shipping routes were determined from data provided by the U.S. Bureau of Census and the U.S. Department of Transportation.

Table 1 summarizes key findings of this study for the three fuel storage scenarios. Total fuel discharged in each of the three cases is 47,000 MTHM. Using currently available spent fuel shipping casks, the Case 1 option results in a cumulative radiation dose of 11,600 man-rem for shipment of all fuel discharged by the U.S. nuclear power plants through 1990. By comparison, logistics Case 3 results in a cumulative dose of 3,400 man-rem. These calculations assume the existence of a repository that accepts spent fuel by 1985. That is, with a repository available in 1985, Case 1 would result in about 95% of the spent fuel being shipped from power plants to an AFR and eventually reshipped to disposal with the other 5% shipped directly to the repository. Case 3 results in 5% of the fuel being shipped to the AFR and 95% directly to the repository. If a repository is not available until 1995, all fuel would be shipped directly to an AFR in both Cases 1 and 3 for eventual reshipment to a repository.

TABLE 1. Cumulative Radiation Dose for Shipment of all Spent Fuel Discharged from U.S. Nuclear Power Plants Through 1990 to a Disposal Facility

Storage Scenario	Total Spent Fuel Shipments Required Through Disposal[a] (MTHM)	Cumulative Dose (man-rem)	Shipping Cost (10^6)
1. FCR, no rerack	92,000	11,600	1070
2. No reserve, no rerack	80,000	5,500	930
3. FCR, rerack to storage 10 years	49,000	3,400	580

(a) Assumes a repository will begin accepting spent fuel by 1985.

Cumulative dose from Case 3 will increase with delays in the availability of a spent fuel repository. The doses in Case 1 will remain larger, however, due to the shorter decay times for fuel shipped to the AFR. A delay in AFR construction would require storage space expansion at many nuclear power plants. Increased onsite storage would reduce dose because of longer fuel decay times prior to shipment and the possibility of reduced shipping volumes due to direct shipment to a spent fuel disposal facility. The transportation cost of Case 3 is about half that of Case 1. This is due to the reduced shipping requirements. Case 2 costs are between those of 1 and 3. Thus, Case 3 is preferable to Cases 1 and 2 on both dose and cost bases. Similarly, Case 2 is preferable to Case 1.

Radiation exposure from shipment of spent fuel will compose only a small amount of the total exposure experienced in the U.S. If current shipping practices are followed, spent fuel shipments will result in total population exposure of a few hundred man-rem/year of radiation exposure. This exposure can be compared to the 40 million man-rem of radiation exposure received annually by the U.S. population from natural sources, medical examination, television and air travel. Of the

additional exposure generated by spent fuel shipments, less than 10% will be received by the general population. The balance will be received by workers in the nuclear and transportation industries.

Current generation spent fuel casks were designed to accommodate spent fuel that had been discharged from a reactor as little as 120 days prior to shipment. Sufficient radiation shielding is provided to maintain external radiation levels below 10 mrem/hr at 2 m from the cask for this fuel. Spent fuel storage/shipment options currently considered in this report result in average fuel ages at shipment from 2.4 to 6.9 years. Surface dose rates from 5-year old fuel in existing casks are less than 1 mrem/hr at 2 m. It should be possible to develop larger capacity casks designed specifically for longer-cooled fuel. The dose and cost of using these redesigned casks for logistics Case 3 is shown in Table 2. Logistics Case 3 is shown using existing casks for comparison. Designing new casks to the existing 10 mrem/hr at 2-m limit would increase dose by 10,100 man-rem compared to shipments in existing casks even though fewer shipments would be required in larger casks. This results because current casks contain substantially more radiation shielding than is required to meet surface dose rate limits with aged fuel. About $150 M (U.S.) in shipping costs ($15,000/man-rem), could be saved by using these redesigned casks. Redesigning casks to 1 mrem/hr at 2 m would allow for a slight reduction in dose due to increased cask capacities and a reduced number of shipments. Cost savings are less than those for casks designed to the 10 mrem/hr limit. Detailed cost analyses are required to further evaluate this concept.

TABLE 2. Summary of Doses for Alternatives in Transportation Equipment Design

Cases	Cumulative Shipment Dose[a] (man-rem)	Cost[c]
Logistics Case 3[b] (Existing Casks)	3,400	580
Redesign Larger Casks for Aged Fuel (Logistics Case 3)		
Current Regulations (10 mrem/hr)	13,500	430
Reduced Dose Rate Requirements (1 mrem/hr)	3,300	460

(a) Total dose for shipment of spent fuel discharged through 1990; assumes repository startup by 1985.
(b) Expand onsite storage to a minimum of ten years at all nuclear power plants (most fuel is shipped from the reactors to the repository, reducing need of the AFR).
(c) Costs are expressed in millions (10^6) of U.S. dollars in 1976.

Current U.S. Nuclear Regulatory Commission rules prohibit the transport of spent fuel through highly populated regions. Analysis of the routing strategy to bypass high population regions were found to have little effect on overall exposure. Alternative routing will reduce exposure to residents and travelers enroute by about 1/3, but the increased dose to the truck drivers and bystanders at truck stops (resulting from the longer shipping distances) will more than offset the reductions.

In conclusion, public exposure to radiation from spent fuel shipments is low. Occupational and transportation worker doses account for the majority of the total dose. Any efforts to reduce dose should be concentrated in this area. Practices that affect the age of spent fuel in shipment and the number of times the fuel must be shipped prior to disposal have the largest impact on accumulated dose. A policy to encourage a 5-year spent fuel cooling period prior to shipment coupled with appropriate cask redesign to accommodate larger loads would be consistent with ALARA principles. Rerouting shipments to avoid high populations density areas will not reduce total shipment dose due to a corresponding increase in occupational dose.

REFERENCES

Andrews, W. B. et al., Pacific Northwest Laboratory (PNL) Spent Fuel Transportation and Handling Facility Models. PNL-3105, Pacific Northwest Laboratory, Richland, Washington, September 1979.

Greenborg, J., Brackenbush, L. W., Murphy, D. W., Burnett, R. A., Lewis, J. R. Application of ALARA Principles to Shipment of Spent Nuclear Fuel. PNL-3261, TTC-0073, Pacific Northwest Laboratory, Richland, Washington, 1980.

DISCUSSION

Question by B. G. Pettersson, Sweden: In assessing the occupational exposure of cargo handlers which were the external radiation levels used: the regulatory maximum levels, or data from actual dose rate measurements?

Answer: Occupational dose levels were calculated based on shielding calculations and mathematical models of cask handling operations for the reference casks. Surface dose rate models were validated with actual measurements. Time and motion observations were made for cask handling procedures.

Question by M. Brown, United Kingdom: It is not clear to me what proportion of the collective dose is due to handling at the power station. Could you clarify this?

Answer: Occupational doses presented in this paper include cask handling at both the power station and fuel destination. Unfortunately differentiation between these facilities is not given in the paper. Additional information on occupational dose is available in the references.

AN ALARA ASSESSMENT OF SPENT FUEL AND NUCLEAR WASTE TRANSPORTATION SYSTEMS*

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The number of spent fuel and nuclear waste shipments per year may be expected to increase dramatically during the next few decades as the nuclear industry matures and waste management endeavors are expanded. Existing regulations specify allowable dose rates in the vicinity of such shipments. For example, under normal operating conditions the acceptable dose rate at 2 m from accessible surfaces of a spent fuel cask is 10 mrem/hr (IAEA Safety Series No. 6, para. 534.c and 537.c). As the number of shipments increases there may be pressure to restrict the allowable dose rates in accordance with the philosophy of "as low as reasonably achievable" (ALARA).

When changes are considered in the regulations, both the costs and the benefits of a given change should be addressed. In the past, the application of the ALARA concept in the transportation area has been investigated with almost exclusive emphasis on reducing occupational and public radiation exposure. Although radiation exposure is of paramount importance, ALARA effects on transportation system costs, including parameters such as physical dimensions and total lifetime shipping costs also must be determined to adequately justify using ALARA in transportation applications. During the past year some preliminary work in this area has been completed at Sandia National Laboratories.

The recent effort at Sandia evaluated the application of the ALARA concept to light water reactor (LWR) spent fuel, high-level commercial and defense wastes, and remote-handled transuranic waste transportation.[1] This evaluation included: (1) obtaining transportation hardware shield designs which comply with the regulatory 10 mrem/h, 2 m dose rate condition; (2) determining the additional shielding required to decrease the dose rate to 5 mrem/h; (3) determining the additional shielding required to decrease the dose rate to 2 mrem/h; and (4) investigating the effects of the additional shielding on system parameters such as cask weight and dimensions. This evaluation was done for wastes and LWR spent fuel of 1, 3, 5 and 10 years age and for casks with 1, 4 and 7 unit capacities. Gamma shield materials investigated were depleted uranium, lead and steel. The neutron shield material was water.

This study considered pressurized water reactor (PWR) spent fuel (SF) which was initially enriched to 3.3 wt.% ^{235}U. The irradiation sequence assumed for the fuel involved 3-year residence in the reactor with three separate burn cycles and 80% capacity factor. The total burnup sustained was about 15,000 MW days per assembly (33,000 MW days/tonne of uranium).

*Work sponsored by the U.S. Department of Energy under Contract DE-AC04-76-KP00789.

Current national policy in the USA does not authorize reprocessing of commercial reactor spent fuel. Since this policy could change in the future, ALARA impacts on the transportation of high-level commercial waste (HLCW) were investigated. The HLCW considered here was assumed to contain 0.5% of the uranium and plutonium and 100% of the fission products and other transuranics originally in the unprocessed LWR SF. Presently, in the USA, there are approximately 2 PWR reactors for each boiling water reactor (BWR). Since the radiation and thermal characteristics of PWR and BWR spent fuels differ, the HLCW considered was formulated to reflect this ratio of LWR types. The concentration of the waste in some suitable matrix (borosilicate glass for instance) was assumed to be such that the HLCW obtained from reprocessing 1 tonne uranium equivalent of the 2:1 PWR/BWR SF mix resulted in 0.085 m^3 (3 ft^3) of waste product.

High-level defense waste (HLDW), which is similar to but less radioactive than HLCW, is a by-product obtained during the reprocessing of military reactor [redacted]. Large quantities of this waste currently are stored as salts, sludges, liquids and calcine at three US Department of Energy sites.[2] This waste will be concentrated and immobilized in a suitable matrix before being transported offsite. The radioactivity of the waste is quite variable from site to site and within a given site. The transportation of the most radioactive waste was investigated in this work.

Remote-handled transuranic waste (RH-TRU) is transuranic and fission product contaminated material which requires some radiation shielding for safe transport. Relatively small quantities of RH-TRU are in temporary storage at DOE sites. Generalizations concerning its physical characteristics are difficult to make since waste package sizes range from 30 gallon drums to round caissons 240 cm diameter by 300 cm high and larger. The volume of RH-TRU will increase considerably as decontamination and decommissioning programs of DOE facilities are begun in the coming decade and beyond. For the purposes of this study, the proposed Waste Isolation Pilot Project (WIPP) definition of an acceptable RH-TRU package[3] was used. The package assumed is 61 cm diameter, 460 cm long and has a maximum surface dose rate of 100 rem/hr.

Table 1 shows some of the physical parameters of the SF, waste material, and waste containers considered in this study. The radiation and thermal characteristics of each waste type were estimated using the SANDIA-ORIGEN[4] isotope generation and depletion code. The diameter of the HLCW canister was assumed to be 30.5 cm based on thermal considerations (i.e., possibility of waste melt along the canister centerline). This canister dimension could be too large for short cooled HLCW. The HLDW canister diameter was also selected to be 30.5 cm since, at one time, WIPP tests with such containers loaded with HLDW were proposed. The HLDW canister diameter could be increased to that of the RH-TRU container without any adverse thermal effects.

The cask hardware investigated in this study included a cask with a cavity sized to accommodate 1, 4 or 7 assemblies or canisters, a 2.5-cm-thick steel inner wall, a gamma shield zone of adjustable thickness, a 5.0-cm-thick steel structural wall, and a neutron shield zone of variable thickness if such a shield was necessary. Figure 1 shows idealized cross sections of the cask for each loading configuration. In each case, an aluminum basket was assumed to provide support for the waste canisters or fuel assemblies in the cask cavity. Table [redacted] lists the cask cavity diameters assumed for each configuration.

The thicknesses of the shield zones were varied for each waste type and cask capacity to obtain conceptual shield designs resulting in 10, 5 and 2 mrem/h dose rates at 2 m from the cask exterior. In order to obtain reasonably balanced shield designs, the acceptable primary gamma and neutron-secondary gamma dose rate contributions to the total dose rate as given in Table 3 were selected. The

XSDRNPM[5] one-dimensional radiation transport code was used to estimate the required thicknesses of the neutron and gamma shield zones. The calculations were performed using an 11-group, P1 primary gamma cross-section set and a coupled 19-neutron, 13-secondary gamma P1 cross-section set. Past experience has shown that one-dimensional transport calculations performed using these cross section sets yield results which agree well with predictions obtained with more sophisticated Monte Carlo analyses using MORSE.[6] All calculations were done in cylindrical geometry using an S_6 quadrature.

Table 1. Physical Parameters of Spent Fuel and Waste

	PHYSICAL DIMENSIONS	WEIGHT (kg)	THERMAL POWER GENERATION RATE (Wth) 1-Yr-Old	3-Yr-Old	5-Yr-Old	10-Yr-Old
PWR Spent Fuel	21.7-cm dia 420-cm length	660	5000	1700	970	580
High-Level Commercial Waste Canister	30.5-cm dia 305-cm length	750	20000	6600	3600	2100
High-Level Defense Waste Canister	30.5-cm dia 305-cm length	750	230	200	180	150
Remotely Handled TRU Waste Container	61-cm dia 460-cm length	3200	100	95	90	80

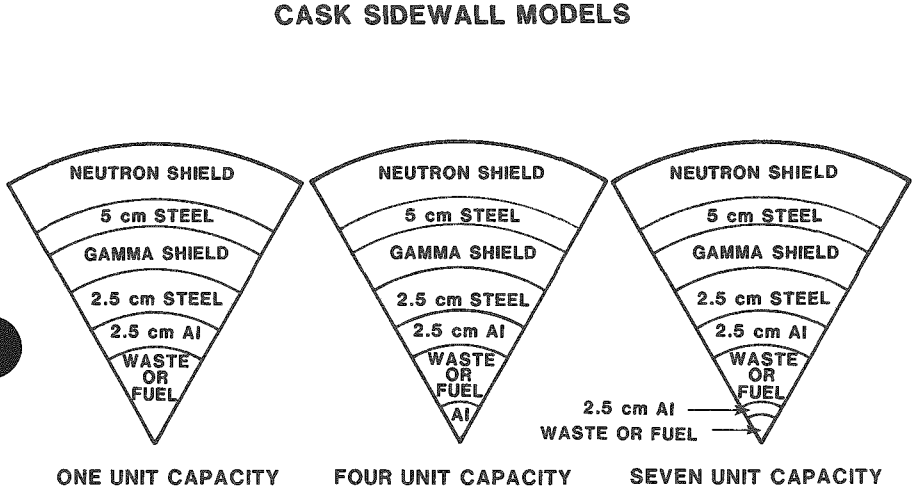

Figure 1. Cask Models for Various Loading Configurations

Table 2. Cask Cavity Dimensions and Capacities

TYPE	CAVITY DIAMETER (cm)	CAPACITY (Number of Assemblies or Canisters)
PWR Spent Fuel	39	1
	75	4
	91	7
HLCW	39	1
	87	4
	107	7
HLDW	39	1
	87	4
	107	7
RH-TRU	66	1
	137	4
	193	7

Table 3. Acceptable Primary Gamma and Neutron-Secondary Gamma Dose Rate Contributions

NOMINAL DOSE RATE AT 2 M FROM CASK EXTERIOR (mrem/h)	PRIMARY GAMMA (mrem/h)	NEUTRON-SECONDRY GAMMA (mrem/h)
10	7.0 ± 2.0	3.0 ± 2.0
5	3.5 ± 1.0	1.5 ± 1.0
2	1.5 ± 0.5	0.5 ± 0.5

The reasons for considering waste and SF cooled only for up to 10 years should be mentioned before considering the results obtained in this study. One reason is that a significant fraction of the SF stored in the USA is approaching 10 years since being discharged. A more fundamental reason is that the radiation source associated with each waste changes dramatically during the first 10 years of cooling but much more slowly thereafter. This is indicated in Figure 2, which shows the ratio of the neutron and gamma sources for each year decay to that at 20 years cooling time as a function of cooling time for PWR SF. The curves are quite steep for the first 10 years of cooling and nearly flat during the subsequent 10 years. Thus the shielding requirements of the material do not change significantly during the second 10 years.

The results obtained in this study for each waste type, cooling time, cask capacity, and dose rate are too extensive to detail in full here. To indicate trends, only the results for the 7 element capacity PWR cask will be given. In Figure 3 are shown the estimated radii (not including cooling fins) of conceptual spent fuel casks designed to transport 7 PWR fuel assemblies as a function of uel age and 2 m dose rate. The left portion of the figure is for casks in which depleted uranium provides the necessary gamma shielding. The center portion illustrates similar results for lead shielded casks. On the right are shown the

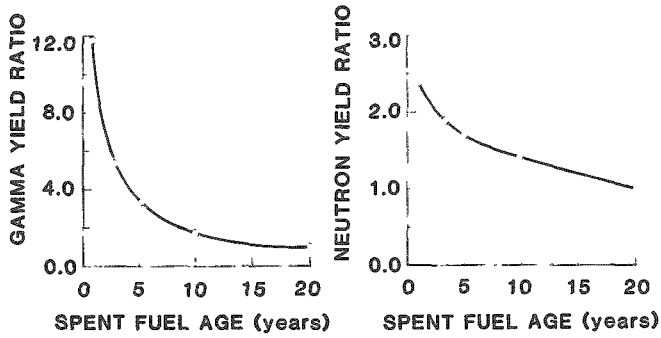

Figure 2. Temporal Variation of Radiation Source Strength for PWR Spent Fuels

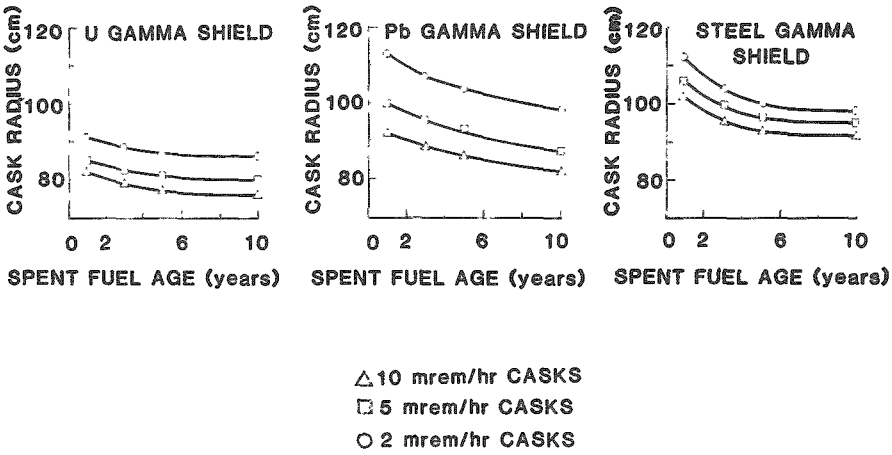

Figure 3. Estimated Radii of Seven Element Spent Fuel Casks

results for steel casks. In general, the radii of the depleted uranium casks are less than the corresponding lead and steel shielded casks because uranium is the best gamma shield material of the three considered. The radius of the cask is important since transportation-imposed constraints limit this dimension to probably no more than 105 cm (without cooling fins). From the figure it is seen that the lead and steel shielded casks for short cooled fuel which yield 2 mrem/h dose rates and the steel shielded cask with the 5 mrem/h dose rate for 1 year cooled fuel do not meet this requirement.

Figure 4 shows the estimated empty weights of the casks indicated in Figure 3. Again, three sets of curves are given: the left for uranium shielded casks, the center for casks with lead gamma shielding, and the right for steel casks. The weights of the lead and steel casks are more sensitive to waste age and 2 m dose rate than are the uranium casks.

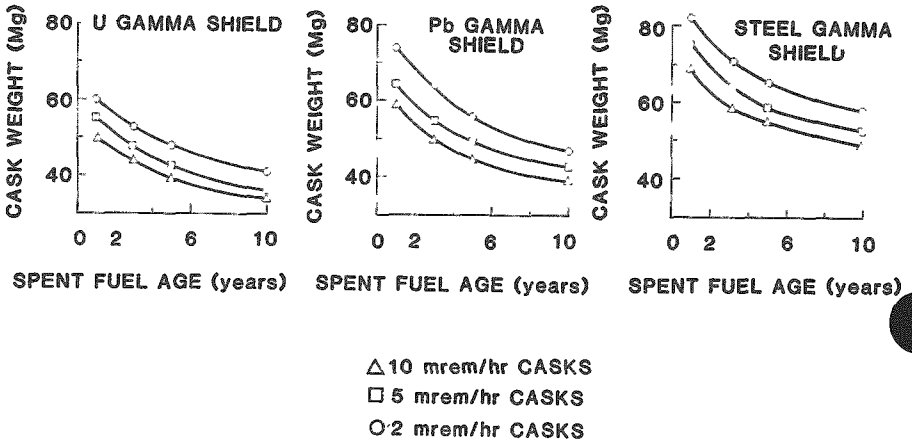

Figure 4. Estimated Weights of Empty Seven Element Spent Fuel Casks

The percentage increases in the weight of each cask when the design dose rate is decreased from 10 mrem/h to 5 or 2 mrem/h as a function of cooling time for each gamma shield material are shown in Figure 5. By halving the regulatory dose rate, the cask weight increases by about 6 to 10% in each case. A further reduction of the acceptable dose rate by a factor of 2.5 causes an additional 10% increase in the weight of the uranium and steel casks relative to the corresponding 10 mrem/h casks. The additional weight percent increase is even higher for the lead shielded casks, being around 15%. The reason the lead casks weight percent increases are higher apparently is because lead is the poorest neutron shield material of the three gamma shields considered. The neutron shielding problem is more important at the lower dose rate.

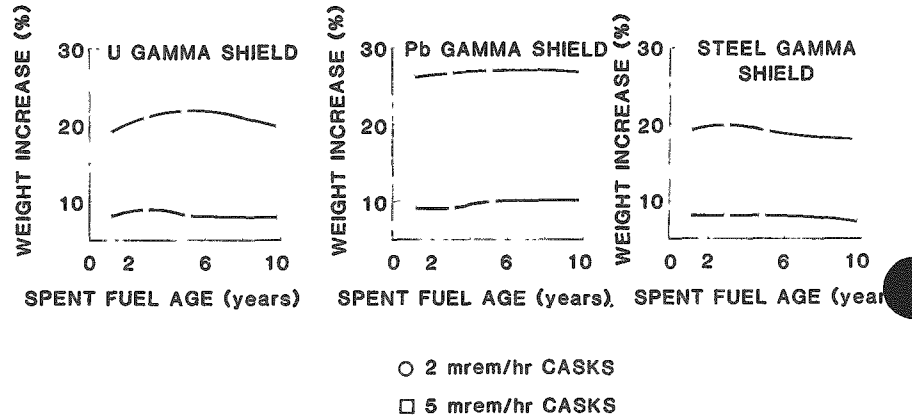

Figure 5. Weight Increase in Empty Seven Element Spent Fuel Casks Due to Reduction in Allowable Dose Rate

Figure 6 gives estimates of the lifetime transportation costs of the casks, in 1978 U.S. dollars, based on the weight estimates of Figure 4. These cost estimates do not include fabrication costs. Assuming a 20 year lifetime and round-trip distances of 3200 km with 15 roundtrips per year at 9.3 cents/Mg-km including both haulage and usage fees, the estimates for lifetime transportation costs, as given in Figure 6, are obtained. From the figure it is apparent that a lead shielded cask would cost approximately $1 million and a steel cask $2 million more to operate over their lifetimes than a corresponding uranium shielded cask. Such additional operating cost may be recovered since lead and, particularly, steel casks are less expensive to fabricate than uranium casks.

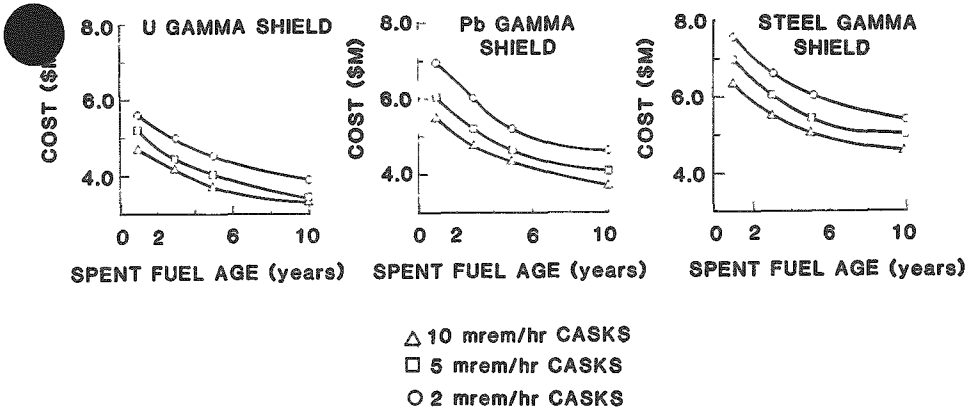

Figure 6. Estimated Lifetime Transportation Cost of Seven Element Spent Fuel Casks

Similar results are obtained for the 1 and 4 element SF casks and for the other waste casks considered in this study. Some of the overall significant results may be summarized as follows: (1) some low dose rate cask designs may be oversize or overweight if lead or steel provide the gamma shielding, (2) designing for 5 instead of 10 mrem/h increases the cask weight by about 10% and for 2 mrem/h adds another 10% or more to the weight, and (3) the increased weight of the low dose rate casks results in an estimated $1 to $2 million increase in the cask lifetime transportation costs.

REFERENCES

1. S. H. Sutherland, ALARA Studies on Spent Fuel and Waste Casks, SAND79-1841, Sandia National Laboratories, Albuquerque, New Mexico, April 1980.

2. D. E. Bennett and S. H. Sutherland, Defense High-Level Waste and Spent Fuel Characterization for Geologic Waste Repositories, Sandia National Laboratories, Albuquerque, New Mexico, September 1979.

3. P. D. O'Brien, Waste Characterization for WIPP Safety Analysis, SAND79-0420, Sandia National Laboratories, Albuquerque, New Mexico, to be published.

4. D. E. Bennett, SANDIA-ORIGEN User°s Manual, SAND79-0299, Sandia National Laboratories, April 1979.

5. N. M. Greene, et al., AMPX: A Modular Code System For Generating Coupled Multigroup Neutron-Gamma Libraries From ENDF/B, ORNL/TM-3706, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1976.

6. S. K. Fraley, User's Guide to MORSE-SGC, ORNL/CSD-7, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1976.

DISCUSSION

Question by K. Schneider, FR Germany: 1. What is the liriting factor for seven fuel assemblies per cask? 2. Is it right that a dose reduction by a factor of ten, as mentioned in the previous paper, would be no more practical? 3. You did not mention the fabrication costs of an uranium shielded cask. Do you think it is possible that the high costs could delete this concept?

Answer: 1. Preliminary scoping studies indicated that casks of larger capacity would exceed weight and size limits for use in the US. 2. Yes, a factor of ten reduction to 1 mrad/h at 2 m would aggravate the situation even further and drive size, weight and costs even higher. 3. At the present time there is no industrial capability in the US for the fabrication of uranium shielded casks. As a result it would be prohibitively expensive to fabricate such a unit. Uranium was included in the study simply because it is the most efficient shield material currently in use.

ASSESSMENT OF RISKS INVOLVED IN HANDLING AND TRANSPORTATION OF SPENT FUEL FROM LOVIISA NUCLEAR POWER STATION

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1. INTRODUCTION

The objectives of this study include the assessment of the radiation doses that the employees and the environment are exposed to during spent fuel handling, storage and transportation operations. Both normal operation and accident situations have been considered. Furthermore one goal of this study was to consider means to decrease the arising risks.

At the Loviisa nuclear power station including two WWER-type units of 440 MWe the spent fuel is transferred during refuelling to fuel pool inside containment and stored there for about two years. After that period the spent fuel assemblies are loaded into transfer baskets and moved in transfer casks to the storage pools in the auxiliary building of the unit 2. After additional storage of about three years transfer baskets containing spent fuel assemblies are reloaded into transport casks and transported first by truck and then by rail out of the country to the USSR.

The main emphasis in this paper is paid on the occupational risk assessment of different spent fuel handling stages /1/. Consequently the consideration of the risks to the population caused by storage and transportation is mainly restricted to the comparisons to be made in section 4 between some fuel cycle stages.

2. DESCRIPTION OF THE REFERENCE SYSTEM

The fuel assemblies of the Loviisa reactors are hexagonal and include 126 fuel rods. Each year about one third of the fuel assemblies in the reactor core are removed and replaced by fresh ones. At discharge the fuel burnup on an average is 28.6 GWd/tU and the spent fuel contains 0.8...1 weight per cent U-235 and 0.6...0.7 weight per cent plutonium. The total fission product contents per ton U initially charged is about $5 \cdot 10^9$ GBq (calculated with the ORIGEN-code) immediately after shutdown. Similarly the total transuranium contents is about 10^9 GBq/tU. The radionuclides contained in the fuel matrix, in the fuel-clad gap and in the hardware components of the assemblies constitute the potential for the risk to be analyzed in this study.

The spent fuel handling operations within the power station are executed using the following sybsystems:

- Fuel handling machine for transfers of fuel assemblies and control rods from the reactor vessel to the refuelling pond
- Refuelling pond inside reactor containment
- Cylindrical pool, well, for transfer basket or cask
- Spent fuel storage pools in the auxiliary building of unit 2
- Transfer cask for transfer of irradiated fuel from refuelling pond to spent fuel storage
- Transfer baskets, each for 30 assemblies, used in the transfer with the cask and during storage in the auxiliary building
- Cranes in reactor hall and in the storage room of auxiliary building for transfers of cask and basket
- Tightness check flasks between the reactor pit and refuelling pond
- Leak tight bottles for leaking assemblies
- Transport wagon for transfer of spent fuel cask from refuelling ponds to spent fuel storage in auxiliary building.

The fuel handling machine carries out unloading of spent fuel assemblies to the refuelling pond, shuffling operations ,loading of fresh fuel assemblies from basket located in the well. Beyond the refuelling the machine transfers the 2 years old assemblies from the refuelling pond to the transfer basket, which is then in the transfer cask in the well. On an average 440 hoistings are made during one refuelling period in the reactor cavity and about 600 hoistings in the refuelling pond. The fuel handling machine has a versatile protection system to prevent damage to the fuel being transferred and to the assemblies in the core or in the refuelling pond.

The refuelling pond is designed to contain about 450 fuel assemblies or control rods and a possibility is provided for storing 60 leaking assemblies in special leak tight bottles. At the equilibrium condition the pond contains just after refuelling three sets of about 120 assemblies having cooling times of about 10 days, 1 year and 2 years, respectively. Before next refuelling the oldest assemblies are transferred away employing a transfer cask containing 30 assemblies in a basket.

The transport wagon having a capacity of 125 tons is designed for transportation of the transfer cask between the reactor containment and the spent fuel storage in the auxiliary building. The transport occurs on rails through a transfer corridor inside the reactor building and over the yard in case of transfers from reactor unit 1.

3. ASSESSMENT OF RISKS INVOLVED IN SPENT FUEL HANDLING DURING NORMAL AND ACCIDENTAL SITUATIONS

3.1. Risk Assessment Methodology

The risk assessment includes the identification of potential accidents and the quantification of the probability and magnitude of the release of radioactive materials associated with the accidents. Based on the estimated amounts of radionuclides released

into the containment or storage room the occupational doses and releases to the environment are determined. As the final step of the risk assessment the probability and consequence estimates are combined.

The quantification of the probabilities is based on failure rate approach and the employment of data applicable to the pertinent handling and storage operations. In addition to this a systematic reliability analysis of the cooling system has been carried out. The estimation of the magnitude of radioactive releases includes generally three major factors: the extent of fuel rods damaged, fraction of radioactive materials available for release in the considered environment and the amount of different radionuclides released through the cooling water. All these factors are difficult to be determined theoretically. For example the fuel resistance is strongly dependent on the direction of the impacts and on the behaviour of energy absorbing structures. Consequently the analysis for this area employs either judgements based on the dimensions and weight of the dropped objects or information available in the literature. In case of dropping accidents during internal transfers, where the damping effect of pool water does not affect, the acceleration forces impacting the cask may be considerably higher. For these type of situations some limited structural analysis was carried out. Concerning the amount of radioactivity released from damaged fuel rods and retention by pool water mostly the information available in the literature has been employed. Because of the uncertainties involved in the evaluation of fuel damage and activity released each accident was analyzed using two alternative sets of input data having different degree of conservatism.

The assessment of occupational radiation doses takes into account external exposure to activity in the room air, in cooling water and on all the surfaces and internal exposure due to inhalation of contaminated air. The radiation doses were estimated separately for the whole body, skin (mainly due to beta-emitters) and thyroid. In the calculation of external gamma-dose from the activity in the air the finite dimensions of the reactor hall or fuel storage room are taken into account, whereas the beta-exposure is based on immersion in a semi-infinite cloud. In the calculation of the releases to the environment the absorption by the filters in the recirculation ventilation system has been considered.

3.2 Description of Spent Fuel Handling Accidents

Besides the inventory of radioactive materials contained in the reactor core the other major sources of radioactivity that have to be handled or stored can be divided into two main categories. During refuelling the spent fuel assemblies just discharged from the core represent the by far greatest amount of radioactivity and remain so for about one month after refuelling. The second category of sources comprises the fuel assemblies stored in the refuelling pond and in the fuel storage.

In this study the consideration of the spent fuel handling accidents was carried out within the following main categories: refuelling, storage and internal transfers. During refuelling the most evident incident to cause fuel damage is the drop of a spent

fuel assembly into the core or the spent fuel pool. In the conservative case all the rods in the dropped assembly are assumed to be damaged and 10 per cent in the realistic case. Some other ways for fuel damage than dropping have been identified, but the consequences are likely to be quite similar and the estimated probabilities of occurence are significantly smaller and consequently they have minor contribution to the risk. An example of these other types of incidents leading to fuel damage is the rupturing of the assembly during tightness check operations. Another type of dropping accidents is the falling of a heavy component on the fuel assemblies in a pool. The number of assemblies to be damaged depends on the weight and size of the dropped component. For example the dropping of one of the four covers for refuelling pond has been assumed to cause damage to claddings in all rods of 25 assemblies. Increased external exposure to workers without fuel damage can be caused in case the electrical switces fail to prevent the lifting of a spent fuel assembly above the predetermined transfer level. The formation of a critical configuration presupposes multiple malfunctions and unauthorized operations or array disruptions in the connection of dropping accidents. Taking into account that the consequences are not likely to exceed significantly those of other incidents and the very low probability of induced criticality the risk contribution is minor.

During storage period either in the refuelling pond inside the containment or in the spent fuel storage in the auxiliary building incidents caused by the drop of heavy component are considered as well; taking naturally into account the longer cooling time. The possibility for loss of cooling accidents has also been studied and the most unfavourable situations in this respect are immediately after the whole core has been discharged to the refuelling pond. Since the descent of water level in the refuelling pond to the top of fuel assemblies by evaporation of water would last at least 10 hours, there is ample time available either to arrange a substitute cooling procedure or at least find the means to maintain sufficient coolant inventory. Therefore no quantitative estimates were made for the occupational exposures due to this kind of events. The accidents brought about by external causes have been considered only succintly in this study.

As concerns the internal transfers the following types of incidents have been considered: dropping of transfer cask or basket, loss of cooling in the transfer cask and lift of spent fuel above the transfer level. Generally the contribution to the total occupational risk caused by internal transfers is inferior to that brought about by refuelling and storage accidents.

3.3 Results of Risk Assessment of Spent Fuel Handling during Normal Operation and Accidents

The occupational exposure caused by the spent fuel handling and storage operations has only a small influence on the total occupational doses. During refuelling the total occupational dose would be about $3 \cdot 10^{-3}$ man Sv/a and that is caused by the external exposure through the pool water both from the fuel assemblies and activity in the water and from contaminated hoist

equipment. The storage phase causes a total occupational dose of about $1.5 \cdot 10^{-3}$ man Sv/a the main contributor being the working close the open refuelling pond. During internal transfers the working close to the cask is estimated to cause an occupational exposure of about $2 \cdot 10^{-3}$ man Sv/a. The total estimated annual occupational dose from all spent fuel handling and storage routines does not exceed 0.01 man Sv/a (whole body).

In Table 1 the different accident types are shown with the estimated total probability of occurrence for the failure sequences In addition the occupational risk and risk for noble gas releases to the environment are shown. Here the risk is defined as the consequence multiplied by the frequency of occurrence and summed within each subcategory. The consequences presented in this table are based upon the assumptions made for the conservative case. The total expectation value for occupational whole body exposure is about $2 \cdot 10^{-4}$ man Sv/a and similarly $2.4 \cdot 10^{-3}$ man Sv/a for skin and 0.14 man Sv/a for thyroid. The case where the more realistic assumptions in the estimation of consequences have been employed is not possible to be discussed in detail within the scope of this paper, but the total expectation values for occupational exposures to whole body, skin and thyroid are in that case $1.4 \cdot 10^{-5}$, $2.4 \cdot 10^{-4}$ and $1.7 \cdot 10^{-3}$ man Sv/a, respectively. Consequently the contribution of the considered incidents during spent fuel handling and storage operations is considerably smaller than that of normal operational routines.

4. EVALUATION OF THE RESULTS

This study indicates that the risk from spent fuel handling accidents is dominated by those potential accidents that are caused by drop as the initial event. It was found that the following factors contributed significantly to the calculated risk: failure rate of transfer and lifting equipment, the amount of radioactivity released from the fuel, and behaviour of the personnel in emergencies. The large number of individual fuel assemblies lifted and transferred annually with the fuel handling machine results in this large contribution from dropping accidents in relation to other incidents to the overall risk.

The normal occupational doses and risks arising from spent fuel handlings should be seen in the perspective of the overall radiation detriments from all similar operational activities in a nuclear power plant. The other important acitivities in this respect are maintenance, repair and inspection works performed during a refuelling period and conditioning of reactor wastes. All these activities expose persons to radiation, and the accumulation of radiation dose in all these ways should be quantified in amount and probability to prepare a sound basis for consideration of ways to decrease the risk from spent fuel handling and storage. However, a substantial feature of spent fuel handling risk is its considerably greater risk potential compared to the risk potential of the other operational activities exposing employees to radiation risk at nuclear power plants.

Concurrently with the assessment of spent fuel handling risks caused by operations within the power station a similar study was

Table 1. Summary Results of Risks to the Personnel and Environmental Risk (Details given only for conservative case)

Accident/Accident type	Probability (1/a)	Personnel risk (man Sv/a/unit) whole body	skin	thyroid	Risk to environmental * releases (GBq/a/unit)
1. REFUELLING					
1.1 Drop of assembly	3.5E-4	1.2E-4	1.4E-3	8.1E-2	7.7
1.2 Rupture of assembly	2.5E-5	2.6E-7	3.1E-6	1.8E-4	1.2E-2
1.3 Drop of component on fuel	5.0E-4	7.5E-5	9.5E-4	5.2E-2	5.0
1.4 Lift above transfer level	3.5E-6	2.1E-9	-	-	-
1.5 Criticality	1.0E-7	-	-	-	-
2. STORAGE					
2.1 Drop of component into pool	3.1E-4	2.0E-6	6.7E-5	4.5E-3	1.3E-1
2.2 Loss of cooling in pool	2.0E-8	-	-	-	1.6E-1
2.3 Releases due to ext. causes	1.0E-8	-	-	-	1.0E-4 (iodine 1.0E-3)
2.4 Criticality	3.0E-7	-	-	-	
3. INTERNAL TRANSFERS					
3.1 Drop of loaded cask or basket	1.2E-5	4.9E-9	2.5E-6	3.0E-9	1.1E-3
3.2 Loss of cooling in cask	4.0E-6	-	-	-	7.6E-6
3.3 Lift above transfer level	8.0E-8	2.6E-11	-	-	-
TOTAL (Conservative)		2.0E-4	2.4E-3	1.4E-1	1.3E+1
TOTAL (Realistic)		1.4E-5	2.4E-4	1.7E-3	1.4

*In most cases iodine is removed by filters; noble gases as Kr-87 eqv.

directed to the assessment of the risks of spent fuel transport to the population /2/. In this paper it is not possible to present the details of that study. The collective dose including both the employees and the population was estimated to be 2 10^{-3} man Sv/a during normal transportation. The risks caused by the transport accidents are shown as a cumulative probability distribution for collective dose in Fig. 1. For comparison the figure also includes the results of a study, where the environmental effects of loss of coolant accidents at spent fuel storage facility were assessed. Even if the possibility of fuel meltdown in connection of that kind of accidents is very remote that type of situation was considered as well. Similarly the results of the assessment of risks of reactor accidents at the same site are shown. It can be clearly concluded that the spent fuel transport represents clearly smaller risk to the population than spent fuel storage or reactor accidents. The similar conclusion can be drawn from the estimated expectation values of collective doses of 2 10^{-6} man Sv/a for spent fuel transport, 1.10^{-4} man Sv/a for fuel storage or 3 10^{-2} man Sv/a assuming fuel meltdown and about 1 man Sv/a for reactor accidents. Concerning both the employees and the population it can be concluded that the spent fuel handling, transfer and transport operations are not the main contributors to the overall risk from the nuclear fuel cycle.

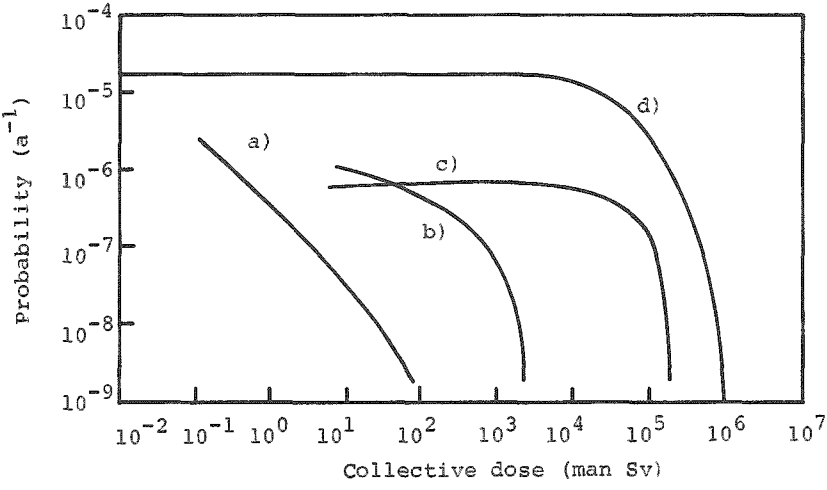

Fig. 1. Comparison of Complementary Cumulative Probability Distributions for Estimated Collective Doses from Some Fuel Cycle Stages.
a) Spent fuel transportation /2/, two units of 440 MWe
b) Spent fuel Storage for 1000 MWe (no fuel meltdown) /3/
c) Same as b) assuming fuel meltdown /3/
d) Nuclear power plant accidents (BWR1...5/1000 MWe) /4/.

REFERENCES

1. KAIKKONEN, H. & VILKAMO, S., Assessment of Risks Involved in Handling Spent Fuel at Loviisa Nuclear Power Stations, Feb. 1980, Technical Research Centre of Finland, Nuclear Engineering Laboratory.

2. RÄMÖ, E., Assessment of Risks Involved in the Transportation of Spent Fuel from Loviisa Nuclear Power Station, Dec. 1979, Technical Research Centre of Finland, Reactor Laboratory.

3. KAIKKONEN, H. et al., Interim Storage of Spent Fuel (in Finnish) Apr. 1980, Technical Research Centre of Finland, Report 48 of Nuclear Engineering Laboratory.

4. ELMAHGARY, Y., KÄRKKÄINEN, S., RANNE, A., SAVOLAINEN, I. & VUORI, S., Economic Analysis and Probabilistic Site Evaluation of Nuclear Plants Producing District Heat and Electricity. Second International Total Energy Congress, Oct. 8-12, 1979, Copenhagen. Miller Freeman Publications, San Francisco, 1979.

DISCUSSION

Question by G. Wehner, FR Germany: The risk spectrum for nuclear reactor accidents you give in Fig. 1 shows collective doses up to 10^6 manrem. This corresponds to about 100 latent cancer facilities. In the German reactor risk assessment study the corresponding value for l.c.f. is at least one order of magnitude higher. Can you explain the reason for this difference?

Answer: The paper includes a c.c.d.f. for collective doses due to reactor accidents and the maximum consequences are about 10^6 manSv not manrem. Consequently this corresponds to about 10,000 cancer fatalities. Nevertheless, it is still lower than the corresponding figures in the German Reactor Safety Study.

Question by K. Schneider, FR Germany: You mentioned criticality accidents. Are they possible during transportation too?

Answer: The possibility of criticality accidents during refuelling or spent fuel storage has been considered to be very improbable, because the formation of a critical configuration requires in addition to severe mechanical deformations, simultaneously lack of boron in the water and low burn-up. During transportation the possibility of criticality requires similarly large deformations to the cask and the fuel and subsequently immersion into pure water. On the basis of a route survey, however, there were practically no places where hard massive targets existed near water. Consequently the probability of criticality situations is very low.

A METHOD TO DEVELOP DATA SUPPORTING CONSEQUENCE ANALYSES OF TRANSPORTING NUCLEAR MATERIALS IN THE UNITED STATES*

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INTRODUCTION

Risk assessment techniques for the analysis of transportation of nuclear materials require supporting data. The data are needed to define the two basic terms of the risk equation -- the probability of and the consequences resulting from an event involving a shipment of nuclear materials. The data include accident frequencies and severities, response of packages, the routes of shipments and road types which indicate possible kinds of accidents and estimates of the consequences resulting from a particular accident. Much of the data is available and can be analyzed for a particular event. However, the response of nuclear materials to transportation environments, the interactions of the materials and packages and measured data describing materials released from packages in any conditions (with emphasis on extreme environments) are not generally available. This paper presents a method to develop the data needed to support consequence analyses.

In the United States when sufficient data are lacking for statistical analysis of an event for purposes of developing an environmental impact statement, it is necessary to postulate worst-case conditions in order to be consistent with the purpose and intent of the National Environmental Policy Act of 1969 (NEPA). Analyses of worst-case accident/incident conditions have resulted in conservative estimates of package response, materials released and, often, predictions of significant consequences. While the implied probabilities of worst case scenarios are very low, these probabilities are often ignored or misunderstood by the public and regulatory agencies. Quantitative data are needed to define possible consequences when shipping system for nuclear materials is subjected to extreme environments, including possible NEPA worst-case environments.

This paper presents observations on the needs for data, the specifications that data obtained must meet and methods currently used to obtain the data. Also included is a rationale for testing shipping systems for nuclear materials in extra-regulatory environments. The emphasis in this paper is on the materials transported in Type B containers particularly commercial spent power reactor fuel.

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

TESTS IN EXTRA-REGULATORY ENVIRONMENTS

The tests of packagings subjected to regulatory requirements are defined in the Code of Federal Regulations (49 CFR 173.398). Figure 1 indicates the conceptual relationship between the environment experienced by the package and its failure to withstand the environment as indicated by the fraction of contents released. Successfully meeting the regulatory test generates the data point which shows compliance with the regulations (Point D). There is a perception of package response that suggests (incorrectly) that at levels or intensities slightly above these regulatory requirements, the containers will fail or exhibit some degradation in which materials would be released to the biosphere (curve A). In environmental impact statements, safety analysis reports for packages and other safety analyses, a release of material is always assumed and the ranges of releases are shown in the shaded area (Area B). This diagram illustrates that: (1) in environments slightly exceeding the regulatory requirements, it is assumed that a release of the major fraction of the contents occur and (2) the release of materials in "realistic" analyses shown in Curve C is much less severe. However, for many packagings these relationships have not been proven to exist. These correlations can be defined only with additional experimental evidence. The shipping systems designed to meet the current requirements have such large margins of safety that the regulatory test requirements do not approach failure thresholds of the containers. This is indicated by Point D in the diagram. As a result, the only method available to provide experimental data to define the limit of the shaded area is to perform tests above the regulatory requirements. Performance of these tests should not be construed as an indication that the current regulation for shipping systems are inadequate or that the packages are not safe. The technical arguments advanced by critics of transportation of nuclear materials need to be addressed in objective terms with data describing the response of packages and contents to a variety of extreme conditions. These extra-regulatory tests will provide data defining failure modes and thresholds for packages and contents. These tests will provide the only technically defendable evidence of whether a release of materials could occur and of the characteristics of the materials that might be released.

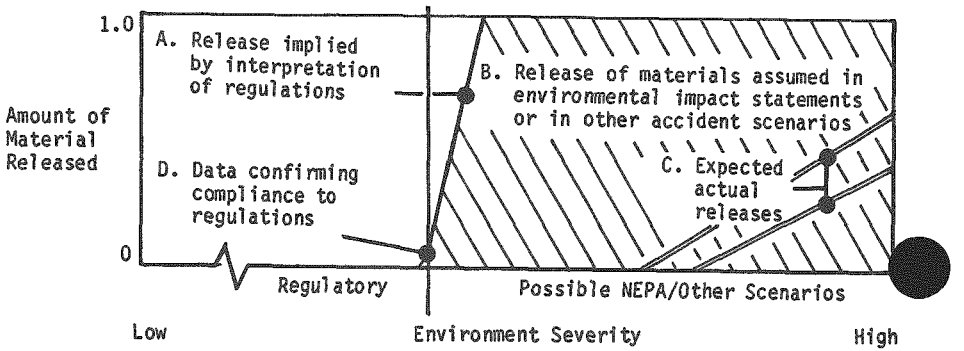

Figure 1. Possible relationships among regulatory conditions and assumed and expected source terms.

DATA NEEDS FOR CONSEQUENCE ANALYSES

The analytical models describing the consequences of a release of nuclear materials require certain basic data. These data define a source term which describes the characteristics and amounts of materials released. The data should be available for possible environments in which releases of materials could occur and should include conditions usually postulated in EIS worst-case scenarios. The following specific information is needed for the consequence models: (1) amounts of materials released, (2) the particle size distribution of the released materials, (3) the chemical species, isotopes and corrosion products (crud) and (4) the rates of release. The particle size distribution ranging from 0.01 m to 2000 m (aerodynamic diameter) is necessary to define and analyze possible health effects, clean-up costs and contamination. Consequences are a function of the elements and isotopes of the materials. The rates of release affect the dispersion and dilution of the materials as they enter the environment. If data can be obtained using experimental procedures which will give test results which bound the ranges of actual conditions, then the data will have broader application and utility.

SPECIFICATIONS FOR DATA FOR CONSEQUENCE ANALYSES

Experimental data describing the response of nuclear materials to severe environments are only partially available. Among the potentially damaging environments likely to be encountered during transportation are shock, vibrations, thermal loadings and pressure (internally and externally applied). The source term data should be correlated to the magnitude and type of environment. Data format, measurement techniques and data acquisition systems used should be consistent among the various research programs. The additional specifications for the data are the following: (1) the response of the contents (e.g., spent fuel) should be documented with particular emphasis on the mechanisms, failure modes and loadings conditions which mitiate the formation of particulates, (2) the response of the packages to extreme environments with a similar attention to failure modes and aerosol formation mechanisms, (3) the interactions of the packages and contents with emphasis on the features offered by the container which mitigate or enhance the formation and transport of particulates within the container and (4) actual measured source terms.

Data are required on the response of contents to determine what conditions cause failures. It is possible that spent fuel may exhibit very similar failure mechanisms for different loading conditions (e.g., externally applied pressure may produce similar results to a shock loading). There is evidence that suggests that UO_2 particle sizes should follow a log-normal distribution at relatively low energy loadings. There is also evidence that UO_2 particle size distributions are bimodal at higher energy loadings. Additional evidence also suggests as the intensity of a particular environment increases (strain rate, thermal loading), that there is a corresponding shift in the weight percentage of smaller particles. However, for spent fuel, limits to the weight percentage of particles smaller than 10 m to appear because of physical limitations in application of a particular loading condition over the length of a fuel rod or adjacent fuel rods contained in an assembly. The resulting relationships are shown in Figure 2 of a log-log plot of particle size as a function of weight percentage of particles that are of a size equal to or smaller than the amount indicated. The various intensities of environments will be related later. It is sufficient for purposes of this discussion to assume that definable relationships between varying strain rates, thermal loadings and externally and internally applied pressure.

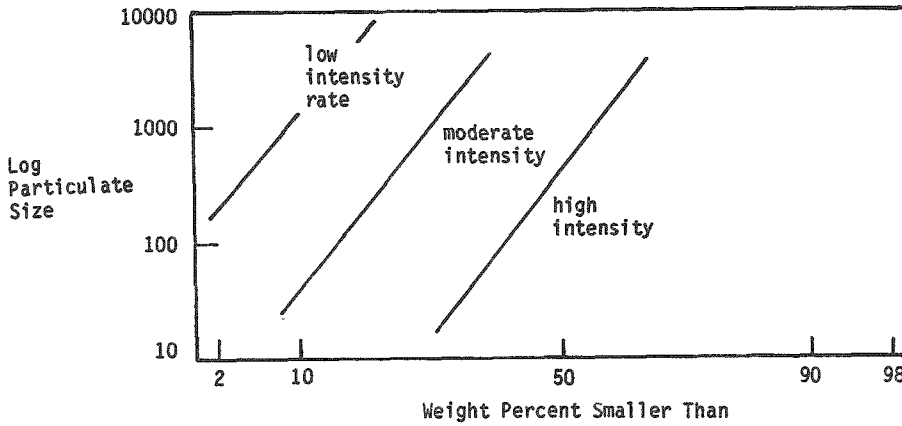

Figure 2. Weight percentage of particles that are of a size equal to or smaller than the indicated value.

In relating the data that would be obtained in experimental programs to actual incidents or accidents involving the transport of nuclear materials, three questions must be answered for the response of a package subjected to extreme environments: (1) is the container breached, (2) if a container is breached, what types of failure modes and release points occur and (3) what environments (mechanical and thermal loadings) are transmitted to the contents? The determination of package and contents interaction is important because the package may mitigate or enhance the effects to the contents as caused by any loading condition. With this package/contents interaction determined, then tests on contents alone subjected to the effects of the particular loading condition will provide data on the materials that could be available for release from a container. These relationships are summarized in Figure 3.

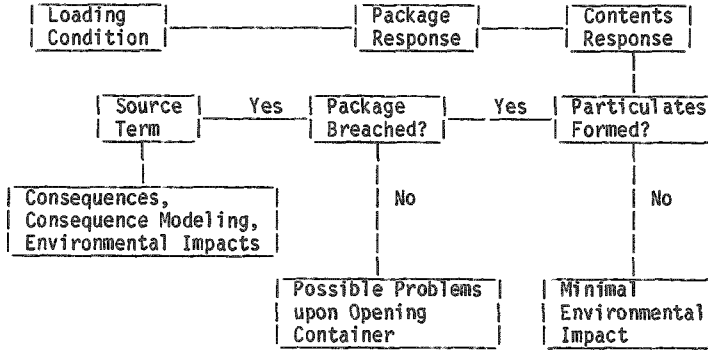

Figure 3. Relationships of loading conditions, package and contents response, particulate formation, source terms and environmental impacts.

The correlation of the test environments to those conditions that would occur in actual incidents or accidents requires evaluation of package responses, mechanical/thermal response of spent fuel or other waste form and determination of the release of radionuclides from the fuel or wastes. This correlation of test and actual environments can be accomplished in a variety of ways including: (1) instrumented model and full scale test in which strain-time and acceleration histories are recorded for packages and contents, (2) tests with ranges of specified loading conditions applied to particular parts of packages and contents to determine failure thresholds and (3) computer-based finite-element and dynamic analysis techniques to evaluate the effects of shock, thermal and pressure loadings on the structure and contents together with evaluation of these models in terms of experimental data. The tests must supply the range of data that will span the possible conditions that could be present.

The following plot relates the the loading conditions to the existence of source terms and outlines the relationships for the source terms that may result from shipping systems for nuclear materials subjected to extreme environments. The abscissa is a series of sliding scales describing the intensity of the loading condition. The information contained in these sliding scales will relate the source terms obtained for the various R&D programs described later from which data will be obtained. These scales will be correctly labeled and located when results are available. The ordinate is the normalized source term.

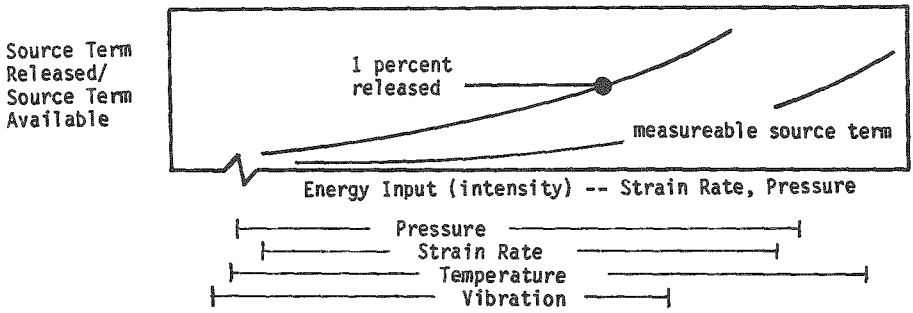

Figure 4. Source Terms for Extreme Environments

METHODS OF DATA ACQUISITION

Within the US, the acquisition of the source term data is part of several programs. One series of programs addressing the "Arrival Conditions" of spent fuel will obtain source term data in normal or near normal transportation environments and possible storage and disposal conditions. Included within this program is the development of criteria describing possible degradation of spent fuel (i.e., effects of temperatures, failure mechanisms of cladding, formation of particulates) in long-term storage and accident-free transportation environments.

Also included in the effort to determine source terms is the work associated with reactor environments. Certain of the reactor environments are similar to or bound effects that would be created in a transportation accident involving spent fuel. An example of the applicability of reactor experiments is the use of reactor loss of coolant data to loss of water in spent fuel casks. The experimental

program has resulted in a series of tests on irradiated fuel rods which varied duration of temperature, time and pressures. The objective of these experimental efforts was to determine the characteristics of the fission products (particulate sizes and chemical species) which escaped from fuel rods in conditions similar to possible transportation accidents and loss-of-coolant accidents (LOCA) in light water reactors. From these tests and from long-term storage tests, the following relationships might be developed for thermal loading as shown in Figure 5.

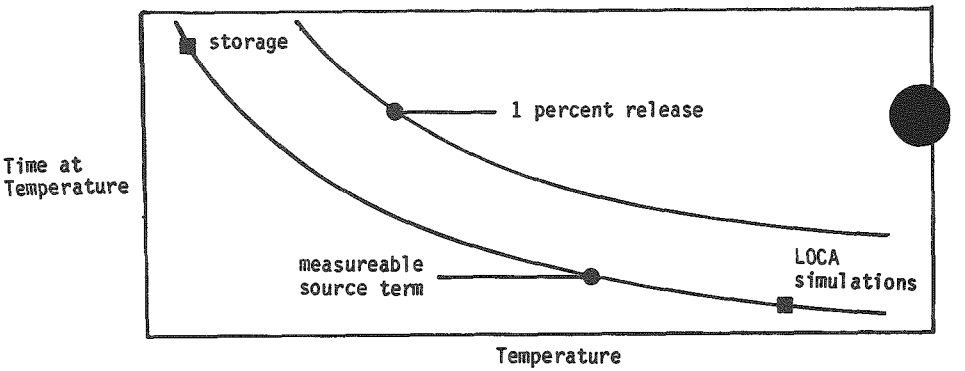

Figure 5. Source term variation with time at temperature.

CONCLUSION

The Transportation System Safety Evaluation (TSSE) program at Sandia National Laboratories' Transportation Technology Center was initiated to provide the necessary information on source terms for nuclear materials subjected to extreme environments as described above. The techniques for derivation of source terms for the fuel alone has been described as well as the outline for package response. An additional facet of this problem is is the development of analytical methods to describe the transport of the released radionuclides from the fuel rods to possible release points. This work is also covered in the TSSE program.[3] Not all the work required will be performed or funded by Sandia; rather existing work will be sought out and ongoing work will be utilized in an attempt to unify the presentation of data and thus increase its usefulness.

REFERENCES

1. R. M. Jefferson, ed., Program Strategy Document, SAND80-0784, TTC-0080, Sandia National Laboratories, Albuquerque, NM, April 1980.

2. A. P. Malinauskas, Quarterly Report on Fission Product Behavior in LWR for the Period April-June 1977, NUREG/TM-139, Oak Ridge National Laboratory, Oak Ridge, TN, September 1977.

3. P. Baybutt et al., "The Determination of Environmental Radionuclide Releases from Breached Shipping Casks Using the TRACK Computer Code," Battelle Memorial Institute, Columbus Laboratories, 6th International Symposium on Packaging and Transportation of Radioactive Materials, West Berlin, Federal Republic of Germany, November 10-14, 1980.

4. R. E. Luna, H. W. Church and J. D. Shreve, "Variability of Air Sampler Data," Atmospheric Environment 5 (February 1971) pp 579-597.

5. W. Davis, G. A. West and R. G. Stacy, Oxide Particle Size Distribution from Shearing Irradiated and Unirradiated LWR Fuels in Zircalloy and Stainless Steel Cladding: Significance for Risk Assessment, NUREG/CR-0866, ORNL/NUREG-60, Oak Ridge National Laboratory, Oak Ridge, TN, October 1979.

SESSION IV
SYSTEMS AND PACKAGE DESIGN

CASTOR CASKS: A NEW SYSTEM FOR TRANSPORT AND STORAGE OF SPENT FUEL ELEMENTS

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The family of CASTOR transport and storage casks (CAST IRON CASKS FOR STORAGE AND TRANSPORT OF RADIOACTIVE MATERIALS) was developed with the aim of making available an economic and reliable cask for the transportation, intermediate storage and possible final storage of highly radioactive materials. The most important step was to considerably reduce the up to now very high costs for type B casks. This was possible by using high quality cast iron (nodular cast iron) for the construction of the cask: when basic costs are low and machinability good, serial production is quickly possible and with further cost advantages. Welding seams are almost completely avoided and therefore, in spite of a high standard, quality control is made easier.

Having recognised the advantages of the system, the DWK (Deutsche Gesellschaft fur Wiederaufarbeitung von Kernbrennstoffen) and STEAG AG made available the means for wide-scale development work. Apart from numerous small pieces the following test casks were constructed, which even during production helped to improve the casting method and quality control.

1. 5 t test cask (MOSAIK I)
2. 18 t test cask 1:1, bottom part CASTOR Ia
3. 9 t model cask CASTOR I, 1:2
4. 14 t model cask CASTOR II, 1:2
5. 10 t model cask CASTOR I, 1:2 (double lid system)
6. 70 t prototype cask CASTOR Ia, 1:1
7. 45 t test cask 1:1, bottom part CASTOR IIa

More than 60 drop tests were carried out using these casks, sometimes under conditions stricter than IAEA requirements, e.g. 10 drop tests at -40°C, drop tests without shock absorbers, repeated drop tests without servicing and renewal of the sealing system. The 5t cask was submitted to drop tests about 30 times, the 9 t model cask times, the 70 t prototype 5 times. Behaviour under accident fire conditions was investigated by a dozen fire tests. As a general result, it is established that all the tested casks are still functioning and there is no loss of shielding nor leaktightness in relation to their previous condition.

For long-term intermediate storage there were a number of important requirements, more stringent than for type B casks, to be fulfilled:

- Behaviour of the fuel elements during intermediate storage up to 30 years had to be investigated. Among other tests, over 100 individual experiments were carried out to exactly determine the thermodynamic cask system, in order to establish the most important value, the maximum permissible fuel rod temperature.

- Proof of the long-term stability of the whole cask, especially the sealing system, was necessary. This was mostly done by theoretical analysis, selection of suitable materials, particularly in the case of gaskets and sealing surfaces, as well as accompanying experiments to investigate the possibility of drying the cask, of filling it with inert gas, etc.

- Minimisation of activity release to values which are too small to be gauged, by a sophisticated multiple lid system which can be repaired without opening the cask and the functioning of which is continuously supervised.

- Incident analysis, as required for nuclear facilities. While some of the incidents are covered by IAEA requirements for transport, others must be taken into consideration. The extreme case of a fast-flying military aircraft crashing directly on to the cask was simulated on a 1:1 scale by missile bombardment.

As a result of this and other tests as well as numerous further analyses, we believe that intermediate storage using CASTOR casks, apart from its economic advantages, offers an optimum of safety:

- short delivery times;
- direct correlation of capacity and demand;
- no secondary waste;
- practically no activity release;
- inherent safety due to dissipation of decay heat by natural air convection without machines;
- maintenance of safety and operational capacity, even in the case of natural and technical disasters;
- possibility of extending the intermediate storage period if necessary;
- easy delivery and removal;
- practically no shut-down costs.

The casks themselves consist of a cavity from a single cast with moulded-on cooling fins and a lid system, which can have one to three lids depending on the use to which the cask is put.

There is also a new development in the heat dissipation and neutron shielding: orignally dry storage casks had no neutron shielding, but today's cask is equipped with an external mantle made of hydrogen-containing material to act as a neutron shielding, through which, however, heat must be transferred. For this purpose there are numerous cooling fins which intersect the mantle, made out of material with high heat conductivity, e.g. copper.

The CASTOR cask contains the necessary moderating material in longitudinal borings in the cask wall. This, on the one hand, ensures an uninterrupted heat flux to the outside, and, on the other, means that the neutron shielding is not destroyed, even under extreme accident conditions.

Normally dry storage casks are used (cooled by air or inert gas), however in special cases water may also be used for cooling the fuel elements.

CASTOR casks of the series I and II weigh about 80 and 120 t respectively and hold 1.5 to 5 t irradiated uranium. The minimum decay time for the fuel elements is 1 year (type I) to 1.5 years (type II). The resulting dose rates are about 20 mrem/h at the cask surface and about 2 mrem/h 2 m from the cask.

The approval procedure takes place just before the end. The type B(U) approval as transport cask according to IAEA regulations was granted for the type CASTOR Ia in May 1980. For the other type B approvals, the technical examination by experts for approval for storage according to the German atomic law, should first take place in order to be able to take into account possible additional technical requirements necessitated by storage and long-term aspects in the type B approval as transport cask.

The following approvals for cask storage using the CASTOR system have been applied for:

1.	Wurgassen	120 t U	AR (at reactor)
2.	Stade	200 t U	AR (at reactor)
3.	Ahaus	1,500 t U	AFR (away from reactor)
4.	Gorleben	1,500 t U	AFR (away from reactor)

In Wurgassen the first CASTOR cask should be loaded in December 1980/January 1981 as part of a preliminary demonstration. Casks and fuel elements will be continuously supervised and have the necessary instrument equipment.

The public hearings required by the authorities for the Ahaus and Gorleben sites are to take place during the first few weeks of 1981, and the experts report for storage approval must be ready by then. We therefore reckon on the first part-approval and hence on the start of construction for one of the larger storage facilities already during the first half of 1981.

When the approval suitability of the CASTOR family as type B(U) transport cask has been proved and the storage approval is expected within a short time, the system should be examined for further possible uses and the cost/benefit aspect futher improved. The following points in particular are being investigated at present:

1. Improvements and simplification of production technique in order to reduce manufacturing costs, especially in the case of larger series.

2. Influence of extending the decay times on the specific costs.

3. Complete development of an inexpensive CASTOR cask optimised for transportation purposes.

4. Use of the CASTOR principle for other fields, especially for transport and storage of highly radioactive waste.

We will soon be able to report on first results.

DISCUSSION

Question by G. Arendasy, Austria: 1. What method has been used for the internal stainless lining of the CASTOR cask? 2. What heat decay is the cask capable of?

Answer: 1. Thick galvanic nickel plating. 2. CASTOR IIa (1.5 a decay time): 45 kW.

Question by H. Diem, FR Germany: The flask can be used for wet transport. Therefore it is a pressure vessel. What is the design pressure?

Answer: The flask is licensed up to now for dry use only. A design pressure of 5-10 bar is no problem, since the flask is calculated for higher pressures and tested before use hydraulically with pressure one order of magnitude higher.

Question by M. S. T. Price, United Kingdom: 1. What non-destructive testing methods are used for Quality Assurance of CASTOR flasks? 2. What thickness are these methods capable of working up to? 3. What defect criteria have been defined for CASTOR flasks.

Answer: 1. Ultra sonic testing. 2. Up to 500 mm without problems. 3. Defect criteria cannot be published, as they are proprietary information, but are available for competent authorities.

Question by S. F. Johnson, Sweden: The CASTOR flask presented does not have a neutron shield which completely covers it. Your neutron shielding consists of neutron absorbing pins. To avoid direct neutron radiation on the outside of the flask, you have to have a minimum distance between the pins. How big is this distance?

Answer: Due to the reflection of neutrons in the iron/carbon alloy there is practically no "minimum" distance to avoid window effects. For example, regular borings with distances of 80 mm show no measurable effect. In fact, the number of borings is determined in accordance with source strength. The material is moderating, not absorbing.

Question by E. Schlich, FR Germany: As far as neutron shielding is concerned, you say that your cask is one of the third generation. You know that only one barrier is allowed in the side and base regions of a cask. 1. Don't you think that the perforations in the thick wall cavity create a risk? 2. How is non-destructive testing possible after boring these holes? Which and how many drop tests have you performed with these borings and how could you be sure that cracks staring in the boring region were excluded and how can you prove this?

Answer: 1. I object to the term perforation. The cask is not perforated but equipped with lengthwise bore holes. 2. The ultrasonic tests are carried out before boring. 3. There have been drop tests, compression tests and calculations controlled and accepted by the competent authorities. In addition it is known among experts as state of the art that fissures are stopped, not initiated, by bore holes.

Question by M. S. T. Price: We have already heard that an ASME ad hoc committee (NUPAC) had been formed to coordinate codes and standards for the construction of transport flasks. Can you comment on the possible interaction of the West Germany work on cast iron flasks with the work of NUPAC (e.g. is there any intention to provide observers to NUPAC if the ASME committee agrees?).

Answer: GNS is investigating via its US representative whether a participation in the NUPAC committee is welcome and feasible.

Question by J. D. Hart, United Kingdom: I presume that you will not be proof testing each flask by drop testing at -40°C. What tests are conducted to ensure quality during manufacture? What inspection techniques are used to detect defects? What defect acceptance limits are set and how are they derived?

Answer: It is true that we do not proof test each flask at -40°C. A number of tests are required to ensure quality of the casting: test bodies are joined to the big casting of the body and examined for quality; test specimens are machined out of the bottom area of the cask, which is most sensitive to faults in the casting procedure. Defect acceptance limits are laid down by the competent authority. They have proprietary character. They have been derived by examining the prototype which has actually been drop tested several times.

Question by H. Rack, USA: What analytic procedures are used in the FR Germany to assure that cask structural dynamics are understood? How do German designers use IAEA drop tests and fabrication quality assurance programs to assure uniform response of a large number of casks?

Answer: The designer's approach to cast iron casks has been a pragmatic one. The approach to establishing a conforming analytical method is underway. The assurance of uniform response of a large number of casks to IAEA drop test requirements is assured by conservative test and control programs: actual casks are drop tested under conditions more severe than IAEA drop tests. The collected data (acceleration, deformation, etc.) are used to set the minimum quality requirements which will be controlled by an elaborate and precise quality assurance program.

CAST FERRITIC STAINLESS STEEL CASK FOR THE SHIPMENT OF IRRADIATED FUEL ELEMENTS

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R. Laug
Transnuklear GmbH

A. Lührmann
DWK

Due to the increasing need of shipping casks for irradiated fuel elements, Transnuklear has worked in cooperation with DWK on their development.
The aim was to produce a cask which would meet the following requirements:

- both a wet and a dry unloading capability
- speedy availability by means of a short manufacturing time
- maximal loading capacity
- fulfilling of the COGEMA acceptance criteria for unloading at th UP 3
- reduction in manufacturing costs.

The starting point was the already well-advanced development of the TN forged steel dry shipping cask. During its development, both alternative materials and alternative production technologies for the cask components were investigated.

In the following one of the development ideas that was pursued will be presendet. This is the concept of a cast steel shipping cask, fitted with a removable cooling fin jacket containing neutron shielding. The reference design is the TN 17 cask with a capacity for 17 BWR fuel elements. The cask body consists of a thick-walled cast cylinder, with an integrally bottom at one end. This bottom can be welded or screwed and welded, or, as is possible with cast casks, be integrally cast (Fig. 1).

The cask body is made from the internationally recognised pressure vessel material G-X5 CrNi 13.4 (SEW 410/70), also listed in the ASME code (ASTM A 296-77 gr. CA-6NM). This ferritic cast steel has not only excellent cold ductitily properties, but also a substantail resistance to corrosion from air, water, and all customary means of decontamination due to the Cr-13 % and Ni-4 % alloys. The technology is available for this casting material to cast transport casks up to a weight of about 120 tonnes in one piece.

The cask is fitted with a 3-piece lid system which satisfies COGEMA's requirements. This lid system essentially consists of 3 components:

- a shielding plug
- a ring flange
- a shipping cover.

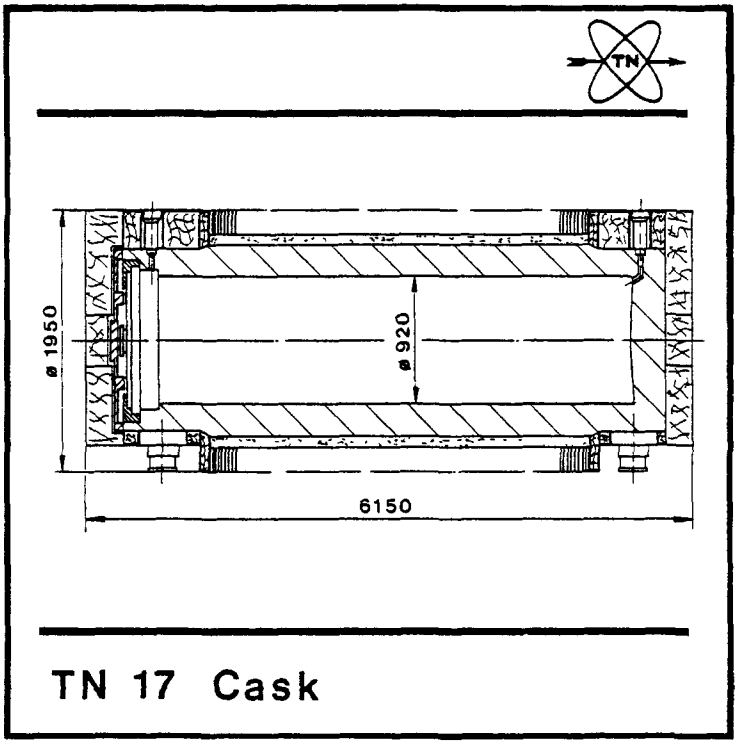

Fig. 1

The shielding plug provides both shielding and sealing, the ring flange fixes the shielding plug in position, and the shipping cover provides extra sealing. In addition, the shipping cover also protects the shielding plug from any external mechanical influences. The shielding plug and the shipping cover are each fitted with a pair of seals to facilitate a leak test on the space between them. The cask is fitted with 2 rinsing orifices, located on the outer casing near the lid and near the base.

As the cask should be suitable for dry transport, a special basket construction (Fig. 2) was required in order to conduct the heat from the fuel elements to the cask wall. This basket consists of a solid aluminium cylinder, into which square tubes of high-grade boron steel are inserted to hold the fuel elements. In order to conduct the heat from the basket to the cask wall, the inner diameter of the cask and the outer diameter of the basket are arranged such that, on heating up, the basket will contact the cask wall.

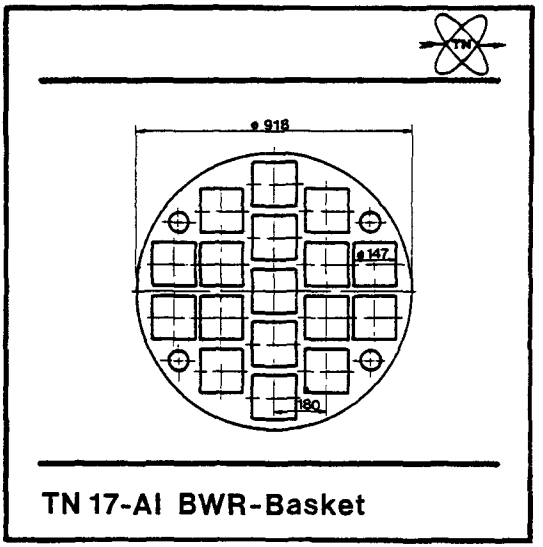

Fig. 2

From the outer cask wall the heat must be dissipated using suitable cooling elements to the sourrounding atmosphere by means of natural convection. To do this the cask is fitted with a removable cooling fin jacket with incorporated neutron shielding.

This means that due to the short manufacturing times (parallel manufacture), a simple exchange of cooling fin jacket elements is possible, thereby avoiding the long unavailability of the cask after an incident.

There are 2 possible materials available for the cooling fin jacket:

- nodular cast iron (Fig. 3)
- copper profile sections (Fig. 4)

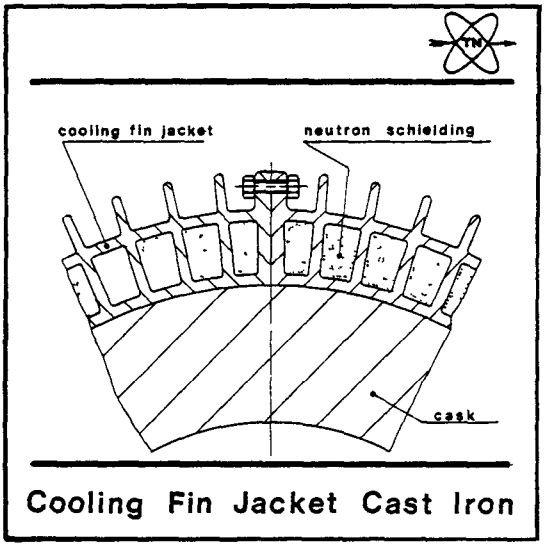
TN
cooling fin jacket
neutron schielding
cask
Cooling Fin Jacket Cast Iron

Fig. 3

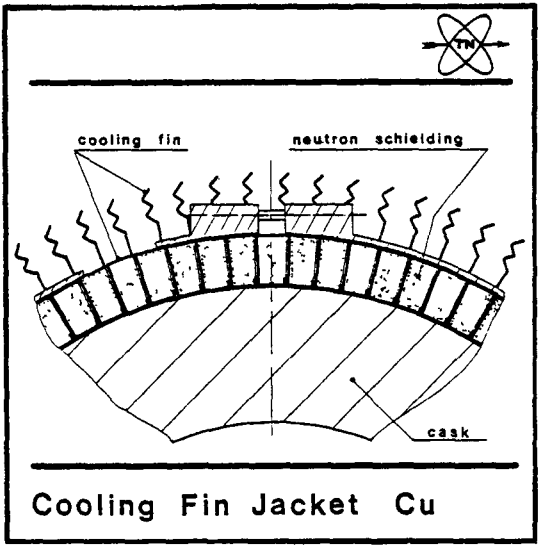
TN
cooling fin
neutron schielding
cask
Cooling Fin Jacket Cu

Fig. 4

The nodular cast iron cooling fin jacket consists of 4 inherently stable half-shells with integrally cast longitudinal cooling fins. The half-shells are fitted with longitudinal chambers for the neutron shielding, and are bonded together on the mounting using cast flanges. The basic cask body and the cooling fin jacket are matched together to such a high degree of accuracy that a good heat transfer coefficient is guaranteed without any further means of contact.

The copper profile sections cooling fin jacket consists of about 100 box sections, fitted with a large number of moulded cooling fins.

The neutron shielding is incorporated into the box sections. The colling fin jacket units are attached to the cask body by clamping straps. Here again, an adequate heat transfer coefficient is guaranteed without any further means of contact.

The neutron shielding consists of a synthetic resin embedded PP-powder. This mixture will be poured into the chambers in the cooling fin jacket, or installed as prefabricated units.

Both for testing and licensing purposes, 2 cask models were prepared along the lines of the TN 17. The first model had a cross-section corresponding to the full scale TN 17 but with a length of only 2 m. The simplification of the casting technology and interior mechanical work could be justified as the base was not integrally cast.
The base was prepared separately and then bolted on and finished with a leakproof weld outside and a leakproof stress weld inside. Outside, this model was fitted with a nodular cast iron cooling fin jacket. The model was subjected to drop tests from 9 and 18 m at both the surrounding temperature and at -40 ^{o}C to test the ability of both welds to maintain their high mechanical integrity at relatively low temperatures.

The welds form 2 barriers (or seals) on behind the other, of which one must remain intact after an accident (9 m drop) in order to safely contain the shipped goods.
The drop test showed that on completion of all 3 drop tests, some partial cracks were apparent on the inner weld, although the outer leakproof weld retained its full integrity.
The cast cooling fin jacket survived 3 drop test without significant damage.

On the 3rd test - the drop onto the connecting flange of fin jacket - however, it was considerably damaged. By determining the cause of the damage - the flange was too rigid - and a survey of the design details supported by a computer program the cast cooling fin jacket was redesigned so that it would meet the recognised requirements. (Further tests have yet to be conducted.)

On completion of this series of tests, the full scale model was fitted with a copper cooling fin jacket. Heat tests were then carried out, observations being made with (Fig. 5):

1. horizontal cooling fins at 0 °
2. vertical cooling fins at 90 °
3. cooling fins inclined at 45 °

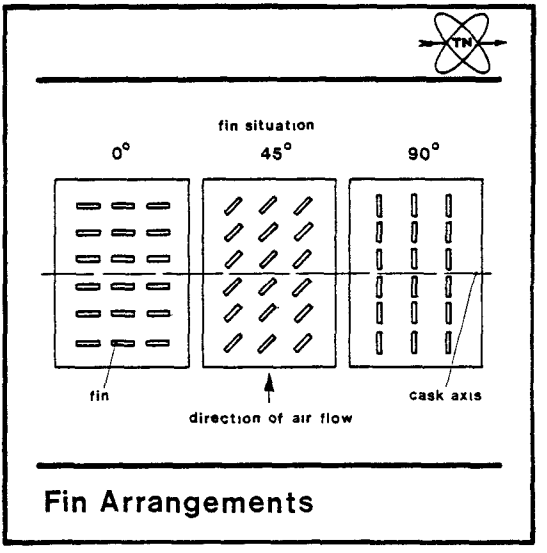

Fig. 5

These showed that the copper cooling fin jacket with vertical cooling fins can better dissipate a heat capacity of 40 - 45 kW (Fig. 6). At this heat capacity a rise in temperature around the gap between the fin base and the cask surface was about $\Delta t \cong 24$ K.

The experiments completely confirmed that, in general, vertical cooling fins give rise to a greater heat dissipation than horizontal ones. This effect only shows itself, however, at higher heat capacities ($\dot{Q} > 10$ kW).

A drop test on the copper cooling fin jacket, which should demonstrate especially the integrity of the clamping straps and clamping locks is planned for early 1981.

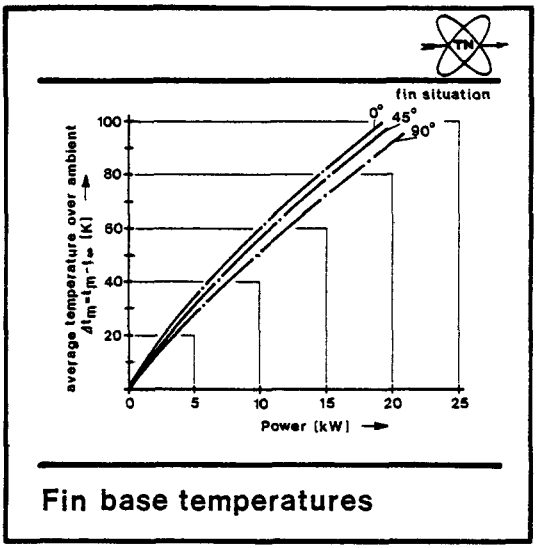

Fig. 6

The shielding capabilities of both cooling fin jackets with incorporated neutron shielding were tested on the cask body. Cf 252 was used as a point neutron source. These results experimentally confirmed the neutron alternation factor used in the cask design calculation.

The second model corresponds to a third scale version of the TN 17. This model is fitted with an integrally cast base. The cooling fin jacket is represented by angle iron as it is not feasible to scale it down. The major object of this model is to test the performance of lid system. It will also undergo a series of drop tests in early 1981.

DISCUSSION

Question by W. R. Taylor, Canada: Could you describe briefly how you cool the cask to -40°C and how long this process takes?

Answer: For cooling down the described test model, we put the flask into an insulated wooden box and sprinkled the flask with liquid nitrogen. The cooling time for the temperature difference of about 60 K and the flask mass of about 33 t was about 20 hours. The temperatures were controlled by several thermocouples

Question by W. Denk, FR Germany: Could you tell me something about the neutron shield construction in the lid and bottom regions?

Answer: If a neutron shielding in the cask presented is necessary, we install it in hermetical closed pockets in the lid and botton of the flask.

Question by M. S. T. Price, United Kingdom: Can you please explain why the copper fins are cranked?

Answer: The heat transfer from the flask to the ambient air depends for example on the fin surface. Therefore, we have cranked the copper fins to obtain a higher surface area of each fin. The fin length is lirited by the outer diameter of the flask.

Question by A. Olivieri, Italy: I would like to know the maximum decay power of the shipped fuel and the maximum temperature which is expected to occur in the fuel bundles?

Answer: The decay heat of burned up fuel elements depends on the decay time. We assumed for the design a decay time of 180 days of the BWR fuel elements that is comparable with $\dot{Q}$ = 2.5 kW per element. The maximum decay power for this flask has not been tested. The maximum temperature of the central rod of the fuel bundle is expected to be $t_{max} \cong 400°C$.

Question by J. D. Hart, United Kingdom: I presume that you will not be proof testing each flask by drop testing at -40°C. What tests are conducted to ensure quality during manufacture? What inspection techniques are used to detect defects? What defect acceptance limits are set and how are they derived? What material property information is sought and how are the acceptance levels derived?

Answer: The casks are manufactured in accordnace with a quality assurance plan which is defined for each type of cask. The quality assurance provisions used and necessary requirements are described in detail. Several inspection techniques are used such as ultra-sonic, magnetic particle and dye penetrant examination. The acceptance limits are specified in German and international standards, e.g. DIN 11 245 and CCH 70-I (RE 70-1). The inspected properties of the materials are cracks, holes, seqregates, etc.

Question by H. Rack, USA: What analytic procedures are used in the Federal Republic of Germany to assure that cask structural dynamics are understood? How do German designers use IAEA drop tests and fabrication quality assurance programs to assure uniform response of a large number of casks?

Answer: The German standards for pressure vessels are the "AD-Merkblätter," which given the analytic procedures for cask design. In addition the IAEA Regulations for the "Safe Transport of Radioactive Materials" Rev. 1973, Series No. 6, form the basis of the tests. The quality of the fabrication of a number of casks is guaranteed by an operating quality assurance system, which includes audits, personnel qualification, fabrication and inspection plans and documentation.

SUITABILITY OF A SPECIAL HIGH-DENSITY CONCRETE AS MATERIAL FOR SPENT FUEL SHIPPING AND STORAGE CASKS

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Transnuklear GmbH

A. Lührmann
DWK

Dependent on the situation emerging in Germany regarding a back end service for nuclear power stations, and the resulting need for intermediate storage capacity, Transnuklear GmbH has, in cooperation with DWK, examined the feasibility of using concrete to build shipping and storage casks for irradiated fuel elements.

For this particular use a special concrete formulation had to be found which, when applied to a suitable cask design, would guarantee not only adequate thermal conductivity and radiation shielding, but also sufficient mechanical strength and stability at relatively high temperatures.

Following numerous studies relating to the formulation, ease of manufacture, and workability of a suitable special concrete, a steel-fiber high density concrete was developed containing, in addition to cement and water, the following aggregates:

- magnetite
- hematite
- iron granules
- steel-fibers.

With an optimal mix of the aggregates, this steel-fiber high density concrete has a relative density of $\gamma \simeq 3.99$ kg/dm^3.
The properties of the material were investigated by means of a series of different loading tests carried out on test cylinders and beams. As a result the compression resistance was found to be 35 N/mm^2.

Furthermore, the pressure test showed that a maximal deformation of 10 % is possible, without any disintegration of the concrete structure. Even after exceeding the breaking strength, no splitting of entire sections from the test block, a characteristic of concrete, was observed.
As casks made of this material will have a relatively high operating temperature (> 100°C), the samples were subjected to heat treatment in order to expel all the water not combined with the concrete.

These samples were heated to 300°C on a weighing scale lying inside an air-conditioned cabinet, and assessed as soon as all the free water was completely expelled. The complete expellation of the free water was identifiable by an initial weight loss followed by a constant weight over a long period.

Following the completion of tests to determine the principal qualities of this material, attention was for the first time directed towards the cask design and its capacity. In parallel to this a model was also prepared.
At this point, to give a better understanding, the cask design will be described with the aid of the reference design for 7 PWR-FE (fig. 1):
The cask shell, as well as the base, are constructed in several layers. The inner layer - the inner liner - consists of a thin-walled, corrosion-resistant steel cylinder, fitted with a corrosion-resistant welded steel base plate. The outer layer - the outer liner - is constructed in the same way but using C-steel. The space between the liners, both around the shell and the base, is filled with a steel-fiber high-density concrete. In addition, the space between the inner and outer liners is fitted with a strong steel reinforcement system arranged radially, annularly, and longitudinally. In order to secure the two liners to the concrete, they are fitted with a large number of radially arranged setbolts. This guarantees a positive locking connection.

At the open upper end of the cask a solid stainless steel ring is welded between the two liners to take the lid or lid system. The rinsing orifices for the cask are accommodated on the flat side of the lid . 2 trunnions are laterally mounted, both near the lid and near the base. To take the fuel elements the cask is fitted with a fuel element basket which, to ensure subcriticality, consists of boron-alloy stainless steel double cross-pieces.
The model is an annular model with an outer diameter of 1860 mm, a wall thickness of 555 mm and a length of 1 m. (Inner liner thickness 25 mm, outer liner thickness 50 mm). The manufacturing technology was then successfully tested, namely to put the concrete into a leakproof steel reinforcement system from a height of 6 m, (this corresponds to the original height of a cask) and to compress the concrete, without any separation occurring in the aggregates. Using a similarly installed drainage system, the free water was expelled by heat treatment.
Determining the thermal conductivity of the cask wall was carried out using heating elements fitted inside the cask. A recalculation of the thermal conductivity from the measured temperatures after reaching the state of inertia gave a value of

$$\lambda \simeq \frac{4{,}8 \text{ Kcal}}{\text{m h grad.}}$$

Projecting this value for a cask with an inner length of 5 m, a thermal conductivity for the fuel pins of ca. 20 KW and an ambient temperature of 38°C, yields the following results:
- cask outer wall temperature 85°C
- cask inner wall temperature 193°C.

A drop test from a height of 9 m onto an unyielding foundation was then carried out on the annular model at the surrounding temperature. The model was made to fall onto its shell, thereby deforming not only the outer but also the inner liner. The maximal deformation of the inner liner amounted to 12 mm. This resulting oval-shape can be avoided by appropriate measures such as modifying the thickness of the inner liner or by putting a buffer layer inbetween the inner liner and the adjacent concrete. On a further test cylinder, an experiment using a neutron source showed that adequate shielding is guaranteed with a wall thickness of about 600 mm.

Based on the studies described so far, the feasibility and limitations of a concrete cask as a shipping and storage cask were now investigated. The following boundary conditions were assumed:

- dose rate limits for transport regulations
  max. 200 mrem/h at the cask surface
  max. 10 mrem/h at a distance of 2 m from the transport vehicle
- diameter and weight limits defined by the conditions in German nuclear power stations (> 600 MW):
  max. radial outside measurement
  a) 2,5 m for Biblis-type PWR fuel element casks
  b) 2,0 m for BWR fuel element casks

  gross cask weight 120 t for Biblis-type PWR fuel element casks
  76 t for Würgassen-type BWR fuel element casks
- permissible cladding temperature of max. 400°C at the beginning of storage.

The study limits itself to the following parameters:

- sub-criticality assurance
- shielding
- heat dissipation.

The cask construction was conducted both for Biblis-type PWR fuel elements and for BWR fuel elements:

	Biblis-type PWR fuel elements	BWR fuel elements
fuel pin arrangement	16 x 16	8 x 8 - 1
no. of fuel pins per fuel element	236	63
pin diameter (mm)	10.75	12.5
cladding thickness (mm)	0.72	0.85
active length (mm)	3900	3760
fuel element length (mm)	4925	4470
cross-section (mm)	230	140
fuel element weight (kg)	790	277
fuel element weight as U (kg)	534	195
specific capacity KW/kg	36.7	25
burn-up (MWd/tU)	36000 max.	30000 max.
irradiation period (d)	1000	1200

Putting these studies and thoughts together led to the following 5 cask variants:

cask variant	I	II	III	IV	V
fuel element capacity	4 PWR	7 PWR	9 BWR	10 PWR	17 BWR
heat capacity (KW)	14	20	14	10	5
fuel element cooling period (years)	2	3	2	5	5

In conclusion it can be established that concrete, together with the appropriate aggregates and bearing in mind the long cooling period for the fuel elements, appears entirely suitable for the construction of shipping and storage casks.

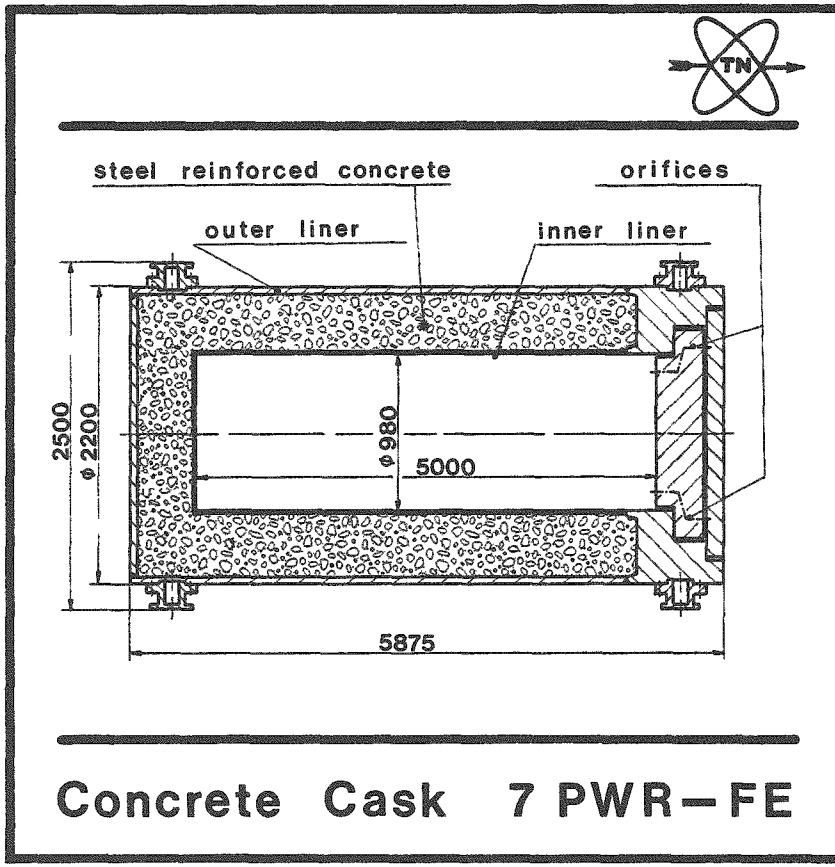
steel reinforced concrete
orifices
outer liner
inner liner
2500
ϕ 2200
ϕ 980
5000
5875
Concrete Cask 7 PWR–FE

Fig. 1

DISCUSSION

Question by H. Eick, FR Germany: Concerning the hot (inner) liner, are the stresses within the elastic or the plastic range?

Answer: For the concrete casks considered now they are in the elastic range.

Question by G. Kaspar, FR Germany: You have explained that the cask concrete is heated to 300°C in order to release water. In time there will no doubt be new humidity in the concrete. How can humidity be prevented from arising in the interior of the cask or excess pressure occurring between the outer and inner liner of the cask during transport when the concrete inner wall reaches almost 200°C?

Answer: Building up of pressure between the two steel walls (inner and outer liner) is prevented by two mechanisms:

1. After heating of the concrete cask the drainage openings are tightly sealed. Therefore it is guaranteed that while the cask remains intact, no water can enter.
2. Should all the water not have been drained out of the cask or should water have penetrated through both liners because of a fault in the casing, inadmissible build-up of pressure is prevented by the pressure relief valves integrated in the outer liner.

HLDW TRANSPORTATION SYSTEM DESIGN BASIS AND CONCEPTS

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INTRODUCTION

A system is being developed to transport solidified High Level Defense Waste (HLDW) from the Defense Waste Processing Facility (DWPF) at Savannah River to a federal repository. This work is being performed under DOE contact and under the technical direction of the Transportation Technology Center at Sandia National Laboratory. A design goal of the project is to develop a High Level Defense Waste Transportation System (HLDWTS) which is compatible with the solidified HLDW forms generated at Hanford and Idaho sites as well as the solidified waste which may be generated at a waste solidification facility to be built at West Valley, New York.

WASTE GENERATING SITES

DOE operates High Level Waste (HLW) management operations at Savannah River Plant, the Hanford site and the Idaho site. In addition there is commercial and defense HLW stored at the NFS plant near West Valley, New York. The locations of these sites are shown in Figure 1. Alternatives for long term HLW management at the SRP, Hanford, and the Idaho site are described in References 1, 2 and 3. The options being studied for disposal of currently stored HLLW and decommissioning of the West Valley (WV) site are described in Reference 4.

High-level radioactive wastes generated at Hanford, SRP and West Valley are similar in chemical and radionuclide composition. Hanford, SRP and West Valley wastes are processed to excess alkalinity and transferred to large underground storage tanks where they are stored as high-level liquid waste (HLLW). In addition, the WV wastes also include a small quantity of acid Thorex waste. At the Hanford site radiocesium and radiostrontium are being removed from the waste and stored in double-wall canisters as cesium chloride and strontium fluoride. Idaho Chemical Processing Plant (ICPP) high-level waste composition varies greatly depending on the type of fuel being processed, the irradiation history of the fuel, and the length of time the fuel is stored before processing (Reference 3). Unlike the alkaline wastes at Savannah River, Hanford and West

FIGURE 1 NUCLEAR WASTE SITES

Valley, ICCP high-level liquid waste is initially stored as an acid solution and contains high fluoride concentrations. After a suitable decay period, the acidic waste is converted to, and stored as, a granular calcined solid.

DESCRIPTION OF THE HLDW FORM

The objective of the current Defense Waste management program is the development of a waste form which can be reliably and safely isolated from the biosphere for extremely long periods of time. The form which is in the most advanced state of development and, to date, has been of the most interest is borosilicate glass. One of the advantages of borosilicate glass is that it can accommodate a large variety of glass formers and waste compositions. The waste glass can thus be tailored to the particular waste composition and to the particular processing equipment and conditions. Work on adapting the borosilicate glass to the site specific waste forms is underway at each of the defense waste sites.

In addition to borosilicate glass many alternate waste forms are under study. These alternate waste forms are not nearly as well developed as the borosilicate glass. The HLDW transportation system is, therefore, being developed using borosilicate glass waste as a reference, however, it is expected that the resulting design will be compatible with most of the alternate waste forms currently under development.

Waste Canister Design

The borosilicate glass waste is poured into stainless steel canisters and allowed to cool slowly to minimize cracking of the glass. Although each of the sites has somewhat different canister design concepts, for the purpose of the transportation system development the dimensions of the SRP canister have been adopted since it represents the most advanced design. A working group has been formed to standardize the canister size for HLDW. A summary of the SRP canister design is given in Figure 2 and Table 1.

Table 1
Reference (SRP) HLDW Canister Design

Diameter	610 mm (24 inches)
OAL	3000 mm (9 feet - 10 inches)
Wall thickness	915 mm (0.375 inches)
Material	Type 304L stainless steel
Head	Welded dished head with combination fill neck and lifting pintle
Bottom	Welded reverse dished head
Working Volume	640 liters (22.5 cu. feet) at 80% fill
Empty Weight	500 kg (1100 lb)
Glass Weight	1482 kg (3260 lb)
Filled Weight	1982 kg (4360 lb)

Following cooldown, the canisters are remote seal welded and leak checked, and the exterior surface is decontaminated. The waste canister may then be transferred to interim storage or to the loading facility for offsite shipment.

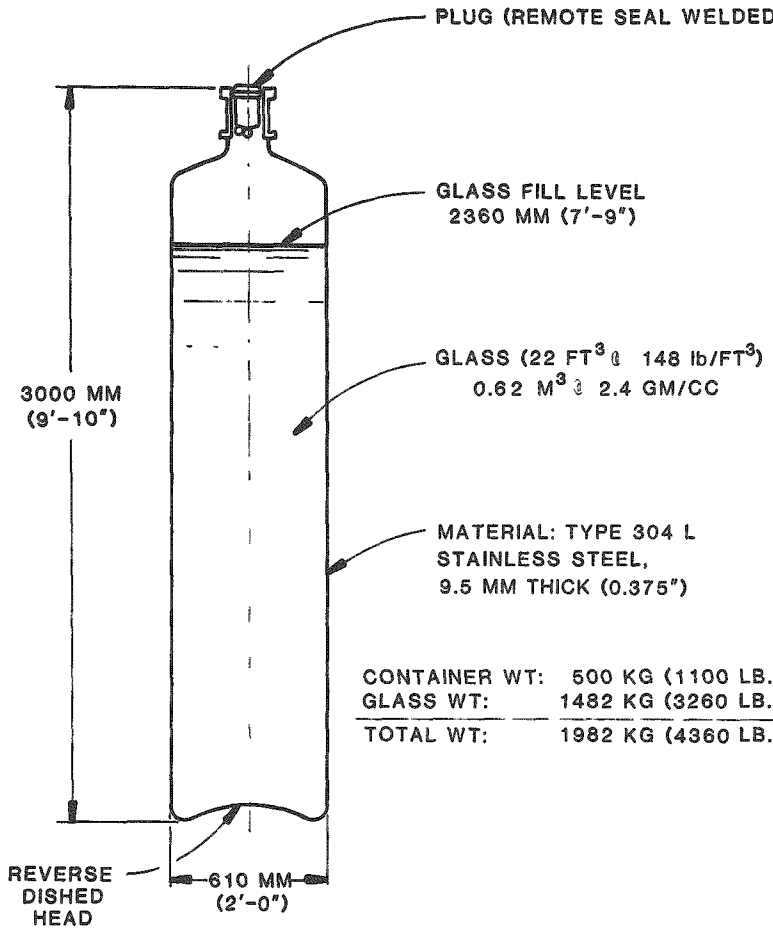

FIGURE 2. REFERENCE HLDW CANISTER

Shielding Requirements

The HLDW glass composition and activity varies from site to site as a result of process and fuel variations. An evaluation of the isotopic composition of the HLDW glass for each site concludes that the neutron dose rate is low compared to the gamma and that no neutron shielding will be required, however, the neutron dose rate is large enough that it must be included in the total dose rate calculation.

For conceptual evaluation dose rate criteria have been established consistent with U.S. NRC and DOT requirements except that the allowable cask surface dose rate was reduced from 200 mr/hr to 50 mr/hr to facilitate implementation of ALARA requirements for cask handling activities at the DWPF or repository. During the more detailed design phase it is planned to do a time and motion evaluation of the cask handling process in order to establish expected crew exposure times. In conjunction with dose criteria established for plant technicians the need for supplementary shielding, or remote handling operations can then be determined.

Table 2 lists the thickness of steel shielding required for each of the waste forms to maintain the dose rate at 50 mr/hr or less on the cask surface or 10 mr/hr or less at two meters from transport vehicle. In all cases the cask surface dose rate controls.

Table 2
HLDW Cask Shielding Requirements

Waste Type (Ref. decay time)	Steel Thickness[a] mm (in.)	Important Contributors[b]
SRP (5 yrs out of reactor)	229 (9.0)	Pr-144, Cs-134, Eu-154, Co-60
Idaho (5 yrs out of reactor)	180 (7.1)	Cs-134, Ba-137m, Pr-144, Eu-154
Hanford Existing (decayed to 1995)	158 (6.2)	Ba-137m, Eu-154, Co-60
Hanford Future (decayed to 2010)	173 (6.8)	Eu-154, Ba-137m, Co-60, Cm-244
West Valley 8D2 (decayed to 1990)	196 (7.7)	Ba-137m, Eu-154, Cm-244
West Valley 8D4 (decayed to 1990)	183 (7.2)	Ba-137m, Eu-154, Co-60

[a] 6.4 mm (1/4 in.) canister wall assumed.
[b] In the order of importance.

Decay Heat

The site specific decay heat generation is shown in Table 3. The maximum thermal activity of 700 watts/canister occurs for the West Valley HLW glass.

Table 3
HLDW Decay Heat

Waste Type (Ref. Time)	Watts/Canister
SRP (5 yrs out of reactor)	540
Hanford Exist. (decayed to 1995)	125
Hanford future (decayed to 2010)	54
Idaho (5 yrs out of reactor)	200
West Valley (decayed to 1990)	700

Since these heat generation values are much lower than either spent fuel assemblies or anticipated processed waste from commercial power reactors the dissipation of heat from a HLDW cask is expected to be non-critical.

TRANSPORTATION SCHEDULES

The DWPF at Savannah River is scheduled to be the first solidification facility on line and will begin operation in 1989. Transportation of West Valley solidified waste can begin in 1990. The Hanford and Idaho facilities will become operational in 1995 and 2000 respectively. The canister shipping requirements are given in Table 4.

Table 4
HLDW Transportation Schedule

Site	Period	Canisters/Yr	Total Canisters
Savannah River (DWPF)	1989-2003	511	7,665
	2004-2006	170	510
West Valley	1990-1995	60	300
Hanford	1995-2010	1282(a)	19,230
Idaho	2000-2014	2164	32,460
	2015-2030	791	11,865
		TOTAL	73,580

(a) Includes shipment of encapsulated cesium and strontium waste.

ASSESSMENT OF EXISTING CASKS

An assessment of truck and rail casks which are existing, or in-license, as potential candidates for the HLWD mission indicate that:

1. Small cavity diameters and excessive cavity lengths produce poor packaging efficiencies for the HLDW canister.
2. Gamma and neutron shielding thicknesses are excessive.
3. Several units employ fin arrays to dissipate much higher thermal loadings than is available with HLDW.

It is concluded that none of the existing, or in-license, systems are suitable for HLDW transportation. This is not too surprising since most are designed to carry spent LWR fuel assemblies which are smaller in cross-section, longer and require greater shielding and heat dissipation when compared to HLDW canisters. It is appropriate therefore to design a transportation system specifically for the HLDW canister.

HLDW CASK DESIGN BASIS

The objective is to design a cask which satisfies the safety requirements for both the normal conditions and the hypothetical accident conditions of transport while maintaining economy of operation, reliability and simplicity. The designers also recognize that the mission life of 41 years (1989 to 2030) represents a very long period in a dynamic regulatory environment. The HLDW cask design is based on the following:

1. Rail will be the principal mode of transport.
2. The maximum GVW will be 263,00 pounds.
3. The shipping and handling operations will be performed "dry."
4. The cask will provide all required containment functions.
5. The cask containment systems will be designed in accordance with ASME code rules for nuclear shipping packages currently being formulated.
6. The cask will meet the normal and accident criteria of the NRC and DOE regulations.

HLDW CASK DESIGN CONCEPT

One of the HLDW cask concepts currently under evaluation is a solid steel body cask of cylindrical design with a sealed, shielded inner lid and an outer impact lid. The cask body may be cast in steel or ductile nodular iron or fabricated by welding together one or more ring forgings to a forged base. The design concept being pursued is identified as a "convertible cask" and is shown in Figure 3.

An internal (replaceable) basket consists of an aluminum structure which contains cavities to receive the HLDW canisters. The cask wall consists of a minimum thickness of 158 mm (6.2 in.) steel corresponding to the shielding required for lowest activity fuel from Hanford. The basket of this cask is configured to carry eight canisters. As the need arises to accommodate higher activity waste a basket of lesser capacity but with supplemental shielding replaces the original basket, as shown in Figure 4. In this way a single, flexible containment structure can be reconfigured so as to optimize the capacity and shielding to the site specific waste characteristics while requiring only one basic certified design.

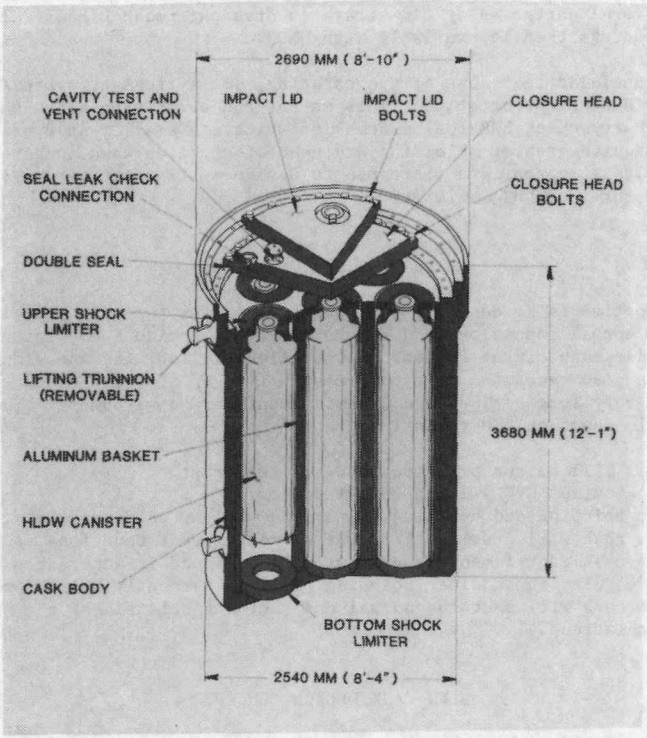

Figure 3. HLDW Rail Cask (8 Canister Capacity)

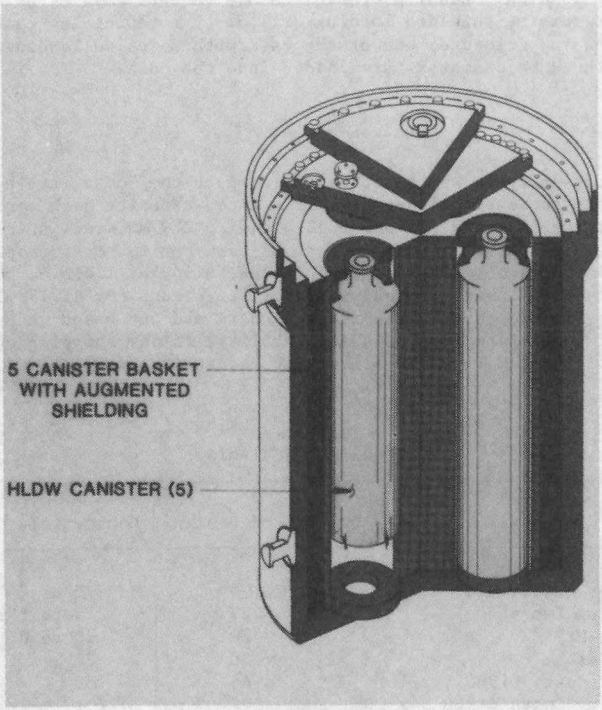

Figure 4. HLDW Rail Cask With 5 Canister Capacity And Augmented Shielding

The solid body cask is of such high integrity that a large "soft" external impact limiter structure is not required, thereby improving the cask handling characteristics and reducing the cask envelope dimensions. Only small internal energy absorbers are provided to protect the canisters from normal transportation and handling loads. Compact fins can be readily added for additional protection against the corner impact condition if required according to detailed analysis of the hypothetical accident condition.

The HLDW rail cask is carried horizontally on a flatbed rail car as shown in Figure 5. Cask handling and loading operations are performed dry. The cask is vertical top loading. Loading operations may be performed by transferring the cask into a remote, shielded loading cell or by rotating the cask 90 degrees and engaging the top (closure) end of the cask with a sealed loading port through which the HLDW canisters are loaded into the cask.

SYSTEM COSTS

A basis cost analysis was performed to compare the "convertible cask" concept to a more conventional design. If a fixed capacity cask were utilized it would have to be designed for SRP fuel, require 229 mm steel shielding and have five canister capacity. A comparison of cask capital costs, operating costs and total number of casks is shown in the following Table 5. Capital costs were calculated based on acquisition of rail casks for $1.15 million each. Operation costs include only transportation costs and are based on an average one-way trip of 1000 miles, 85% cask availability, 15 day shipping cycle and a shipping cost of $8.50/CWT - 1000 miles with return included.

Table 5
System Cost Analysis

Item	Fixed Design	Convertible Cask
Capacity (Canisters)	5	5 to 8
Shielding (mm steel)	229	229 to 158
Weight, Mt	80	80 to 87
Total Casks	38	27
Capital Costs (1980 $ x 10^6)	44	31
Operating Costs (1980 $ x 10^6)	297	201
Total Cost (1980 $ x 10^6)	341	232

Based on this simplified analysis, the convertible cask design reduces peak cask fleet requirements from 38 to 27 casks, and results in a 29% savings in capital cost and a 32% reduction in transportation costs. Overall cost advantage over the life of the program would amount to more than $100 million.

TOTAL CASK WEIGHT	= (174,500 LB.)	79,320 KG.
TIEDOWNS & PERSONNEL BARRIER	= (6,500 LB.)	2,955 KG.
RAILROAD CAR	= (68,000 LB.)	30,910 KG.
APPROX G.V.W.	= (249,000 LB.)	113,185 KG.

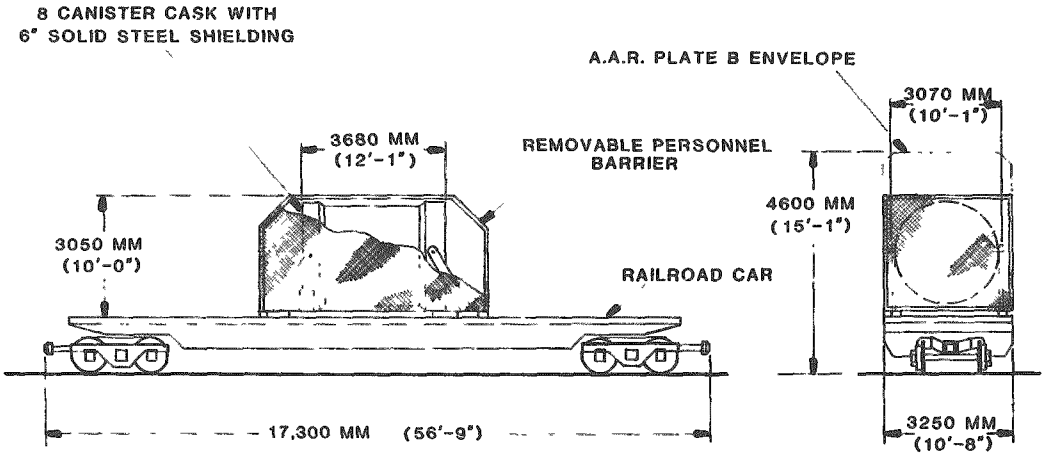

Figure 5. High Level Defense Waste Cask And Rail Transporter

CONCLUSIONS

A reference design concept has been described for a system to transport High Level Defense Waste from solidification and packaging facilities to a federal repository. This system is characterized by the following features:

1. High integrity solid body cask with dual lid.
2. 80 metric ton rail cask suitable for a four-axle flat bed transporter.
3. Uncomplicated design and dry handling permit efficient cask loading and unloading operations.
4. A convertible capacity design permits flexible and efficient packagi for multi-site utilization and lower overall costs.

Future program activity will be focused upon more detailed conceptual design, development of design criteria and system descriptions, systems safety analysis, and studies of in-plant handling operations to establish requirements for contract or remote handling.

REFERENCES

1. ERDA 77-42 (2 Vols.)., "Alternatives for Long Term Management of Defense High-Level Radioactive Waste, Savannah River Plant, Aiken, South Carolina," Energy Research and Development Administration, May 1977.

2. ERDA 77-44, "Alternatives for Long Term Management of Defense High-Level Radioactive Waste, Hanford Reservation, Richland, Washington," Energy Research and Development Administration, September 1977.

3. ERDA 77-43, "Alternatives for Long Term Management of Defense High-Level Radioactive Waste, Idaho Chemical Processing Plant, Idaho Falls, Idaho,": Energy Research and Development Administration, September 1977.

4. TID-28905 (2 Vols.), "Western New York Nuclear Service Center Study," U.S. Department of Energy, Washington, D.C., November 1978.

5. Johanson, S. W., and S. Su, "Consolidated High-Level Defense Waste (HLDW) Description," DC951001 (Draft), General Atomic Company, San Diego, California, June 1980.

6. GA-A16124, "HLDW Transportation System Reference Design Selection - Final Report for FY-80," Project Staff, U.S. Department of Energy Contract No. DE-AT03-80SF10791, General Atomic Company, San Diego, California, September 1980.

TRU WASTE TRANSPORTATION PACKAGE DEVELOPMENT*

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Contact-handled transuranic (CH-TRU) waste is generated at numerous locations in the United States as a by-product of defense related programs. Some of these facilities are incapable of storing large quantities of waste and must periodically transport Type B quantities to other Department of Energy (DOE) sites. At some time in the future the majority of the existing waste will be placed in a federal waste repository. CH-TRU waste will require transportation from its present storage locations to the repository. If processing is required prior to disposal, an additional transportation step will be added for those sites which cannot process their own waste.

The Transportation Technology Center at Sandia National Laboratories is responsible for the development of new generation transportation systems to be utilized in the management of defense related CH-TRU waste. The development program places particular emphasis on the compatibility of transportation hardware with both waste generating and receiving facilities. An initial design concept has been completed and scale models are being fabricated and tested. Concurrent structural analysis, thermal analysis, and material test programs are being conducted to support the design effort. This paper presents a summary of the activities involved in the development program and a brief overview of the progress to date.

A list of the inventory of the transuranic waste at a number of DOE sites[1] in the United States is given in Table 1. Both the buried and retrievably stored waste as well as the projected annual storage rates are given for each site. The buried waste mentioned in the table was accumulated at the sites prior to 1970. In 1970 the United States Atomic Energy Commission initiated a policy whereby TRU waste with actinide activities greater than 10 n Ci/gm must be retrievably stored. Much of the stored waste and buried waste (were it retrieved and packaged) indicated in Table 1 is CH-TRU. Since there is a possibility the buried material will be exhumed, packaged, and handled along with the stored waste, the need for safe, cost-effective transportation hardware is obvious. Even if only the stored waste is sent to a repository, the waste volumes involved are still formidable.

Retrievably stored CH-TRU waste typically consists of plutonium contaminated metal scrap, sludge, paper, filters, and other materials resulting from weapons production and reprocessing operations. The activity of the waste is low with the maximum dose rate at the surface of a container of CH-TRU being less than 200 mrem/h. Most of the waste has a plutonium content of less than 18 Ci/m^3.

*Work sponsored by U. S. Department of Energy under Contract No. DE-AC04-76-DP00789.

Table 1. Inventory and Storage Rates of Transuranic Waste at DOE Sites

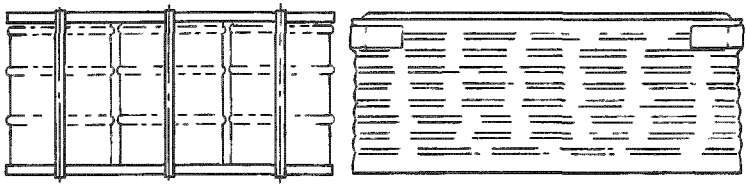
Site	Inventory (m^3) Buried	Inventory (m^3) Stored	Estimated 5 Year Storage Rates (m^3/year)
Oak Ridge National Laboratory	6300	1200	70
Los Alamos Scientific Laboratory	11500	3500	550
Savannah River Plant	27000	2400	150
Nevada Test Site	5700	240	25
Hanford Site	225000	8000	1400
Idaho National Engineering Laboratory	57000	40000	2800

The stored waste is contained primarily in steel drums and fiberglass reinforced plywood (FRP) boxes, although a small amount is being stored in rectangular steel boxes known as M-3 bins. Table 2 itemizes some of the waste units with which the new generation transportation systems will need to be compatible. The term "six-pack" in the table refers to a proposed efficient method of handling six 200-litre (55-gallon) steel drums. In this method a rectangular array of six drums is placed between two welded steel frames (a frame being placed at the top and bottom of the drum array) and the assembly is then banded together into a unit with steel strapping. A side view of an assembled six pack is shown in Figure 1. Also shown in Figure 1 is a side view of the modular box. The modular box is currently being developed at Sandia. It is constructed of corrugated sheet metal and is meant to serve as a primary waste container or as an overpack for up to six damaged or otherwise unserviceable 200-litre drums. Used as a primary waste container, the modular box is obviously more efficient volume wise (i.e., can contain more waste) than the six drums it replaces. The box is also estimated to be less expensive than six drums with rigid polyethylene liners. The modular box is scheduled to be tested to Type A requirements (IAEA Safety Series No. 6) during the 1981 fiscal year. Once the box is approved, it will be recommended as the new standard container for CH-TRU waste.

When the Type-A containers are assembled into groups large enough to be transported efficiently, Type-B quantities of some radionuclides (particularly plutonium) may be present in the accumulated volume. Type-B packaging is therefore required and the TRUPACT must provide double containment as specified by the

Table 2. Containers of Waste Requiring Transport

Container	Container Dimensions (m, LxHxW)	Container Mass (kg)
Six-Pack	1.3 x 1.0 x 1.9	2300
Modular Box	1.3 x 1.0 x 1.9	4500
FRP Box	1.2 x 1.3 x 2.1	2300
M-3 Bin	1.2 x 1.8 x 1.5	1500



Six Pack Modular Box

Figure 1. Six-Pack and Modular Box CH-TRU Containers

United States Nuclear Regulatory Commission as well as provide structural and thermal protection to the waste in accident environments as specified in Appendix B, Hypothetical Accident Conditions, of 10CFR71. (These accident environments are equivalent to those found in IAEA Safety Series No. 6.) No radiation shielding or internal heat removal mechanism is necessary for this waste.

Concepts for safe, cost-effective Type-B packagings that can handle drums or boxes are currently being developed and assessed. The leading conceptual design is known as the TRansUranic PACkage Transporter (TRUPACT). As illustrated in Figure 2, the TRUPACT is envisioned to be a large metal container consisting of inner and outer tubular steel frameworks which are separated by rigid polyurethane foam and sheathed with steel plate. To minimize the package weight and maintain ductility at low temperatures, a high-strength, low-alloy steel is used in both frames and the outer skin. The sides of the cargo compartment are constructed of carbon steel clad with stainless steel such that the stainless steel surface faces inward. This construction facilitates decontamination if required. The inner structure is made secure with a hinged, 13-cm-thick door that has a tubular steel framework and is filled with foam. This door is fitted with dual elastomeric seals and bolted in place during transport. The exterior door, which is attached to the outer frame with hinges, is 90 cm thick, utilizes a single elastomeric seal and

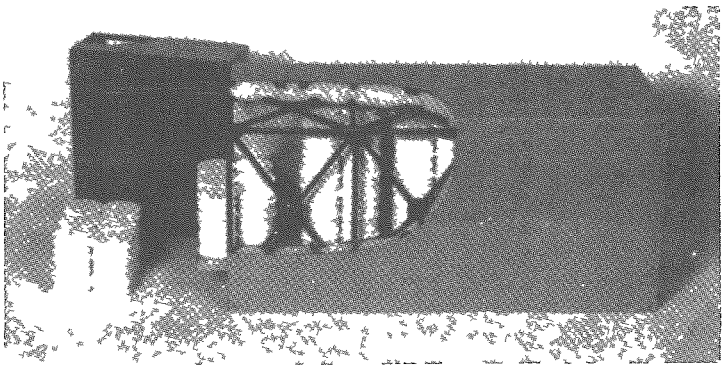


Figure 2. TRUPACT

is also bolted in place during transport. There is approximately 90 cm of foam in the package ends and 36 cm in the walls to provide impact energy mitigation and thermal protection.

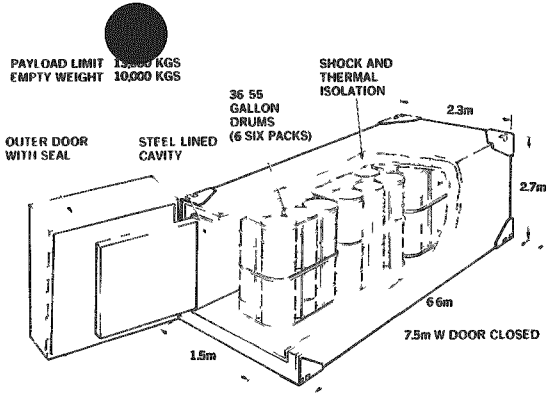
Although making the TRUPACT transportable by rail and truck was a design goal early in the program, transportation constraints and payload maximization considerations suggested the need for a different version of the package for each mode. The rail transported TRUPACT has approximate external dimensions of 7.3 x 2.7 x 3.0 m (L x W x H) and an internal cargo compartment measuring 5.5 x 2.0 x 2.4 m. The smaller truck version measures 7.5 x 2.3 x 2.7 m externally and 5.7 x 1.5 x 2.0 m internally. Figure 3 illustrates the six-pack loading orientations to be used for each version. Six six-packs (or modular boxes) may be transported in the truck version of TRUPACT and eight can be accommodated in the rail version. The empty rail verson weighs 11 tonne and carries a maximum payload of 19 tonne. The truck version weighs 10 tonne and carries a 13 tonne payload. The weights and dimensions of the two TRUPACT versions are such that one truck TRUPACT may be transported by legal weight truck and two rail versions by one railcar (see Figure 4).

Concurrent with the structural analysis of the TRUPACT, full- and quarter-scale tests were completed on containers filled with various forms of simulated CH-TRU waste to assess the cargo response in accident environments as well as its effect on the response of the loaded TRUPACT. The tests were conducted in both static and dynamic environments. Static tests consisted of crushing individual drums and modular boxes. The dynamic tests involved dropping drums individually, in arrays, and in modular boxes from various heights. Modular boxes containing loose simulated waste were dropped singly and in arrays from 9 m. The data generated from these tests have led to the development of realistic analytical cargo models.

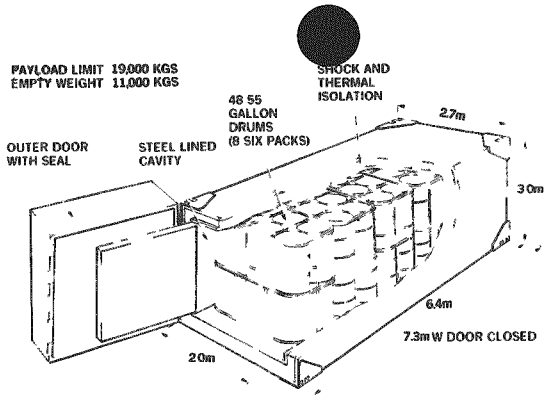
The vulnerability of the TRUPACT to puncture is being evaluated both analytically and experimentally for various wall configurations. Results of this investigation, together with information from the assessment of the mechanical and thermal response of candidate foams, are expected to require some modification of the existing wall cross sections to optimize punch resistance.

The structural response of the TRUPACT to the 9 m end-on, side-on, and center-of-gravity-over-corner drop tests has been evaluated analytically. Each case is being compared to the results of quarter-scale model drop tests as the tests are conducted. To date, comparisons have been made only with the end-on 9 m drop test. The extent of the damage sustained by the quarter scale model during this test is shown in Figure 5. The average reduction in length of the outer box was seen to be 3.2 cm, which corresponds to a full-scale crush of 12.8 cm. This agrees with the analysis which predicted a crush of 13.2 cm. The package has thus far performed satisfactorily and no significant changes to the design are expected as a result of the structural evaluation for impact environments. A detailed description of the TRUPACT structural analysis program may be found in the companion PATRAM '80 paper, "Contact-Handled Transuranic Transportation Systems Structural Analysis."

The TRUPACT conceptual design essentially is complete. Additional work on the conceptual design and thermal and puncture tests will be completed during 1981. Following DOE approval of the concept, a package of system specifications and drawings will be prepared and a contract placed in private industry for the completion of a detailed design in FY 1982. Activities during FY 83 and FY 84 will mainly involve construction, test, and possible redesign of ptototype hardware. Production units should be available in 1985. Figure 6 shows the current schedule for the remainder of the program.



TRU PACT CONTAINER
SHOWING TRUCK CONFIGURATION WITH SIX PACKS



TRU PACT CONTAINER
SHOWING RAILROAD CONFIGURATION WITH SIX PACKS

Figure 3. TRUPACT Truck and Rail Version

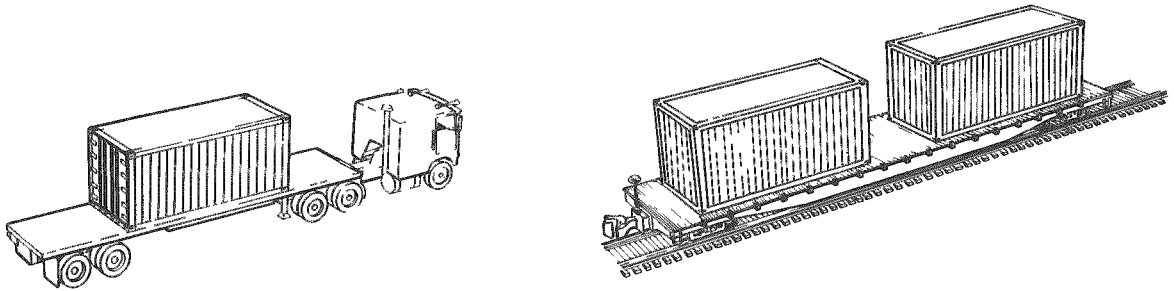
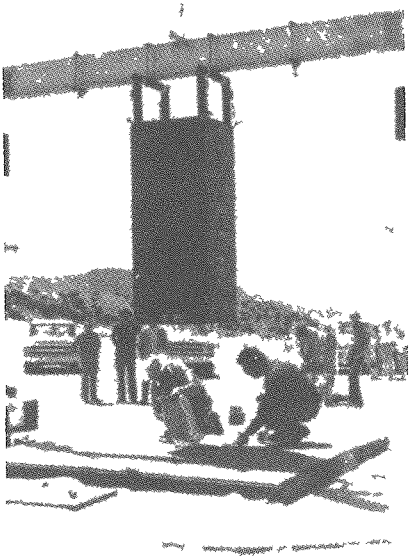
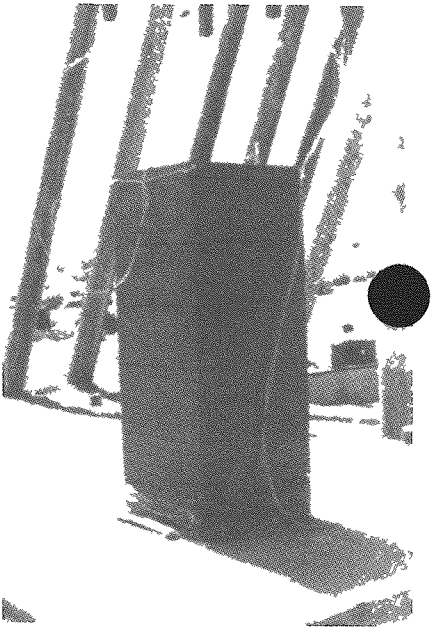


Figure 4. Configuration for TRUPACT Truck and Rail Transport



Quarter-Scale TRUPACT Being Readied for End-on 9 m Drop Test



Quarter-Scale TRUPACT Following End-on 9 m Drop Test

Figure 5. Quarter-Scale 9 m Drop Test of the TRUPACT

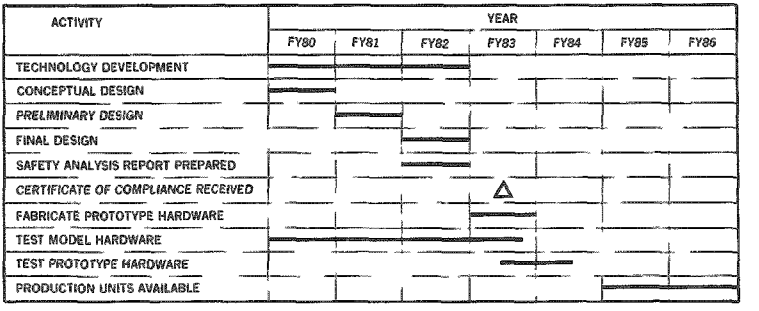


Figure 6. Schedule for CH-TRU Transportation System Development

REFERENCES

1. U.S. Department of Energy Acceptance of Commercial Transuranic Waste, DOE/AL/TRU-8001, Rockwell International-Rocky Flats Plant, Boulder, Colorado, February 1980.

2. H. C. Shefelbine, Preliminary Evaluation of the Characteristics of Defense Transuranic Waste, SAND78-1850, Sandia National Laboratories, Albuquerque, New Mexico, November 1978.

DISCUSSION

Question by M. L. Bell, USA: What are the expected costs of the TRUPACT and type A package? Were burial leads considered in design of the type A package for interim storage?

Answer: We expect the cost for the TRUPACT to be approximately $ 200,000/unit and for the type A packaging (modular box) to be less than $ 500/unit (in US dollars). Please be advised that these figures are very rough estimates and are affected greatly by the quantities produced and by manufacturing methods. Repository loads were considered in the design of the type A packaging. This is described in the design criteria for the Waste Isolation Pilot Plant (WIPP).

Question by W. R. Taylor, Canada: Could you explain how the foam might behave in the thermal test? Will it char?

Answer: We have an analytical and experimental program to investigate the response of foam in a thermal environment. A discussion of the results from the preliminary phase of the program is described in paper 142 in Poster Session B entitled "One-Dimensional Thermal Response Modeling of a Transuranic Foamed Overpack Shipping Container System to a Fire." Generally speaking, the foam will decompose, forming pyrolysis gases and a char residual. The char residual forms a thermal barrier which slows the thermal penetration rate, hence reducing the heat load to the contents.

A NEW CONCEPT OF PACKAGING, TRANSPORT AND DISPOSAL OF INTERMEDIATE LEVEL WASTE IN THE UNITED KINGDOM

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Introduction

The concept outlined in this document is part of system engineering studies, sponsored by the Department of the Environment, to aid the formulation of possible future practices in the United Kingdom.

The methods used in the United Kingdom to package, transport and dispose of radioactive waste over the last 30 years have derived mainly from methods of dealing with laboratory wastes. Viewed on a national basis disposals have only been a small fraction of the arisings, so that the majority of radioactive waste generated by the UK nuclear power programme remains stored at the sites where it originated. Whether such wastes are disposed to sea or to land or to an interim store is outside the scope of this paper but for the purposes of this study it was assumed that the amount transported is likely to increase.

For sea disposal the transport package has been, almost without exception, exactly the same as the disposal package. When intermediate level wastes, (i.e. those requiring shielding), have been disposed to sea it has necessitated the introduction of internal radiation shields into existing types of disposal package, as appropriate,to reduce the surface dose rates to acceptable levels. Since the existing packages are small, the introduction of shielding significantly reduces the payload and the cost of the disposable shielding is a dominant item.

Consideration of the cost components of small shielded drums containing intermediate level waste therefore raises the question of how to reduce the fractional cost of the shielding. A preliminary study of intermediate level waste streams in the UK has indicated that many could potentially benefit from the development of returnable shielding concepts. The nearest term development which it appeared useful to study was the disposal of spent ion exchange resins, used for irradia[redacted] fuel storage pond clean-up and effluent treatment, particularly from the Trawsfynydd and Hinkley Point nuclear power stations of the CEGB. Plant design studies to deal with the accumulation of resin at these sites have been aimed at the selection of a disposal route which can be operational in 1983. Faced with this situation the designers can only reasonably choose from within the range of already approved packages. The present solution is, therefore, to use a Type 1803 (formerly designated 0094) sea disposal package with it being transported under the Low Level Solids category. The dimensions and volume of this type of package are given later. To aid compliance with the Ministry of Agriculture, Fisheries and

Food Code of Practice for sea disposal packages it has been necessary to consider the incorporation of iron radiation shields into the package (lead being considered inadmissible for disposal to sea). The cost of the shielding has been estimated to be 80% of the overall cost of packaging, transport and disposal.

The spent ion exchange resin streams therefore provide a practical case for evaluation of the potential for returnable shielding. The scope of the initial feasibility study, reported here, was to prepare a conceptual design of shielded flask system for packaged spent ion exchange resin from Trawsfynydd and to compare the economics of the chosen design with the use of a UK Type 1803 sea disposal package.

The work, which was carried out under the aegis of the Department of the Environment was co-ordinated by AEE, Winfrith. A design study contract was placed with NEI Projects Ltd. (who were supported by International Research and Development Company Ltd.). A separate contract covering ship design and marine transport aspects was placed with Burness, Corlett & Partners Ltd.

Package Design Considerations

Before commencing work on possible design solutions, packing practice in the USA was reviewed to assess possible application in the UK. It was found, however, not to be strictly relevant for three reasons:-

(a) road rather than rail transport is the primary mode of transport and different constraints apply, (e.g. a lower all-up weight for road transport);

(b) the IAEA transport classification of 'Low Level Solids' is not used at present;

(c) sea disposal is not currently used by the USA.

It was considered from the outset that a heavy radioactive load in the UK would be transported for the majority of its journey by rail. Since the railways are extensively used for the transport of irradiated fuel in flasks weighing up to 50 tonne, it appeared appropriate to plan to use the shielded radwaste flask transport on the same routes using existing cranes and designs of railway wagon wherever practicable. Nevertheless, minor road journeys from nuclear sites to associated railheads are still required. The main constraints implied by the use of irradiated fuel flask transport routes are a gross weight of 55 tonne and external dimensions in conformity with the British Rail W5 structure gauge for rail freight traffic.

The basic design concept is a returnable shield or flask containing unshielded drums. To minimise the surface to volume ratio it is desirable that the flask shall be as large as possible within the constraints of size and weight. The evaluation of these constraints suggests that if the flasks are designed within the following dimensions they should not present practical handling difficulties.

SHAPE	SHAPE CODE	EXTERNAL DIMENSIONS (m)
Rectangular Maximum Height	RMH	2.44 (long) x 2.21 (wide) x 2.21 (high)
Rectangular Maximum Width	RMW	2.59 (long) x 2.59 (wide) x 2.06 (high)
Cylindrical	C	2.59 (long) x 2.59 (diameter)

The initial approach was to study the conveyance in a shielded flask of standard waste drums whose dimensions are:-

Type	Nominal Internal Diameter (inches)	Nominal Height (inches)	Capacity (litres)
1801	22	36	220
1802	26	42	360
1803	33	45	640
1804	30	60	730

Assuming a nominal 127mm wall thickness for shielding purposes the following payloads were then established:-

Drum	Package (see Note)	No. of Drums	Vol. (litres)	Note
1801	RMH	18	3,960	C4 = drums in 4 columns C7 = drums in 7 columns
	RMW	9	1,980	
	C4	8	1,760	
	C7	14	3,050	
1802	RMH or RMW	9	3,240	
	C4	8	2,880	
	C7	14	5,000	
1803	RMH or RMW or C4	4	2,560	
1804	RMH or RMW	4	2,920	
	C4	4	2,920	
	C7	7	5,510	

Because of the relatively poor payloads achieved by using standard drums, t study was therefore extended to non-standard drums ~360 litre capacity. By va ing the height and diameter of the drum the payload could be increased by about 40%. In consequence it was decided to specify a non-standard drum of 360 litre volume to investigate shielding requirements and establish the conceptual design of shield. With this non-standard type of drum the payloads which can be achieved are illustrated below:-

SHAPE CODE	CONFIGURATION OF DRUMS	No. OF DRUMS/FLASK	TOTAL PAYLOAD VOLUME (m^3)
RMH	3 x 3 x 2	18	6.5
RMW	3 x 3 x 2	18	6.5
C4	4 x 4	16	5.8
C7	7 x 3	21	7.55

Disposal Ship Design Considerations

For completeness the study was taken through to disposal. A working assumption was made that disposal would be to sea. However, it was appreciated that the development of returnable shielding (whether the material is carried as Low Level Solids or in a Type B flask) was likely to be equally applicable to land burial. A principal reason to expect commonality is that the railway freight gauge is a key constraint.

It was not possible to make a rational choice of the ship cargo carrying capacity. Nevertheless, certain major design principles were established:-

1. a variety of types of package, not only those discussed in this paper, ranging in gross volume between 0.2 and 10 m^3, with disposable or returnable shielding, is to be expected. For ease of handling it is desirable that the packages are handled in modules, the module size being dictated by the railway freight gauge;

2. the ship would have damage survival capability of a standard similar to the IMCO "Code for the Construction and Equipment of Vessels Carrying Dangerous Chemicals in Bulk" by inclusion of side and double bottom tanks;

3. to avoid restriction on possible loading ports the ship draft should not exceed 5 metres.

Selected Design of Returnable Shielding

Shielding calculations were carried out for three drum arrangements (RMH, C4 and C7) by the Radiation Physics and Shielding Group, Reactor Physics Division, AEI, Winfrith, using the RANKERN computer code and input data appropriate to Traws-fynydd resin incorporated in a polymer/cement matrix.

The results of these calculations are given on the next page - (Table A).

Given the assumption that the fabrication costs are £1000/te for lead shielding and £500/te for steel shielding it is possible to derive the comparative costs of the above three configurations - (next page Table B).

The results of these calculations indicate that the cylindrical configuration C7 of 21 Drums (each 0.76m diameter, 0.76m long, capacity 360 litres) arranged in a hexagonal close-packed array to give 7 drums in each of 3 layers is the most cost effective of those studied.

Selected Handling Procedure

In selecting the preferred method of handling and discharge, simplicity,

TABLE A	Configuration of Drums in Shielded Flask		
	RMH	C4	C7
	Type 1801 Drum	Non-standard Drum (see text)	Non-standard Drum (see text)
No. of drums	18	16	21
Total drum vol. (m^3)	5.22	5.83	7.6
Total drum weight (te)	6.95	7.75	10
Wall thickness of lead (cm)	6.5	6.1	6.5(ends), 5.2(sides)
Wall thickness of steel (cm)	13.2	12.2	13.2(ends), 10.5(sides)
Flask weight (te) (lead shielding)	21.5	21	20
Flask weight (te) (steel shielding)	28.4	27.7	26
Overall weight (te) (lead shielding)	28.5	29	30
Overall weight (te) (steel) shielding)	35.4	35.4	36
Weight of lead/m^3 waste	4.1	3.6	2.46
Weight of steel/m^3 waste ...	5.45	4.57	3.4

TABLE B	CONFIGURATION OF DRUMS IN FLASK		
	RMH (3x3x2)	C 4 (4x4)	C 7 (7x3)
Cost of lead overpack (£K)	21.5	21	20
Cost of steel overpack (£K)	14.2	13.8	13
Contribution of lead shielding to package cost (£K/m^3)	4.1	3.6	2.6
Contribution of steel shielding to package cost (£K/m^3)	2.7	2.4	1.7
Activity of Contents (Ci)	503	562	733

reliability and a minimum of handling operations were considered most important. These factors therefore were given the highest weighting in the subjective system used to select the preferred system. Other factors which were brought into account included radiological impact and safety. It is not considered that a more

extensive assessment system would have changed the conclusions. The chosen system employs an inner disposable frame which is filled at the nuclear site with unshielded drums from behind a radiation shield. The frame is then inserted into the flask, the lid fitted and the flask transferred by road to the railhead. The flask is then lifted by crane onto the rail wagon, conveyed to the port and transferred, using the ship's derrick crane, to its hold. When the ship reaches the sea dump area the shielded flask is placed alongside a moonpool in the ship, the lid removed and the frame complete with drums lifted out into a shielded area, then lowered into the moonpool and released with the load under water.

Conceptual Ship Design

Consideration of the design constraints and requirements and the evolution of the flask design and handling arrangements resulted in the development of possible design characteristics for a ship to convey packaged waste. It would have a cargo capacity of 68 modular packages. Due to the modular arrangement a range of packages can be accepted. The lower hold would be arranged for the carriage of 20 modules and the upper hold 48 modules. The overall dimensions of the vessel would be largely dictated by the volumetric requirements for cargo stowage and the provision of wing and double bottom tanks. The principal particulars of such a design are:-

Length overall	106	m
Breadth	20	m
Design Draft	4.5	m
Service Speed	14	knots (from twin screw propulsion)

The design has the accommodation and machinery space aft and separated from the cargo by a water-filled coffer-dam. Immediately forward of the coffer-dam would be the moonpool through which the cargo is discharged. Within the cargo holds the modules would be moved by overhead gantry cranes with longitudinal and transverse movement capability. The modules would be located in standardised corner fittings within the holds and secured by tensionable lashings.

Cost Estimate

Relative Cost

Because CEGB North Western Region Scientific Services Department had already carried out an estimate of the cost of disposal of Trawsfynydd resin using Type 1803 packages with lost radiation shields made from lead or steel these cost data were used as a major input. Additional data came from British Rail and AERE, Harwell. Although data on the cost using lead shielding was prepared there was no expectation that this would be permissible.

The cost estimate for sea disposal using Type 1803 packages was revised and this was then compared with the cost of disposal of the same amount of radioactive waste using the returnable shielding system. In this latter case the assumption was made that the operator owned the ship and that the rates cover both capital and operating costs. It was also necessary to decide on a system of apportionment of costs between different types of waste package. The estimated costs are given on the next page - Table C.

More recent work by CEGB on the single drum disposal route has indicated that the costs could be reduced to $\sim$3.4 £K/m^3 of solidified waste by the use of iron shot concrete as shielding.

Assessment of Radiological Detriment

The operations involved in manufacturing and disposal of solidified

TABLE C System	Disposal Cost £K	Specific Disposal Cost £K/m^3 of Solidified Waste	Remarks
Single Type 1803 drums	916	5.5	(Steel Shielding. (Lead Shielding (assumed to be inad- (missible would cost (~42% less.
As above but handled in crates	916.5	5.5	
Unshielded drums in returnable shielded flask	((312.25 (400 (	1.88 2.35	Flasks at £ 40K each Flasks at £100K each

Trawsfynydd resin matrix in single self-shielded Type 1803 drums are:-

1. transfer drum to plant
2. insert matrix
3. insert radwaste
4. mix
5. transfer to lidding
6. cap
7. transfer to store
8. monitor/label
9. load drums for road transport
10. transport by road to railhead
11. transfer to rail
12. transport by rail to port
13. transfer to ship
14. dispose to sea

The times for these operations and the dose rates have been assessed, leading to an overall estimate of the dose equivalent for these operations of 15 mrem/drum, or 12 rem for the whole consignment of 795 drums.

With the returnable shielded flask system there is no internal shield in each drum and a simple adaptation of the currently proposed operations would be quite inadequate because the dose/drum would increase to an estimated 48 mrem. Fitting an external transfer shield around each drum before filling would reduce the radiation dose to ~ 1.75 mrem/drum. The use of remote operations could further reduce this value to ~1 mrem/drum. Virtually all of the dose is incurred when close to the drum for operations such as capping, slinging for transfer to store loading onto a transport vehicle at the road/rail transfer and the rail/ship transfer. It is probable that by improved plant design, embodying greater use of remote control facilities and partition shielding that the assessed dose would be considerably reduced for both the currently proposed system and the returnable shielded overpack. In consequence, it is not very profitable to compare the relative radiological detriment of two moving targets. Nevertheless, it can be expected that a returnable shielded flask system would incur a lower dose than the currently proposed internally shielded drums.

It may be noted that the doses in the transport sector are estimated to be less than 0.5 mrem/drum.

Conclusions & Comments

1. The amount of medium and low level radioactive waste which is transported in the United Kingdom is a small fraction of the annual arisings from the nuclear power programme. It has been assumed for the purpose of this study that this traffic will increase and require the development of modern cargo handling methods.

2. In support of more economical bulk handling methods the study has indicated, by concentrating on one particular waste stream, that the concept of using returnable shielded flasks for the transport and disposal of intermediate level waste, which is novel in the UK, is potentially cost effective.

3. Were sea disposal to be used, the introduction of such a system would probably necessitate the use of a special purpose ship. A conceptual design of such a ship has been prepared.

4. The development of returnable shielded flasks is also of benefit to the economics of land disposal where it will be important to minimise the volume placed in the repository.

5. Not all waste will be able to be transported under the Low Level Solids classification but the concept of using a shielded flask is equally valid for transport as a Type B package.

Acknowledgements

The work was carried out under the sponsorship of the Department of the Environment. It relied heavily on input data from the CEGB, Barnwood and CEGB NW Region Scientific Services Department, and the authors would particularly like to thank Mr A S D Willis and Dr C Smitton for their contributions. The authors would also like to thank Dr P C Miller, of AEE, Winfrith, for the shielding calculations.

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DISCUSSION

Comment by W. Kolb, FR Germany: During the discussion of the 1983 revision of IAEA-SS No. 6 you mentioned that it would be very restrictive for the shipment of waste to require that the dose rate 1 m distance from the unshielded content of the LSA package should not exceed 1 rem/h. The kind of transportation you describe in your paper seems to be a typical case of a special arrangement procedure in which case the package requirements can be arranged with the competent authority. Therefore, you should not be concerned about the proposed provision mentioned before which is intended for industrial type LSA packages mainly under non full load conditions.

Answer: It is one possible way of dealing with the problem.

Comment by M. S. T. Price, United Kingdom: There may be a case for waste disposal being dealt with under the proposed new LSA-III transport category as a Special Arrangement; but it might be useful to anticipate extensive use of Special Arrangements for radioactive waste transport to storage or disposal by making a specific sub-category of LSA-III within the regulations to cover this.

Answer: I agree that the regulations should provide flexibility for design options for waste which is transported prior to disposal.

Comment by G. Cohendy, France: My comment is not a question but an observation. I do not think you are right when you say that you would like the regulations to take into account problems particular concerned with waste to be disposed of at sea. To me this seems to be a rather dangerous course, as it would then be a pretext for other "special" problems to be considered and would then lead without doubt to the disappearance of the main strength of the regulations, i.e. their uniqueness

CONCEPTS FOR PACKAGING RADIOACTIVE WASTE GENERATED BY DECOMMISSIONING OF NUCLEAR POWER REACTORS

H.J. Engelmann, V. Duill, R. Köster, KfK
P. Petrasch, NIS

Introduction

Within the frame of investigations concerning the suitability of the Konrad mine as a storage facility, wastes arising from decommissioning were taken into account, KfK and NIS elaborated a study on radioactive wastes generated during the decommissioning of nuclear power stations and suggested a packaging concept.

The study was based on the quantities received from 29 power stations with 30 GWe presently operating or in the construction and planning stages, respectively. A 30 year period of normal operation was assumed (planned decommssioning).

Moreover, the following boundary conditions were taken into account:

- The study was restricted to commercially operated power reactors.
- Complete removal of the nuclear reactors was assumed.
- Waste from normal operation, as mobile components, coolant and off-gas filters were not considered as wastes arising from decommissioning.
- Shaft capacity of the Konrad mine.
- Packaging in "lost containers".
- The time between reactor shutdown and the start of removal was assumed to be 1, 10, 30 and 50 years respectively.

2. Classification of Decommissioning Wastes into Activity Categories

The amount and type of wastes generated during decommissioning were determined for all 29 power stations by studies made on typical reference power stations such as Biblis A as regards a PWR, and Brunsbüttel as regards a BWR.

Installed Power	Total Mass	Mass of activated parts	Mass of contaminated parts	Mass of secondary waste	Mass of handling waste
MW_e	Mg	Mg	Mg	Mg	Mg
29,395	$8 \cdot 10^6$	$3.2 \cdot 10^4$	$1.6 \cdot 10^5$	$4.3 \cdot 10^4$	$1.8 \cdot 10^4$

Table 1: Total amount of calculated decommissioning wastes in Mg from the 29 nuclear power stations

In order to suggest a suitable packaging system for the wastes generated during decommissioning we classify into 3 categories (related to Co-60).

- Category 1: > 3×10^{-4} Ci/g
  (e.g. reactor pressure vessel internals)
- Category 2: 3×10^{-4} to 10^{-8} Ci/g
  (components and structural elements affected by the neutron flux, but not directly exposed to it)
- Category 3: < 10^{-8} Ci/g
  (contaminated parts)

Decommissioning of a nuclear power station and its subsequent removal takes period of about 6 years according to data from NIS-studies.

Accordingly, different types of masses must be handled every year, depending on the sequence of disassembly, assuming a decommissioning period of between 1 year and 50 years after reactor shutdown.

Time schedule for decommissioning	Percentage of Waste Generated % per Year		
	Category 1 Steel Components	Category 2 Steel Components	Category 3 60 % Steel Components 24 % Concrete 16 % Burnable Waste
1	-	-	29
2	100+	50	29
3	-	50	7
4	-	-	7
5	-	-	14
6	-	-	14

Table 2: Generation of relative amounts of waste categories as function of time schedule for decommissioning

+Will be produced regardless of time between reactor shutdon and the start of decommissioning, required by the dismantling itself, always during the second year of decommissioning.

3. Proposal for Elaboration of a Container Concept

It is evident from the estimated decommissioning wastes that about 2.6×10^5 of radioactive wastes are generated, which must be transferred to a final stor These decommissioning wastes must be put into containers so that a minimum of handling procedure, dose burden and final storage volume as well as a maximum in safety aspects are guaranteed.

3.1 Boundary Conditions

The concept of a container for final storage exerts a marked influence on de-

commissioning of a power station. Conditioning, crushing of components etc. are determined by the container structure (shape, dimensions etc.). Contrarily internal transport, winches etc. will influence the structure of the containers.

The boundary conditions for transport have been defined for those transports routes eligible here (rail, road, waterways) in the relevant ordinances.

Further boundary conditions result from transport aspects in the final storage itself.

3.2 Container Design

The structure of the containers was designed under above mentioned boundary conditions, like total mass and statutory dose burden with reference to dimensions of the containers, wall thickness, net loaded masses and specific activities.

3.2.1 Dimensions and Net loaded Masses

The total mass of a container is presented in the following way:

$$M_{\text{Total Mass}} = M_{\text{Container}} + M_{\text{Waste}} + M_{\text{Fixing Material}}$$

equally the composed total volume of:

$$V_M = V_C + V_W + V_F$$

Further the filling factors were defined as follows:

$$f = \text{filling factor} = \frac{\text{effective filled volume of waste}}{\text{theoretical fill volume of waste}}$$

S = wall thickness

ϱ_i = theoretical density of the internal of containers

$$\varrho^x = \frac{\text{theoretical density of the internal containers}}{\text{density of container material}}$$

Out of that the maximum dimensions of the container result as cylinder or rectangle.

Cylinder:

$$r^3 \cdot x \cdot \varrho^x + r^2 [2s(1+x)] + r[s^2(4+x)] - \left[\frac{M_M}{\varrho_c} - 2s^3\right] = 0$$

Rectangle:

$$l^3 \cdot y \cdot z^2 \cdot \varrho^x + l^2 [2s(y \cdot z + y \cdot z^2 + z)] + l[4s^2(1 + y \cdot z + z)] - \left[\frac{M_M}{\varrho_c} - 8s^3\right] = 0$$

l, x, y, z are the geometric factors of the containers (length, width, diameter, and height).

The results of equations were presented graphically in curves, whereby the wall thickness and the geometric factors were put as variables.

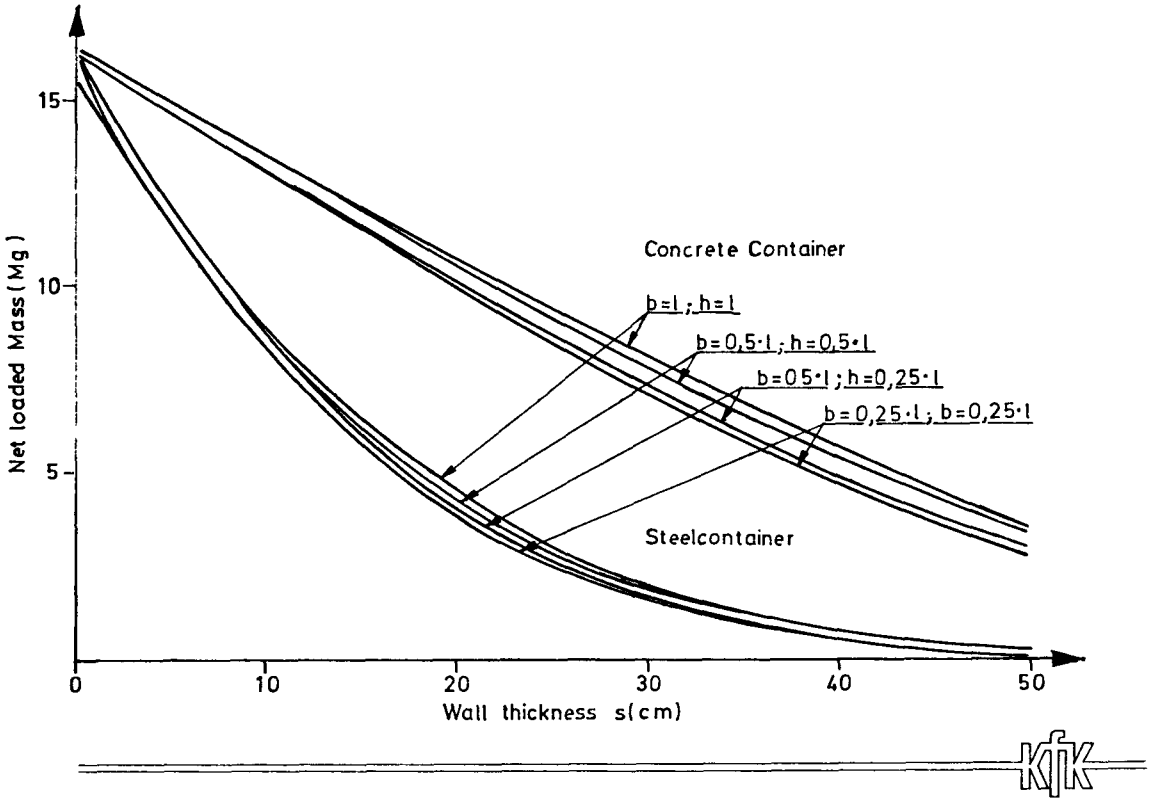

Fig. 1: Net loaded Mass in rectangle Containers (Steel and Concrete)
Total Mass: 20 Mg ; Space factor 0,6

In figure 1 we can find the net loaded mass of a container with a weight of 20 Mg for a certain wall thickness and an adequate geometry. The wall thickness will be definited by the specific activity.

3.2.2 Specific Activity

The maximum input of the specific activity in a container results from the statutory dose burden at the surface and at 1 m distance.

The following model is assumed for calculation:

- A part of waste contacts the inside wall of the container, the shielding effect of fixing material can be neglected.
  The shielding of the container is composed of shielding effect of container wall and "geometric attenuation".
- The "geometric attenuation" will be assumed as relation between surface of waste (= inner surface of the container) and theoretical surface in 1 m distance.
- Related to the important nuclides Co-60 and Mn-54.

In pursuance of these modul calculations for each of these three categories it was determined one type of container.

Category 1 includes materials with specific activities > 3×10^{-4} Ci/g. So as to form with the regulations applicable to the individual transport routes a type B container must be used for this category.

The selected container material is cast steel. The most favorable container geometry seems to be a cylinder which fulfills most conveniently by required properties of a type B package.

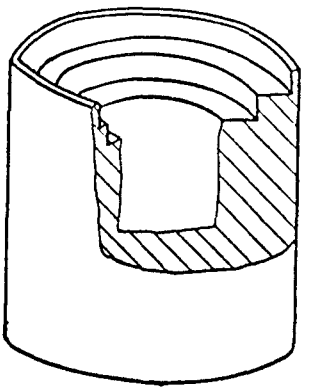

Fig. 2: Cast steel container of category 1 (20 tons)

The containers for the transport and final storage of category 2 wastes are likewise made as cylindric cast steel containers. To cover the large specific range of activities the container is divided into a primary container with three internals. For the primary container a type A design is sufficient. Respectively the primary container including the internals represents a type B container. The

shell construction of containers was tested and then successfully used with the S7V-container of the KfK-transportation system for medium level waste. The cast lid will be screwed onto the primary container.

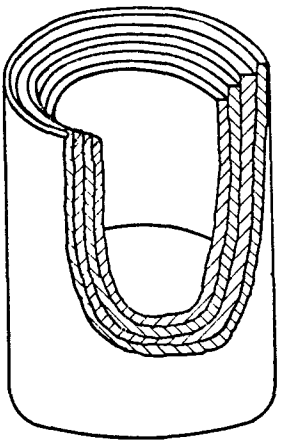

Fig. 3: Variable cast steel container of category 2 (12 to 18 tons)

The containers for the activity category 3, accomodate waste with only a small activity inventory. The container could be designed as a non-standardized package although the licensing authorities, such as the mining authority have indicated that they will approve only a type A container for the handling of wastes in mines.

A rectangle is used for the container geometry so that long slim components (piping) can be packed conveniently. It may be constructed to the existing design specifications of the ISO-containers.

4. Transport Volume

The net loaded mass to be filled into the individual containers can be determined from the filling factors evaluated empirically. From previously estimated mass value, the number of containers needed for each of the three activity categories has been calculated and listed in table 3.

Over a time span of 27 years during which 29 power stations mentioned above are to be decommissioned, a maximum number of 7000 containers per annum from mass value is calculated, whereby a maximum volume for final disposal of 23.000 m^3 per year will be necessary.

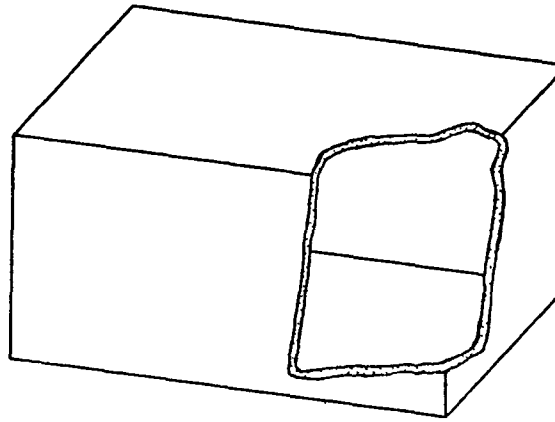

Fig. 4: Lost steel containers of category 3
(7 to 18 tons)

Waste categories	time			
	1 a	10 a	30 a	50 a
	Number of containers			
1	5500	2400	880	-
2	3100	2880	1950	850
3	78660	79080	79760	80460

Table 3: Number of containers after different times prior to dismantling

Question by M. S. T. Price, United Kingdom: I understand that your study was constrained by the dimensions of the shaft of the KONRAD mine. If that constraint were removed (i.e. for a country which may drill a new mine) what is likely to be the first constraint which will be met? Is it still likely to be the dimensions of the shaft on cost optimisation grounds? Have you carried out a general cost optimisation exercise for land burial of decommissioning wastes?

Answer: We have optimised only the dimensions of the container for capacity of the shaft "KONRAD". When we have to drill a new mine, we should take a higher capacity (40-50 tons). Later on we shall optimise costs for several capacities.

Question by J. D. Hart, United Kingdom: I appreciate that this was a study but were you seriously considering cutting up the primary circuit components and putting them into your tiny category 1 flask? Does it not lead to the conclusion that you should wait 50 years to avoid this predicament?

Answer: Today there are many possibilities of cutting big components after a short time, further there are other important reasons for a short time between reactor shut-down and start of decommissioning.

Question by H.-J. Obst, FR Germany: 1. Does fixed conditioning mean total encapsulation? If so, how thick? 2. What is the percentage of fixed conditioning for containers, in particular of category 3?

Answer: 1. Fixed conditioning is necessary to delay washout when flooding of the mine occurs. If it could be proved that the packaging forms a large enough barrier, fixed conditioning would not be necessary. The competent authority has the last word. 2. This depends on the type of waste, in the case of pipes about 5.5% fixed conditioning, concrete waste does not require this conditioning.

DEVELOPMENT OF A VERSATILE DEPARTMENT OF TRANSPORTATION SPECIFICATION 7A METAL WASTE CONTAINER FOR INTERIM AND TERMINAL STORAGE/BURIAL

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Beginning in 1970 the Atomic Energy Commission, now the Department of Energy (DOE), initiated a program of retrievable storage of defense waste contaminated with transuranic (TRU) material to a level of greater than 10nCi/g. These wastes were to be packaged in containers which could be retrieved intact after a 20 year period and transported to a terminal repository, the plan being that some type of terminal storage facility would be operational by 1990 or before. DOE contractors with their own burial sites, such as Oak Ridge, Savannah River, and the Hanford Operations, provided the area for this interim retrievable storage. Contractors without burial sites packaged and shipped to other locations. For example, Mound Facility and the Rocky Flats Plant send their TRU waste to the Idaho National Engineering Laboratory for retrievable storage.

When the task force began work on criteria for a terminal storage facility, specifically the Waste Isolation Pilot Plant (WIPP), it was known that there existed in the DOE complex a wide variety of TRU retrievable package modes. Mound Facility was given the assignment to provide information on current packaging systems and their compatibility with the WIPP criteria. This task initially involved the generation of input from the other waste generators from survey reports and telephone contacts, followed up by site visits.

It was determined that more than 16 different packaging configurations were being used for interim storage. During the compilation and analysis of this data from the questionnaires and site visits, some important pieces of information were developed. For example, for the period of time surveyed over seventy percent of the low level TRU waste was packaged in a box geometry. It was also noted that the movement of this waste was highly dependent upon the ATMX railcar system. Another interesting piece of data was the relatively large volume of equipment waste (decommissioned gloveboxes, process piping, building rubble) that was generated exceeding the next waste form category by more than 100 percent.

The next phase of the study involved taking all of the above input and developing some basic design criteria for a standardized low level TRU waste container. Important topics requiring consideration included

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*Mound Facility is operated by Monsanto Research Corporation for the U.S. Department of Energy under Contract No. DE-AC04-76-DP0-0053.

- Current and proposed waste processing systems.
- Waste forms to be packaged.
- Material handling systems.
- Current modes of transportation.
- Safety.
- Strength.
- Cost Effectiveness.

In addition, the proposed criteria for the WIPP and existing interim storage criteria played a major role including such items as 20-year retrievability, combustibility, and storage efficiency. Appropriate Department of Transportation criteria were, of course, included. In reviewing all of the input to the design criteria, analysis of data led to a durable, metal container of rectangular shape. It had to be capable of being handled by conventional equipment such as fork lifts, capable of accepting a variety of waste forms and be cost effective. Preliminary designs were drawn up for a family of rectangular metal containers.

Several of these packages were studied in terms of their packing efficiency in two current transport systems used in the U.S.A.: the Super Tiger, (a Type B overpack); and the ATMX railcar. From the family, a container which appeared most versatile was selected for detail design, fabrication and testing. The container size selected had nominal internal dimensions of 4 ft wide by 4 ft high by 7 ft long. This size was selected for a number of reasons. First it was capable of overpacking the 4 x 4 x 7 fiberglass reinforced plywood box used extensively for retrievable storage of TRU waste. The WIPP criteria had called for these boxes to be overpacked before going into terminal storage. Secondly, the metal container was sized such that it could be readily transported in the ATMX railcar and even double stacked, if required, in the 600 series ATMX. This first prototype was fabricated of 3/16 in. steel sheet stock and formed 3/16 in. angles. It was of welded construction and of a design that lent itself well to high volume production. The structural members were formed angles onto which the sheet stock was welded. Formed channels were used as stiffeners. The container was top loading or could be rolled on its side and loaded with a fork lift. The empty weight of the container was 1750 lbs. The material selected for construction was Corten A (A606) Type 4 alloy steel. This alloy was selected for its strength properties and outstanding resistance to corrosion as compared to mild steel. Corten has five to eight times the corrosion resistance of low carbon steel besides having a higher yield strength. The cost penalty over use of mild steel is about 10 percent.

Lanson Industries, located in Cullman, Alabama, was chosen as the vendor based upon extensive experience in package development and testing, primarily in the area of providing weapons containers for the Department of Defense. Their facility has equipment designed to produce containers of one of a kind or several hundred per day, with a capable staff and a strong quality assurance program.

Figure 1 shows the results of Mound/Lanson efforts on the first prototype.

After fabrication, the prototype was subjected to the DOT Specification 7A series of tests as listed in 49 CFR 173.398(b).

The results of these tests were as follows:

a) Water Spray - Exempt from this test.
b) Drop test - failed. See below.
c) Corner drop - Did not test as it was not anticipated that this container would be used for Fissile Class II shipments.
d) Penetration test - Passed. No noticeable effect.
e) Compression Test - 84,000 lb. was applied for 24 hr with no deformation or damage.
f) Environmental Tests
    1) Heat - 130°F - Passed.
    2) Cold - -40°F - Passed.
    3) Reduced pressure - 7.3 psia - Passed.
    4) Vibration - Passed.

For the 4 ft drop test the container was loaded with 14,908 lbs. of wet sand to give a total gross weight of 16,658 lbs. The container was dropped and passed. However, it was determined that the angle of impact did not satisfy the DOT criteria calling for "striking the surface in a position for which maximum damage is expected." Therefore, a second drop was done on another corner of the same container and a flow path near a corner did indeed develop as evidenced by water dripping out from the wet sand. This failure is shown in Figure 2.

As a result of this failure, the development, fabrication and testing of a second prototype was initiated. Some changes were made to increase strength at the corners. These included:

a) The use of 3/8 in. structural angles and channels rather than the formed 3/16 in. angles.
b) Placement of some 1/2 in. steel bolts which also assisted in positioning the lid for the welding operation.

In addition, the flanged sealing surface for the lid was turned in instead of out and the side and end stiffeners were moved to the inside of the box. These changes are illustrated in the line drawing of Figure 3.

The result was a strong high quality package. The heavier structural members increased the container weight to 2002 lb empty. This container was then subjected to the DOT Specification 7A tests.

The entire series of Specification 7A tests was easily passed including the 4 ft drop test at a gross weight of 18,407 lb and the 1 ft corner drops. The orientation for the 4 ft drop test and resultant damage is shown in Figures 4 and 5. The only visible damage after the drops was to appurtenances such as fork wells, lifting rings or roll bars. In addition, reduced pressure tests were successfully performed after the drop tests at both 7.3 psi and 11.1 psi. The 11.1 psi requirement is part of the proposed DOT rulemaking designed to bring these regulations in closer agreement with the IAEA regulations. Subsequent to the tests, a SARP on this container has been prepared. The box is shown in Figure 6 after going through the entire test series.

As noted earlier, this container had been sized to overpack the 4 ft x 4 ft x 7 ft fiberglass coated plywood (FRP) boxes used extensively for packaging TRU defense waste, since these combustible containers required overpacking in the

WIPP criteria. Although WIPP may be dying or dead, other terminal storage criteria will, no doubt, be similar. In addition this is well suited for use as a primary container, particularly for heavy items resulting from decommissioning and decontamination operations (D&D) such as gloveboxes and related equipment. A number of DOE contractors are interested in the container for D&D and as a direct replacement for the FRP boxes. The shape lends itself well to both efficient storage and transport utilizing existing transport systems.

Another benefit of this design involves the modular concept possibilities. For example, the NRC requirement in 10 CFR 71.42 for shipment of plutonium could be met by two nested containers.

Cost is an important factor. The simple design of this container (all welded sheet stock and angles) lends itself well to a mass production operation. In large quantities (several hundred), the metal boxes could be competitive with the FRP box of the 4 ft x 4 ft x 7 ft size. They are competitive when one evaluates the cost per lb for disposal of D&D waste. Mound Facility has developed some larger FRP boxes for D&D wastes with containers up to 9 ft x 6 ft x 11 ft. In these sizes, the metal container is definitely less expensive. Mound is, in fact, currently developing and certifying a replacement metal container for these oversize FRP boxes.

Not only can defense waste be packaged, but commercial applications for this container also exist as evidenced by numerous contacts from this sector.

The metal container development program will continue as additional needs and new requirements are developed. The next phase of the project involves the design and testing of a more economical metal container suited for lighter weight loads.

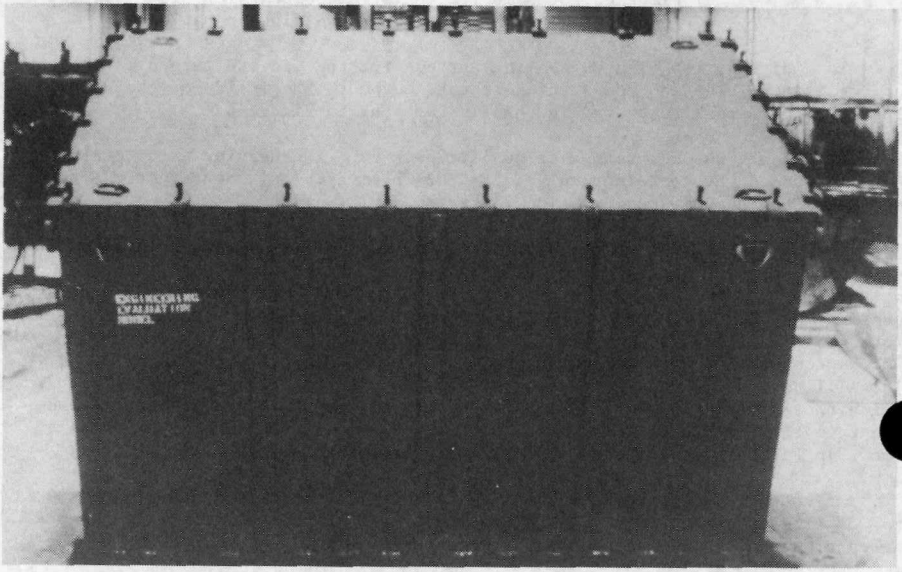

Figure 1

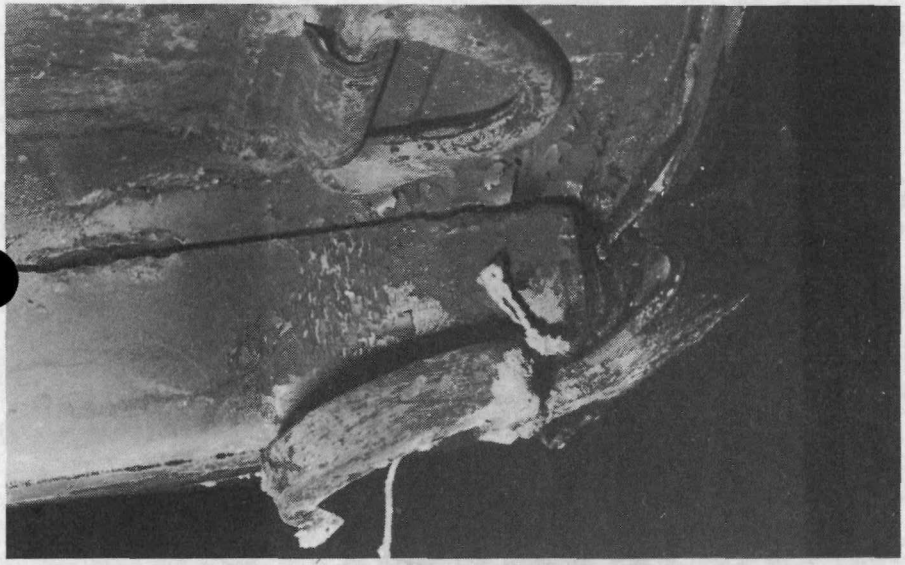

Figure 2

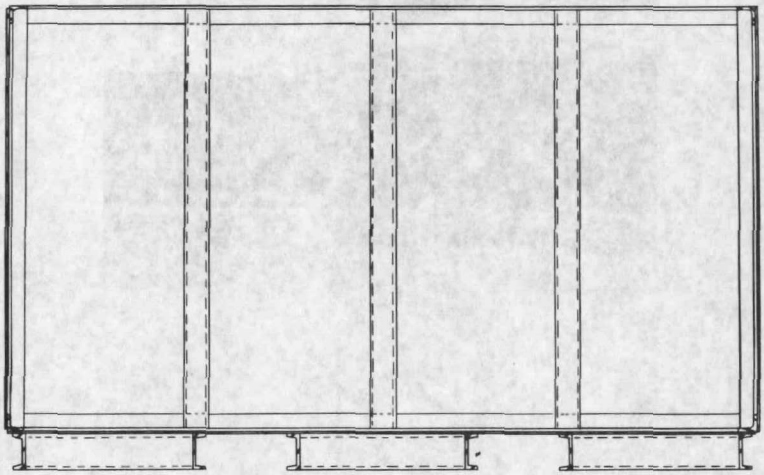

Figure 3

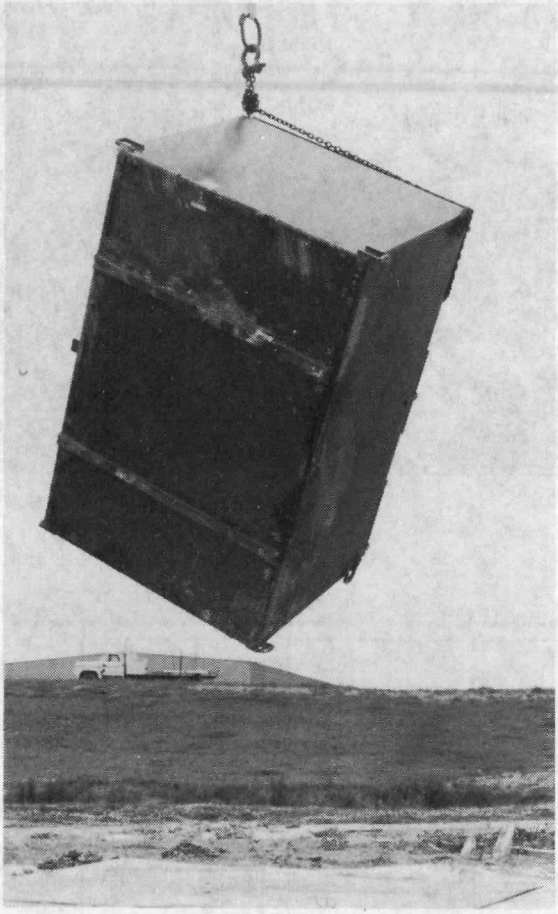

Figure 4

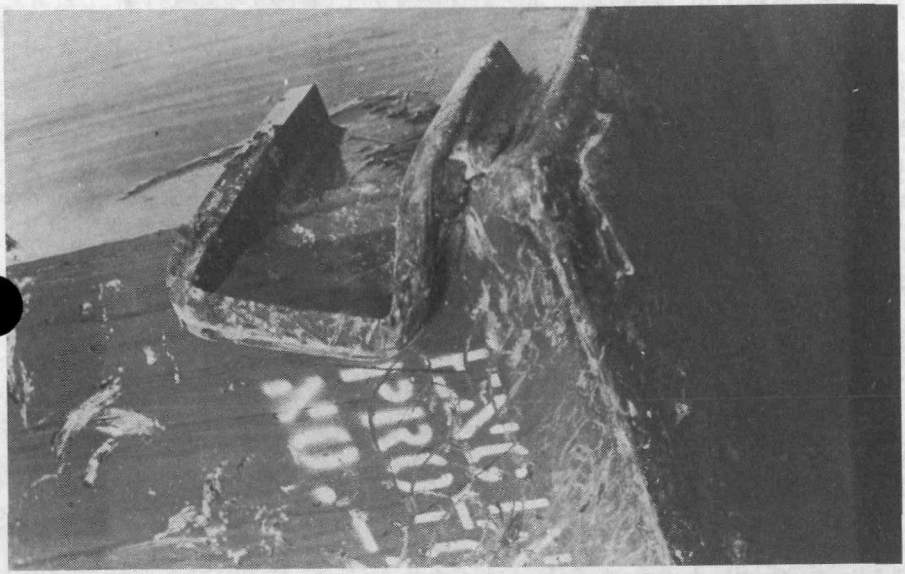

Figure 5

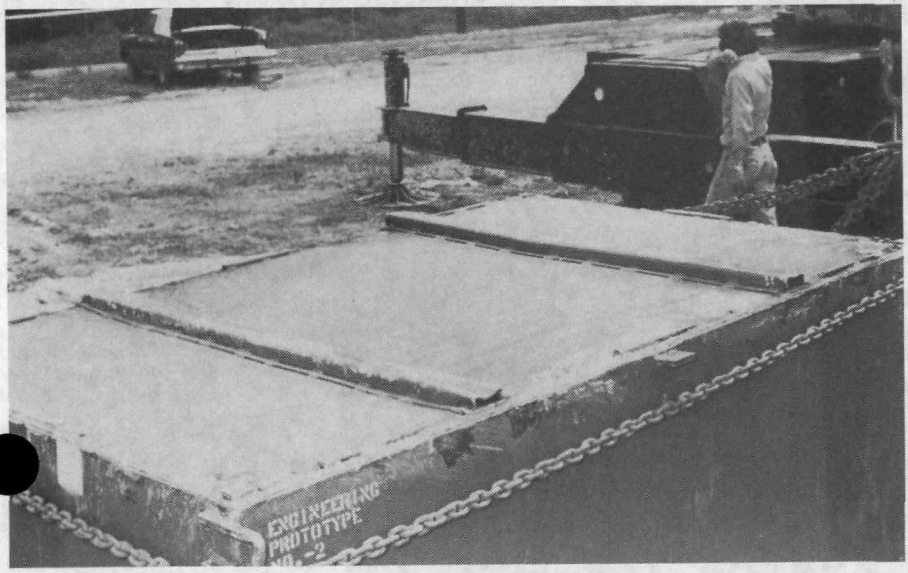

Figure 6

DISCUSSION

Question by M. L. Bell, USA: What is the expected production cost of the 4' x 4' x 7' box?

Answer: The cost of the original container (which failed) was expected to be approximately $ 1000 each. After the modifications, the second design will cost approximately $ 2000 each. On a cost per pound basis this is competitive with existing packagings; however we will be developing a container for lesser gross weights and a corresponding reduction in cost.

DEVELOPMENTAL TESTING OF PARTIALLY VOLATILE NEUTRON SHIELDS FOR HIGH-PERFORMANCE SHIPPING CASKS*

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INTRODUCTION

Many spent-fuel shipping casks in use throughout the world utilize water or ethylene glycol/water mixtures as the neutron shielding material. In either case the shield material is usually contained in a cavity near the outer surface of the the cask. As such, the likelihood of retaining the liquid neutron shield in a severe accident is low. For such casks, the designer usually is required by his certificating authority to assume in his safety analysis that the shield material is lost during an accident. The post-accident evaluation of such a package when used to transport spent light-water reactor fuel usually shows that the increase in radiation level external to the package does not exceed that which is allowed by regulation. For example, in the IAEA Safety Series No. 6, paragraphs 229 and 532, an increase in external dose from 10 mrem/h, 2 m from the external surface of the package to 1000 mrem/h, 1 m from the surface of the package is allowed following the "tests for demonstrating ability to withstand accident conditions in transport".

At least three problems associated with the use of volatile liquids as neutron shielding materials in shipping casks have been identified. First, for casks such as might be needed for the shipment of high-burnup, short-cooled spent fuel from liquid metal fast breeder reactors (LMFBR), the external dose rate following exposure to the accident-related tests may exceed that allowed by the regulations if all of the neutron shielding material is lost. Specifically, this occurs if the design of the cask has been optimized about a balanced shielding of both gamma rays and neutrons. Complete loss of the neutron shield following an accident requires that the designer add extra gamma shielding. This results in significant increases in cask weight and overall dimensions. Second, the use of volatile neutron shields may often complicate the packaging design and operation, since expansion volumes and pressure relief systems must be provided to prevent over-pressurization during normal use and exposure to fire environments. Third, irrespective of the material being carried, the loss of all the neutron shielding material in an accident results in a more difficult and potentially more hazardous accident recovery procedure.

A study was initiated in 1977 at Sandia National Laboratories (SNL) to evaluate the feasibility of fabricating neutron shields for shipping casks using commercially available materials which would retain at least part of their shielding capability following exposure to an 800°C, 30-minute thermal test. This paper describes the results of the first phase of this developmental program, and outlines the plans for the second phase of the program.

*Work was supported by US Department of Energy Under Contract DE-AC04-76DP00789.

DESIGN CONCEPT

The neutron shield concept investigated in this program utilizes interstitially located layers of a neutron shielding material and metallic heat transfer fins. Materials which can be preformed to specific shapes and inserted into neutron shield cavities, and materials which can be "injected" into the cavities and solidified in place were studied. Only materials which were commercially available in the United States were considered. In phase one of the program it has been assumed that the heat transfer fins could not allow a "neutron window" through the shield and therefore the design concept provided for four distinct bends in the fins as they pass through the shield area to prevent neutron streaming.

Other criteria for selecting candidate materials, in addition to the fabrication procedures described above, included: cost, neutron shielding effectiveness and thermal stability of the material in both long-term temperature environments up to 225 °C and short-term exposure to higher temperatures. Initially the literature was reviewed and potential suppliers contacted for information not available in their literature. This "screening" process was continued by selecting candidate materials and examining their thermal stability using thermo-gravimetric analysis (TGA). As a result, three candidate materials were selected for the first phase of a more detailed evaluation: (a) Boro-silicone rubber 236, a product of Reactor Experiments Inc.; (b) Boro-silicone rubber, BISCO NS-I, a product of Brand Industrial Services Inc. (BISCO); and (c) Borated beechwood, Permali JN, a product of The Permali International Group.

The problems associated with fabrication were specifically addressed early in the program. For simplified fabrication procedures, it was desirable to locate a suitable material which could be injected into existing cavities in the cask structure and have it solidify in-situ. This fabrication procedure allows the designer to easily provide a protective outer metal cover over the neutron shield, which would prevent significant contamination uptake during in-plant handling and could also serve as a barrier to oxygen during fire exposure. At the time of initial material selection, only the NS-I material could be injected into a cavity and cured in place. However, it now appears that the Boro-silicone 236 material can also be fabricated in place.

MATERIAL PROPERTIES EVALUATION

Various basic material properties, including chemical composition (important neutron shielding components only), mechanical behavior and thermal stability of the three candidate materials, were evaluated experimentally.

The chemical analysis of the as-received materials showed that the Boro-silicone rubber 236 contained 1.6 wt. percent B in the form of elemental boron, the NS-I contained 16.1 wt. percent in the form of B_4C, and the Permali JN 2.1 wt. percent in the form of B_2O_3. In addition, the Boro-silicone 236 contained 16 wt.percent aluminum oxide filler.

Measurements of the compressive properties of all three materials at a strain rate of 10^{-4} sec^{-1} over the temperature range -40°C to 200°C were made. The compressive strength of the boro-silicone 236 varied from approximately 3 MPa at -40°C, to as low as 1.5 MPa at 200°C, whereas the BISCO NS-1 material had a compressive strength which varied from 15 to 27 MPa at -20°C to as low as 4 MPa at 200°C. In contrast, the borated beechwood material had compressive strengths typically 10 to 40 times higher than the rubber based materials. Also, the energy absorbed during compressive straining to a total strain of 15 percent was determined.

It was shown that (a) the energy absorption capability of the silicone rubbers was essentially independent of test temperature, (b) the Boro-silicone 236 material will absorb approximately twice the energy that the BISCO NS-I material will absorb, and (3) aging for 1000 h at 150°C does not affect the energy absorption capability of either of the silicone rubber materials.

THERMAL PROPERTIES EVALUATION

The TGA studies indicated that the silicone rubber based materials were stable to temperatures of approximately 240°C, and lost mass on a nearly linear basis to a temperature of approximately 700°C. At this temperature, approximately 40 percent of the initial mass had been lost. Tests were run in both air and argon, and at "heating rates" from 1°C per minute to 10°C per minute. The results for Boro-silicone 236 were essentially independent of cover gas and TGA "heating rate". In contrast, the Permali JN material experienced small amounts of degradation at temperatures as low as 25°C, but the onset of significant degradation began at approximately 200°C. The degradation continued such that at temperatures exceeding 700 °C essentially all of the material had been volatilized.

A further indicator of the relative thermal stability of the candidate materials was obtained by determining the temperature to which the materials must be heated at a constant heating rate to achieve a pre-selected percent weight loss. Table I gives a comparison for these materials at a 5 percent weight loss. This table shows that, although BISCO NS-I appeared to be more stable than Boro-silicone 236, which in turn appeared to be more stable than Permali JN, the presence of the cover gas atmosphere was far more important for BISCO NS-I and for Permali JN than for Boro-silicone 236. This observation is germaine to the results of the fire exposure tests discussed later.

Table I. Temperature for Achieving 5 Percent Weight Loss in Constant Heating Rate (10°C/min) Evaluation of Candidate Neutron Shield Materials

	AIR ATMOSPHERE	ARGON ATMOSPHERE
CANDIDATE MATERIAL	TEMPERATURE (°C)	TEMPERATURE (°C)
BORO-SILICONE 236	279	274
BISCO NS-I	368	421
PERMALI JN	115	230

One final method to assess the expected behavior of these materials during normal use with spent-fuel shipping casks was explored in which samples were exposed to isothermal aging at various temperatures. Weight-loss measurements were made as a function of aging time. Selected samples were then chemically analyzed following exposure to the isothermal conditions. The weight loss as a function of temperature, after exposure for 2040 hours, is shown in Table II. These data show that the silicone-based materials are much more stable than the beechwood material. Again, the BISCO NS-I material appeared to be the most stable. However, additional data indicated that the weight losses for the Boro-silicone 236 and BISCO NS-I at the highest temperature had reached an asymptotic value, whereas weight loss was continuing

Table II. Weight Loss of Candidate Neutron Shielding Materials Following Exposure for 2040 Hours at Constant Temperature

TEMPERATURE (°C)	PERCENT WEIGHT LOSS Boro-Silicone Rubber Boro-silicone 236	BISCO NS-I	Borated Beechwood Permali JN
102	0.3	0.5	8.0
152	3.3	1.2	20.
202	8.5	2.3	46.
227	18.0	5.2	70.

Results of chemical analyses following long-term isothermal exposure are given in Table III. Although all chemical constituents were determined, hydrogen is addressed here since it is the important constituent for neutron shielding. These results indicate that, on the basis of percent loss of initial hydrogen, the ranking of materials suggested above is still valid. However, after exposure at high temperatures, the total hydrogen content of both silicone-based materials appear comparable. If the material had been confined in a closed cavity, the loss of mass probably would have been lower than that shown above.

Table III. Hydrogen Content of Candidate Neutron Shield Materials Following Exposure for 2040 Hours at Constant Temperature

TEMPERATUR (°C)	WEIGHT PERCENT OF HYDROGEN Boro-silicone Rubber Boro-silicone 236	BISCO NS-I	Borated Beechwood Permali JN
20	4.5	3.3	5.6
102	4.5	3.2	5.5
152	4.1	3.1	4.7
202	4.1	3.0	2.3
227	3.2	3.0	1.8

NEUTRON SHIELDING AND THERMAL TESTS

Slab sections with full-scale thickness (200 mm x 660 mm x 660 mm) and with copper segments simulating the heat transfer fins of the spent-fuel shipping cask were fabricated. Figure 1 shows a three dimensional sketch of the test sample as it was instrumented for the thermal (fire) test. These test samples were first tested to determine their neutron shielding capability, were exposed to a fire exceeding 800°C, 30-minute thermal test required by regulation, and were then tested again to determine the degradation in neutron shielding capability.

The neutron attenuation measurements were made by the Los Alamos Scientific Laboratory, Los Alamos, NM, USA, using a ^{252}Cf source. The thermal tests were performed in a natural gasfired furnace at the Portland Cement Association Research and Development Construction Technology Laboratories in Skokie, Illinois, USA.

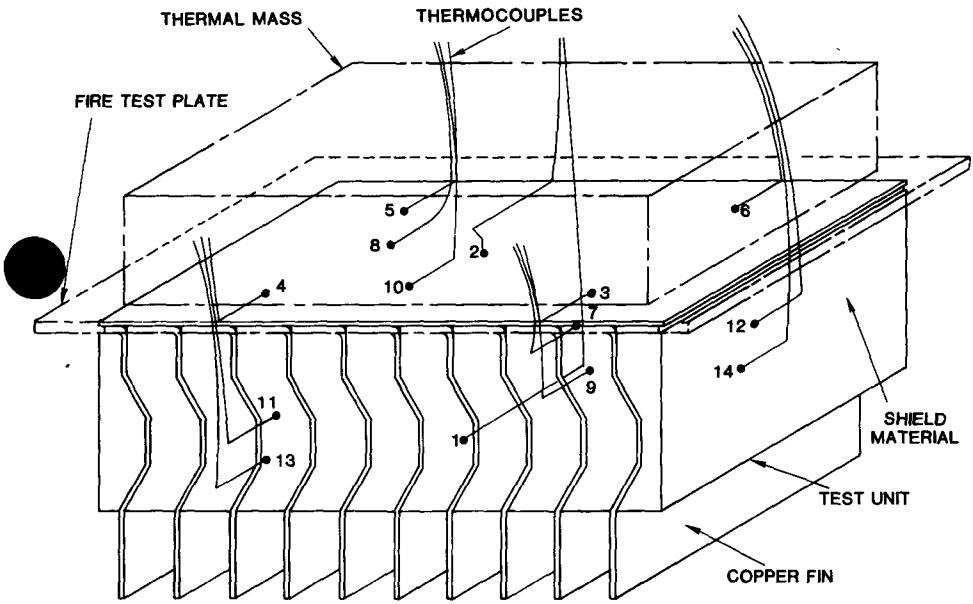


Figure 1. Schematic showing the neutron-shielding test samples as instrumented for thermal exposure tests. (Numbers designate thermocouple locations.)

The thermal tests, which involved measurements of heat penetration during fire exposure, were performed by mounting the samples in the furnace with a 130-mm-thick stainless steel thermal mass on the shield backface, simulating internal cask structure (see Figure 1). Initial preheating of the samples with small "pilot" burners for approximately 3 hours was used to simulate normal operating temperatures. The main burners were then ignited. The furnace required approximately 15 minutes to reach a temperature of 870°C. Fuel and air were adjusted to maintain this temperature for 30 minutes. Combustion of the samples was then allowed to proceed with the fire source extinguished. The temperature responses during the thermal test are summarized in Table IV.

The Boro-silicone 236 shield experienced the least heat gain in the sample self-extinguished within minutes following termination of exposure. The [redacted]mali JN shield experienced the greatest heat gain at the end of exposure, although it self-extinguished approximately 30 minutes following exposure. In contrast, the BISCO NS-I shield initially showed low heat gain at the backface. However, an exothermic reaction of the BISCO NS-I caused the temperature to increase above the desired 870°C to a maximum of 900°C at the end of the 30-minute exposure despite extinguishing the main burners 15 minutes into the test. The temperature continued to increase to 945°C at 67 minutes. The shield at this point was readily sustaining combustion and was "dripping" into the furnace. It was removed and

despite extinguishing the main burners 15 minutes into the test. The temperature continued to increase to 945°C at 67 minutes. The shield at this point was readily sustaining combustion and was "dripping" into the furnace. It was removed and the fire was extinguished with extensive water spray. Final weight losses for the test samples resulting from the fire exposure were: Boro-silicone 236, 15 percent; Permali JN, 40 percent; and Boro-silicone NS-I, 100 percent.

Table IV. Measured Temperatures (°C) of Neutron Shield Samples in Thermal Test

	END OF 800 °C, 1/2 HOUR EXPOSURE		MAXIMUM
	50 mm From Surface	Shield Backface	Shield Backface
BORO-SILICONE 236	244 °C	194 °C	208 °C at 53 minutes
BORO-SILICONE NS-I	748 °C	198 °C	299 °C at 67 minutes*
PERMALI JN	514 °C	304 °C	321 °C at 60 minutes

*BISCO NS-I sample removed from furnace, continued to burn

The measured neutron attenuation factors (the ratio of dose rate unattenuated and dose rate attenuated) preceding and following thermal test exposure are shown in Table V. Note that the pre-test attenuation for the Boro-silicone NS-I is lowest of the three, correlating with the hydrogen contents reported in Table III. The significant amount of the neutron shielding capability retained by the Boro-silicone 236 material was comparable to the pre-test value for BISCO NS-I. A moderate amount of the shielding capability was retained by the Permali JN material following exposure.

Table V. Neutron Attenuation Factors of 200-mm-Thick Slab Test Samples of Candidate Neutron Shielding Materials Using ^{252}Cf Source

	AS RECEIVED	AFTER EXPOSURE TO 800°C, 1/2 HOUR THERMAL TEST*
BORO-SILICONE 236	17	11 to 12.5
BORO-SILICONE NS-I	13	0
PERMALI JN	17	3.5 to 4.5

*Two sets of tests, each material

On the basis of these tests, a second series of tests was performed to evaluate the thermal behavior of the Boro-silicone 236 and the Permali JN in more severe thermal environments. The shield models were exposed to a 1000°C, 60-minute fire source. The Boro-silicone 236 was the only sample which retained a significant quantity of material following this exposure, with a post-test exposure attenuation factor of approximately 2.

FUTURE EFFORTS

In view of the success achieved in the first phase of testing with the Boro-silicone 236 material, a second phase test program is underway.

Test samples have been fabricated of steel, some having bent internal fins and others having straight-through fins. These samples are shown in the simplified engineering drawings of Figure 2. Small holes were left in the outer surface of the samples, three holes per cavity, and Boro-silicone 236 has been successfully injected through these holes and solidified in the cavities. Fusible nylon plugs have been inserted in the holes to provide weather and contamination protection for the neutron-shielding material. The test samples, in contrast to the sample shown in Figure 1, have small external heat rejection fins (finlets) 76 mm high, 55 mm wide and 6 mm thick.

The goals of the second phase effort are:

(1) to demonstrate that the material can be formed in place inside of fabricated cavities,

(2) to demonstrate that the fabricated shield will withstand long-term high temperature heat soak without significant degradation,

(3) to measure the effective thermal resistance of the neutron shield layer,

(4) to assess the effect of bent fins versus straight fins inside the neutron-shield cavities on neutron shielding capability,

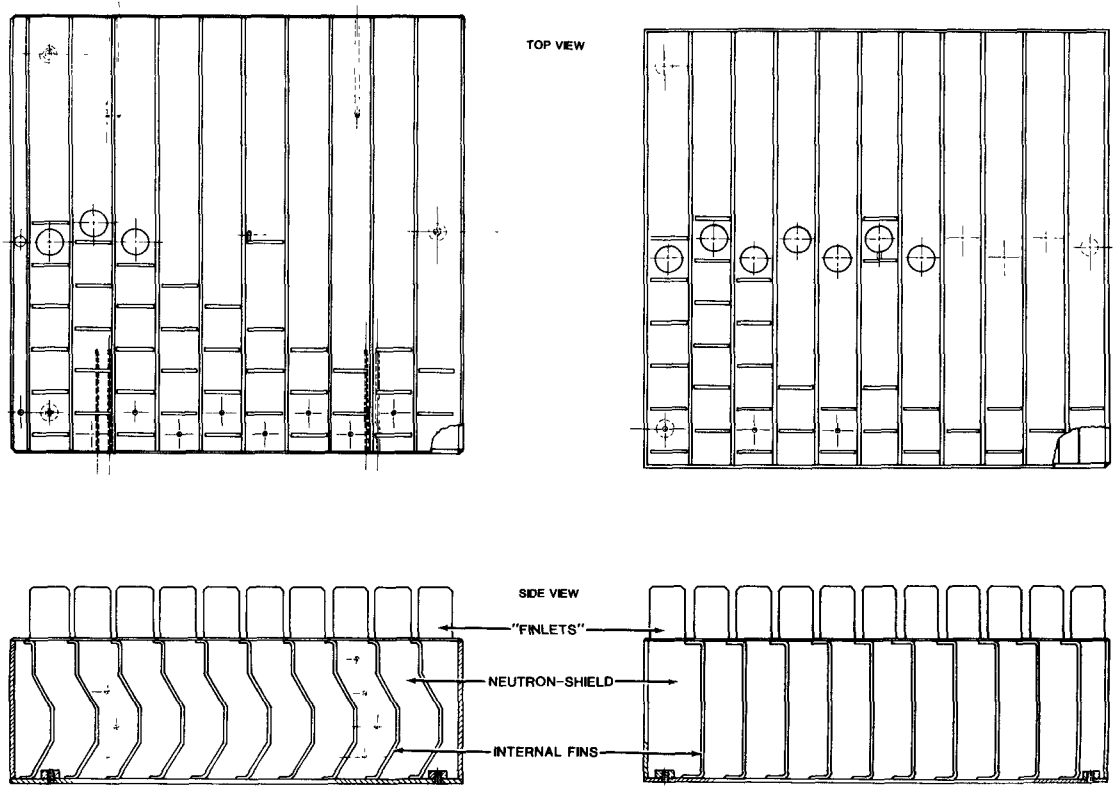
(5) to determine the feasibility of fabricating the internal fins from steel rather than copper,

(6) to assess the heat rejection performance of small "finlets" on the external surface of the shield as compared with large lateral or circumferential fins, and

(7) to assess the behavior of the material in an enclosed cavity when exposed to the thermal test environment.

CONCLUSION

The results of the phase one tests have demonstrated that the neutron-shielding concept described in this paper is a viable design option for spent fuel shipping casks. The tests have shown that the Boro-silicone 236 shield is superior to the other shield materials considered. Repeated TGA, aging and fire tests demonstrated the reliability of the data discussed in this report. A second phase of the test program is now being pursued where the Boro-silicone 236 is "injected" into all-steel slab sections, and cured in place. These samples will be tested to assess the adequacy of a simpler design concept than the one evaluated in the first phase.



ure 2. Engineering Sketches of the Phase Two Neutron-Shield Test S les

DISCUSSION

Question by M. S. T. Price, United Kingdom: 1. Why did you originally go for cranked fins? 2. Did you carry out any shielding calculations to estimate the differences in shielding efficiency between straight and cranked fins?

Answer: To prevent neutron streaming. We found that, by using copper bent fins, the gross neutron shielding was enhanced by about 10% due to inelastic scattering of the neutrons. 2. Calculations of that detail were not made.

SESSION V
US INSTITUTIONAL ISSUES

INSTITUTIONAL ISSUES AFFECTING TRANSPORTATION OF NUCLEAR MATERIALS*

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INTRODUCTION

The shipment of nuclear materials in the US nuclear waste management program will require the solution of both technical and institutional problems. The technical problems associated with the design and development of systems and packagings for shipments of nuclear materials can be solved. However, the institutional problems require different approaches, assessments and evaluations. An issue develops when there is disagreement or conflict within a single institution or with another institution over a particular problem, concern, action, legislation or regulation. Using this definition of an institutional issue, an outline of the program to address such issues, for nuclear materials, has been developed at the Transportation Technology Center (TTC). The TTC program consists of procedures to identify the issues, the methodology developed to assess and evaluate what effects are created by an issue and what, if anything, can be done about a particular issue. By outlining this program the relationships of the issues to the transport of nuclear materials will be clarified. The information developed in this program describing the issues, is made available to the US Department of Energy (DOE) and others through TTC report publication. In performing this function the TTC does not try to establish DOE policy but instead supplies objective information and recommendations on particular issues affecting transportation of nuclear materials. The aim in this paper is to select a few current problem areas and to provide statements of the issues in question form, indication of the institutions involved, discussions of the problems and why the issues exist and recommendations regarding particular issues where they have been developed. Among the issues outlined are the following: (1) perceived safety of transportation including risk assessments, public acceptance and related requirements; (2) legal and financial responsibilities; (3) emergency preparedness and response capabilities, (4) effectiveness of regulations and regulatory processes; (5) legislative and regulatory initiatives and preemption of state/local laws by federal regulations; (6) the rail transport option; (7) personnel training responsibilities and (8) uncertainties in federal policies. These issues are currently being studied and analyzed in the United States; however, they are so basic as to be considered international in nature. These issues may be cast into different contexts or have different emphasis in various countries, but the questions, problems and concerns relative to each issue have been, are or will be addressed in the world-wide arena of nuclear materials transportation.

TTC PROGRAM FOR INSTITUTIONAL ISSUES

The issues affecting transportation of nuclear materials usually develop from several sources: (1) new national policies (or changes in them), (2) actions intended to enhance safety or other aspects of transporting nuclear materials resulting from actual incidents or postulated conditions, (3) legislation and (4) regulations. An example of the issues originating from each one of these sources

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

will be explained. It must be emphasized that the issues often overlap and may have their origin in one or more of these sources. A new national policy in the United States to suspend reprocessing of spent light water reactor fuel affected nuclear institutions in the following ways: (1) industry -- utilities are required to store additional spent fuel at reactor sites or storage pools pending development of other disposal facilities, (2) shippers -- less material to transport initially, and older, less radioactive material to be transported in the future and (3) carriers -- use of present packages either as is, modified to carry additional fuel, or development of alternative packaging consistent with anticipated reductions in shielding requirements. As a result, institutions found it necessary to modify their business operations and adapt to the effects created by this policy. One result of this policy was to reduce the interest and commitment by industry to design, develop and manufacture packagings and shipping systems for nuclear materials because of uncertainties in the future needs for transporting materials and the associated revenues.

An example of a policy to enhance public safety by increasing the security of transporting nuclear materials is the physical protection requirements for shipments of spent power reactor fuel recently promulgated by the US Nuclear Regulatory Commission (NRC) in the Code of Federal Regulations (10 CFR 73.37). These regulations are based upon consideration of projected consequencces created by actions directed at shipping systems where the intent is to disperse radioactive materials to the atmosphere. These regulations require escorts, prenotification of shipments, specified routings, two-way communication between the transport vehicle and a central location and other provisions. The basic question (issue) is whether these requirements are necessary and do they enhance safety? Arguments for and against these requirements are advanced on subjective grounds because only limited data exist regarding the amounts and types of materials that could be released. Further, there are no definitive indications that such an action would ever occur. Some feel that the requirement for prenotification and specified routing offer greater access by those who may act malevolently toward a shipping system.

Every action affecting transportation of nuclear materials may not be a transportation issue but may instead be a means of addressing some other aspect of the nuclear waste management program. An example of this type of institutional issue is the Louisiana ban on shipments of certain nuclear materials within the state. The principal intent of this legislation is not aimed at transportation, but at denying the use of salt domes in that state as possible storage/disposal sites.

The US Department of Transportation (DOT) is in the process of adopting routing requirements for shipments of nuclear materials. These requirements would (when adopted) preempt many of the state and local laws which have been adopted in the last few years. This preemption may lead state and local governments to feel that they have lost a measure of control over how and what materials are moved through a particular jurisdiction. However, these routing requirements and accompanying procedures would provide for a uniform set of operating conditions for shipments over the entire country.

In terms of evaluating and assessing institutional issues, it is first necessary to define or describe the issue. Before this can be done, an understanding of the problems and concerns must be obtained. Based upon discussions with, and relevant information from, the institutions involved (testimony before legislative committees, public hearings, reports, documents, notices on regulatory processes, etc.), concerns and conflicts are identified. A variety of means can be used to develop interchanges with institutions: workshops, seminars, technical meetings, symposia and direct contact. With the problems, concerns and conflicts of a similar

nature identified, a particular issue can be defined. It is important to emphasize that each issue is unique and requires separate evaluation of the problems and conflicts involved. In each of the preceding examples, TTC has initiated studies to address the issues involved.

Once an issue is defined, it can be categorized by the type of solution anticipated. There are two basic categories of possible solutions: (1) those issues over which there is little control require adaptation to the effects or requirements created and (2) those issues in which the effects can be mitigated/resolved or alternative actions, policies or regulations developed. For both categories, it is necessary to understand a particular issue in sufficient depth so that its essential constituents are defined. Once an issue of the second type is understood, possible solutions can be developed. There can be more than one solution to a particular issue. Sometimes a technical solution can be developed. However, because an institutional issue is usually a dynamic situation in which new data emerge, policies change and regulations are promulgated, the solution may also be dynamic in nature. This last type of solution should be able to survive the continued interactions of the institutions involved. Therefore, rather than a closed form type of technical solution, an institutional issue is often open-ended. A continuing observation program must be developed in which changes in policies, regulations and cyclic impact of specific issues are evaluated. Then the developments observed must be met with alternative actions or policies.

INSTITUTIONAL ISSUES AFFECTING TRANSPORTATION OF NUCLEAR MATERIALS

Safety

The first issue is that of the perceived safety of transportation. The basic questions (issue) describing the conflict are (1) can transportation of nuclear materials be performed safely and (2) will the transportation processes become publicly acceptable? The risks associated with transportation can be quantified, but acceptance by the public cannot be obtained simply by analyses of available data or postulated conditions. There are many factors involved in the overall determination of safety including (1) shipment information (amounts, types of materials, frequencies, packagings and routes), (2) transport incidents (leaking containers, defective transport systems and rates thereof), (3) accidents involving shipping systems (frequencies and severities), (4) emergency preparedness and response capabilities, (5) availability of necessary information on transportation of nuclear materials and its dissemination to the public and the media, (6) the degree of enforcement of regulations, (7) compliance by industry to regulations, (8) design testing and licensing of packagings, (9) performance of packages in accident or other extreme conditions, (10) response of the contents in extreme conditions with emphasis on the formation of respirable particulates, (11) possible consequences to the public from the release of materials, (12) technical measures (risk assessment, hazards, consequences and environmental impact statements) and (13) perceptions of transportation hazards and how these perceptions are factored into the regulation and acceptance of transportation. It is questionable that transportation of nuclear materials can ever be proven to be completely safe to the public and regulatory bodies. Therefore, a pattern of consistent presentation of risk assessments, shipment information and analyses at appropriate forums needs to be developed. Only through continued education and a willingness to answer all questions on safety will evaluation of the risks and consequences of the transport of nuclear materials possibly develop a measure of acceptance by the public.

Legal/Financial

Legal and financial responsibilities are an integral part of the transportation of nuclear materials. The institutions involved are the industry (shippers, carriers, utilities and insurance companies) and the federal/state/local governments. The basic issue is defined from the various concerns and problems in the current system as follows: Is there an assured third party liability coverage in the event of a release of radioactive material during transport? There are a number of concerns associated with the legal and financial activities resulting from an accident involving nuclear materials, including: (1) Who is responsible for a spill of material? (2) Who is responsible for the consequences? (3) Will cleanup require extended periods of time? (4) Can the facilities adjacent to the route be used and can the route itself be used after an accident? and (5) Which shipments are covered under what conditions? Transportation of nuclear materials in the United States is covered by insurance in one of three ways: (1) under normal commerce of hazardous materials (e.g., radiopharmaceuticals), (2) under the omnibus provisions of Price-Anderson coverages (e.g., fuel cycle materials) and (3) through Federal Torts (e.g., DOE shipments). The ground rules governing applications of the Price-Anderson and Federal Tort system are written to define the action of the entity that is financially responsible for the consequences of an accident. However, determination of who is responsible and under what conditions are not clearly defined. The legal and financial responsibilities and how they are applied to and provide for the various transport systems are an issue in any country. There have been so few incidents in transporting nuclear materials where legal and financial responsibilities have been questioned that the possibilities of problems of incomplete legal definitions, complicated procedures for reimbursement and other complications developing certainly exists.

Emergency Preparedness

Another example of systems that are rarely exercised are emergency preparedness and response procedures. As a result, these systems have not been thoroughly evaluated under actual emergency conditions. Thus, the major question associated with this phase of transportation is really one of assuring capability. The issue is whether emergency preparedness (plans, equipment and training) and response procedures (personnel, notification, detection capabilities) will function and perform in an effective and competent manner in every emergency involving the transportation of nuclear material transportation. The plans are tested on varying schedules and the involved agencies receive training and exercise their functions. However, because of the infrequent use of the systems, it has been observed that the actual implementation of plans may be different (in varying degrees) from the original plans. While there is a distinct correlation between plans and the results of their actual implementation in all forms of emergency conditions (disaster response), emergency response plans should be designed to accomodate malfunctions and contain procedures to minimize their effect. Jurisdictions of the agencies and institutions involved should be clearly defined because there will likely be some critical time lost in an emergency in determining who is in charge and what actions should be performed.

Regulatory Efficiency

The real value of regulations is to provide for safe day-to-day operation of transportation systems for nuclear materials. The institutional issue affecting regulatory agencies, shippers and carriers is defined as follows: Are regulations reasonable, understandable and enforceable? In terms of regulatory processes, a regulation is "reasonable" if it provides the necessary protection for the public, provides procedures for compliance by industry with appropriate considerations for cost/benefit analyses, is capable of being enforced and cannot be

misinterpreted or conflict with existing regulations. An example of legislation/regulation which initially appears to offer significant local control with high political leverage is a ban on shipments of certain materials through communities. Unfortunately, some of these bans have been enacted, but have provided no capability for local officials to stop transport vehicles and inspect cargoes, thus these bans are probably not enforceable.

Preemption

Legislative and regulatory initiatives are increasing at federal, state and local levels with particular emphasis on the latter two. Directly affected by the actions of these goverment institutions and capable of having significant interactions with them are industry, the media and the public. The institutional issue can be stated in two questions: (1) Do state and local governments have the right and power to limit the types of shipments of nuclear materials or ban shipments of some or all types of materials through their jurisdictions? and (2) Can the Federal government through its regulatory agencies preempt laws and regulations developed by elected local officials concerning transportation of nuclear materials? The residents of local communities have become increasingly skeptical of federal government procedures and regulations and it becomes more difficult for them to be convinced that people and agencies outside their community have acted responsibly to minimize hazard to their community. Currently the DOT is in the process of developing routing requirements for shipments of certain types of nuclear materials (Docket HM-164). The major disagreement between the states/local governments, the public and DOT is that the states/local governments are requesting prenotification of certain shipments of nuclear materials through their communities. This advanced notification is not part of the DOT plan. The use of prenotification as one of the procedures for contesting preemption has developed.

Rail-Option

Another example of an institutional issue is the question of rail transport of spent LWR fuel in the US. There have been a number of hearings at various levels of government and considerable litigation involving the railroads. They claim that Type B packagings do not offer sufficient protection from release of radioactive materials in an accident and, therefore, require special trains (low speed, few cars and sole use of right-of-way) to move spent fuel. The contention by the railroads is that rapid reopening of a transport route may not be possible following an emergency involving nuclear materials. As a result, the issue affecting the utilities, the DOE, regulatory agencies and the railroads, can be expressed in two questions: (1) Are the emergency response plans and procedures currently in place adequate to permit rapid reopening of a route? and (2) Would financial considerations of generating sufficient revenues and having adequate insurance coverages interest the railroads in shipping nuclear materials? Since only part of the litigation has been settled and many disagreements still exist, these questions will have to be answered before the rail option can be completely examined.

[redacted]w Training

Training of operational personnel for carriers is a complicated task, particularly with the additions of the physical protection requirements for shipments of certain materials. The question (issue) involved is what type of training and certification are needed and who should bear the cost of such a program? Because the requirements for training are viewed as initiated by the government, industry feels that federal/state agencies should bear the costs. Government agencies feel that it is in the best interest of industry to provide training for its

personnel. Another aspect of this issue is the liabilities involved as a function of training and certification as well as who the employer would be.

Changes in Policy

Uncertainties at the federal level of government regarding clearly defined policies on waste storage and disposal exist because final plans have not been developed. As a result, the industry in unwilling yet to commit capital and resources to manufacture shipping containers. The institutional issue involved is whether decisions on storage and disposal of nuclear materials will be made in time for the industry to respond and still meet the requirements of the shippers and carriers. Decisions are needed soon on the type, kinds and locations of storage and disposal facilities so that industry can respond in a timely manner.

CONCLUSION

The institutional issues affecting transportation of nuclear materials in the United States represents significant barriers to meeting future needs in the transport of radioactive waste materials to their ultimate repository. While technological problems which must be overcome to perform such movements seem to be within the state-of-the-art, the timely resolution of these institutional issues seems less assured. However, the definition of these issues, as attempted in this paper, together with systematic analysis of cause and possible solutions are the essential elements of the TTC's Institutional Issues Program.

REFERENCES

1. R. M. Jefferson, ed., Program Strategy Document, SAND80-0784, TTC-0080, Sandia National Laboratories, Albuquerque, NM, April 1980.

2. T. L. Montgomery and D. J. Rose, "Some Institutional Problems of the U.S. Nuclear Industry," Technology Review (March/April 1979) pp 53-62.

3. Docket HM-164, 49 CFR Parts 173 and 177, Federal Register 45(22), January 31, 1980.

4. Physical Protection of Shipments of Irradiated Reactor Fuel, NUREG-0561, U.S. Nuclear Regulatory Commission, Washington, DC, June 1979.

5. Report to the President...on Nuclear Waste Management, Draft, Interagency Review Group, October 1978.

6. "Hazardous Materials Transportation, U.S. National Strategy Policy to be Defined at February 1981 Meeting," Traffic Management, August 1980.

7. B. S. James and S. P. Rochereau, Uncertainty and Capital Formation in the U.S. Spent Fuel Transportation Industry, IEAL-124-A, TTC-0033, International Energy Associates, Ltd., Washington, DC, March 1980 (to be published).

8. A. H. Schilling et al., Emergency Response in Transportation of Radioactive Materials. An Evaluation Methodology, B-HARC-311-044, TTC-0045, Battelle Memorial Institute, Human Affairs Research Centers, Seattle, WA, November 1979 (to be published).

9. F. A. Morris, Federal Preemption of State and Local Nuclear Transportation Regulation, B-HARC-311-043, TTC-0046, Battelle Memorial Institute, Human Affairs Research Centers, Seattle, WA, November 1979 (to be published).

10. L. McGinty and G. Atherley, "Acceptability versus Democracy," New Scientist (May 17, 1977) pp 323-325.

11. J. A. Herbert et al., Nontechnical Issues in Waste Management: Ethical, Institutional, and Political Concerns, PNL-2400, Battelle Memorial Institute, Human Affairs Research Centers, Seattle, WA, May 1978.

12. "Government Vetoes Tighter Controls on Rail Shipments of Nuclear Wastes," Congressional Quarterly (April 15, 1978) pp 880-883.

13. Ten Most Critical Issues in Hazardous Materials Transportation, ISS N0097-8515, No. 219, Transportation Research Board, National Academy of Sciences, Washington, DC, July 1980.

DISCUSSION

Question by G. L. Dooley, Canada: 1. Am I right in understanding that the United States Department of Transport is actively working on regulations to preempt state and local regulations? 2. What is your best estimate of a time frame before this goal will be realized?

Answer: Yes, through Docket HM-164 and the Hazardous Materials Transportation Act in terms of advancing a legal basis for preemption and actually initiating preemptive actions. 2. There can be only estimates or guesses of time frames because many of these regulations will require judicial decisions before they are final. An institutional issue is an open-ended problem in many cases and the solution is too.

FEDERAL LEGAL CONSTRAINTS ON STATE AND LOCAL REGULATION OF RADIOACTIVE MATERIALS TRANSPORTATION*

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INTRODUCTION

Through the Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT), the federal government of the United States regulates such aspects of radioactive materials transportation as physical security, packaging, vehicle safety and placarding. Within the past decade, state and local governments have also become involved in the regulation of radioactive materials transportation. This state and local involvement gives rise to important policy questions: (1) To what extent should the federal government regulate the transportation of radioactive materials uniformly throughout the country? (2) To what extent is state and local regulation appropriate? At the same time, state and local involvement raises the legal question: To what extent is state and local regulation permissible under federal law? This paper addresses this legal question. It summarizes the relevant legal doctrines (invalidation under the commerce clause and preemption), places these doctrines in the context of the federal regulatory framework and reaches conclusions about what forms of state and local regulation may be subject to possible preemptive initiatives and what regulations are unlikely candidates for federal actions. This paper also discusses an example of a preemptive initiative and a federal action. The initiative is contained in DOT's proposed rule on Highway Routing of Radioactive Materials (Docket No. HM-164). DOT's first general preemptive action under the Hazardous Materials Transportation Act is described with respect to decisions on Rhode Island's regulations regarding transportation of liquified natural and petroleum gases.

THE FEDERAL LEGAL DOCTRINES: INVALIDATION UNDER THE COMMERCE CLAUSE AND PREEMPTION

The commerce clause of the United States Constitution gives Congress the power "to regulate commerce...among the several states...." This affirmative grant of power has been judicially interpreted to limit state and local interference with interstate commerce. (In this paper, the term state regulations, actions or initiatives includes corresponding activities of local government.) Under current doctrine, state regulation is valid under the commerce clause only if the regulation is rationally related to a legitimate end and the burden imposed

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by the regulation on interstate commerce is outweighed by the state's interest in enforcing the regulation. In short, state regulation that unreasonably burdens interstate commerce is invalid. State regulation can be invalid under the commerce clause whether or not the area or activity is also regulated under federal law if states erect barriers to interstate commerce.

When there is federal regulation in the area, the doctrine of preemption comes into play. Preemption derives from the supremacy clause of the United States Constitution, according to which validly enacted federal law prevails over contradictory state law. Thus, preemption analysis requires a two-stage inquiry: first, to determine whether the federal law is valid; and second, if the federal law is valid, to determine whether state law interferes with the federal design. Because federal regulation of radioactive materials transportation is unquestionably valid, it is the second stage that is critical. In determining whether state law interferes with federal law, three doctrines are relevant: express occupation of the field, implied occupation of the field and the principle of conflict. Under the doctrine of express occupation of the field, federal law displaces all regulation in a particular area when Congress expressly manifests an intent to do so or has already enacted laws in the area. Under the doctrine of implied occupation of the field, federal law displaces all state regulation in a particular area when Congress manifests an implied intent to do so. Indicators of such implied intent are the pervasivenes of federal regulation in the field and a dominance of federal over state interests in the field, including the federal interest in uniformity. When Congress has clearly not intended to occupy a field to the exclusion of the states, federal law still preempts particular state laws that conflict with the federal scheme. Such conflict may be either "actual" or "potential." Actual conflict arises when compliance with both federal and state law is physically impossible. Potential conflict arises when state law stands "as an obstacle to the accomplishment and execution of the full purposes and objectives of Congress."

THE FEDERAL REGULATORY FRAMEWORK

As indicated above, NRC and DOT are the principal federal regulators of radioactive materials transportation. Pursuant to the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 (collectively, the AEA), NRC imposes physical security requirements on the transportation of special nuclear material and sets packaging standards for certain radioactive materials. Pursuant to the Hazardous Materials Transportation Act (HMTA), DOT sets packaging standards for certain other radioactive materials, in accordance with a memorandum of understanding between NRC and DOT. In addition, DOT develops standards for the classification of materials; for the external radiation fields, labeling and marking of packages and vehicles; for carrier equipment; for carrier personnel qualifications; for loading, handling and storage procedures; for nonsafeguards-related special transport controls and for all other safety standards not developed by NRC. The Environmental Protection Agency (EPA) may also have a limited role by virtue of its authority to regulate radioactive air emissions pursuant to the Clean Air Act. The Resource Conservation and Recovery Act of 1976 (RCRA) also has some limited relevance. While RCRA does not apply to source, by-product or special nuclear materials, states have used it as a model for the regulation of hazardous waste, including nuclear waste in terms of transportation, storage and disposal.

Potentially, either NRC regulation under the AEA or DOT regulation under the HMTA could preempt state regulation of radioactive materials transportation. The AEA does not explicitly address the question of preemption. As a consequence, the AEA's preemptive effect is solely a matter of judicial precedent. Significantly,

every federal court that has reviewed the issue has concluded that Congress has occupied the field of nuclear energy regulation. Unfortunately, however, the breadth of this field remains unclear. Accordingly, the AEA's preemptive effect on particular types of state regulation must hinge on a judgment about whether that regulation is within the field preempted. In contrast, the HMTA does explicitly address the question of preemption. The HMTA preempts any state or local requirement that is inconsistent with a requirement in the Act or a regulation promulgated thereunder, unless upon application the Secretary of Transportation determines that the state or local requirement affords at least as much protection as the corresponding federal requirement and does not unreasonably burden commerce.

APPLICATION TO PARTICULAR TYPES OF STATE REGULATION

In summary, federal law may invalidate state regulation either because the state regulation unreasonably burdens interstate commerce under the commerce clause or because the state regulation is preempted. State regulation may be preempted by the AEA because it is within the field occupied or because it conflicts with a particular federal requirement imposed under the AEA. State regulation may also be preempted by the HMTA according to the statutory test requiring inconsistent regulations to afford protection equal to that of the federal regulation and not to unreasonably burden interstate commerce. Under these tests, some types of state regulation are likely to be invalid, some are of uncertain validity, and some may be valid.

State and Local Regulation That is Likely to be Invalid

One type of state regulation that is likely to be invalid is that restricting or prohibiting shipments of certain materials. These requirements either limit shipments to specifically authorized routes or require shippers or carriers to obtain permits that impose routing restrictions. Such restrictions are probably invalid under the commerce clause because they interfere with radioactive materials transportation and thus unreasonably burden interstate commerce. Such restrictions on the shipment of special nuclear materials and spent fuel are also likely to be preempted by the AEA, which has occupied the field of physical security, of which routing restrictions are a part. Finally, as discussed below, DOT would expressly preempt most such restrictions through its proposed rule on highway routing of radioactive materials.

Another type of state regulation that may be invalid is the imposition of personnel qualifications on carriers. NRC has established requirements for escorts and guards. DOT has established personnel qualifications. Together, these requirements probably preempt most state and local requirements.

Federal law also probably preempts the imposition of packaging requirements by states. The AEA and the HMTA probably occupy this field and thus preempt state requirements. In addition, a diversity of state requirements would probably unreasonably burden commerce, rendering them invalid under the commerce clause or preempted under the HMTA's statutory standard. Similar conclusions apply to state requirements for labeling, handling, storage in transit, loading, unloading and packing or unpacking procedures.

State and Local Regulation of Uncertain Validity

Efforts by states to require registration, assess fees, issue permits and define licensing requirements (other than those related to packaging) are not within the fields occupied by the AEA or the HMTA and thus, for commercial shipments, would not be preempted. However, if unduly burdensome as determined case

by case or under existing precedents, such requirements could be invalid under the commerce clause.

Financial liability is within the area occupied by the AEA. State regulation of a type covered by Price-Anderson provisions would be preempted. However, other types of state financial liability requirements may be valid, unless they impose unreasonable burdens on interstate commerce.

State requirements in the area of monitoring, surveillance and inspection can probably be treated as adjuncts to the types of regulation whose compliance they are designed to ensure. If state substantive requirements are valid, so are the corresponding compliance and enforcement provisions, and vice versa.

State and Local Regulation That is Probably Valid

Neither NRC nor DOT presently require advance notification of radioactive materials shipments, although NRC receives such notification in connection with requests for route approvals for spent fuel. Accordingly, such requirements are probably not preempted. For certain materials and transportation systems, state requirements for advance notification could be invalid under the commerce clause. For example, requiring such notification for shipment of a radiopharmaceutical with a short half life imposes a burden incommensurate with the state's interest in enforcing the regulation. In general, however, advance notification requirements are probably valid. (See also the discussion of this issue below, as it is affected by the DOT rulemaking on routing.)

Accident notification and emergency response requirements are almost certainly valid because they serve federal purposes.

PREEMPTION AND HM-164

The DOT's Hazardous Materials Docket Number HM-164 proposes additions to 49 CFR parts 173 and 177. The major aspect under consideration in this paper is that portion of the docket which advances a legal basis for DOT inconsistency rulings and subsequent preemption of state laws restricting and/or banning the transport of nuclear materials. In addition, the reactions of state jurisdiction to possible preemptive initiatives are discussed. The origin of this federal effort to preempt state regulations was in response to the regulations and procedures developed by New York City's Bureau of Radiation Control (BRC) in January 1976, which created a two-level system of restrictions on transporting nuclear materials. Low level radioactive materials including most medical shipments were allowed entry into and passage through New York City along specified routes but without advanced notification. However, shipments of high level radioactive materials (e.g., spent reactor fuel and enriched uranium) were required to obtain a certificate of emergency transport from the BRC which, in effect, created a ban on these shipments because the certificates were not issued.

Associated Universities (AU) operates a reactor which requires fuel to be discharged and, because of geographical constraints, shorter transport distances over land must be through New York City. AU requested that DOT determine whether this ban might be preempted under the provisions of the Hazardous Materials Transportation Act (HMTA). DOT concluded that the HMTA could preempt local requirements, in general, but, because highway routing had not been included within HMTA, the regulations developed by New York City's BRC could not be preempted by HMTA requirements. DOT reached this decision internally and the ban was not challenged in the courts. With the enactment of this BRC emergency transport certificate (ban), numerous state and local jurisdictions have passed similar ordinances

which restrict in various ways the transport of radioactive materials.

In August 1978, DOT opened Docket HM-164 and requested public comment to the advance notice of rule-making intent. DOT received more than 550 written comments and subsequently released a draft of the proposed rule in January 1980, under the auspices of the Materials Transportation Bureau (MTB). This proposed rule disallows state (and local) jurisdictions from establishing prohibitive restrictions on the transport of nuclear materials by declaring those that do so to be inconsistent with the rule. The rule develops guidelines for routing which is in agreement with federally perceived health and safety considerations and yet allows some state participation in the selection of preferred routes.

In paragraph 177.825 of the proposed rule it is stated:

State and local requirements which apply to any person because that person transports radioactive materials are inconsistent with this subchapter if they have any of the following effects:

(1) Completely prohibiting travel between any two points serviced by highway;

(2) Prohibiting the use of an Interstate highway, including prohibition of travel based on time of day, without designation of an equivalent preferred highway as a substitute in accordance with the provisions of this section;

(3) Requiring use of a preferred highway except in accordance with the provision of this section;

(4) Requiring prenotification escort requirements, except as established under paragraph (b) (4) of this section (NRC or equivalent requirements for spent reactor fuel shipments), or

(5) Requiring special personnel or equipment.

The response of many state jurisdictions is to oppose preemption of their regulations. Local elected officials feel they have a legal right to enact laws to insure the health and safety of their constituents regardless of how restrictive these laws are or the fact that transportation may be included. As a result, there has been an escalation of the preemption arguments.

South Carolina has also challenged the exemptions under which the Department of Energy (DOE) ships nuclear materials. DOE shipments are not required to meet NRC regulations, but DOE has stated that its procedures provide for equivalent protection for the public.

State jurisdictions are also stressing their desire to be prenotified of the shipments of nuclear materials. In the case of radiopharmaceuticals, the prenotification requirements would be considered a burden to commerce. For higher level sources, prenotification of shipment and specific routing could enhance the possibility of malevolent acts and diversion of the shipment. However, because state jurisdictions feel they have lost a measure of control they feel the need to prepare emergency response capabilities in advance of shipment. DOT's Hazardous Materials Branch has proposed to post-notify state agencies of shipments which have occurred to construct a data base on shipments. Some form of notification of the shipment of nuclear materials may be a likely future requirement for transport of nuclear materials.

DOT AND PREEMPTION OF RHODE ISLAND REGULATIONS

In January 1980, DOT announced its first general pronouncement applying the preemption section of HMTA, and the corresponding procedures in 49 CFR Part 107. DOT found that five substantive Rhode Island state requirements were inconsistent with HMTA and its regulations and thus declared the Rhode Island regulation to be preempted. This ruling was a result of an application filed to DOT by the National Tank Truck Carriers (a for-hire bulk motor carrier trade association), regarding transport of liquified natural and petroleum gases. An appeal filed by Rhode Island regarding the preliminary injunction was rejected by the First Circuit Court of Appeals in Boston.

The Rhode Island requirements consisted of eight imposed rules: (1) a load-specific permit which required each specifically identified truck to carry a permit along a predetermined route, (2) a rush hour travel curfew between the hours of 7 and 9 a.m. and 4 to 6 p.m., (3) immediate telephone and follow-up written accident reports, (4) required headlights to be turned on during all transport operations, (5) inspection at sites where loading and unloading occurs, (6) two-way radio communications, (7) illuminated sign on the rear bumper indicating KEEP BACK 500 FEET and (8) frangible locks for valves. DOT found that the following five requirements were inconsistent and thus preempted: (1) the permit, (2) the curfew, (3) the written accident report, (5) the bumper sign and (5) the valve locks. DOT found that the following were not inconsistent with HMTA: (1) immediate accident reporting, (2) driving with headlights on, (3) vehicle inspections and (4) two-way radios.

DOT found that the Rhode Island load-specific permit required much of the same information as already contained on a DOT shipping paper but presented possible delays in transportation by requiring (1) detailed shipment information, including identification number of trailers, (2) material quantities and (3) certifications by shippers and carriers that the loads meet federal rules. DOT found this permit to be inconsistent with the need for shipments of hazardous materials because they could not be accomplished in an expeditious manner.

The curfew appears to be significantly troublesome because of the possibilities of lines of trucks waiting for a curfew to end or trucks hurrying to complete transport of their commodities prior to the start of a curfew. By initiating a curfew, the transport risks to adjacent states or cities within Rhode Island would increase. How curfews affect the risks of transporting hazardous materials has not been adequately evaluated.

The additional written accident report is covered in the provisions of the Hazardous Materials Incident Report (HMIR) system. The illuminated bumper sign conflicted with placarding requirements. The requirement of valve locks were inconsistent with packaging requirements.

SUMMARY AND CONCLUSION

Within the last five years, the transportation of nuclear materials has experienced a rapid growth of state/local regulations. The federal government is responding to develop a legal basis for declaring these state/local regulations inconsistent and has proceeded to declare certain state regulations invalid. However, there are some issues that have not been clarified -- the role of the federal government in the development and support of emergency response capabilities for nuclear and other hazardous materials, detailed shipment information and state requirements for prenotification.

REFERENCES

1. F. A. Morris and M. A. Condon, Federal Preemption of State and Local Nuclear Transportation Regulations, B-HARC-311-043, TTC-0046, Battelle Human Affairs Research Center, Seattle, WA, to be published.

2. Federal Register, January 31, 1980 and May 15, 1980.

3. Hazardous Materials Transportation, Vol. 2, No. 11, November 1979.

4. Inside D.O.E. (September 12, 1979) p. 7.

DISCUSSION

Question by W. G. Bevan, USA: What is the status of the Bridge Toll Authorities (Ogdensburg, N.Y.) who have closed bridges between Canada and the US to radioactive materials?

Answer and Comment by R. Chitwood, USA: I recently looked into the status of this action relative to the Ogdensburg bridge and the decision of the authorities to prohibit certain shipments of radioactive materials. I know of letters sent to the authority from the Department of Energy and the Department of State objecting to the proposed prohibition. I understand that the Bridge Authority has delayed implementation of the proposed action for 30 or more days pending their review of the matter.

Question by G. Jack, Canada: 1. Is it not likely that a state wishing to ban radioactive material transport will be successful in practice, despite any federal preemptive action, because of the state laws which may be held to be valid (e.g. on notification, permits, etc.), and because of the time delays involved in courts disposing of legal issues? 2. Is there a difference in the legal status between a state that is the end-point of transport of radioactive material, and a state that is a transit state during inter-state commerce?

Answer: 1. There has been a significant escalation in the preemptive arguments which will be resolved only in judicial proceedings. The Rhode Island actions which were preempted by DOT/HMTA is significant. I think it is too early to provide a definite answer, but the trend is definitely moving towards the more uniform federal control 2. No, although in actual practice the answer is dependent upon the state/local laws in existence and how long it will take to resolve conflict.

Question by A. L. Kaplan, USA: In many cases, states impose requirements on transportation of radioactive material (particularly low-level radioactive waste) which appear to contravene applicable federal laws or regulations. Of course, the individual or industry so adversely affected by such state requirements always has recourse to legal action. However, the length of time for resolution of such legal action could be many years. Have any federal agencies or contractors to such agencies been considering ongoing actions to be taken by the federal government, which could bring relief to transporters of radioactive materials faster than the time for legal actions to transpire?

Answer: Yes, the progress made by the US DOT in the last few years through the initiation of Docket HM-164 and the preemptive actions nullifying Rhode Island requirements are significant. The reduction of time to resolve judicial proceedings is beyond the capabilities of the author. However, the actions of the agencies and at the congressional levels indicates significantly greater interest and support for nuclear materials transportation specifically and hazardous materials transportation in general.

U.S. INTERGOVERNMENTAL AND STATE RESPONSIBILITIES AND CARRIER PARTICIPATION IN THE ROUTING OF NUCLEAR MATERIALS

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The network of regulations and regulatory authorities within the United States federal government and between the federal agencies and the states on the subject of transportation of nuclear materials is as intricate, as complexly patterned, and sometimes as redundant and conflicting as the road map of transportation routes affected. A grasp of the troubled U.S. transportation environment for radioactive commodities must begin with, or at least include, some understanding of this scheme of authorities and responsibilities. This paper will highlight the regulatory dilemma; the safety and economic controversies that affect the free flow of commerce in nuclear materials within the United States and abroad; and the quasi-judicial and judicial mechanisms that are being, or may be, brought to bear on the problem.

1. The Regulatory Framework.

Looking first at the federal agencies provides the outline of the major regulatory framework of transportation in the United States. These agencies include the Department of Transportation, (including the Federal Highway Administration, the Federal Railroad Administration, the Federal Aviation Administration and the U.S. Coast Guard, among others), the Interstate Commerce Commission, the Nuclear Regulatory Commission, the Civil Aeronautics Board, the National Transportation Safety Board and the Federal Maritime Commission. Where energy resources, such as nuclear fuels and wastes, are concerned, this scheme might also include the Department of Energy, the Environmental Protection Agency, the Federal Energy Regulatory Commission, ithe Occupational Safety and Health Administration, and the Federal Emergency Management Agency.

Superimposed on this national system is the entire panoply of state and local governmental agencies responsible directly or indirectly for transportation of hazardous materials within or through the states. With the increasing emergence of state statutes and local ordinances dealing with nuclear facilities and nuclear materials transportation, the number of state and local agencies has increased dramatically in the last five years. A large number of states and many local governments have adopted or have under consideration some law which will, to a greater or lesser extent, affect nuclear materials transportation.

Initially, the regulation of hazardous materials transportation was the exclusive domain of the U.S. Interstate Commerce Commission (ICC) 1/ with some overlapping responsibility by the now defunct U.S. Atomic Energy Commission in matters nuclear. But in the last two decades much has changed. The ICC yielded its safety authority to the U.S. Department of Transportation (DOT) in 1967. 2/ The Atomic Energy Commission was replaced in 1975 by the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Energy Research and Development Administration

(ERDA) 3/. The latter agency is now subsumed into the Department of Energy created in 1977. 4/ This evolution of governmental regulatory and research responsibilities has not clarified the federal role or satisfied the public need for uniformity and consistency in nuclear transportation. Prompted by a desire to isolate regulatory control from economic and programatic incentives, these changes evoked a salutory division between regulation and the fostering of growth in the industry regulated. But a certain decentralization and complexity naturally followed the creation of more governmental components. New legislation since these changes, however, affords the machinery to put the national regulatory network in order. 5/ As will be discussed, that mechanism is now in operation.

Enabling statutes governing the ICC and the Atomic Energy Commission, as well as the U.S. Constitution, provided the basic ingredients for addressing these problems, i.e. federal preemption and federal supremacy. Recent law is even more explicit. The Transportation Safety Act of 1974, 6/ and more particularly, Title I of that law, the Hazardous Materials Transportation Act, provided the announced policy and the express regulatory authority for ordering the regime of interstate transportation of hazardous materials. This responsibility was given to DOT. The Nuclear Regulatory Commission, having inherited the regulatory authority of the Atomic Energy Commission, controls transportation of radioactive materials to the extent that it regulates the packaging of nuclear commodities under the authority it is given by the Atomic Energy Act. 7/ But in practice, the principal authority on transportation is DOT.

Added to the federal and state governmental entities responsible for regulating nuclear materials transportation are the actual participants in such transportation, the carriers and the shippers. Their responsibilities of course are directly affected by the burgeoning interest in the federal, state and local arenas. These regulatory activities have placed additional, sometimes conflicting, burdens on shipper and carrier industries which exacerbate the often strained and adversarial relationships between these industries, and in turn swell the tensions in the regulatory process.

2. Economics and Safety.

The principal underlying issues in the regulatory scheme are safety and economics. The safety issue directly threatens the existence of such transporation, while the economic issues can whittle away at efficiency and availability and so thwart commerce. The safety question itself has stirred the regulatory waters to their present uneasiness, and has prompted the imposition of severe economic penalties by some carriers on shippers.

As the debate over the safety of nuclear power advanced, the primary focus was on the nuclear power generating facilities with their store of high level radioactive fuel. In the last half decade, concern over this commodity has steadily been channeled toward the movement of all nuclear materials in the fuel cycle.

The first major transportation conflict arose in the State of New York. City of New York, despite some 25 years of safe nuclear transportation within its boundaries, adopted a regulation which banned virtually all such transportation within or through the City. 8/ This precipitated a lawsuit between the United States and the City. 9/ Then the State, within a matter of months sued the United States to prevent the air transport of special nuclear materials. 10/

New York City's rule affected transport of fresh fuel, spent fuel and most other nuclear materials from or to the Brookhaven National Laboratory, a federally owned facility on Long Island, and future transport from commercial nuclear facilities under construction there--the City being the only land route off the Island. The State's court action was directed chiefly at the movement of plutonium and highly enriched uranium fuel in international commerce pursuant to agreements for cooperation between the United States and other countries. The New York State attempt failed in court, 11/ but Congressional action imposed additional constraints on air transport of plutonium. 12/

The litigation with the City has never been prosecuted to conclusion. Proceedings were suspended. Though refusing to void the City's ban, the Court suggested at the hearing an application of the new and yet untried Hazardous Materials Transportation Act, noted above, which gave DOT the authority to preempt inconsistent state and local regulation of the transportation of hazardous materials. 13/ This event started the sequence of regulatory actions that resulted in the principal governmental proceeding that may promise a solution to much of today's nuclear transport controversy. More about that below.

Although New York City was not the first locale to impose local restrictions on the interstate commerce of radioactive materials, its action and the consequent notoriety prompted other cities and states to adopt regulatory constraints, often variations on the New York City requirements, ultimately leading to the present confusion.

Miami was the first port city to adopt severe restrictions on nuclear materials transportation in its harbor and that example too has resulted in several additional ports refusing access to ships carrying these commodities. 14/ The most serious instance of such difficulties involved the shipment of spent fuel being transported from Japan to England. The ship, while off the coast of Hawaii, was denied entrance to the Port of Honolulu to refuel. Only because of its dangerously limited fuel supply, did it finally gain entrance to Pearl Harbor. Litigation ensued, although it was dismissed before it could decide any of the fundamental underlying transportation issues. 15/

From the economic perspective, almost simultaneous with the promulgation of a ban on nuclear transport by New York City began the withdrawal by United States railroads of routine freight service for nuclear materials. The Western and Southern railroads generally imposed inordinately burdensome tariffs and the Eastern railroads perserved in their long-term refusal to move these commodities at all. 16/ While these developments were threatening for some time, their sudden maturity, coincidental with the other transportation difficulties, perhaps also marks in time an important turning point in public perception and sentiment on nuclear materials transportation and nuclear power generally.

Although the rail tariff controversies were on their face economic, it must be understood that the rationale for such changes was safety--not costs and profits. This was brought clearly to the fore in the lengthy administrative and court proceedings that have followed. 17/ Not all of these suits are finally resolved, although in all cases thus far, unduly restrictive tariffs or refusals to publish reasonable tariffs have been defeated. 18/

Currently, the U.S. trucking industry is studying proposals to withdraw from the publication of common carrier tariffs or to impose additional constraints on the movement of nuclear materials. Their concern, however, seems to stem as much from the confusion and excessive administrative burdens generated by the

multiplicity of state and local laws as from safety. With the exception of two minor proposals adopted by a joint committee of the Southern Motor Carriers Rate Committee, the East-South Interritorial Committee and the South-Western Interritorial Committee, the two attempts to alter routine tariff arrangements has been deferred. 19/

3. Judicial and Quasi-Judicial Remedies.

Presently, the mire of confusing, often conflicting, federal and state regulation, coupled with the strife this has generated between the carriers and shippers, both government and private, has resulted in court contests and protracted administrative proceedings on tariff and safety issues. These processes may be the only realistic remedy for the structural, if not the attitudinal, dilemma in U.S. nuclear transportation.

Although air and water transport modes are not entirely secure, ground transportation, chiefly the movement of radioactive materials by road and rail, offers the best opportunity and the most defined avenue for the realignment and harmony of this commerce in general.

As for rail, the tariff controversies just mentioned are slowly evolving into a hard-wrought conclusion. But this is not a solution. Again underlying this conflict is the safety issue and, as presented in these rate proceedings, it reveals its true complexities. Even within the federal government, this rail litigation evidenced redundancy and confusion as to the principal expert and the final arbiter of the entire transportation question. Because of the prominence of the safety issue, the ICC had to grapple with the respective roles and rules of several agencies, but especially those of DOT and NRC, combined with the uncertain limits of its own responsibility. 20/ Naturally, it being only an equal to its sister agencies, and not the principal safety regulator, 21/ it could not speak finally on the divisions of authority and responsibility on this pivotal issue.

The Department of Transportation, however, may have the key to shaping a remedy. DOT, after some preliminary proceedings, 22/ is now fully engaged in the important rulemaking proceeding on the routing of radioactive materials, 23/ spawned by the court controversy in New York City and nurtured by the proliferation of state and local regulation. Despite its somewhat limited scope--highway routing--the rulemaking does deal with the underlying concerns of federal and state government, of industry, of shipper and carrier, and of the public: a safe, secure, uniform and efficient federal system of moving nuclear materials. As focused, the DOT proceeding will not settle all concerns directly. But its initiative in meeting the problem and providing a forum for debate from all quarters and especially its assertion of primacy may, if forcefully implemented, settle even collateral economic and safety issues.

Despite the hope thus expressed for the rulemaking process, the protracted history of this proceeding is perhaps the only characteristic of it that could imperil its effectiveness. The problems it must solve, especially the culling and distribution of federal, state and local responsibilities, with the concommittent resolution of shipper and carrier duties, can not wait much longer. 24/ Patchwork solutions, such as rerouting, changes in transport mode and temporary storage, are limited and often unavailing. And new court litigation, while always a possibility, is a difficult choice and a less complete solution.

While judicial resolution is not necessarily undesirable, it provides much less flexibility, less opportunity for full participation by a multitude of interests and less chance of a final disposition of the entire safety question. For example, Missoula, a city in Montana, recently adopted an ordinance 25/ banning transportation of radioactive materials within its borders. One shipper, apparently the only one directly affected, brought suit and succeeded in gaining an injunction against the operation of the provision of local law. 26/ While nonetheless indicative of the possible success such contests may have elsewhere, the ruling is not final and only applies to the one ordinance, and that ordinance will differ in one degree or another from other local rules. Final court disposition of the litigation could take considerable time; appellate review by higher courts could add years. Ultimate success will probably prove too narrow to be applicable to the resolution of the spectrum of local and federal governmental responsibilities nationwide.

But this is not to say that a court contest could not finally dispose of the issue, should the proper vehicle for review present itself. Also, of course, DOT's resolution of these issues is subject to court challenge. However, DOT can look at the issue in its proper dimension as a national question. Its rule can have, if properly constructed, the breadth of application and depth of participation such that it will succeed in providing a national solution whose rationality would likely be sustained in court, if appealed.

4. Conclusion.

Because it is in the nature of transportation that many and often different communities are weaved together by these threads of commerce, so too it is essential that the pattern that emerges be one that is carefully tailored by the affected governmental and industrial entities affected. Such is not yet a reality. This handiwork warrants the attention of the international nuclear transportation community because our experiences in the U.S. are not unique, 27/ and our solution, if we can fashion one, will provide an example to other

Footnotes

1/ Established in 1887, the Interstate Commerce Commission's enabling statute is now codified in Title 49 of the United States Code (hereinafter cited as USC), Sections 10101 et seq..

2/ This authority was transferred to the Department of Transportation at its inception. 49 USC 1655(e)(4). DOT was created by the Department of Transportation Organization Act of 1967, 49 USC 1651 et seq..

3/ The Atomic Energy Commission, established by the Atomic Energy Act of 1954, as amended, 42 USC 2011 et seq., was superseded by operation of the Energy Reorganization Act of 1974, 42 USC 5801 et seq.. This law created NRC and ERDA.

4/ The Department of Energy was created by the Department of Energy Organizati Act of 1977, 42 USC 7101 et seq..

5/ The U.S. General Accounting Office, an arm of the U.S. Congress, published a report in 1979 concluding that the proliferation of multiple and inconsistent regulatory requirements created both an economic burden and a safety hazard. The report recommended an expeditious and aggressive effort by DOT aimed at melding the divergent federal, state and local regulatory endeavors into an harmonious scheme. USGAO, Federal Actions Are Needed To Improve Safety and Security of Nuclear Materials Transportation, EMD-79-18, May 7, 1979.

6/ 49 USC 1801 et seq..

7/ See 10 Code of Federal Regulations (CFR) Part 70. NRC's Coordination with DOT is articulated in a Memorandum of Understanding, published in the Federal Register of the United States Government, 44 Fed. Reg. 38690 (1979).

8/ New York City Health Code, Section 175.111.

9/ United States v. The City of New York, Civil Action No. 76 Civ. 273, (S.D.N.Y., filed January 15, 1975).

10/ State of New York v. The Nuclear Regulatory Commission, et al., Civil Action No. 75 Civ. 2121, (S.D.N.Y., filed May 5, 1975).

11/ The Federal District Court entered orders on September 9, 1975 and May 7, 1976 dismissing plaintiff's petitions for preliminary injunctions. The Second Circuit Court of Appeals affirmed the District Court on February 14, 1977. 550 F.2d 745. The State withdrew its suit on August 29, 1979, but later instituted a similar action under the same caption. Civil Action No. 79 Civ. 4568.

12/ On June 20, 1975, Congressman Scheuer of New York offered an amendment to the authorization bill for appropriations for NRC (H.R. 7001). This amendment was passed that day and prohibited plutonium shipments by air until NRC could certify to the Congress that a safe container had been developed and tested which would not rupture under crash and blast testing equivalent to the crash and explosion of a high flying aircraft. This provision became part of Title II of P.L. 94-79, August 9, 1975, 42 U.S.C. ¢5841, note.

A similar amendment to S.598, the bill to authorize ERDA's appropriation, was offered by Senator Jackson and received Senate approval on July 31, 1975, as Title V, and was adopted on December 31, 1975. This amendment became section 502 of P.L. 94-187, 42 U.S.C. 5817, note.

13/ There was no published opinion but only a brief handwritten note, dated January 30, 1975, by the Judge finding no grounds at the time for granting the injunction. No appeal was taken.

14/ The ban was imposed by Memorandum of December 29, 1978 from Carmen J. Lunetta, Acting Port Director, to All Agents, Port of Miami.

15/ Life of The Land v. Brock Adams, Civil Action No. 79-249 (D. Hawaii, filed June 6, 1979). An injunction against entry into port was sought on June 7, 1979 and denied on the same day. An appeal in the United States Court of Appeals for the Ninth Circuit affirmed the denial on June 14, 1979, finding that the possible harm to the ship would be more hazardous to the public safety than any other course of action. The suit was finally dismissed in December, 1979, by court order with agreement of the parties. None of the court opinions were published.

16/ 1. Missouri-Kansas-Texas Railroad Company: Uniform Freight Classification 12, I.C.C. 8, Supplement 3, J.D. Sherson, Tariff Publishing Officer, Item 80762-A (Note 5); Item 80764-A (Note 5); Item 80768-A; Item 80769.5, effective date December 31, 1975.

2. The Special Train tariff of the Western and Southern railroads: Uniform Freight Classification 12, I.C.C. 8, Supplement 3, J.D. Sherson, Tariff Publishing Officer, Item 80768-A (Note 5); Item 80764-A (Note 5); Item 80768-A; Item 80769.5, effective date December 31, 1975.

3. Eastern Railroads Flag-out: Uniform Freight Classification 12, I.C.C. 9, supplement 3, J.D. Sherson, Tariff Publishing Officer, Item 80708-A (Note 3). Once the Eastern railroads Flag-out was defeated, these roads filed the following: Eastern Trainload - Uniform Freight Classification 13, I.C.C. 9, Supplement 4, G.F. Earl, Tariff Publishing Officer, Item 80768-A, effective date December 9, 1978.

17/ United States Energy Research and Development Administration v. Akron Canton & Youngstown Railroad Company, et al., I.C.C. Docket No. 36312; Allied-General Nuclear Services, et al. v. Akron, Canton & Youngstown Railroad Company, et al., I.C.C. Docket No. 36313; G.P.U. Service Corporation, et al. v. Akron, Canton & Youngstown Railroad Company et al., I.C.C. Docket No. 36330; Commonwealth Edison Company et al. v. Akron, Canton and Youngstown Railroad Company et al., ICC Docket No. 36335; General Electric Company v. Akron, Canton & Youngstown Railroad Company, et al., I.C.C. Docket No. 36336. These cases were consolidated under ICC Docket 36312.

Radioactive Materials, Missouri-Kansas-Texas Railroad Company, I.C.C. Docket No. 36307; Empty Containers for Radioactive Materials, Missouri-Kansas-Texas Railroad Company, I.C.C. Docket No. 36307 (Sub. 1); Restricted Usage of Containers and Cars, Non-irradiated Cores, I.C.C. Docket No. 36307 (Sub. 2); Restricted Usage of Cars, Radioactive Materials, I.C.C. Docket No. 36307 (Sub. 3); Radioactive Materials, Illinois Terminal Railroad Company, I.C.C. Docket No. 36307 (Sub. 4).

Radioactive Materials, Special Train Service, Nationwide, I.C.C. Docket No. 36325.

Trainload Rates on Radioactive Materials, Eastern Railroads, I.C.C. I & S Docket No. 9205, 9205 (Sub. 1).

U.S. Department of Energy, et al. v. The Baltimore and Ohio Railroad Company, et al., I.C.C. Docket No. 37076. The Akron, Canton and Youngstown Railroad Company et al. v. I.C.C., No. 78-3425, (6th Cir., filed August 3, 1978).

Consolidated Rail Corporation, et al. v. Interstate Commerce Commission and United States of America, Commonwealth Edison Company, et al. - Intervenors, No. 80-1709 (D.C. Cir., filed June 26, 1980); Consolidated Rail Corporation, et al. v. Interstate Commerce Commission and United States of America, No. 80-1804 (D.C. Cir., filed July 15, 1980) consolidated with Burlington Northern Inc., et al. v. United States of America and Interstate Commerce Commission, No. 80-1843 (D.C. Cir., filed July 23, 1980).

Where these cases have resulted in final decisions, those decisions are cited in footnote 18.

18/ (a) ICC Decisions: Radioactive Materials, Special Train Service, Nationwide, 359 I.C.C. 70 (1978); Trainload Rates on Radioactive Materials, Eastern Railroads, 362 I.C.C. 756 (1980) (not a final decision); Radioactive Materials, Missouri-Kansas-Texas Railroad Company, 375 I.C.C. 458, (1977); U.S. Energy Research and Development Administration v. The Akron, Canton and Youngstown Railroad Company, et al., 359 I.C.C. 639 (1978).

(b) Court Decision: The Akron, Canton and Youngstown Railroad Company v. I.C.C., 611 F.2d 1162 (6th Cir. 1979), cert denied, 49 U.S. Law Week 3246 (1980)

19/ Appeal No. 748, Radioactive Materials Docket 798, Subject 17(B) Meeting of the National Classification Committee, October 9, 1980, at the Capital Hilton Hotel, Washington, D.C. The Committee voted to refer the proposal (to remove ratings and minimums under Item 164901 of the National Motor Freight Classification) to a committee for further amplification.

Southern Motor Carriers Rate Committee, Docket Item No. 1128-540-190(G) (10-80) SMCRC Tariff 190-E effective July 23, 1980; Docket Item 1129-540-190(G) (11-80) (AMD) SMCRC Tariff 190-E effective August 25, 1980.

20/ A subject of much contention during the proceedings, the question of I.C.C. safety jurisdiction was resolved first in the Commission's decision of February 24, 1978, rendered by its Division 2 in the Missouri-Kansas-Texas Railroad case. The U.S. Court of Appeals for the Sixth Circuit, in The Akron, Canton and Youngstown Railroad Company et al. v. I.C.C., concluded that there is some residual safety jurisdiction in the ICC despite the evolution of such responsibilities in other agencies.

21/ See footnotes 2 and 6; note especially 49 USC 1803 and 1804.

22/ At the behest of Brookhaven National Laboratory, acting on behalf of its operator, Associated Universities Inc., and as a federally owned facility, DOT undertook the first proceeding of its kind, DOT Docket IR-1, under the authority granted DOT by the Hazardous Materials Transportation Act to determine whether the New York City ordinance restricting nuclear materials transportation was inconsistent with federal law and so preempted. 49 USC 1811. This proceeding did not get underway until early 1977 when the regulations for the conduct of such proceedings were first promulgated in the Title 49 of the Code of Federal Regulations, Section 107.203 et seq.. This proceeding, in any event, resulted in a stalemate. In its decision of April 20, 1978, DOT explained that it had no regulation of its own exactly on point and so had no basis for comparison to determine consistency. It therefore resolved to conduct a rulemaking and develop such a requirement. See 43 Fed. Reg. 16954.

23/ On August 17, 1978, DOT published an "Advance Notice of Proposed Rulemaking" Docket HM-164, to elicit comments and determine "the need, and possible methods for" a federal highway routing rule for radioactive materials. 43 Fed. Reg. 36492. Some sixteen months later, DOT announced a proposed rulemaking now underway. 45 Fed. Reg. 7140 (January 31, 1980).

24/ Recently, DOT announced that it expects to have its final rule published this November, 1980. 45 Fed. Reg. 56607, August 25, 1980. However, it appears at present that this deadline may be slipping.

25/ An Ordinance Concerning The Transportation of Radioactive Materials In The City of Missoula, Ordinance Number 2104, adopted February 4, 1980.

26/ Chem-Nuclear Systems, Inc. v. City of Missoula, Civil Action No. 80-18, (D. Montana, filed March 5, 1980). On March 7, 1980, the District Court issued a Temporary Restraining Order suspending the City Ordinance. On May 22, 1980, the court stayed further proceedings pending either the modification or repeal of the Ordinance or the conclusion of the DOT rulemaking proceeding, HM-164.

27/ Britian is presently beginning to experience surprisingly similar troubles: questions of packaging safety; the dangers of highway and rail transportation; and local community opposition to routing through or near urban areas. Bugler, Nuclear Waste Through Britian's Cities, New Scientist, June 26, 1980, pp. 406-408.

Question by C. B. G. Taylor, United Kingdom: You have expressed the problems you put forward in terms of safety, of legal liability, and of the traditional conflict between the states and the federal government. But the underlying problem is opposition to any use of nuclear power by people who do not want to live in a "nuclear" society. Is the debate not being carried on in those terms?

Answer: Such a debate can be conducted in the political arena, but as for the legal process, such motivations cannot be handled directly but only in their manifestations, that is: the safety issue and the economics issue. The legal forum is not conducive to open debate on philosophical differences. Rather it must be confined to the specific problems and challenges arising within the legal framework in existance.

Question by K. Gablin, USA: Currently there is a possibility of a Grand Jury in the state of New Jersey bringing criminal charges against a trucking firm's officer. Will the adoption of DOT's HM-164 preempt the action in New Jersey? Reference: accident of two casks of IR-192 on Rt. 17 without filing proper notice paper prior to shipment with the state of New Jersey.

Answer: HM-164 will not directly preempt such a proceeding. But if time permits, it could be brought into the case as a defense. It is also possible that the constitutionality of the New Jersey law could presently be raised as a defense to the proceeding, even without HM-164. But there is no way to predict the outcome of such a defense without a thorough examination of the applicable New Jersey statute.

Question by M. Grenier, France: How are the responsibilities shared, in the USA, between states and federal authorities in the special field of compliance and quality assurance for packages and packagings?

Answer: Responsibilities between the states and federal government are evolving and are in fact also a subject for consideration in the Department of Transportation rulemaking, HM-164. As to the regulation of packaging, there is little, if any, room for state involvement because such packages must meet federal standards and must obtain federal licenses. It of course should be noted that the Atomic Energy Act does provide for a state role in nuclear energy regulations in certain ways specified by the law. States wishing to participate do so by entering into an agreement with NRC.

Question by W. G. Bevan, USA: Even though the Department of Energy might prevail in the present litigation, do you foresee that the railroads will attempt to reimpose economic sanctions against radioactive materials under the deregulation legislation recently passed by Congress?

Answer: Despite the success so far in defeating restrictive rail tariffs for nuclear transportation, many railroads have persisted and much of the litigation has not been finally concluded. The recent legislation providing for deregulation does provide some protection for captive shippers, such as nuclear transporters, but there will be opportunity for some penalties to be imposed on nuclear materials transport as the deregulation scheme unfolds. However, it is hoped that the exchange of information and opinions within and outside the ongoing proceedings will solve the differences between the shippers and the railroads.

ASSESSMENT OF INSTITUTIONAL AND TECHNICAL MEASURES TO REDUCE THE RISK OF POTENTIAL ACCIDENTS AND SABOTAGE OF NUCLEAR SPENT FUEL DURING TRANSPORTATION

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The transportation of spent fuel within the United States (as within other countries) is an activity which is presently under considerable scrutiny by policy makers and the public. A major cause for such close examination is the possible consequences to public health and safety and to the environment resulting from a serious accident or sophisticated sabotage attempt, however small the risk. Further, lack of faith by the public in the ability of local authorities to effectively respond to nuclear spent fuel transportation emergencies has gained considerable attention. The purpose of this paper is threefold. The first is to examine the potential risks associated with spent fuel in transit. The second is to discuss institutional and technical measures in the United States to reduce transportation risks due to serious accidents or malevolent activities. The third is to provide an assessment of these measures by discussing their effects on spent fuel transportation activities in the United States.

RISKS ASSOCIATED WITH SPENT FUEL IN TRANSIT

Risk due to the occurrence of an event is traditionally defined in the following manner:

$$R_1 = P_1 \times C_1$$

where: R_1 = risk due to event "1"
P_1 = probability of occurrence of event "1"
C_1 = consequences of event "1"

An exact determination of the risks associated with spent fuel in transit is elusive since probabilities and consequences are neither black nor white, particularly when a "what if..." approach is used. The report, "Transport of Radionuclides in Urban Environs: Draft Environmental Assessment" (July 1980, SAND 79-0369), focuses on the risk statement. The probability term may be expressed in three parts: 1) the expected number of accidents or malevolent activities, of a given severity, per year for each transport mode; 2) the package or container response to these occurrences; and 3) the anticipated dispersal of radioactive materials. The consequences of an accident or malevolent activity relates to the effects of the estimated quantities of released radioactive material on local population and environment.

Closer examination reveals additional external variables that may also affect P_1 and C_1. Examples include mode of transport; condition of highway or railroad track; cask integrity; severity of accident or sabotage event; fraction of the total package contents which might be released during an accident or sabotage event of a given severity; number of people in exposure area and their proximity to the accident; meteorological conditions; and evacuation capability.

The Sandia report categorizes spent fuel shipments as low-probability/high-consequence events. An approach used to deal with such events is called "mini-max." This technique focuses on the calculation of consequences separate from probabilities, recognizing that even at extremely low event probabilities the consequences may be intolerably severe. Spent fuel used for this risk analysis was in shipment sizes of 154 curies dispersible and 2.17×10^5 curies nondispersible and was transported in casks by highway motor carrier. Ensueing analysis of low-probability/high-consequence accidents involving spent fuel yielded the following health effects:[1]

- early fatalities - 0
- early morbidities - 0
- latent cancer fatalities - 1

In addition, the direct economic impact involving both the dispersible and nondispersable forms of spent fuel accidents is extimated to be $\$2 \times 10^9$, and is broken down as follows:[2]

	% of $\$2 \times 10^9$
• on-scene costs	0.1%
• survey costs	0.7%
• street cleanup	0.1%
• building cleanup	0.1%
• evacuation	1.0%
• security costs	2.0%
• land-use denial	95.0%

In analyzing the roles played by health effects and economics in defining consequences, the social impacts resulting from the transportation of spent fuel through urban environments should not be ignored. An accident or act of sabotage, even if only perceived, stimulates responses aimed at preventing the occurrence of such events. Categories of social impacts include:

- Psychological impacts -- defined as psychological distress and changes in an individual's beliefs, attitudes, and behavior.

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[1] "Transport of Radionuclides in Urban Environs: Draft Environmental Assessment," NUREG/CR-0734, SAND 79-0369, July 1980, p. 66.

[2] Ibid.

- Sociological impacts -- defined as changes or attempted changes in social structure, the organization of social life, and collective behavior.

- Political impacts -- defined as changes or attempted changes in those activities by which opposing viewpoints are reconciled, including both governmental politics and public interest group activities.

- Legal impacts -- defined as changes or attempted changes in statutes, regulations and adjudicatory decisions.

- Organizational impacts -- defined as changes in organizations with respect to their existence, structure, procedures, and capabilities.

These categories are not mutually exclusive.[3]

According to the study, the severity of sociological impacts may vary according to the cause of transportation events; for example, caused by vehicular malfunction, caused by human errors and deviation from accepted quality assurance practices, or a result of malevolent acts such as sabotage or theft.

To summarize, risk is expressed as the product of probabilities and consequences. The consequences of an accident or act of sabotage can be expressed as the sum of health effects, economic costs, and social impacts. In the case of spent fuel transportation, accident and sabotage events may result in high consequences. Since the probability of these events occurring is low, however, the "Urban Environs" study concludes that overall risk attendant to spent fuel shipments is small.

The following two sections; 1) institutional measures and 2) technical measures taken in the United States to reduce transportation risks due to serious accidents or malevolent activities, are based upon the low-probability/high-consequence approach to risk assessment analysis. Institutional and technical measures focus on reducing risk by minimizing both the probability and consequence factors.

INSTITUTIONAL MEASURES TO REDUCE TRANSPORTATION RISKS

The assessment of institutional measures is based on existing and proposed federal and state laws and regulations that govern spent nuclear fuel shipments. At the national level, the Nuclear Regulatory Commission's (NRC) final ruling (Physical Protection of Irradiated Reactor Fuel in Transit, 45 FR 37399 amended from 44 FR 34466) of June 3, 1980, and July 27, 1979,

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[3]Ibid, pp. L-3-L-6.

respectively and the Department of Transportation's (DOT) proposed ruling (Highway Routing of Radioactive Materials, HM-164, 45 FR 7140) of January 25, 1980, illustrate institutional measures. Further, the Federal Emergency Management Agency (FEMA) was created in 1979 to better organize the federal government's preparedness for and response to national emergencies ranging from natural and manmade disasters to nuclear attack.

To date, a number of legal and regulatory measures that impact nuclear spent fuel transportation are being passed by state legislatures and local jurisdictions. South Carolina's and Kentucky's laws are primary examples of measures taken at the state level. A bill recently introduced in Montgomery County, Maryland, illustrates efforts taken at the local level to regulate nuclear materials transportation. All three of these measures will be discussed later. The fact that regulations on radioactive materials transportation are being instituted at all levels of government represents a conflict of authority between the federal government and state and local governments.

Insitutional Issues at the Federal Level

Nuclear Regulatory Commission's Final Ruling. Following review of the first working draft of Sandia's "Urban Environs" report in 1978, the NRC established requirements for physical protection of spent nuclear fuel shipments in transit. The requirements are based on data which show that malevolent activities directed against such shipments have a potential for producing serious radiological consequences in areas of high population density. The final rule is contained in Title 10, Code of Federal Regulations (CFR) 73.37, July 27, 1979.

Major requirements of the rule include the following:

- NRC approval of the route in advance. The route is to avoid high population density areas wherever possible.
- Prior arrangements with law enforcement agencies along the routes for their response to an emergency or a call for assistance.
- No intermediate stops except for refueling and reprovisioning.
- Improved escort system, including trained guards and radiotelephone voice communication.

Department of Transportation's Proposed Rulemaking. The DOT introduced its proposed rulemaking on the Highway Routing of Radioactive Materials (HM-164) after DOT concluded that a federal routing requirement was needed to standardize interstate routing requirements for radioactive materials transportation by highway and to prohibit any outright state or

local ban on such shipments. From written comments and public hearings on HM-164, additional issues have been raised. Some of these are:

- Federal preemption of state and local laws - is it legal?
- Emergency response and preparedness - can first responders cope with a spent fuel accident?
- Prenotification to state and local authorities - specifically prohibited by HM-164 but strongly desired by state and local governments.
- Enforcement - who enforces the regulation?
- Impact of DOT proposed ruling on other industries unrelated to nuclear materials use (chemicals, liquified natural gas, propane, and other hazardous materials) - will there be other federal routing regulations covering these materials?

The Department of Transportation, Materials Transportation Bureau, is presently analyzing all comments on HM-164 and may incorporate new provisions that address some of these comments in a final ruling that has yet to be enacted.

Federal Emergency Management Agency. As a result of Reorganization Plan No. 3 of 1978 (effective April 1, 1979, pursuant to Executive Order 12127 of March 31, 1979), the Federal Emergency Management Agency (FEMA) was created. President Carter's reorganization plan merged into FEMA five federal agencies involved with emergency preparedness and mitigation and response to national emergencies ranging from natural and manmade disasters to nuclear attack.

The five previously existing agencies were:

- Federal Disaster Assistance Administration
- Federal Preparedness Agency
- Defense Civil Preparedness Agency
- United States Fire Administration
- Federal Insurance Administration

Furthermore, FEMA consolidates into a single structure a number of federal emergency related activities. Those dealing with nuclear issues are:

- Preparedness-planning assistance to state and local governments in developing a capability to respond to all types of emergencies and disasters.
- Coordination of preparedness activities for response to national crises, including peacetime radiation hazards and nuclear warfare.

- Coordination of federal activities in national and multinational civil-emergency planning.
- Operation of a national emergency-broadcast system.
- Coordination of all emergency warnings.
- Practical application of research to mitigate damaging effects of emergencies and disasters.
- Coordination of federal, state, and private activities for response to and recovery from emergencies and disasters.
- Response to consequences of terrorist activities.
- Emergency response training.

FEMA regulations, set out in Title 44, Chapter I, of the U.S. Code of Federal Regulations, outline the procedures to be followed for disaster assistance and emergency preparedness in the event of radioactive releases. These guidelines establish terms and conditions under which FEMA provides federal funding to states for procurement of civil defense equipment and for personnel and administrative expenses. The guidelines also focus on concerns related to federal-state disaster assistance agreements, emergency shelter, temporary housing, emergency support teams, use of local firms and individuals, emergency communications, assistance by federal agencies, legal services, and emergency public transportation.

Institutional Measures at the State and Local Levels

On January 15, 1976, truck shipments of irradiated reactor fuel from Brookhaven National Laboratories' facilities on Long Island, New York, were interdicted by an amendment to the New York City Health Code. In effect, the amendment created a two-tiered system of restrictions on nuclear materials transportation in New York City. First, low-level radioactive materials are allowed entry, but are required to be transported over specified truck routes. High-level radioactive materials, including spent reactor fuel, specified quantities of plutonium, enriched uranium, other actinides, and certain teletherapy sources, require a Certificate of Emergency Transport.

Since enactment of the New York Code, numerous state and local jurisdictions have passed similiar ordinances restricting or banning nuclear materials shipments. Some of these are modeled nearly word-for-word after the New York amendment. For example, laws recently implemented in the states of South Carolina (HB-3350) and Kentucky (HB-863), and a bill introduced in Montgomery County, Maryland, exemplify state and local initiatives to control and regulate the movement of radioactive materials, including spent fuel. These laws establish requirements for carrier permits, prenotification, routing, emergency response procedures, and vehicle insurance. Oftentimes, state and local regulatory initiatives introduce requirements that may be

stricter than federal transportation guidelines, and may even conflict. Thus, shippers and carriers are faced with a myriad of regulations that can vary significantly from one state or local jurisdiction to another.

South Carolina. HB-3350, the South Carolina "Transportation of Radioactive Waste Act," regulates the transportation of any radioactive waste into and within the state. It requires shippers of radioactive waste, including the U.S. Department of Energy, to:

- post bond in an amount that ensures adequate liability insurance;
- purchase a permit;
- comply with state and federal rules and regulations regarding packaging, transportation, storage, disposal, and delivery;
- certify that it will not hold the state responsible for any claims, actions, or proceedings arising out of radiological injuries or damages to persons or property;
- provide a manifest for each shipment;
- provide certification with each shipment that all regulations have been met;
- provide any other information requested by the state Department of Health and Environmental Control;
- state a primary route; and
- provide 72-hours advance notice of any shipments.

Carriers of radioactive waste are required to meet certain provisions as well, including:

- successful completion of operator training as prescribed by the U.S. Department of Transportation;
- reporting any variances from original routes or estimated times of arrival;
- certifying that the shipment has been inspected and meets all state and federal requirements; and
- carrying a copy of the shipper's manifest.

In addition, the shipment must be properly placarded and all papers put in order. Penalties for non-compliance may be up to $25,000 per day. This law became effective in September 1980.

Kentucky. HB-863 authorizes the Secretary of Kentucky's Department of Transportation to control and regulate the movement of all radioactive materials and the intrastate transportation of other hazardous materials within the state. In doing so, the

federal hazardous materials transportation regulations may be adopted in part or in entirety. All carriers of radioactive materials are required to obtain a permit and provide certain information annually, including emergency procedures to be followed in the event of an accident. In addition, this law details the types and minimum amounts of insurance required for each vehicle carrying radioactive materials and other hazardous materials. HB-863 became effective in July 1980.

Maryland. The Montgomery County bill is designed to provide minimum standards and regulations to ensure the safe shipment and transportation of radioactive materials (including spent fuel) into, within, through, and out of Montgomery County; and to monitor shipments that pose a significant threat to public health and to the environment. The bill stipulates the following:

- A "Certificate of Emergency Transport," issued by the County Executive (or designee), is required prior to transporting certain nuclear materials, including spent fuel.
- Carriers must provide the County Executive with specific information in order to receive a certificate; e.g., name of carrier; type and quantity of nuclear material in the shipment; proposed date and time of shipment; starting point, route schedule, destination, place and time of stops; name of receiver; etc.
- Transport of nuclear waste or spent fuel will not be allowed into or through the County for the purposes of storage, disposal, or reprocessing in the County or elsewhere.
- The certificate will not be given unless proof exists that the shipment is necessitated by urgent public policy interests of the County which shall include medical, nondestructive radiographic testing, and educational interests only.
- The shipments must meet all federal regulatory standards on packaging and labelling.
- No certificate shall be issued until the applicant files with the County Executive a bond sufficient to protect the county from the costs of cleanup, decontamination, and immediate and residual health costs arising from radiation exposure. The amount of the bond shall be determined by the County Executive.

If one compares these requirements with one another and with the proposed federal regulation HM-164, discrepancies and conflicts are evident. A major unanswered question is to what extent state and local institutional measures will ultimately be able to regulate spent fuel shipments.

Technical Measures to Reduce Transportation Risk

Technical measures designed to reduce the overall risk of accidents and sabotage focus on consequence reduction through design of transportation hardware, including packaging, and additional physical security measures. Technical measures currently being employed to limit consequences of potential accidents or malevolent acts include:

- High integrity casks which meet NRC specifications regarding high temperature, impact, and pressure.

- Physical security measures, including 1) provisions to immobilize the cab or cargo-carrying portion of the vehicle, 2) highly trained and armed escorts within heavily populated areas, and 3) communications.

Spent fuel shipping casks currently offer significant resistance to accidental or intentional damage which would result in the release of radioactive materials of any consequence. Casks licensed by the NRC must be designed to meet the following hypothetical accident conditions with only minor external effects (i.e., external dose or release):

- Thirty foot free drop on an unyielding surface striking in a position expected to produce maximum damage;

- puncture test involving a free drop of 40 inches striking a 6-inch diameter mild steel cylinder;

- simulated fire test with heat input from a radiation environment of 1,475°F for 30 minutes with no artificial cooling of an additional three hours; and

- submergence test such that cask is under 3 feet of water for a period of not less than 8 hours.

In addition, Sandia Laboratories has completed successful full-scale tests involving a railroad locomotive crashing into a spent fuel cask (with the cask on a highway trailer), and involving a truck crashing into a rigid concrete barrier. These tests demonstrate the integrity of typical spent fuel casks against damage, and verifies mathematical models of postulated accidents.

Of the recent NRC physical security requirements for spent fuel, immobilization of the cab or cargo-carrying portion of a vehicle is probably the most unique. While several options for immobilization are available (e.g., disable ignition system in tractor, puncture several tires, etc.), they are all intended to render the spent fuel transport immobile such that neither it nor the spent fuel can be easily moved from a location from which hijacking might be more probable to one where higher consequences might be produced. Training of guards in escort and defensive tactics may be considered if spent fuel shipping becomes more frequent. Use of sophisticated weapons (e.g., automatic rifles, etc.) are generally restricted by federal,

state, and local laws in the U.S. Nonetheless, weapons suitable for use by escort personnel are available. Finally, a range of communications options are available to keep spent fuel shipments in contact with emergency response personnel should an accident or sabotage attempt occur. Two-way radiotelephones or mobile transceivers linked to local and/or state police are examples. Sophisticated communications/tracking systems such as DOE's SECOM have been proven technically feasible, although not currently available for private carrier use.

The NRC is sponsoring research to analyze the behavior of spent fuel within a shipping cask during and after explosive attacks. Depending on the outcome of these tests, NRC may either increase or perhaps relax some of the physical security requirements for spent fuel during transportation.

INSTITUTIONAL AND TECHNICAL MEASURES AFFECTING SPENT FUEL TRANSPORTATION

The overriding purposes of the institutional and technical measures are to ensure maximum safe transportation of nuclear materials by minimizing the risks involved, and to improve public acceptability of such transport. Both measures compliment one another, as laws and regulations are the means by which governments impose technical requirements. Stronger shipping casks, improved safeguards, and physical protection systems, combined with the best possible routes and emergency management training systems are expected to minimize both the probability and consequence terms of the risk equation, and thus will minimize the risk itself.

In the United States, a situation of controversy and confusion has developed during the past several years consequent to state and local governments' enforcement of regulations to control nuclear materials shipments into and through their jurisdictions. Shippers and carriers are increasingly forced to comply with a myriad of requirements that may vary from state to state and locality to locality. As a result, the regulatory environment has become uncertain and nuclear materials transportation has been curtailed to some degree.

The Federal Government's institutional measures are aimed at imposing standards in a unified, national regulatory structure that applies to all interstate nuclear materials transportation, yet does not ignore the concerns of state and local jurisdictions regarding such shipments. However, the issue of states' rights versus federalism in the United State has yet to be resolved.

The effects of institutional and technical measures on spent fuel transportation are varied. Both types of measures focus on reducing risk, yet arguments frequently occur among transportation policymakers over the ability of technical measures to keep risks at a minimum. Therefore, institutional measures, which also focus on reducing risk, tend to reflect the controversy among policymakers at varying levels of government.

THE OAK RIDGE SPENT FUEL LOGISTICS MODEL*

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In response to continuing interest in the shipment of radioactive materials and as a mechanism for detailed transportation analyses, a Spent Fuel Logistics Model[1] (SFLM) has been developed at the Oak Ridge National Laboratory. The model was designed to evaluate the impact of waste management policy decision on the transportation requirements for moving spent fuel from the reactors to alternative away-from-reactor (AFR) storage facilities and geological repositories. This effort is currently being funded by the Department of Energy (DOE) through the Transportation Technology Center (TTC) at Sandia National Laboratories.

The model has three basic functions: (1) to prepare a shipment schedule that depicts the rate at which the fuel assemblies will enter and leave the transportation system; (2) to evaluate the appropriate shipment destinations (if multiple storage facilities are in operation, the model selects the appropriate destination to minimize transportation costs); and (3) to calculate and report the transportation data describing the particular scenario under consideration.

The transportation network being analyzed includes shipments between reactors and AFRs, AFRs and repositories, and direct reactor-to-repository shipments. The model is currently designed to handle problems that could involve ten AFRs, ten repositories, and a study period of 80 years.

A computerized nuclear reactor data base, which is included as part of the SFLM, contains information for all reactors in the United States which are either operating, under construction, or are firmly planned through 1990. Among other information, the data base contains a summary of historical and projected spent fuel discharges, historical fuel shipments, and onsite storage pool capacities. The data are based on information collected during the summer of 1979 in the Department of Energy Spent Fuel Survey. The basic data have been modified to include more recent information relating to delayed startup dates and cancellation of proposed nuclear plants.

In order to make long-range spent fuel shipment projections, it is necessary to estimate how the nuclear industry will change after the early 1990s. Since detailed reactor information is not available for this time period, the model has been designed to automatically schedule and site a series of standard size expansion reactors to span the gap between the nuclear data base and the capacity projection of interest. The user does have control over how the expansion reactors will be geographically distributed across the country.

*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.

**System Development Corporation, Oak Ridge, TN (formerly with the Chemical Technology Division, Oak Ridge National Laboratory).

Once all the reactors have been defined, the spent fuel shipping schedules are generated. For the reactors in the data base, it is assumed that a full-core reserve will be maintained at each reactor site. Options are available which control the onsite storage capacity and whether interreactor shipments will be allowed. Two onsite storage capacities are included in the nuclear data base: the first represents the current storage capacity, which reflects any announced additions; the second includes the utilities projection of the maximum storage capacity available at that site. If interreactor shipments are permitted, the user can specify which set of reactors will share storage facilities. This option delays the necessity of making a shipment to an AFR but fills the combined storage facilities selected more rapidly. Figure 1 shows two separate shipping projections. The solid line represents the case where there are no interreactor shipments; the dashed line indicates the case with interreactor shipments. All interreactor shipments were assumed to remain within a given utility. Notice that when interreactor shipments are included, the shipping rate is lower prior to 1998. After this date, however, a greater percentage of the reactors have filled their storage facilities, and this option actually produces an increased number of shipments each year. Since onsite storage capacities for the expansion reactors are not known, shipments from these reactors are based on a specific cooling time of the discharged assemblies in the storage pool.

The second basic function of the SFLM is to determine the destination of the projected fuel shipments. If only a single destination is available, no decisions are required. However, if multiple destinations happen to be available, the optimal destination will be selected based upon minimizing transportation costs or public radiation exposure. The optimization techniques can be controlled to minimize shipments to a set of AFRs only or to include AFR-to-repository shipments in the objective function. This latter option will have a significant impact on projected AFR inventories. In the first case, the reactors will make fuel shipments to the nearest AFR. However, in the second case, the total transportation costs are minimized, and reactors could be directed to ship to a more distant AFR if the subsequent cost of the AFR-to-repository shipment would result in an overall savings.

Two different optimization techniques are available. The global optimization technique, which utilizes linear programming, operates upon all of the independent variables simultaneously and generates the lowest cost solution. The second approach is a year-by-year optimization technique where a number of small transportation problems are solved sequentially. Each technique has its own advantages and disadvantages. While the particular problem being analyzed dictates which optimization technique will be selected, both techniques will give identical results in most cases.

In order to reflect the actual operational characteristics of the transportation system and receiving facilities (AFRs or repositories), a number of constraints have been incorporated into the model. These constraints regulate the opening and closing dates, fuel receiving rates, and storage capacities at the various receiving facilities. The numerical values of the constraints can be changed annually to reflect changing characteristics as a function of time. Any of the transportation links can be blocked to allow the analysis of regionalized AFRs and repositories. In this type of study, a particular AFR is allowed to receive shipments from a selected set of reactors. While a specific set of regional boundaries can be imposed for a particular scenario, the model has been designed to calculate the optimal regional boundaries that minimize transportation costs. The mode of transportation used to make the shipments can also be influenced. The model is designed to select the optimal transportation mode.

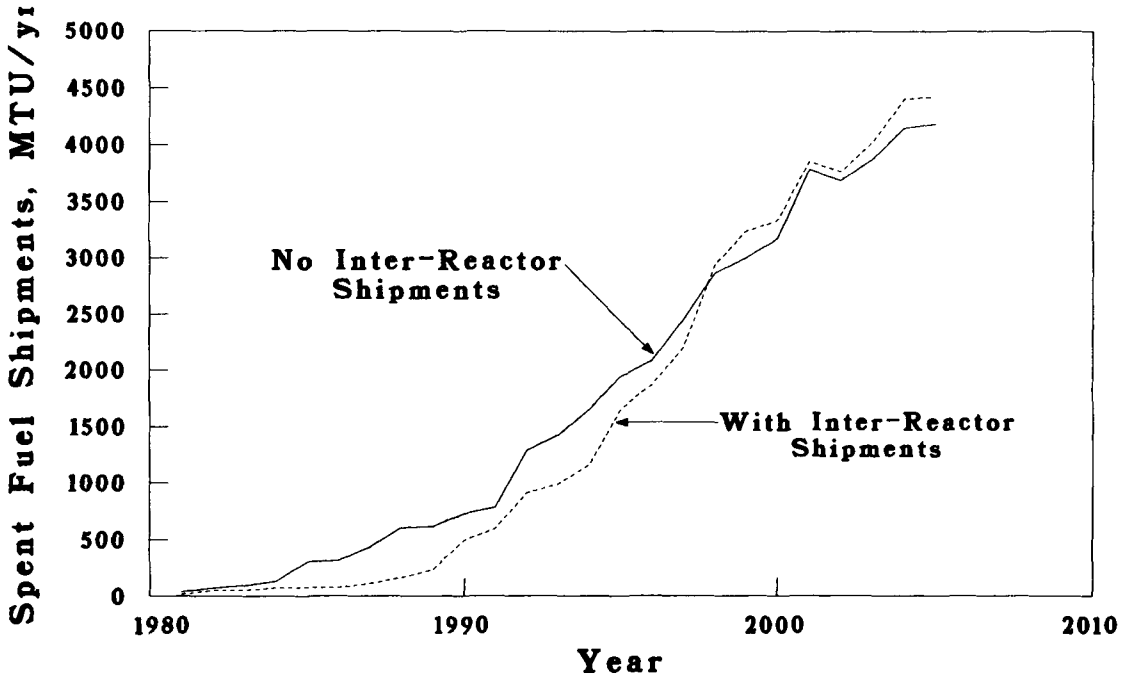

Fig. 1. Projected spent fuel shipments 1981 - 2005.

However, it is possible to maximize rail shipments by requiring all reactors with direct rail access to make all shipments by rail. The relative amount of rail or truck shipments which will be used to transport fuel to a particular receiving facility is calculated in the model. The modal mix is a dependent variable and not an independent variable to be specified at the start of a run as is done in some other transportation models.

A large amount of transportation data is tabulated for each run, and the output report supplies the following information for each year covered in the study: (1) a schedule and destination for all shipments, (2) inventories of fuel at reactors, AFRs, and repositories, (3) transportation distance and cost, (4) radiation exposure to the public, (5) cask fleet requirements, (6) ratio of rail and truck shipments received at an AFR or repository, and (7) the age distribution of fuel being shipped or stored.

An example of some of the output information is shown in Figs. 2-4. The projected spent fuel shipments (assuming no interreactor shipments) were shown as the solid line in Fig. 1. For these shipments, the estimated rail cask fleet size by year for transporting all of the fuel to an AFR in the southeastern United States or all of the fuel to an AFR in the northwestern United States is shown in Fig. 2. A similar estimate of the truck cask fleet requirements is given in Fig. 3. The data used in generating this information assume a rail cask capable of moving 10 pressurized-water reactor (PWR) or 24 boiling-water reactor (BWR) assemblies and a legal-weight truck cask with a capacity of 1 PWR or 2 BWR assemblies. In Fig. 4, the relative amount of fuel transported by rail is outlined. Since only a single AFR site was assumed (either in the southeast or the northwest United States), this information is not dependent upon AFR location. If more than one AFR was available, the relative amount of rail or truck shipments would become a function of the individual reactors making shipments to a particular receiving facility.

Over the past two years, the SFLM has been used in several studies for the Department of Energy (DOE), the Transportation Technology Center (TTC) at Sandia Laboratories, the Office of Nuclear Waste Isolation (ONWI), and Savannah River Laboratory (SRL), which are reported in refs. 2-5. The specific studies include (1) an AFR siting study,[2] (2) a logistics support for SRL participation in congressional hearings, (3) an ONWI regional repository study,[3-5] and (4) a logistics analysis for an AFR Environmental Impact Statement.

1. D. S. Joy and B. D. Holcomb, Logistics Models for the Transportation of Radioactive Waste and Spent Fuel, ORNL/TM-6192 (March 1978).

2. D. S. Joy and L. B. Shappert, The Effects of AFR Storage Location on Spent Fuel Transport, presented at the American Nuclear Society meeting, Atlanta, Georgia (June 1979).

3. K. D. Kirby et al., Evaluation of the Regional Repository Concept for Nuclear Waste Disposal, SSA-123, Southern Science Applications, Inc. (October 1979).

4. D. S. Joy and B. J. Hudson, Transportation Analysis for the Concept of Regional Repositories, ORNL/TM-7170, TTC-0057 (June 1980).

5. D. S. Joy and B. J. Hudson, Logistics Characterization for Regional Spent Fuel Repositories Concept, ONWI-124 (November 1980).

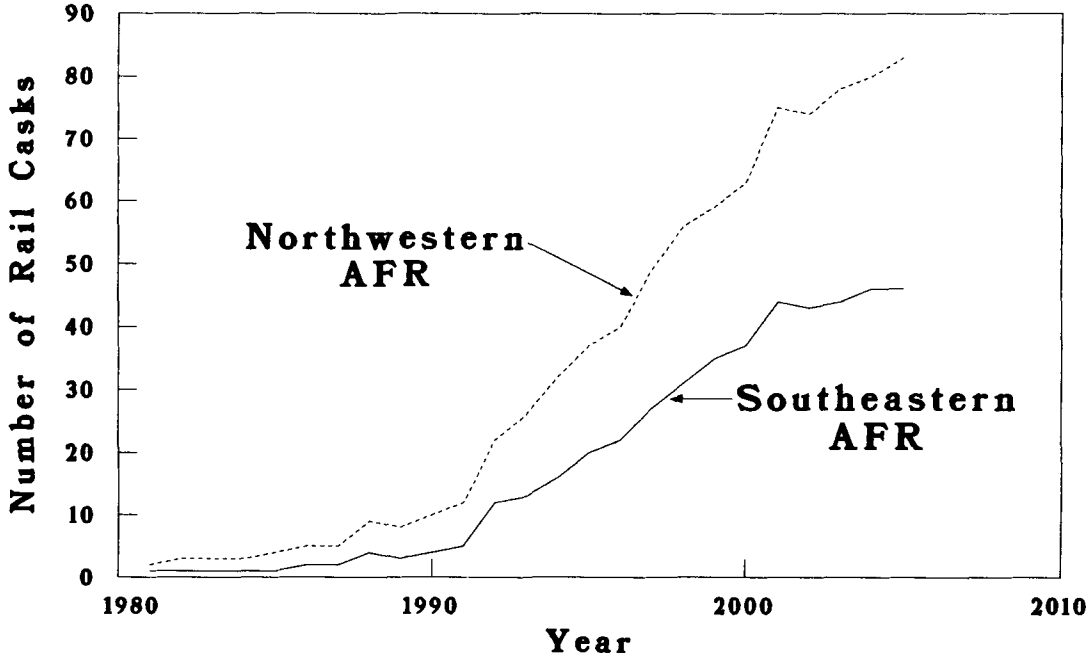

Fig. 2. Projected rail-cask requirements 1981 - 2005.

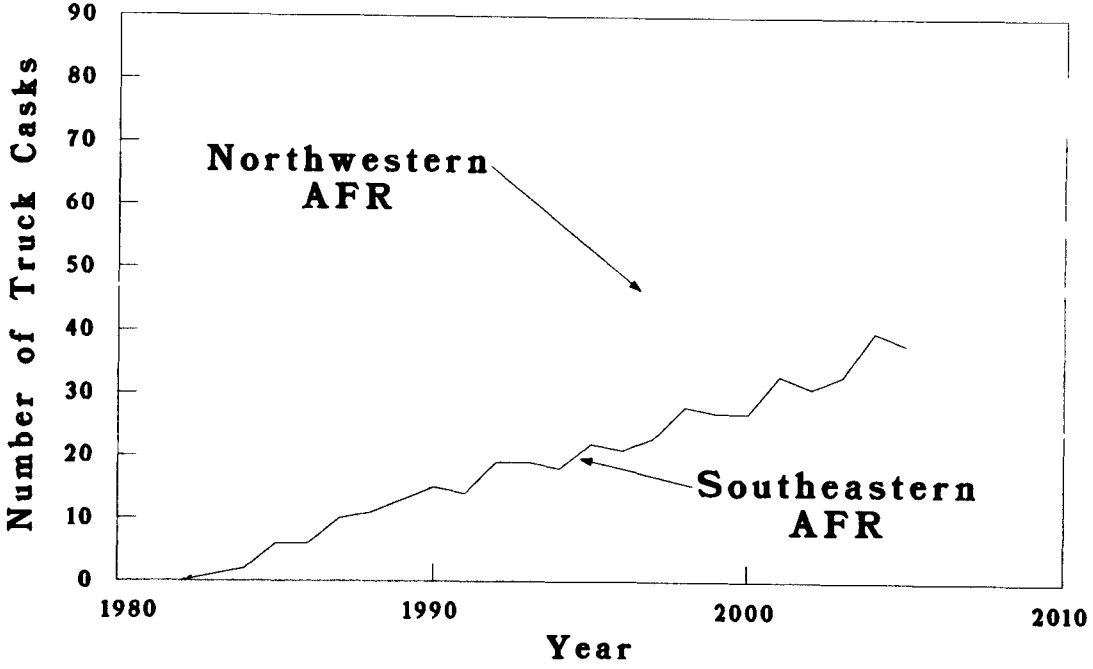

Fig. 3. Projected truck-cask requirements 1981 - 2005.

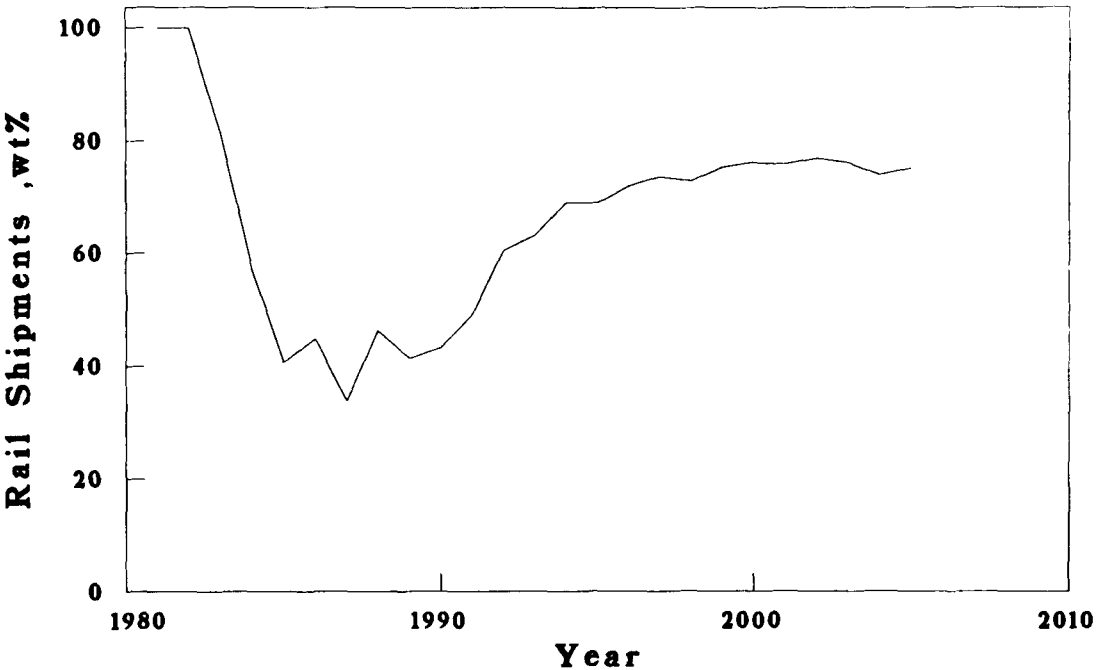

Fig. 4. Relative proportion of fuel shipped by rail 1981 - 2005.

MOTOR CARRIER ROUTING USING THE COMPU.MAP PROGRAM*

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A computerized motor vehicle routing program is being utilized by the Oak Ridge National Laboratory (ORNL) to investigate potential problems associated with the movement of nuclear waste by truck from one point to another throughout the United States. The basic system, COMPU.MAP, was developed by Logistics Systems, Inc., and was tailored to our specific needs by ORNL personnel. The data base contains approximately 12,500 road segments and 9500 nodes, or intersections. This encompasses the entire interstate highway system, the U.S highway system, and all of the principal state highways.

The original program has assigned a specific distance and velocity value to each road segment. Additional blank identifier fields are associated with each segment and these can be used, for example, to specify population density values or other important transportation factors. In the future, this will permit the program to optimize routes based on these or other transportation factors.

Additional links are being added to the program to permit routing from each reactor site. These sites are typically located some distance away from the routes originally specified in the program. Another improvement involves the modification of the computer algorithms for the rail routing model[1] such that they may be applied to the truck routing program. Completion of this work will permit additions and corrections to be made to the data base in a more efficient manner.

Two other significant changes have been or are being made to the program. First, the data base has been carefully checked, and those metropolitan areas whose populations exceed 100,000 persons have been identified; routes passing into these areas can be excluded if necessary. Such restrictions have been promulgated by the Nuclear Regulatory Commission (NRC) and apply to spent fuel shipments. They forbid the transport of such material within 5 km of any metropolitan area whose population exceeds 100,000, and there are now more than 165 such areas in the United States. Second, the longitude and latitude of each node is being determined. This latter step is being taken to improve the accuracy of the maps produced for displaying the routes, but more importantly, to permit the determination of accurate population density values from U.S. census information.

*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.

**System Development Corporation, Oak Ridge, TN 37830 (formerly with the Chemical Technology Division, Oak Ridge National Laboratory).

In order to determine a route, the program calculates link impedances using assigned weights as shown in the following equation:

$$i_n = TB_n\ (\alpha T_n + \beta D_n), \qquad n = [1,2,\ldots.\ k] \tag{1}$$

and

$$I = \sum_{o}^{n} i_n, \tag{2}$$

where

i = impedance for each link;

I = total impedance;

TB = toll weight;

α = time weight;

β = distance weight;

T = time; and,

D = distance.

The toll weight value factor provides a mechanism for including additional operating costs associated with travel on toll roads and toll bridges, thus increasing the impedance to the appropriate link.

With this method of calculation, any one of three criteria can be used to predict a route: the shortest, the fastest, and the most practical. The shortest route will tend to use more state highways and secondary roads, where speeds must be reduced. The fastest route generally utilizes the Federal Interstate Highway system or other high-speed roads. The most practical route is defined as a weighted average of the shortest and fastest routes with a 0.7 factor for time (α) and a 0.3 factor for distance (β), specified in Eq. (1). These factors have been determined by experience to provide routes that are often driven by haulers of hazardous materials and can therefore be considered the most practical. As might be expected, it is not unusual for the fastest and most practical routes to be identical, particularly for long transcontinental movements.

Once population density values are placed into the program and associated with each individual link, a fourth routing criteria will be possible: the lowest radiation exposure to the general population.

A hypothetical case illustrating the current capabilities of the system would be the following: Assume a shipment of waste is to be moved from Morris, Illinois, to Richland, Washington, and no routing restrictions are imposed. Figure 1 identifies both the fastest and the shortest route as determined by the program. The former would take 35 h 25 min of driving time over a distance of 3129 km, whereas the latter would take 36 h 45 min over a distance of 3024 km.

The computer printout of the shortest route is shown in Table 1. These detailed data include information on the road segments, including identification of toll roads and bridges, incremental distances, cumulative distances, cities traversed, and incremental and cumulative driving times.

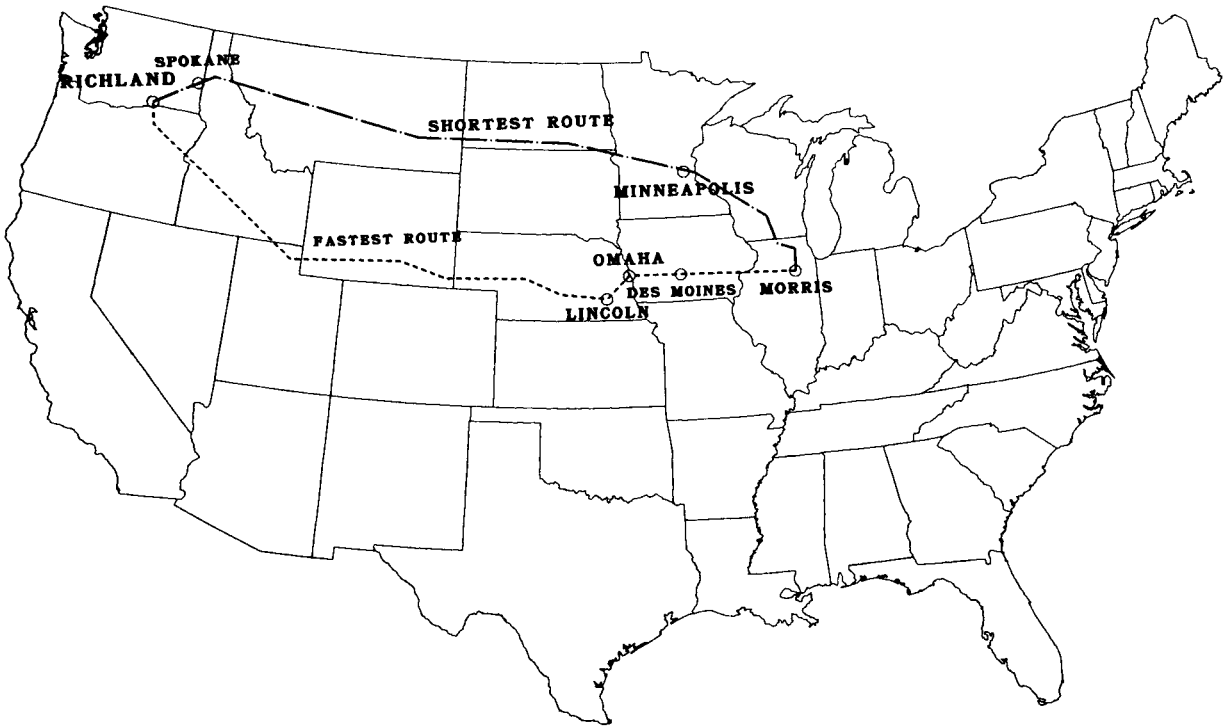

Fig. 1. Comparison of fastest and shortest route between Morris, Illinois, and Richland, Washington.

TAKE ROUTE			ROAD JUNCTION				KM	TIME (HRS)
			I80,X112	NORTH OF	MORRIS	IL	0	0:00
S47	38	KM TO	U30,S47	SOUTH OF	SUGAR GROVE	IL	37	0:35
U30	5	KM TO	U30,S47	NORTH OF	SUGAR GROVE	IL	42	0:39
S47	38	KM TO			STARKS	IL	79	1:15
U20	10	KM TO	I90,U20	S EAST OF	HARMONY	IL	89	1:22
I90*	68	KM TO	I90,U51	EAST OF	S BELOIT	IL	156	2:08
I90	127	KM TO	I90,S78	SOUTH OF	PORTAGE	WI	284	3:33
S78	8	KM TO	U16,U51	N WEST OF	PORTAGE	WI	292	3:38
U16	24	KM TO	I90,U16	WEST OF	WISC DELLS	WI	316	3:57
I90	66	KM TO	I90,I94	EAST OF	TOMAH	WI	382	4:42
I94	281	KM TO			MINNEAPOLIS	MN	663	8:24
U12	1066	KM TO	I94,U12	N EAST OF	MILES CITY	MT	1729	38:14
I94	66	KM TO			FORSYTH	MT	1806	39:07
U12	560	KM TO			GARRISON	MT	2366	46:36
I90	376	KM TO	I90,X15	S EAST OF	COEUR-D'ALENE	ID	2742	58:50
I90B	5	KM TO	I90,X11	N WEST OF	COEUR-D'ALENE	ID	2747	58:59
I90	145	KM TO	I90,X220	S WEST OF	RITZVILLE	WA	2892	61:07
U395	118	KM TO	U12,U395	NORTH OF	PASCO	WA	3010	62:35
U12	15	KM TO	U12,S240	S EAST OF	RICHLAND	WA	3024	62:45

Table 1. Shortest route from Morris, Illinois, to Richland, [illegible]gton.

As mentioned previously, the fastest route is frequently identical to the most practical route. This proved to be the case for the example given above when the most practical unrestricted and the most practical NRC restricted routes were developed. Details of the former are identified in Table 2 and depicted on the map shown in Fig. 2; note it is the same as the fastest route in Fig. 1. For comparison, an NRC restricted route avoiding movement through the larger population centers typically encountered in unrestricted routing is also indicated. The three urban areas which were avoided included Des Moines, Iowa, and Lincoln and Omaha, Nebraska. The distance of the restricted route was 3077 km and had a driving time of 36 h 49 min.

Interestingly, the most practical unrestricted route was 52 km longer, but since the restricted route was more circuitous and kept to somewhat lower-speed roads, the driving time required an additional 1 hr 24 min.

Potential uses of the program involve risk analysis and logistics studies. For example, as shipments approach a receiving site they tend to congregate on relatively few main highways leading directly to the site. This program is capable of predicting how and where this might happen. Logistics studies can use the accurate mileages generated by the program to improve their results. The ORNL logistics program[2] can be used in conjunction with this routing program to display the transportation patterns generated by the predicted movement of large numbers of shipments on maps of the United States.

The program is continuously being updated and improved. As previously mentioned, the mechanism is being developed to add population density information to each link which will permit the determination of radiation exposure to the population. Plans are also being developed to use this program in conjunction with the ORNL Legislative Data Base[3] which, when completed, would permit the identification and evaluation of the impact of federal, state, and local transportation restrictions on specific routes and the results displayed on maps depicting the route.

1. E. L. Hillsman, P. E. Johnson, and B. E. Peterson, Predicting Routes of Radioactive Wastes Moved On the U.S. Railroad System (presented at the Sixth International Symposium on Packaging and Transportation of Radioactive Materials, Berlin, West Germany, November 10-14, 1980; published in the Proceedings).

2. D. S. Joy and B. J. Hudson, The Oak Ridge Spent Fuel Logistics Model (presented at the Sixth International Symposium on Packaging and Transportation of Radioactive Materials, Berlin, West Germany, November 10-14, 1980; published in the Proceedings).

3. C. S. Fore and M. M. Heiskell, Transportation of Radioactive Materials: Legislative and Regulatory Information System (presented at the Sixth International Symposium on Packaging and Transportation of Radioactive Materials, Berlin, West Germany, November 10-14, 1980; published in the Proceedings).

TAKE ROUTE		ROAD JUNCTION			KM	TIME (HRS)
		I80 ,X112		MORRIS IL	0	0:00
I80	453 KM TO	I235,I35	NORTH OF	DES MOINES IA	453	5:34
I35	23 KM TO	I235,I35	NORTH OF	DES MOINES IA	476	5:49
I80	194 KM TO	I29 ,I80	S EAST OF	COUNCIL BLFS IA	670	8:00
I29	5 KM TO	I29 ,I80	S WEST OF	COUNCIL BLFS IA	675	8:03
I80	1440 KM TO			ECHO UT	2115	40:45
I80N	60 KM TO	I15 ,I80N	SOUTH OF	OGDEN UT	2175	41:26
I15	56 KM TO	I15 ,I80N	SOUTH OF	ELWOOD UT	2231	42:34
I80N	826 KM TO	I80N,X188	S EAST OF	STANFIELD OR	3057	59:51
U395	18 KM TO	U395,U730	WEST OF	MC NARY OR	3075	60:04
U370	3 KM TO			UMATILLA OR	3078	60:06
BRDG•	3 KM TO			PLYMOUTH WA	3081	60:12
S14	40 KM TO	U12 ,S14	N WEST OF	KENNEWICK WA	3121	61:20
U12	8 KM TO	U12 ,S240	S EAST OF	RICHLAND WA	3129	61:25

Table 2. Most practical route between Morris, Illinois, and Richland, Washington.

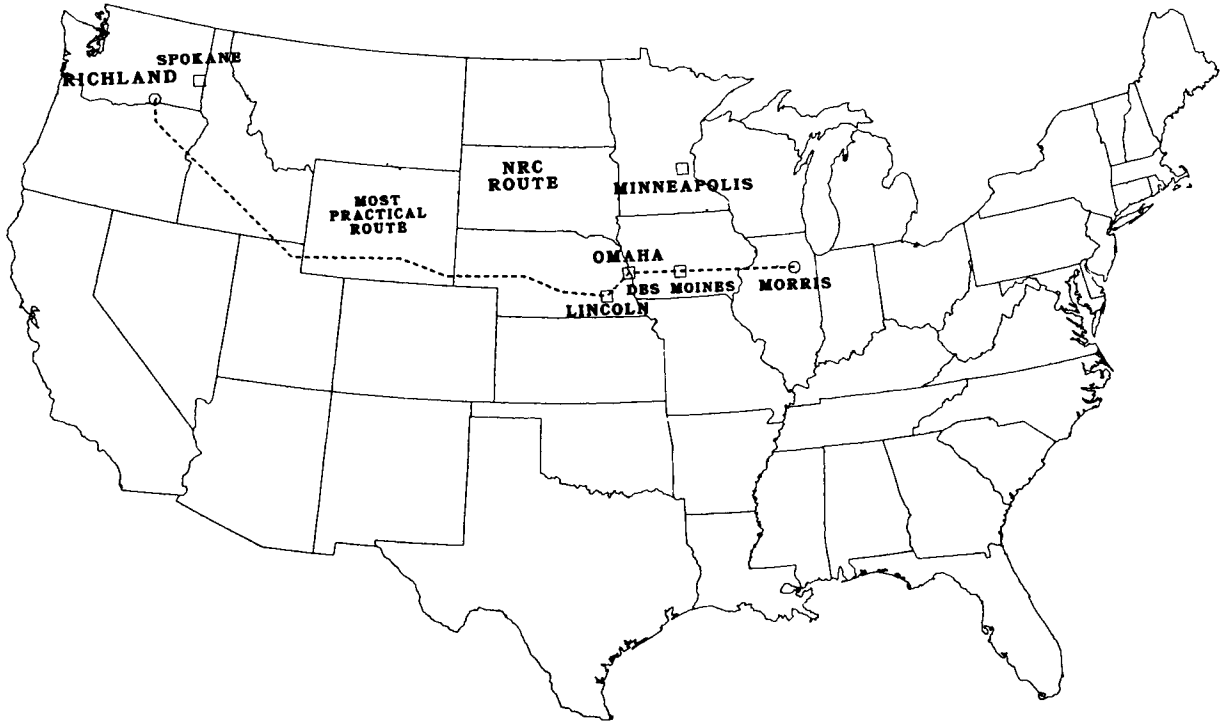

Fig. 2. Comparison of practical route and restricted (NRC) route between Morris, Illinois, and Richland, Washington.

Question by A. A. Anselmo, USA: 1. Since I am responsible for routing truckloads of radioactive material throughout the United States and Canada, could I take advantage of the use of the COMPU.MAP program to track my trucks? 2. What facilities are available to retrack or reroute a truck from original routings due to road construction, wrecks, snow problems, etc?

Answer: 1. The license for the COMPU.MAP program limits the use of the model to DOE transportation projects. I would be happy to put you in contact with the vendor of the COMPU.MAP program. 2. The data base associated with the COMPU.MAP program can be modified easily and quickly to reflect current road conditions. Once this information has been entered a new route can be calculated in a few seconds. Commercial trucking companies using this program continuously update their data base from information received from their drivers.

Question by B. Teer, USA: Can the effects on travel time of restrictions imposed by overweight permits be calculated by the program?

Answer: This is a difficult quesiton for a computer oriented person. Basically the answer is yes. However the restrictions must be expressed mathematically or be capable of being organised in a logic flow chart so the computer program can access the proper information.

PREDICTING ROUTES OF RADIOACTIVE WASTES MOVED ON THE U.S. RAILROAD SYSTEM*

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Railroads offer several advantages for the movement of radioactive wastes when compared to truck transport. These include the ability to accommodate large, heavy containers, the ability to transfer large volumes in a single movement, and the safety of a dedicated, centrally controlled right-of-way. The principal disadvantage of railroads is that the shipper has considerably less freedom to specify the route a shipment will take. The lower density of the railroad network, the need to coordinate waste shipments with other traffic in the system, and the private ownership of the U.S. network — all reduce the shipper's routing power. Although the shipper can dictate an exact route, this could require the use of a special train at an increased cost. Before considering the general use of special trains to move radioactive wastes, it is first desirable to determine how these materials would move as general rail traffic. This paper describes the development of a system to predict routes of general rail traffic at the Oak Ridge National Laboratory (ORNL).

THE U.S. RAILROAD SYSTEM

Several other characteristics of the U.S. railroad network occur rarely, if at all, in the rail systems of other nations, and these complicate the task of estimating the routes to be taken by any given shipments. The most important feature of the network is its ownership by approximately 100 intercity carriers and a number of intracity switching and terminal companies. On most intercity routes, several carriers have the track and equipment needed to provide service, and these companies compete for available traffic. On other routes, such as the one between Philadelphia and Los Angeles, no single company serves both the origin and the destination, and carriers must cooperate if the shipment is to go through. Even for such shipments, a competitive atmosphere still exists due to the presence of alternative routes. In the Philadelphia-Los Angeles example, a shipment could be on the Chessie System to St. Louis and transfer to the Southern Pacific to complete the trip, or it might go to Chicago on Conrail and switch there to the Santa Fe company for the rest of the trip. Other pairings, perhaps involving more railroad companies, could provide comparable service on this shipment.

*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.

**Energy Division.
†Computer Sciences Division.

The fragmentation of ownership in the U.S. network has two implications for the route taken by a shipment. First, changing from one company to another involves some delay — from a few hours to a few days, depending on the degree of cooperation between the companies involved. A shipper who desires a quick delivery will thus prefer to reduce the number of companies involved in the shipment. The second implication is that, where several carriers are needed, the originating rail company usually has a financial incentive to keep the shipment in its portion of the network for as much of the distance as possible before relinquishing it to the next carrier. The industry is characterized by declining marginal costs, making it advantageous to any particular company to increase the total traffic on its system even if the extra revenue is insufficient to cover the fully distributed average cost of the additional traffic. The originating railroad on a shipment is usually in a better position than other companies on a route to obtain this long-haul advantage because its ability to affect service levels gives it greater leverage with the shipper. In general, the shipper does not specify the route but has the option of both specifying which companies will carry the freight and where the transfers will be made.

A final characteristic, and one not peculiar to the U.S. system, is the tendency of railroads to concentrate traffic on a few well-maintained lines, even though this may lengthen the typical route. Again, this is a consequence of declining marginal costs.

IMPLEMENTATION OF A PREDICTIVE ROUTING MODEL

The model uses a shortest-path algorithm on a modified coding of the U.S. intercity railroad network. The network in use was coded by the U.S. Federal Railroad Administration (FRA) in 1975 and subsequently corrected and updated by Princeton University[1] and ORNL. It contains 17,105 links and 14,428 nodes. The rail network has been processed to reduce its size by roughly one-half without affecting the choice of routes by pruning branches that do not serve prospective sites involved in the nuclear fuel cycle and by combining links where appropriate. Ninety-three separate railroads are used, along with an additional category of all other small, intercity railroads, and a category of all terminal and switching companies. This yields 95 separate subnetworks.

To model the route selections of the U.S. railroad industry, the routing algorithm identifies paths of minimum impedance rather than paths of minimum physical length. Within each of the 95 subnetworks, link impedances are calculated by multiplying the physical length of each link by a penalty factor that reflects the historic volume of traffic on the link. The FRA has classified all links in the network into four categories: A and B mainlines and A and B branchlines. Traffic volumes were the most important variables in the classification,[2] with A mainlines tending to be high-quality, high-volume links, B mainlines being high-quality, moderate-volume links, and the two branchline classes having lower volumes. The term "branchline" may indicate either a low-volume "through" link or a true branch on the network. The penalty factors used in computing the impedance of any route are lower for the higher classes of links, thereby encouraging the use of such links on a route. At the present time, the routing model uses default penalty factors based on expert judgement — 1.0, 1.[illegible] 1.9, and 4.0 for A mainlines through B branchlines, respectively. These values differ from those used by Princeton — 1.0, 2.0, 3.0, and 4.0 — which also derive from their judgement. The ORNL values appear to reproduce actual company routes more accurately, but a formal comparison of the two sets of values has not been made. Research is needed to estimate the appropriate values for the penalty factors in a rigorous manner. The model user may override the default if desired.

The algorithm starts a route on the originating railroad and computes paths to the nodes where that railroad transfers freight to other companies. It then continues the paths on each of these additional railroads, transferring the paths at their interchange points to adjoining railroads. This continues until the destination has been reached. A route incurs a penalty each time it switches companies. This represents the actual delay in traffic interchange. At present, all interchange penalties are equal and are roughly equivalent to 500 km of A-mainline travel, or a 1-day delay. Actual delays vary from place to place, depending upon agreements between companies. The algorithm has been designed to use these place-specific penalties when they are known, and future work will refine these transfer penalties to reflect these differences.

In addition to using line classification penalties and transfer impedances, the model assigns a favorable advantage to all links on the originating railroad to reproduce the long-haul phenomenon. The advantage is assigned by multiplying all impedances on the originating company's subnetwork by 0.75. This reduces the impedance of these links and makes them more attractive to the algorithm. As in the case of the line classification penalties, the value of 0.75 is based on expert judgement. Lowering the impedance on the originating line in this way appears to reproduce observed routes more accurately, but the user may change this value if desired.

The routing model contains several additional features to assist in analyzing specific problems. For example, a user can remove nodes from the network to examine proposals to prohibit shipments of radioactive wastes through large urban areas.[4] A user may also remove specific links to examine the effects of weight restrictions, proposed abandonments, or temporary blockages. Other features allow users to examine effects of mergers between railroads on traffic flows and to produce machine-readable files for mapping, as indicated in Fig. 1.

EXAMPLES

Figure 1 is a demonstration of the capabilities of the routing program. Two routes have been computed and plotted between the Hanford Plant in south central Washington State and the proposed defense waste repository at Carlsbad, New Mexico. The solid line represents a normal route. This route follows a mainline of the Burlington Northern railroad through Washington, Idaho, Montana, Wyoming, Colorado, and New Mexico, to Amarillo, Texas, where the shipments would transfer to the Santa Fe Railway; the route then continues southwest into New Mexico to Carlsbad. The total length of this route is 3430 km and uses two railroad companies. An alternative route, which avoids major cities (shown on the figures as stars), is presented as a dashed line on Fig. 1. This route originates on the Burlington Northern and transfers shortly thereafter to the Union Pacific. The shipment would follow the Union Pacific's northwestern mainline through Oregon, Idaho, and Wyoming into southern Nebraska, where it would transfer to the Rock Island Railroad. From here, it takes the Rock Island through Kansas and Oklahoma into New Mexico, where it would transfer again for a short distance on the Southern Pacific before finally transferring to the Santa Fe to reach the destination at Carlsbad. This alternative route specifically avoided four major metropolitan areas but required the use of five different railroad companies and traveled 4275 km. The industry's high-volume mainlines are those directly connecting large cities, and by necessity the route that avoids these cities uses lower-class branchlines.

NORMAL AND ALTERNATIVE ROUTES TO CARLSBAD
ORIGINATING ON THE BURLINGTON NORTHERN

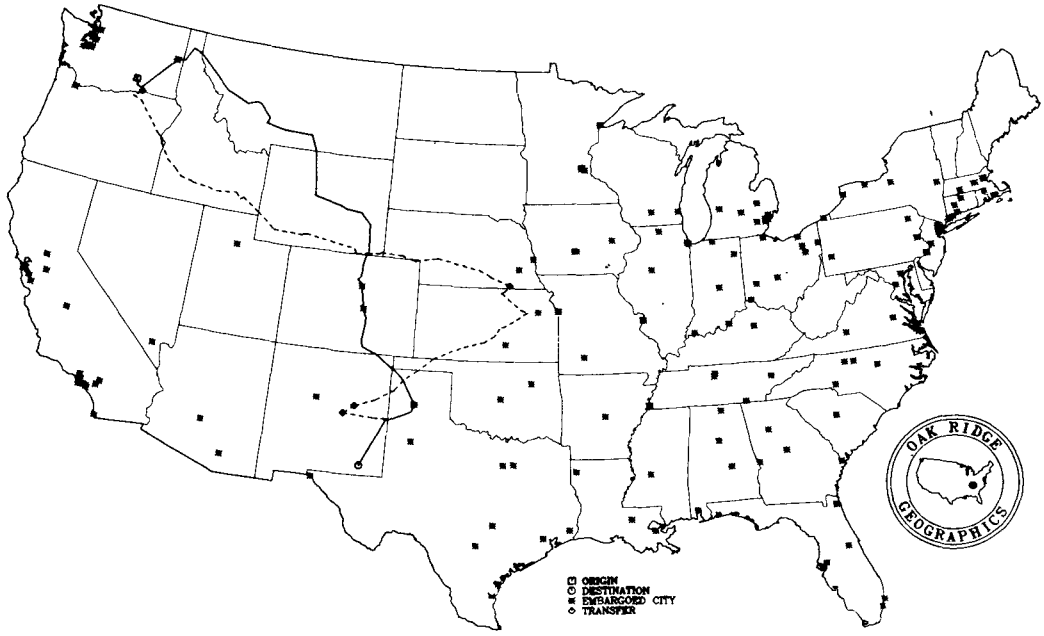

Fig. 1. Sample routes from Richland, Washington, to Carlsbad, New Mexico.

The second example (see Figs. 2 and 3) illustrates the predicted normal routes from each of the 69 existing or planned reactor sites with rail access to a hypothetical repository in the Gulf Coast region. To operate the model, it was necessary to assume a more specific location, and Hattiesburg, Mississippi, was chosen to be the destination for computational purposes only. The volumes of spent fuel were estimated for a year near 2000, using the logistics modeling system of Joy and Holcomb.[5] In this year, each reactor is producing a regular flow of spent fuel waste. The model projects tons of waste, and these projections were converted to a standard-sized cask load for use in mapping the rail movements and interpreting the results.

The selection of Hattiesburg permits illustration of the effect of the terminating railroad on the pattern of shipments. Hattiesburg is served by two different railroad companies, the Illinois Central Gulf (ICG) and the Southern Railway System. One set of routes was run from the reactor sites to Hattiesburg with termination on the ICG (see Fig. 2), and one was run with termination on the Southern (see Fig. 3). In Fig. 2, spent fuel from the northern half of the country would travel to Illinois, transfer to the ICG at various points, move south along the Mississippi River, and then turn southeast to Hattiesburg. This well-developed corridor reflects the network structure of the ICG, which requires all of its cross-country, north-south traffic to pass through one node in southwestern Kentucky. Material from the southern half of the country would enter the ICG system at several points south of Kentucky.

Using the Southern as a terminating railroad would yield a more dispersed pattern of shipments. The major corridor develops in southwestern Ohio, continues south through central Kentucky, eastern Tennessee, and northern Alabama, where a second smaller corridor merges with it for the remaining distance to Hattiesburg. Again, this pattern reflects the network structure of the Southern. Hattiesburg is on the Southern's main line between Cincinnati and New Orleans, and the main corridor follows this route.

As in the example presented in Fig. 1, the program can be directed to find routes that avoid major cities whenever possible. The effect of such restrictions on corridor development varies greatly. New corridors develop in some regions, while corridors such as those in Figs. 2 and 3 may shift, intensify, or fail to form at all. In general, restricting the routes in this way tends to increase the density of traffic in the major corridor that leads to the destination. That is, traffic tends to enter the destination node concentrated on fewer lines than it does when routes are unrestricted. Although the density in this approach corridor may be higher when routes are restricted, however, the corridor itself is sometimes shorter.

FUTURE APPLICATIONS

Future use of the program will be to investigate radiation exposure to the public along routes by applying accurate population densities to the various links. In addition, improved accuracy of rail mileages and speeds from source to destination will permit much more accurate evaluations of shipping cask fleet requirements. The program may also facilitate risk studies where shipments of radioactive material by rail are involved. These studies may involve away-from-reactor storage sites or regional repositories as well as a single national repository. The structure of the routing program is such that, with appropriate data and instructions, it can be used to examine movement of material by special train as well as general freight if this should be desired in the future.

ORNL-DWG 80-14378

FOR DEMONSTRATION PURPOSES ONLY
MOVEMENT OF SPENT FUEL FROM EXISTING OR PLANNED REACTORS WITH RAIL ACCESS
TO A POSSIBLE REPOSITORY SITE NEAR HATTISBURG
SHIPMENTS TERMINATING ON THE ILLINOIS CENTRAL GULF RAILROAD

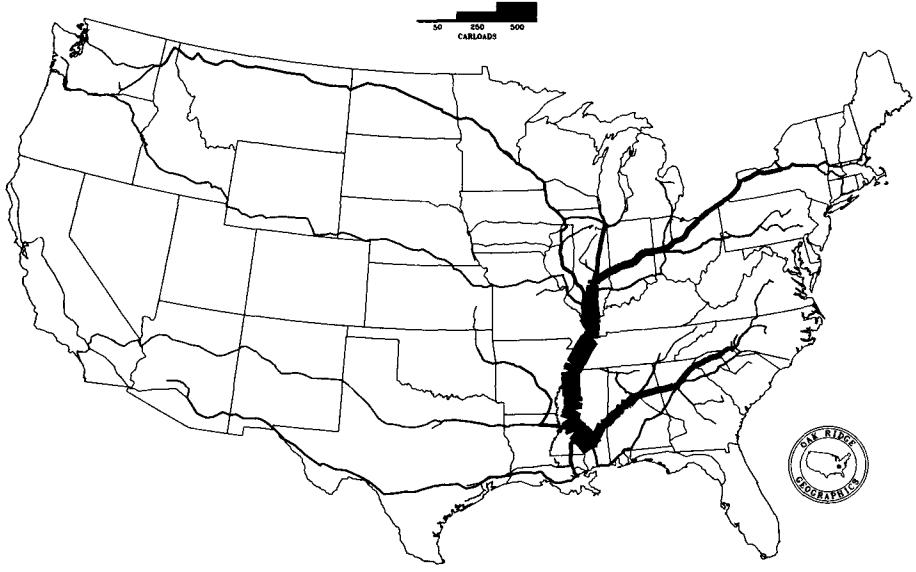

Fig. 2. Pattern of shipments from existing and planned reactors to a hypothetical repository in the Gulf Coast region, with shipments terminating on the Illinois Central Gulf Railroad.

ORNL-DWG 80-14375

FOR DEMONSTRATION PURPOSES ONLY
MOVEMENT OF SPENT FUEL FROM EXISTING OR PLANNED REACTORS WITH RAIL ACCESS
TO A POSSIBLE REPOSITORY SITE NEAR HATTISBURG
SHIPMENTS TERMINATING ON THE SOUTHERN RAILWAY

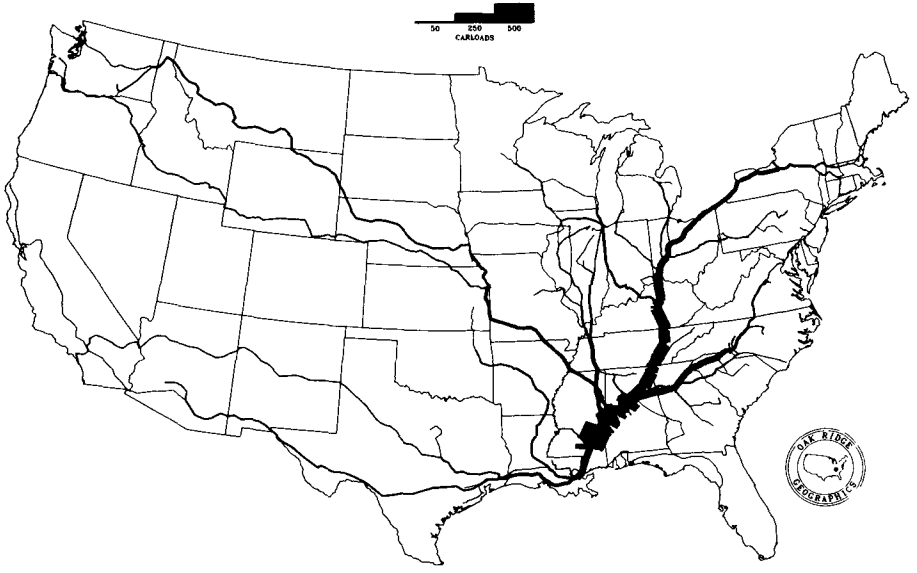

Fig. 3. Pattern of shipments from existing and planned reactors to a hypothetical repository in the Gulf Coast region, with shipments terminating on the Southern Railway System.

1. Jerome M. Luten, Yehonathan Hazony, and Alain L. Kornhauser, An Interactive Network Model for Railroad Traffic Flow Analysis and Cartographic Display, Princeton University Transportation Program, unpublished paper, 1978.

2. U.S. Dept. of Transportation, Final Standards, Classification, and Designation of Lines of Class 1 Railroads in the United States, Vol. 2, A report by the Secretary of Transportation, Washington, D.C. (June 30, 1977).

3. Mark A. Hornung and Alain L. Kornhauser, An Analytic Model for Railroad Network Restructuring, Princeton University Transportation Program, Report 79-TR-11 (1979).

4. Donald J. Kasun, Physical Protection of Shipments of Irradiated Reactor Fuel, U.S. Nuclear Regulatory Commission, Report NUREG-0561 (June 1979).

5. D. S. Joy and B. D. Holcomb, Logistics Models for the Transportation of Radioactive Waste and Spent Fuel, ORNL/TM-6192 (March 1978).

AN ECONOMIC ANALYSIS OF THE TRANSPORT OF RADIOACTIVE MATERIALS IN LWR AND LMFBR FUEL CYCLES IN THE UNITED STATES*

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INTRODUCTION

A study has been performed (Reference 1) to estimate the costs associated with the transportation of radioactive materials in three nuclear fuel cycles. The fuel cycles studied are:

(1) the light-water reactor (LWR) once-through fuel cycle,
(2) the LWR fuel cycle with Pu and U recycle, and
(3) the liquid metal fast breeder reactor (LMFBR) recycle fuel cycle.

For each of the fuel cycles, the shipment of heavy-metal-containing materials (including high-level waste for cycles (2) and (3)) was considered. The shipment of low-level waste was considered in cycles (1) and (2) only. Costs were estimated for existing or planned transportation systems and for alternate systems judged to be economically attractive.

Results are given on an annual basis in terms of 1979 United States (US) Dollars ($) for each shipment leg, normalized in terms of each 1,000-MWe reactor and also in terms of total cost for the reactor population supporting a 1500 metric tonnes of heavy metal (MTHM) per year flow rate. The results of this study permit assessment of the differences in costs between the fuel cycles considered and definition of the transportation systems for which packaging technology improvements will be economically beneficial.

COST BASIS

All costs were based on the assumption of mature fuel cycles, each defined to have a flow rate of 1500 MTHM/year. For low enriched UO_2 this flow rate is equivalent to the annual fuel requirements of 61 LWRs, each producing 1,000 MWe with a 70-percent capacity factor, an average nominal burnup of 33,000 MWD/MTU, and an average annual reload of 25 MTHM. For the LWR fuel cycles, this provides a fuel cycle with approximately 20-percent greater capacity than presently exists in the US (51,100 MWe). The reactor population is assumed to consist of 65 percent pressurized water reactors (PWR) and 35 percent boiling water reactors (BWR).

*Work sponsored by U. S. Department of Energy under Contract No. DE-AC04-76-DP00789.

The economic factors studied for these fuel cycles included packaging development costs where needed; transport system fabrication, maintenance and package decommissioning costs; and shipping tariffs by rail or road in the US. The costs of a cask maintenance facility were included. Where possible, the existing geographical distribution of nuclear facilities in the US was used as the basis for estimating the distances for each transportation segment in the analysis. Generic locations were assumed for those facilities, such as high-level waste repositories, cask maintenance facilities, and reprocessing facilities, that do not presently exist. The distances used in the analyses are representative of the current and projected status of facilities in the U.S.

The analysis of the once-through fuel cycle considered the shipment of natural UF_6 from a mill to an enrichment plant, enriched UF_6 to a conversion/fabrica[redacted] facility, fresh UO_2 fuel to a reactor, and spent fuel to a terminal reposit[redacted] Waste shipments considered were for low-level waste from the UF_6 conversion facility, from the UO_2 fuel fabrication plant, and from the reactor to a low-level-waste shallow-land burial site.

The LWR recycle fuel cycle analyzed is based on forty-three nominal 1,000-MWe UO_2 reactors and eighteen nominal 1,000 MWe mixed oxide fuel (MOX) reactors. Shipping segments (legs) considered in this fuel cycle included all of those mentioned above for the LWR once-through cycle, excluding the spent-fuel to a repository, plus the spent UO_2 fuel to a reprocessing plant, the recycle UF_6 to the enrichment plant, natural UO_2 to a MOX fabrication plant, fresh MOX fuel to a reactor, spent MOX fuel to a reprocessing plant, recycle PuO_2 to the MOX fuel fabrication facility, and high-level waste to a waste repository. Additional waste shipments considered include MOX fuel fabrication waste, MOX reactor waste, clad waste and reprocessing waste from their respective sources to a waste repository.

Figure 1 shows a flow diagram of the heavy-metal and waste materials in the LWR fuel cycle with uranium and plutonium recycle. Similar flow diagrams were developed for the other two fuel cycles considered in this study. Each shipment leg is identified in the flow diagram by either "HM" for a heavy-metal shipment leg or by a "W" for a low-level waste shipment leg. In addition, the waste shipment legs are subdivided into transuranic (TRU) low-level waste and non-TRU low-level waste shipment legs. The once-through LWR fuel cycle encompases only shipment legs HM1, HM2, HM3 and HM4 (with HM4 shipments terminating at a terminal storage or disposal facility rather than a reprocessing facility) and also includes the W1 and W2 waste shipment legs.

The distances and volumes to be transported which are projected for the LWR recycle fuel cycle are shown in Table I. As was noted above, the distances shown are based upon the existing geographical distribution of nuclear facilities in the US. For those facilities which do not presently exist, generic locations were assumed. Similar data were used for the LWR once-through fuel cycle and the LMFBR fuel cycle modeling.

The LMFBR fuel cycle analyzed is based upon forty-three 1,000-MWe LMFBR[redacted] twenty 1,000-MWe LWR MOX reactors. The latter again had a population of 65 p[redacted] PWR and 35 percent BWR. All plants had a 70-percent capacity factor. The [redacted] reactors were assumed to operate with a burnup of 80,000 MWD/MTHM. Only the shipment of heavy metal was studied for the LMFBR cycle. This included, in the breeder portion of the fuel cycle, shipments of natural UO_2 from conversion to fabrication; fresh MOX LWR assemblies and fresh LMFBR assemblies to the reactors; spent MOX LWR and LMFBR assemblies to a reprocessor; recycle PuO_2 to a fabricator; recycle UF_6 to conversion; and clad waste, high-level waste, and control/shield assemblies to waste repositories.

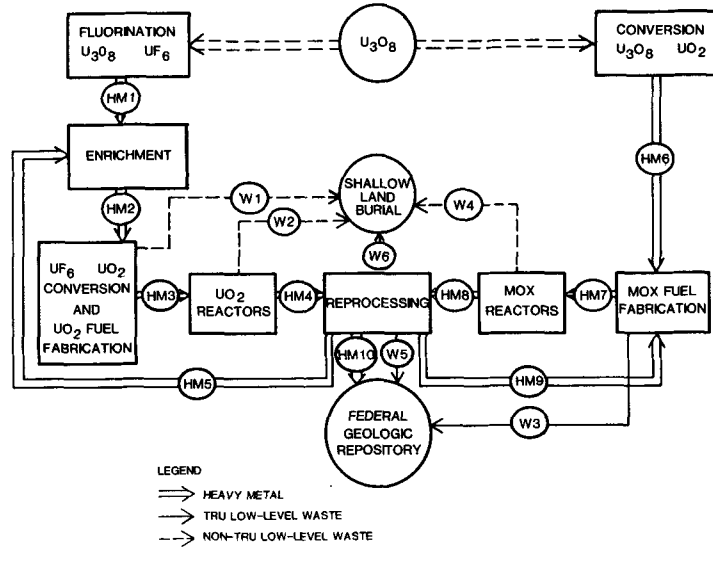

Figure 1. Transportation Links in a Light-Water Reactor Fuel Cycle with Uranium and Plutonium Recycle

Table I. Shipping Distance and Annual Volume Production for LWR Once-Through Fuel Cycle Heavy Metal and Waste Flows

Transportation Segment	Average One-Way Distance (km)	Average Annual Volume Per 1000-MW_e Reactor
Natural UF_6 (HM1)	1000	130 MTHM
Enriched UF_6 (HM2)	1400	25 MTHM
Fresh UO_2 Fuel Assemblies (HM3)	2100	25 MTHM
Spent UO_2 Fuel Assemblies (HM4)	1000	25 MTHM
Recycle Uranium (HM5)	1300	34 MTHM
Natural UO_2 (HM6)	2250	24 MTHM
Fresh MOX Fuel Assemblies (HM7)	2400	25 MTHM
Spent MOX Fuel Assemblies (HM8)	1000	25 MTHM
ycle Plutonium (HM9)	2250	1.4 MTHM
n-Level Waste (HM10)	1800	0.9 MTHM
UF_6 Conversion and UO_2 Fuel Fab. Waste (W1)	500	85 m^3
UO_2 Reactor Waste (W2)	1000	600 m^3
MOX Fuel Fabrication (W3)	500	135 m^3
MOX Reactor (W4)	1000	600 m^3
Reprocessing TRU Waste (W5)	1800	11 m^3
Reprocessing Non-TRU Waste (W6)	800	63 m^3

RESULTS

The annual costs determined for the various shipping legs are shown in Tables II, III, and IV for the LWR once-through cycle, the LWR recycle fuel cycle, and the heavy metal shipments in the LMFBR cycle, respectively. The results presented in these tables assume the use of current technology and/or design philosophy for the transportation systems used.

Comparison of the results from Tables II and III shows that the total transportation costs (including both heavy-metal and waste shipments) are approximately 60 percent higher for the recycle LWR fuel cycle than they are for the once-through LWR fuel cycle.

Since only the heavy-metal shipments were studied for the LMFBR fuel cycle, the comparison of costs between the three fuel cycles must be performed on this basis. For each of the three mature fuel cycles, the heavy-metal transportation costs, using current technology, are shown in Table V. Here it can be seen that the heavy-metal only transportation costs for the LWR recycle fuel cycle are 55 percent higher than the LWR once-through fuel cycle, and the costs for the LMFBR fuel cycle are 400 percent higher than the LWR once-through fuel cycle.

Estimates were made of the economic benefits which could be derived from improving technology and/or design philosophy. It was determined that significant transportation cost savings can result from using advanced systems designs that involve simpler, and therefore less costly, fabrication methods compared with traditional concepts. Also, in the United States, fuel will probably be stored at the LWR reactor sites for five years or more for many years into the future, and costs can be reduced by using shipping casks that are designed to transport long-cooled fuel. Using these suggested alternatives, the total heavy-metal transportation costs are reduced as shown in Table VI. Here, the costs are shown, not per 1000-MW_e reactor, but are shown on the basis of a mature fuel cycle of approximately 62,000 MW_e.

Table II. Summary of Total Radioactive Material Transportation Costs for the LWR Once-Through Fuel Cycle Using Current Technology

Segment	Average Annual Cost $/Reactor*	Percent of Total
Natural UF_6	26,000	5
Enriched UF_6	12,000	2
UO_2 Fresh Assemblies	26,000	5
UO_2 Spent Assemblies	336,000	58
UO_2 Fabrication Waste	6,000	1
UO_2 Reactors Waste	166,000	29
Total Annual Cost	$572,000/Reactor	
Total (Mill/kWh)	0.09	

* Results are normalized to an average 1,000-MWe Reactor.

Table III. Summary of Total Transportation Costs for the LWR Recycle Fuel Cycle Using Current Technology

SEGMENT	AVERAGE ANNUAL COST $/Reactor*	PERCENT OF TOTAL
Natural UF_6	14,000	1
Enriched UF_6	8,000	1
UO_2 Fresh Assemblies	18,000	2
UO_2 Spent Assemblies	256,000	28
Recycle UF_6	3,000	-
Recycle PuO_2	34,000	4
Natural UO_2	1,000	-
MOX Fresh Assemblies	56,000	6
MOX Spent Assemblies	166,000	18
High-Level Waste	64,000	7
UO_2 Fabrication Waste	4,000	-
UO_2 Reactor Waste	116,000	13
MOX Fabrication Waste	2,000	-
MOX Reactor Waste	50,000	5
Reprocessing Clad Waste	92,000	10
Reprocessing TRU Waste	2,000	-
Reprocessing Non-TRU Waste	29,000	3
Total Annual Cost	$915,000/Reactor	
Total (Mill/kWh)	0.15	

* Results are normalized to an average (70-percent UO_2 and 30-percent MOX) 1,000 MWe Reactor.

Table IV. Summary of Heavy Metal Only Transportation Costs for the LMFBR Fuel Cycle Using Current Technology

SEGMENT	AVERAGE ANNUAL COST $/REACTOR*	PERCENT OF TOTAL
Natural UO_2	1,000	-
LWR MOX Assemblies	60,000	3
LMFBR Assemblies	231,000	11
Spent LWR MOX Assemblies	175,000	9
Spent LMFBR Assemblies	1,128,000	56
Recycle PuO_2	113,000	6
Recycle UF_6	4,000	-
Clad Waste	150,000	7
High-Level Waste	88,000	4
Control/Shield Assemblies	62,000	3
Total Annual Cost	$2,011,000/Reactor	
Total (Mill/kWh)	0.33	

* Results are normalized to an average (68-percent LMFBR and 32-percent MOX) 1,000-MWe Reactor in the fuel cycle.

Table V. Comparison of Heavy-metal Transportation Costs for the Three Fuel Cycles

ALTERNATIVE	NO. OF REACTORS IN MATURE FUEL CYCLE	HEAVY METAL TRANSPORTATION COSTS PER NOMINAL REACTOR PER YEAR	HEAVY METAL TRANSPORTATION COSTS PER YEAR PER 1500 MTHM
LWR Once-Through	61	$400,000	$24,400,000
LWR Recycle	61	620,000	37,800,000
LMFBR Recycle	63	2,000,000	126,000,000

Table VI. Comparison of Annual Heavy-Metal Only Cost Savings for each Fuel Cycle Derived from the Development of Advanced Technology*

ALTERNATIVE	TOTAL HEAVY METAL TRANSPORTATION COSTS PER YEAR PER 1500 MTHM FUEL CYCLE		NET ANNUAL SAVINGS
	Current Technology	Advanced Technology	
LWR Once-Through	$24,400,000	$19,500,000	$4,900,000
LWR Recycle	37,800,000	32,800,000	5,000,000
LMFBR Recycle	126,000,000	91,000,000	36,000,000

* Net savings do not include the cost of developing the advanced technology.

Based upon these results, the annual cost savings from the use of advanced casks should allow the costs of developing these casks to be recovered in a very short period of time. For example, for an advanced LWR spent fuel shipping cask, capable of carrying more spent fuel which has been cooled 5 years, the design is estimated to cost approximately $2,700,000. The net annual savings for the mature LWR once-through cycle using this cask is estimated to be $4,900,000. This allows the development costs to be recovered in just one year of "mature" operation. It is estimated that the design of an advanced cask for transporting short-cooled LWR spent fuel would cost more than this; however, the unit capital cask cost could be much less than that of current designs. The net annual savings here would be $5,000,000, also providing for quick recovery of the development costs (no more than two or three years). Reductions in the costs of shipping low-level waste are also possible. For example, advanced waste shipping systems for clad waste could result in annual savings of approximately $4,000,000 for the mature LWR recycle case.

For the LMFBR fuel cycle, major cost savings can result from the design of a cask having a large capacity, while simultaneously providing for a short in-plant turnaround time. For both recycle fuel cycles, the advancement of technology for transporting high-level and cladding waste forms is also suggested.

CONCLUSION

In conclusion, the costs associated with the transportation of heavy-metals in two LWR fuel cycles and one LMFBR fuel cycle have been estimated, both for existing levels of technology and for advanced shipping technology. The costs of transporting low-level wastes in the two LWR fuel cycles have also been estimated. The cost assessment included not only the package capital costs and the operating transportation costs (tariffs), but also included such items as technology development, fleet servicing and maintenance costs, and package decommissioning costs.

Areas in which transportation costs can be reduced through the use of advanced packaging designs have been identified, and the amount of such cost reduction has been estimated. Savings in transportation costs of approximately 20 percent for the LWR once-through fuel cycle, of 13 percent for the LWR recycle fuel cycle, and of 29 percent for the LMFBR recycle fuel cycle can result from the use of advanced shipping systems instead of the use of current package designs or current technology. Identification of those transportation segments in a given nuclear fuel cycle which offer the possibility of significant cost reduction should permit orderly progress toward a more efficient nuclear transportation industry.

REFERENCE

1. Dupree, S. A., et al., "Economics of Radioactive Material Transportation in the Light-Water Reactor Nuclear Fuel Cycle," SAND80-0035, TTC-0064, Sandia National Laboratories, Albuquerque, NM, USA (to be published).

DISCUSSION

Question by B. Teer, USA: For the once-through LWR fuel cycle you show shipments of 600 m^3/year at a cost of $ 166,000 which is the second most significant item after spent fuel shipments. What volume reduction factor was used betwwen waste generated at the reactor and waste shipped?

Answer: Historical data for US reactors was used.

Comment by B. Teer, USA: If the amount of waste shipped was based on historical data, I suggest that future shipments of waste will be lower, because volume reduction will be used to the maximum possible extent in the future.

Answer: I agree with this statement.

U.S. CASK REQUIREMENTS AND INDUSTRY CAPABILITY SURVEY*

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Shipments of commercial nuclear fuel cycle materials have been safely transported in the United States for many years. However, the deferral of reprocessing and delays in developing geologic waste disposal repositories have resulted in only limited quantities of some fuel cycle materials being shipped in the U.S. In 1979, for example, approximately 50 shipments of commercial power reactor spent fuel were made in the U.S. High-density storage racks have been or are being installed to the extent possible because reactor discharge basins are starting to fill up. This has delayed the transportation of spent fuel to an uncertain future date and has resulted in the existing small fleet of spent fuel casks now available in the U.S. not being fully utilized.

The present lack of definitive information on the volumes and destinations of spent fuel and high-level waste has created uncertainties for private companies which would provide nuclear material transportation services. While it is expected that private U.S. industry will design, license, fabricate, own, and operate all commercial fuel cycle transportation systems, the U.S. Department of Energy (DOE) is charged with the responsibility to assure the availability of systems required to meet U.S. policy. Specifically, transportation systems must be available for supporting future away-from-reactor (AFR) and geologic nuclear waste repositories.

The purpose of this paper is twofold. The first objective is to provide an estimate of spent fuel shipping cask requirements for reactor to away-from-reactor (AFR) storage facility shipments from the present time until late in this century. These estimates will provide a basis for assisting government agencies and industry in assessing transportation alternatives. The second objective is to determine and document the willingness and capability of private industry to provide required future transportation services. In order to meet this objective, the Transportation Technology Center at Sandia National Laboratories sponsored Teledyne Energy Systems to conduct a survey of U.S. industry. The results of tasks completed to carry out the stated objectives will be reviewed.

In order to determine future shielded cask requirements, Oak Ridge Natio Laboratories has developed transportation logistics models to simulate operat waste transportation systems,[1] which operate using a spent fuel data base[2] consistent with the information provided to the U.S. Department of Energy Spent Fuel Storage Program. The data base describes historical discharges and projects future discharges of spent fuel from operating and proposed reactors. Spent fuel shipments

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from U.S. reactors to AFRs represent only one of the paths shown in Figure 1 that will require shielded shipping casks. Details of the cask requirements for many of the other paths are not yet fully developed, and depend upon the date when waste repositories become available. For path B (domestic power reactor to AFR) there are three estimates of quantities of material to be moved (shown in Figure 2), and there are four possible destinations under consideration (AFR potentially located in the southeast, northeast, midwest, and western U.S.). The hardware requirements for each of the twelve possible scenarios have been calculated. The spent fuel logistics programs developed at ORNL have been used to predict the number of both truck and rail spent fuel casks required as a function of time for these specific scenarios. The cask estimates for the U.S. reactor to AFR shipments (Path B) do not include the equipment required for reactor to reactor shipments where transshipment is an option. Results for one of the twelve scenarios is shown in Table 1, which represents the reactor to AFR cask requirements for the planning base with the AFR located in a northeastern U.S. site. The effects of the different volume options on total (rail and truck combined) cask requirements are shown in Figure 3. The impact of different mutually exclusive location options for the planning base volume is shown in Figure 4.

Table 1

Reactor to AFR Shipments Cask Fleet Requirements
Projected Planning Base
AFR Located in Northeastern U.S. Site

Year	Number of Rail Casks	Number of Truck Casks
1981	1	2
1982	1	1
1983	1	1
1984	2	2
1985	2	5
1986	2	7
1987	2	11
1988	3	9
1989	4	9
1990	5	12
1991	8	12
1992	11	18
1993	11	20

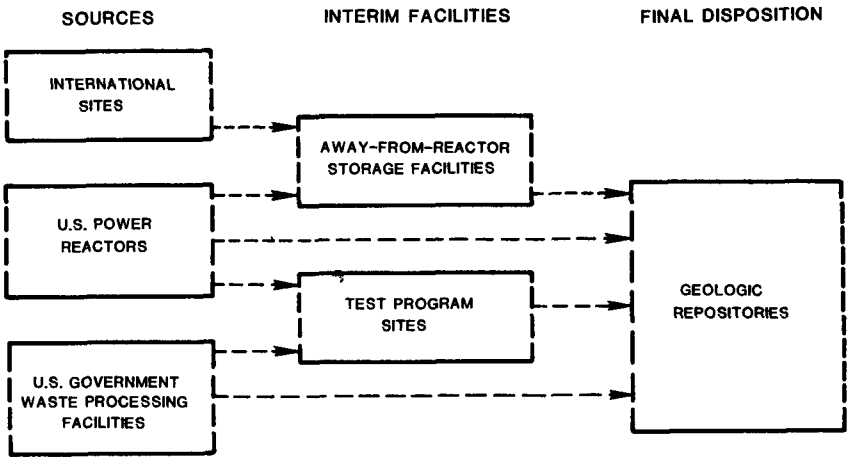

Figure 1. U.S. Transportation Paths for Shielded Spent Fuel and High-Level Waste Shipping Casks

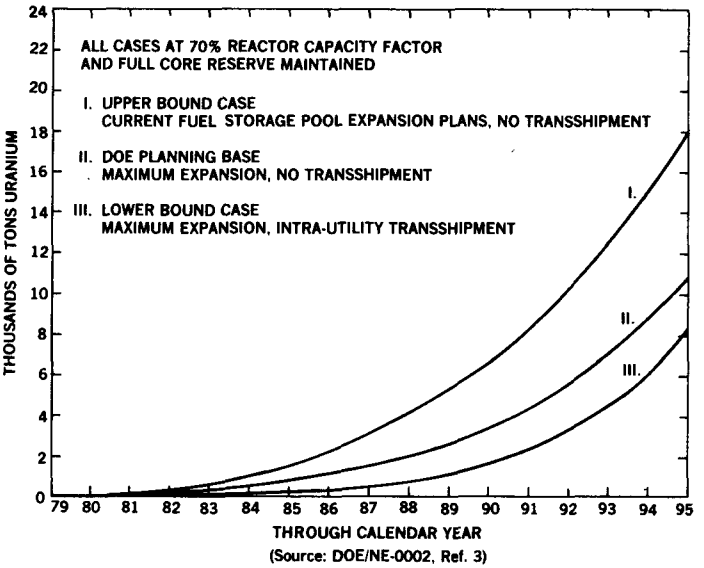

Figure 2. Cumulative Spent Fuel Shipment to AFR Storage Facilities from U.S. Power Reactors

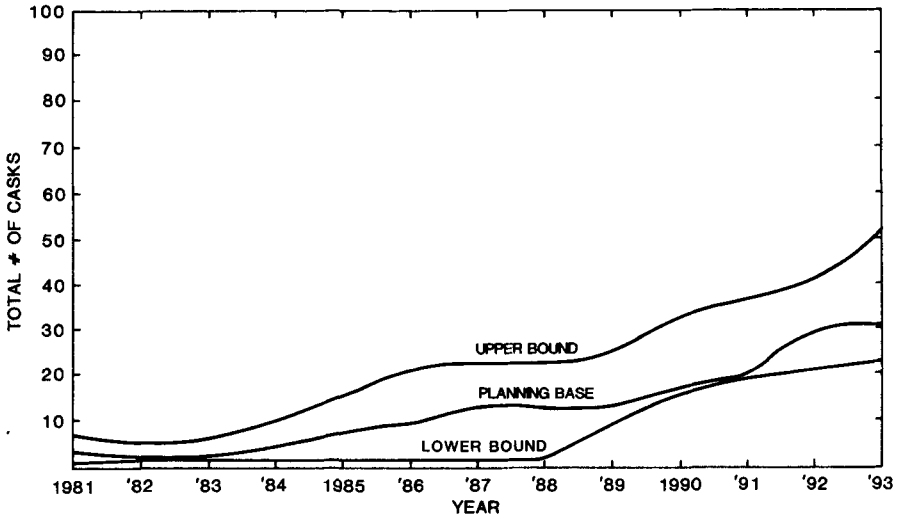

Figure 3. Total Cask Fleet Requirements for Shipments from Reactor to AFR by Data Base AFR Located in Northeastern Region

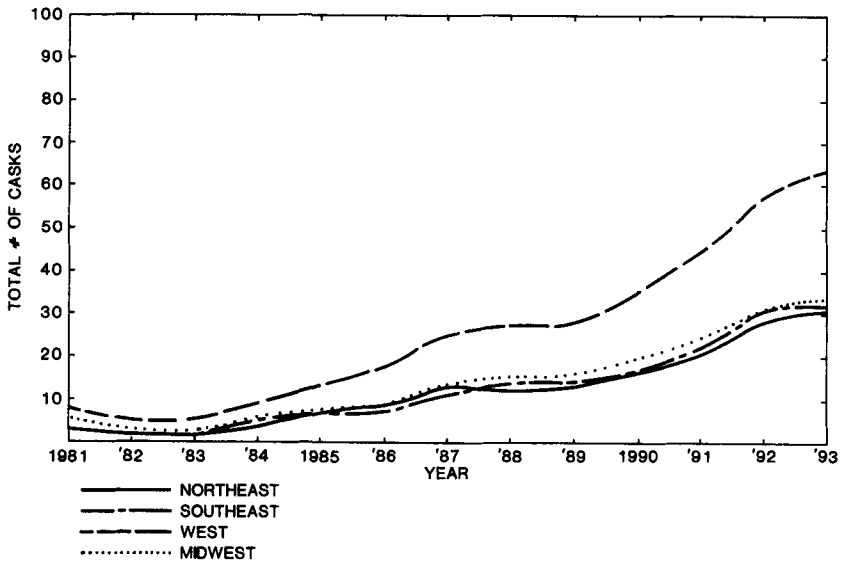

Figure 4. Total Cask Fleet Requirements for Shipments from Reactor to AFR by AFR Location Option Planning Base

As noted earlier, definition of casks required for the other shipment legs depends upon the date when a repository for these materials becomes available in the U.S. Repository startup dates range from as early as 1989 to as late as 2006. The current reference time frame used by the U.S. National Waste Terminal Storage program is for the first repository to become operational between 1997 and 2006. If a repository was available in the 1990s, the number of shipping casks required to meet the requirements of U.S. Government waste processing facilities up to the year 2000 would be less than 40 rail cask equivalents.

To determine and document the willingness and capability of private industry to provide required future transportation service, the Transportation Technology Center at Sandia National Laboratories sponsored Teledyne Energy Systems to conduct a survey of U.S. industry. The objective of the survey is to gather information that can be used in the development of future Federal policy in the transportation area. The survey scope is limited to the back end of the nuclear cycle and emphasis is on shipment of spent fuel and wastes other than non-transuranic low-level waste. The survey program consists of four major tasks: (1) Development of survey questions and background information, (2) Collection of survey data from approximately 135 private companies, (3) Evaluation and assessment of the survey data, and (4) Documentation of results.

The background information prepared for distribution to industry in the survey package included a summary of the overall nuclear material transportation system, a supplier matrix of transportation services in the U.S., descriptions and projected requirements for shipping packages. Survey questions were directed at defining capabilities of package suppliers, transport service groups, carriers, special vehicle suppliers, and testing organizations. Questions regarding general issues (such as finance, marketing, personnel, and government involvement) that affect the U.S. nuclear material transportation industry were also included. The survey questionnaire was approved by the U.S. Department of Energy and the U.S. Office of Management and Budget.

The evaluation of survey data was performed by summarizing written questionnaire responses and evaluating the summarized data. Evaluation was performed with the assistance of a team of consultants to Teledyne, each having long association with U. S. nuclear transportation activities. About 50 companies provided usable responses to the survey questionnaire. This return is clearly less than anticipated since the approximately 150 questionnaire recipients were pre-selected on the basis that their capabilities matched the requirements of a nuclear transportation system. Reasons for lack of participation in the survey cannot be completely determined. However, it is clear from survey results that the current U.S. marketplace for nuclear transportation activities is generally not offering an attractive opportunity for private industry. Those who are presently most heavily involved offer considerable skepticism for future business.

The actual survey results indicate that substantial capabilities exist in every transport function except finance and insurance (see Figure 5). These latter two represent a problem in response, not capability, since the companies falling in these two categories felt they had little constructive information to offer now but their services were available as required. Of the companies responding, the majority derive less than 10% of their revenue from nuclear material transportation with over 20 responses falling in the 0 to 1% category. Even for package fabricators who responded, all but one derive less than 10% of their business from nuclear material transportation. It is interesting to note that all but one respondent anticipated an increased percentage in the future, yet, by a 5 to 1 ratio, the same group felt that a substantial number of new employees would not be required to support a future role. This apparent inconsistency is probably explained by the small amount of total business that nuclear transportation provides to domestic industry.

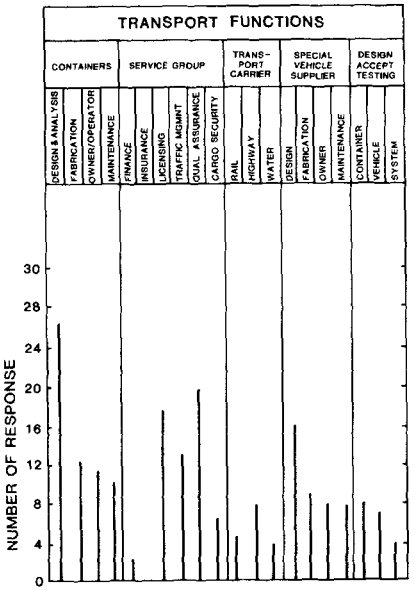

Figure 5. Present Capabilities of U.S. Companies Responding to the Industry Survey

Each company was asked to rate the effect that public concern has on their role in transportation. In evaluating responses to this and similar questions, it is apparent that public concern and acceptance is regarded as an important factor. Most responding companies provided answers to the general questions which addressed critical problem areas, policy, motivation and deterrents to participation in transportation activities. The three problems most often cited were regulatory uncertainty, lack of spent fuel and high-level waste facilities, and public acceptance. For those companies having container fabrication capabilities, regulatory uncertainty was the problem most often cited.

Areas of government policy most affecting a supplier function were generally responded to in a negative context. Lack of a clear overall government policy was most cited and included a perceived lack of support for industry. Other policy areas frequently mentioned were Nuclear Regulatory Commission (NRC) regulatory indecisiveness and problems with specific regulations. The latter category includes time consuming environmental regulations, NRC spent fuel safeguard regulations, mandatory quality assurance documentation and Occupational Safety and Health Adminstration regulations.

There was very definite agreement on what would motivate industry to be more involved in nuclear transportation. Industry wants a clear view of the market so that business prospects can be reasonably assessed. As a corollary to a better market, industry desires clear Federal and state policy with emphasis on government commitment toward nuclear power. DOE's expressed philosophy is that transportation capability will be provided by private industry. Furthermore, government policy will strive to reduce risks and improve incentives for private industry participation. The survey included questions on incentives the government could offer

in the transportation business area. Here responses were more diverse. Most frequent responses mentioned were a dependable licensing schedule (regulatory stability) and government programs that provide R&D funds and opportunities for hardware fabrication. Funding by the Federal government was usually tied to statements about implementation of a national waste storage policy. A question on deterrents to industry participation yielded similar results to the earlier question on industry willingness to participate in this activity. At present, industry can not assess risks and the profit and market growth potentials in nuclear material transportation. Company responses indicated that these problems, coupled with poorly defined national waste policy and the psychological impact of changing political positions over the last ten years, makes the market relatively unattractive. These points were emphasized by those companies with a history of activity and an understanding of the market.

Although the current nuclear transportation market leaves much to be desired from domestic industry's viewpoint, there is little question that ample private industry capability exists to meet the marketplace's needs through the year 2000. The survey asked a number of specific questions related to container fabrication capabilities for truck and rail spent fuel casks, high-level and intermediate waste casks, and TRU overpacks. A key result of the survey is the composite response to the question on domestic manufacturing capability to build various containers for spent fuel and waste transport. Table 2 shows this capability based on the thirteen responses received from companies with presently existing production facilities.

Table 2

Number of Units Per Year That Could Be Produced
With Present Capabilities of Responding Companies

	Spent Fuel Casks		High-Level Waste Casks		Intermediate Level Waste Casks	TRU Overpacks
	Truck 20-30T*	Rail 50-100T	Truck 20-30T	Rail 50-100T	Truck & Rail 15-25T	Truck & Rail 5-15T
Business as Usual	63	28	60	24	113	40
Priority Over Other Jobs	117	61	89	33	159	63

* T = tons; 1 ton = 0.91 tonne

The near-term U.S. cask needs are based on reactor-to-reactor, reactor-to-AFR and waste disposal test program requirements. Shipments to geologic repositories are not expected to occur until approximately the year 2000. It is clear that domestic capability far exceeds the near-term national requirements based on only the thirteen responses. Many companies with known manufacturing capability did not choose to respond to the industry survey and the survey was only mailed to a very small fraction of the overall potential manufacturing population.

Consultants currently evaluating the survey data agree that existing manufacturing capability far exceeds the known hardware requirements of the transportation system. Utilization of this capability must, of course, take into account the problem areas, deterrents and incentives which have been described earlier.

REFERENCES

1. D. S. Joy and B. D. Holcomb, Logistics Models for the Transportation of Radioactive Waste and Spent Fuel, ORNL/TM-6192, Oak Ridge National Laboratory, Oak Ridge, TN, March 1978.

2. DISFUL, Computer Code, S. M. Stoller Corporation, New York, NY, September 1979.

Spent Fuel Storage Requirements -- The Need for Away-From-Reactor Storage, DOE/NE-0002, U.S. Department of Energy, Washington, DC, January 1980.

DISCUSSION

Question by P. F. Highberger, USA: In your estimate of potential cask manufacturers what type cask did you assume, e.g. lead, uranium, stainless steel?

Answer: All three cask types were assumed. The data presented here represent a compilation of the responses for the different cask types. Information exists in the survey for each of the individual types.

Question by M. Grenier, France: Is there any legal obligation, in the USA, for training drivers of trucks shipping radioactive materials, as is the case now for the most dangerous materials in France?

Answer: At present there is no such requirement in effect. However, the proposed new routing rules being studied by the US Department of Transportation in its rulemaking (HM-164) includes provisions requiring training and certification of drivers for the more dangerous nuclear materials transport.

Question by K. Gablin, USA: Can the "industrial survey" be used to verify ample and existing m.f.g. capabilities for LLW shipping casks for ion exchange resins? That is, can the 13 responses meet these needs easily?

Answer: The industry survey was directed at the transportation requirements of spent fuel, high-level, intermediate-level and transuranic contaminated wastes and thus did not specifically address low-level waste shipping systems. However, some of the results, such as in the area of manufacturing capability, would be applicable to low-level waste systems.

SESSION VI
RISK/SAFETY ASSESSMENT II

TRANSPORT OF SPENT LIGHT WATER REACTOR FUEL BY GOODS TRAIN - ESTIMATION OF POTENTIAL RELEASE IN THE EVENT OF ACCIDENTS

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Introduction

As part of the R + D programme called "Projekt Sicherheitsstudien Entsorgung", which is sponsored by the "Bundesminister für Forschung und Technologie" of the Federal Republic of Germany, a study is being carried out concerning the risk due to the release of radioactive material from spent light reactor fuel shippped in casks by goods train.
At present only a description of provisional results is possible, ie. details on the various types of railway accidents as well as the failure thresholds of the cask. A final evaluation of the analysis will be available soon.

Methodology

For estimating accident risks probabilistic methods are applied of which the fault tree analysis permits a detailed investigation of accident possibilities. The following steps have proven to be most suitable for the treatment of a safety analysis:

1) preliminary analysis
2) identification of accidents
3) estimation of accident frequencies
4) estimation of accident consequences
5) assessment of risk

Preliminary analysis

The esential component of the preliminary analysis is the system description, ie. stipulation of reference scenario.

At the onset of the study the projected German reprocessing plant at Gorleben was planned to go into operation at the end of the present decade; so 1990 was chosen to be the reference year for the study with the corresponding reprocessing capacity.

Recently only nuclear power plants of the 1300 MW-class are being planned and built in Germany. Therefore it was decided to consider only LWR's of this class.

Fig. 1 shows the LWR's considered in the study with their geographical location. The shipment of spent fuel from the 17 nuclear plant sites would require approx. 130 transports per annum conveying a total of Uranium of approx. 720 t. The casks would cover an estimated distance of 72 000 km.

The number of annual shipments is based on the following assumptions:

1) annual discharge of BWR core is 25 %
2) annual discharge of PWR core is 33 %
3) the spent fuel is to be transported in the TN 12 cask from the reactor site to the reprocessing plant.

The TN 12 cask (Fig. 2) was selected due to the fact that in the future casks of this category will be used as a standard cask in Europe for shipment of spent fuel from LWR's of the 1300 MW class. The TN 12 is a type B (U) fissile classe II package.
The cask body with cooling fins and external neutron shielding is made from forged steel; the lid is also made from forged steel.
The decay heat is removed by natural convection.
A capacity of up to 6 t Uranium is achievable. As the total weight of the loaded cask is approx. 105 t transportation by rail requires a special wagon.

For the safety analysis assumptions and simplifications concerning the transported fuel assemblies were made:

1) the fuel elements are transported in a dry atmosphere.
2) the fuel assemblies are shipped after a minimum of one year after discharge from reactor.
3) there is no differentation between BWR and PWR fuel as fission product composition is nearly identical after a cooling period of 1 year.
4) the reference fuel assembly is a PWR assembly with an enrichment of 3,5 w/o U-235 and a burn-up of 36 000 MWd/tU.
5) the average reference cask load is about 5.6 t Uranium per shipment.

Further tasks of the preliminary analysis were:

1) definition of risk; in our case the release risk
2) definition of the undesired event; in the study the release of radioactive material from the cask under accident concitions
3) determination of the barriers of the system.
These are the fuel matrix, fuel cladding and cask. The state of each barrier after an accident is relevant for determining the release fractions.

Identification of accidents and determination of accident frequency

In order to identify the possible accidents a fault tree was constructed, the TOP-event being the release of radioactive material from the cask caused by accident conditions. A decisive factor regarding the construction of a fault tree is a detailed knowledge of the concerned system, in this case the cask and the railway-network. The availability of the base data influences the structure of the fault tree considerably. At first we will deal with the data of the "Deutsche Bundesbahn" (DB) before discussing the fault tree.

Since 1974 the DB compiles data on train accidents in respect to type and cause. Fig. 3 shows the types of railway accidents during goods transport and shunting with the corresponding percentages. It should be noted that approx. 95 % of all accidents occur during shunting; accidents occuring in shunting yards are not grave, mainly due to the maximum speed being 25 km h^{-1}.

One of the most interesting data is the general accident rate:

$$2.65\ E\text{-}5 \text{ accidents } km^{-1}a^{-1}$$

This value is obtained by dividing the sum of all possible accidents during goods transportation by the total number of travelled kilometers.

Using the technical information of the DB it is possible to divide a route into various track sections. For the reference route the following division was obtained:

65 %	open railway line
32 %	railway station
2 %	bridge
1 %	tunnel
0.1 %	railway crossing

An allocation of railway accident type to various track sections is not possible.

The DB can provide the statistical data on the speed of goods trains. For instance, the allowed speed of goods trains exceeds 80 km h^{-1} on approx. 92 % of the railway routes.

Analogous to the speeds at which various mechanical loads can occur due to an accident various bridge heights were selected from which a drop can result in an equivalent mechanical load.

For the reference system the following bridge height intervals were analysed and their percentages ascertained:

0 m to 9 m	approx. 71 %
9 m to 25 m	approx. 26 %
25 m to 40 m	approx. 3 %

Bridges higher than 40 m are not present on the reference route.

The cask can loose its property as a barrier when either the cask body or lid fails due to one of the following load modes:

1) mechanical load as function of speed internal or drop height resp.
2) thermal load as function of the fire duration and temperature
3) explosion or sundries
4) combination of mechanical and thermal load
5) loads in connection with material failure or human error.

With the collected data the fault tree can be constructed; the h[illegible] of the tree is shown in Fig. 4. Below the shown three levels the tree branches out to include all the above load modes for each track section.

For each base event a probability of occurence has to be derived or calculated. Using a programme which is based on Boolean Algebra the large number of combination of base events leading to the TOP-event can be evaluated; ie. the track section "open railway line" contains approx. 300 accident possibilities.

Accident consequences

Parallel to the construction of the fault tree calculations were carried out to determine the failure thresholds of the cask under mechanical and thermal load. One result of the investigation of mechanical load was that a cask drop from a height of 40 m on the lid edge or from a height of 60 m on the body axis onto a surface of sand or gravel is comparable to the type B test requirements. The cask sinks one meter into the ground but still remains gas-tight.

Furthermore calculations showed that at the considered speeds a spike of 15 cm diameter is not capable of penetrating the lid or cask body. A deformation of the lid does not occur which can lead to a deterioration of gas tightness. Before such a failure can occur the spike itself is plastically deformed.

In the case of thermal loads the following models were investigated. The cask is exposed to a fire of 800 ^{o}C 1/2 and 2 hours as well as to a fire of 1 000 ^{o}C for 1 hour. Special attention was paid to the seal surface temperature. The calculations showed that the critical seal surface temperature is not attained through a 1/2 hour fire. On the other hand a fire of 800 ^{o}C for 2 hours as well as a fire of 1 000 ^{o}C for 1 hour lead to a seal temperature which causes seal failure.

Closely related to the mechanical and thermal loads and the resulting cask damage are the estimations of the fractions of released radioactive material. In the case of spent fuel the following can be released:

1) radioactive fission gas due to reactor fuel rod failures

2) radioactive fission gas from damaged fuel rods due to accident

3) fuel particles from damaged fuel rods due to accident.

A release of the last two materials is only then possible when cladding and lid seal fail simultaneously. The conclusion of the investigation is that only particles of grain size less than 15 µm can be released from cask to surroundings.

The calculated release fractions will be presented graphically in Farmer digrams, ie. released quantity of radioactive material versus probability of accident causing release. This allows a statement to be made on the risk of every possible railway accident.

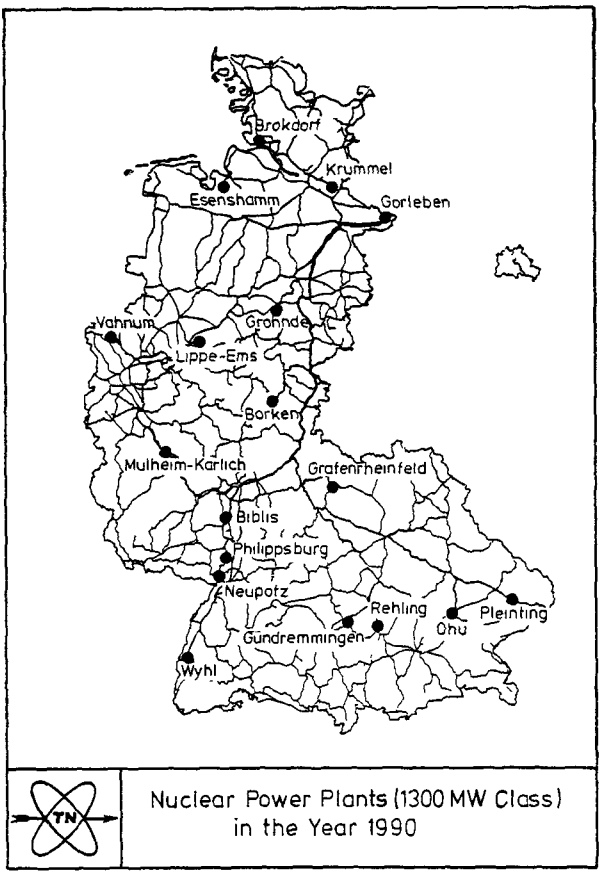
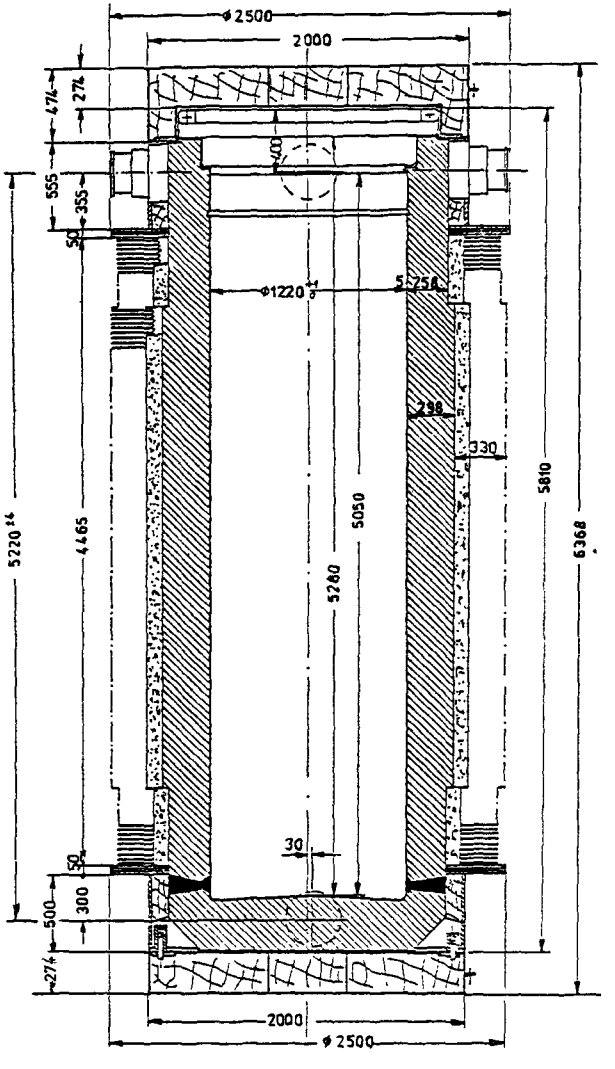


Fig. 1



TN

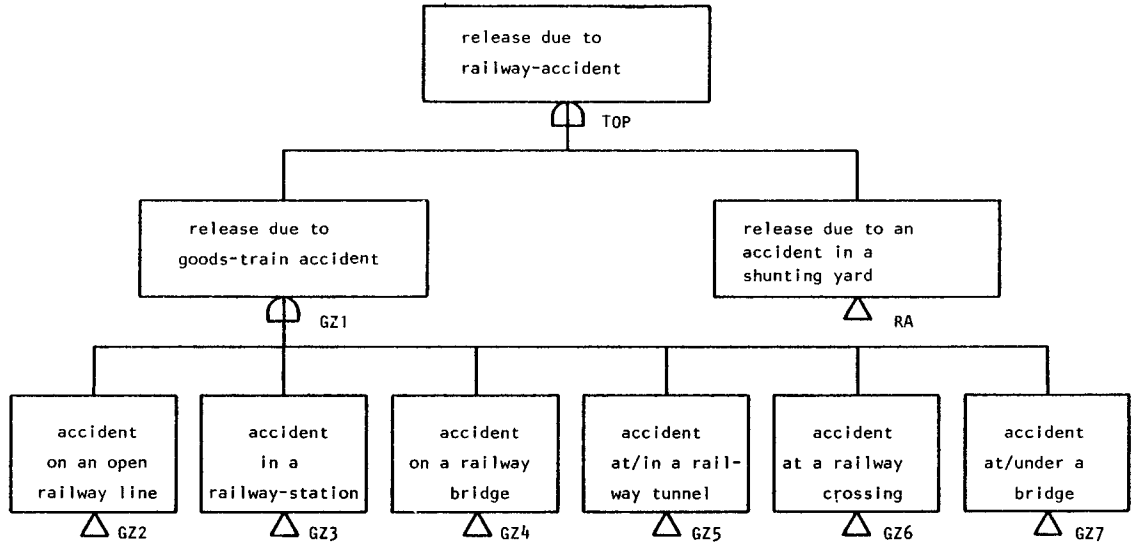
TN 12 Cask

Fig. 2

event	percentage of railway accidents during	
	goods transport (%)	shunting (%)
derailment	29,90	59,92
collision with another train	22,09	29,43
collision with an obstacle	22,13	9,98
collision at a railway crossing	13,49	-
fire	0,99	0,05
explosion	0,01	6 E-4
sundries	11,39	0,62
percentage to total number of accidents	5,13	94,87
general accident rate:	2,65 E-5 accidents km^{-1} a^{-1}	

Survey of railway accidents

Fig. 3



Head of Fault-Tree

Fig. 4

Question by A. Muller, FR Germany: 1. What about the assumptions on fire conditions? Do they result from real fire conditions from experiments, or are they assumed on a theoretical basis? 2. Is it right that the leakage is given by failure of a Viton-O-ring?

Answer: 1. We calculated the failure threshold under themal accident conditions taking into account fires with temperatures of 800°C and 1000°C. 2. Yes.

Question by R. W. Klingensmith, USA: You stated that in an accident condition with spent fuel, only particles of size less than 15 μ are released. Is this size a function of the particle size available or a function of the size of the failure opening in the clad tube or container?

Answer: This particle size is a function of the lid gap assumed under accident conditions.

Question by B. Balmer, Switzerland: If a serious accident were to take place during transportation of spent fuel elements in a type TN-12 packaging, there would probably be partial loss of neutron shielding. Should such an event happen, would danger to the public and environment be considerably increased?

Answer: Total loss of neutron shielding is taken into account during construction of a cask of this type. The dose rates are well within the permissible IAEA values for transportation accidents. It should not be assumed that the share of the total risk is thus considerably increased.

SAFETY ANALYSIS ON TRANSPORTATION OF RADIOACTIVE MATERIALS BY TRUCK IN JAPAN

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Y. Ichikawa, and S. Fukuda
Central Research Institute of Electric Power Industry

1. Introduction

In Japan, 12 BWRs, 9 PWRs, 1 GCR and Advanced Thermal Reactor are now in operation, and 1 BWR and 3 PWRs are under construction. Nuclear source materials are imported by sea-going vessels and transported on road to fuel fabrication plants. LWR's fresh fuel assemblies are mainly transported on road from fuel fabrication plants to nuclear power plants. And LWR's spent fuels are transported by sea-going vessels from reactor sites to the PNC reprocessing plant at Tokai-mura and also to reprocessing plants in Europe. The risk assessment of nuclear materials has been performed for the sea transportation in 1978.

In 1979 and 1980, risk assessment of road transportation of nuclear fuel materials has been performed.

The safety assessment of road transportation is very important as one of versatile safety assessments concerning peaceful uses of nuclear power in the narrow Japan because the road transport of nuclear fuel materials is performed in the area where the general public live.

This safety assessment is performed by dose assessment for the normal moving and the accident considering the results of a truck crash test for safety margin of packagings on board and the countermeasures for the safety of transporting nuclear fuel materials.

2. Actual Condition of Transport

The nuclear fuel for LWR, considering in this assessment research, is imported as UF_6 or UO_2 powder, made into fuel assembly at conversion and fabrication plants and used in power plants. The nuclear fuel transport is almost carried out on road, namely, expressway, national road or local road. Transport of spent fuel derived from LWR plants is nearly performed by ship. In case of power plants which have no ports within their premises, however, road transport is used between the port and the power plant. For example, very short road transport is made between the port of Tokai and the PNC Reprocessing Plant of PNC at Tokai Mura. Population density and traffic condition in the peripheral area of the transport roads are shown in Table 1 and Table 2, respectively.

The package of these nuclear fuel materials is the package approved by the Japan Competent Authority and several 11-ton class trucks for the full load (exclusive use) are employed by moving in a formation accompanied by vanguard car and a rear guard car.

The transport quantity, frequency and distance of these nuclear fuel materials during the period from 1975 to 1976 are shown in Table 3 and the dose rate evaluated in the vicinity of transported materials, number of package per one transport and one transport truck formation are show in Table 4.

Table 1. Population density in the vicinity of transport roads

Classification	Population density (person/km^2)
Large cities	10^4
Urban area	10^3
Country side	10^2

Table 2. Traffic condition of transport roads

Classification of road	Traffic quantity (vehicle/day)	Mean moving speed (km/h)	Width of driveway (m)	Remarks
Expressway A	$10^4 \sim 10^5$	50	15	Principally in large cities
Expressway B	$10^4 \sim 10^5$	10 ∿ 100	40 ∿ 60	Principally between cities and towns
National road A	$5.0 \sim 10^4$	20 ∿ 40	10 ∿ 15	Four-lane road
National road B	10^4	30 ∿ 50	6 ∿ 10	Two-lane road
Local road	10^4	30 ∿ 40	6 ∿ 15	

Table 3. Annual quantity of transport of nuclear fuel materials

Contents	1975			1976			Mean transport distance in 1975 and 1976
	Transport weight (t)	Transport frequency (time)	Number of vehicles	Transport weight (t)	Transport frequency (time)	Number of vehicles	
UF_6	162	8	30	57	5	11	4,000 km
UO_2 powder	166	63	108	346	63	176	20,000 km
Fuel assembly for LWR	157	20	98	239	20	151	85,000 km

Table 4. Dose rate of nuclear fuel materials in transport, number of packagings and transport formations

Content	Dose rate (measured value) (mrem/hr)			Number of packagings per unit transport	Transport formation x: guard cars, o: trialers (Arabic numeral below 'x' or 'o' indicates the number of crews)
	Packaging surface	Driver's seat	Number of packagings		
Fuel assembly	1.5	0.5	0.1	16	x o o o o x 4 2 2 2 2 4 (Two formation)
UF_6	1.0	0.5	0.1	9	x o o o x 2 1 1 1 2
UO_2 powder	1.0	0.2	0.1	152	x o o o o o x 5 2 2 2 2 2 3
PuO_2	5.0	2.0	0.1	7	x o o o o x 5 2 2 2 2 3
Spent fuel	22.0	2.4		2	x x o o x x x 4 6 2 2 5 5 3

In order to derive the probability of transport accidents, the number of vehicles, actual state of running, and type and number of traffic accidents have been studied. Traffic accidents include crash, overturning, downfall, deviation from roads, and fire and accident probability for all vehicles was order of 10^{-6} case/km-vehicle. The probability of accident that may allow serious effect on the package, was estimated to $10^{-8} \sim 10^{-10}$ case/km-vehicle. In the case of a long distance transport of nuclear fuel materials performed once a year moving in a formation consisting of plural number of vehicles, expectation number of accident was estimated to the order of 10^{-6} case/year.

3. Exposure Dose Assessment in Normal Condition

3.1 Assessment object

The assessment in normal moving was made separately in shipment moving and the rest and inspection of the vehicle.

The assessment objects in shipment moving are:

i) General public living in the peripheral area along the transport route.

ii) General public in the other vehicles (persons travelling in the direction opposite to shipment and in the same direction as shipment) on the transport route.

The assessment objects in shipment stops (rest and inspection time):

i) General public living in the peripheral area of the rest and inspection place.

ii) General public on road in the vehicles passing through roads in the vicinity of the rest and inspection place.

3.2 Assessment method

Assessment was performed by obtaining the collective dose from external exposure due to γ-ray emitted out of the transport material for one transport according to each transport route.

The dose evaluation in the period of car running was performed by a subroute unit, which was made by dividing one transport route into approximately 30 at the maximum according to the type of road (expressways, national roads and local roads) and the size of cities and towns (large or small population density). The collective dose for all over the transport routes was obtained by summing up the dose rate obtained from each subroute. As for the γ-ray dose evaluation measured dose rate value was adopted.

3.3 Results

The collective dose per one transport varied according to the transport material and the route, however, its order was 1 man-rem or less. A great part of collective dose was due to the exposure which general public in other vehicle travelling on the route. As the annual transport frequency of nuclear fuel materials for power generation in Japan was approximately 100 times (in 1976) the annual collective dose in conjunction with the transport of nuclear fuel materials by truck would be 10^2 man-rem or less. In comparison with the collective dose of Japanese people of 10^7 man-rem due to the natural radiation (approximately 0.1 rem/yr) in our country, this value is sufficiently small.

4. Dose Assessment in Case of Accident

4.1 Release

Release of radioactive materials from package in case of traffic accidents during a road transportation may be changed according to vehicle structure, packag- structure, binding and fixing methods and accident mode. As the packaging used nuclear fuel transport in Japan is in conformity with the IAEA standard and quality control and maintenance control of packagings in the storage are performed sufficiently, it is considered that possible release would not occur in the accident considered under the Japanese regulations of transportation.

In addition thereto, sufficient safety countermeasures are taken to prevent accidents during transport in compliance with the type of nuclear fuel. Therefore, as for the release of nuclear fuel material in case of accidents, consideration should be taken based on the package to be used for each nuclear fuel material and the actual state of the safety countermeasures therefor.

The results of consideration are as follows:

As for UF_6, a package to be used in type A, however, 1) the packaging is proved by test that it has the allowable quality for safety equivalent to type B, 2) the transport distance is comparatively short (approx. 150 km) in Japan, 3) there were no long tunnels nor high places on the way; and 4) transport is performed in relatively low speed of 40 km/hr at the maximum according to the running program. Therefore it was supposed that there was utterly no chance to encounter any accidents followed by release of nuclear fuel material.

In the case of spent fuel transport, it was almost performed by ship and the transportation by truck was always performed only in a short distance (approx. 10 $\sim$ 40 km) as a coupling between maritime routes, the maximum speed on road transport was limited to 10 km/hour practically and the packagings in use was type B, it was supposed that the release did not occur.

As for the new fuel assembly, it is sometimes transported by passing through an expressway with high speed for a long distance. As there are long tunnels and high bridge on the way, it might encounter fire in the tunnel, and/or downfall accidents from higher places. As the fuel is, however, protected with refractory sheath tube, it is difficult to release the nuclear fuel. With the considerations mentioned above, it was supposed that the release did not occur.

Finally, as for the transportation of the UO_2 powder, as well as new fuel assembly, the transport is made on expressways and others. But the type A packaging whose shape is like a drum is used as package for transport. Therefore, it might be possible that the packaging is broken thus, resulting in the release of the UO_2 powder in case of falling-down from a higher place.

Accordingly, in this analysis, the environmental conditions in the peripheral area of the transport roads of UO_2 were taken into consideration and the evaluation of release quantity was made supposing the rigidity of packagings, impact acceleration and barrier effect against release when falling down on ground from the point of 50 m and 15 m high with the upside-down state of a transport vehicle (a state in which the transport packaging is at the lower side). The estimated release quantities are shown in Table 5.

Table 5. UO_2 release in a accident

Item	Area classification	
	City	Urban Area
Supposed height of downfall	15 m	50 m
Number of pieces contributing to release	1.6 pieces	3.25 pieces
Release from packagings	0.945 mci	1.9 mci

4.2 Methodology for assessing the exposure dose

The outline of assessment method in case of release accidents is shown in Fig. 1. Chemical toxicity and the internal exposure due to food chain were eliminated from this assessment. The assessment object was decided as general public living in the peripheral region of the accident place. The exposure pathway is different according to the mode of release and the behavior of released radioactive materials after accidents. The release mode was classified into two cases. The first case is that wherein the radioactive material is dispersed on ground in the peripheral region of the accident point. The second case is that wherein the radioactive material floats and diffuses in the air. In the first case, the following exposure pathway was considered:

(1) external exposure by γ-ray due to the radioactive materials, dispersed on the ground.

In the second case, the following exposure pathways were considered;

(2) external exposure by γ-ray due to the radioactive cloud.

(3) internal exposure due to the inhalation of radioactive materials.

(4) external exposure due to the γ-ray deposited on the ground surface in the diffusing process of the radioactive materials.

The model drawing from (2) $\sim$ (4) is shown in Fig. 2. In case of external exposure, whole body was decided as the objective organ. The dose assessment was made by means of the unit release, the unit population density and the mean weather conditions along the whole transport route. In addition thereto, the release quantity at the accident time, the weather conditions in the peripheral region of accident occurrence point and the population density were considered to assess the maximum individual dose and the collective dose of the critical group.

4.3 Results of the dose assessment

Based on the consideration taken in Paragraph 4.1, assessment was made only in the case of UO_2 powder.

As a result of the study on the contribution of each of four exposure forms mentioned in Paragraph 4.2, exposure other than the internal exposure due to inhalation is sufficiently small to be ignored.

As a result of this assessment, the following is concluded:

i) The maximum individual dose commitment (50 years' dose) was the order to 10 mrem and is smaller in comparison with the individual dose limit of 500 mrem/yr in Japanese regulations.

ii) The collective dose commitment of a critical group (50 years' dose) is 0.1 man-rem or less for whole body, and this value is sufficiently small in comparison with the commitment dose of Japan due to the natural radiation of 10^7 man-rem.

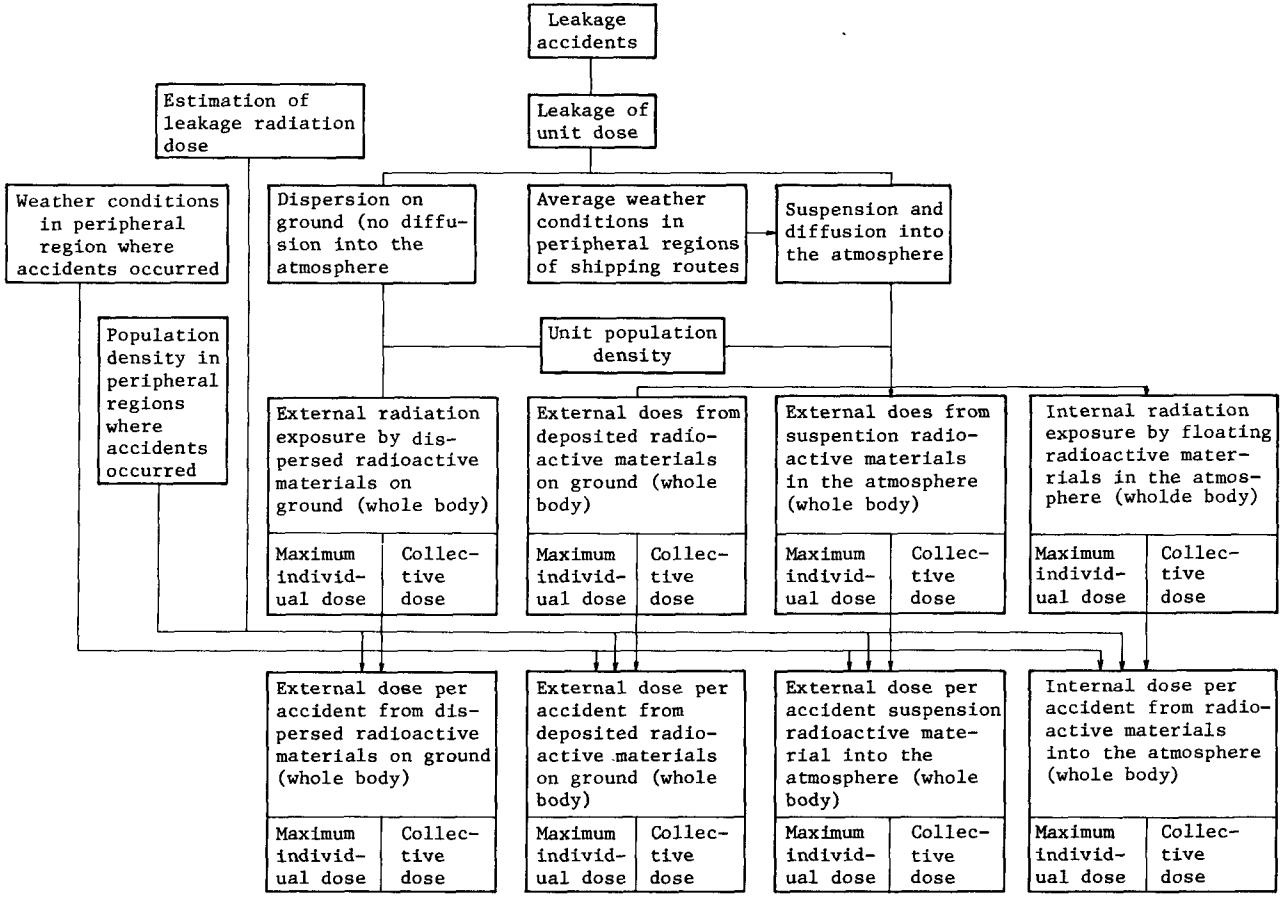

Fig. 1 Methodology for assessing radiation exposures in accide

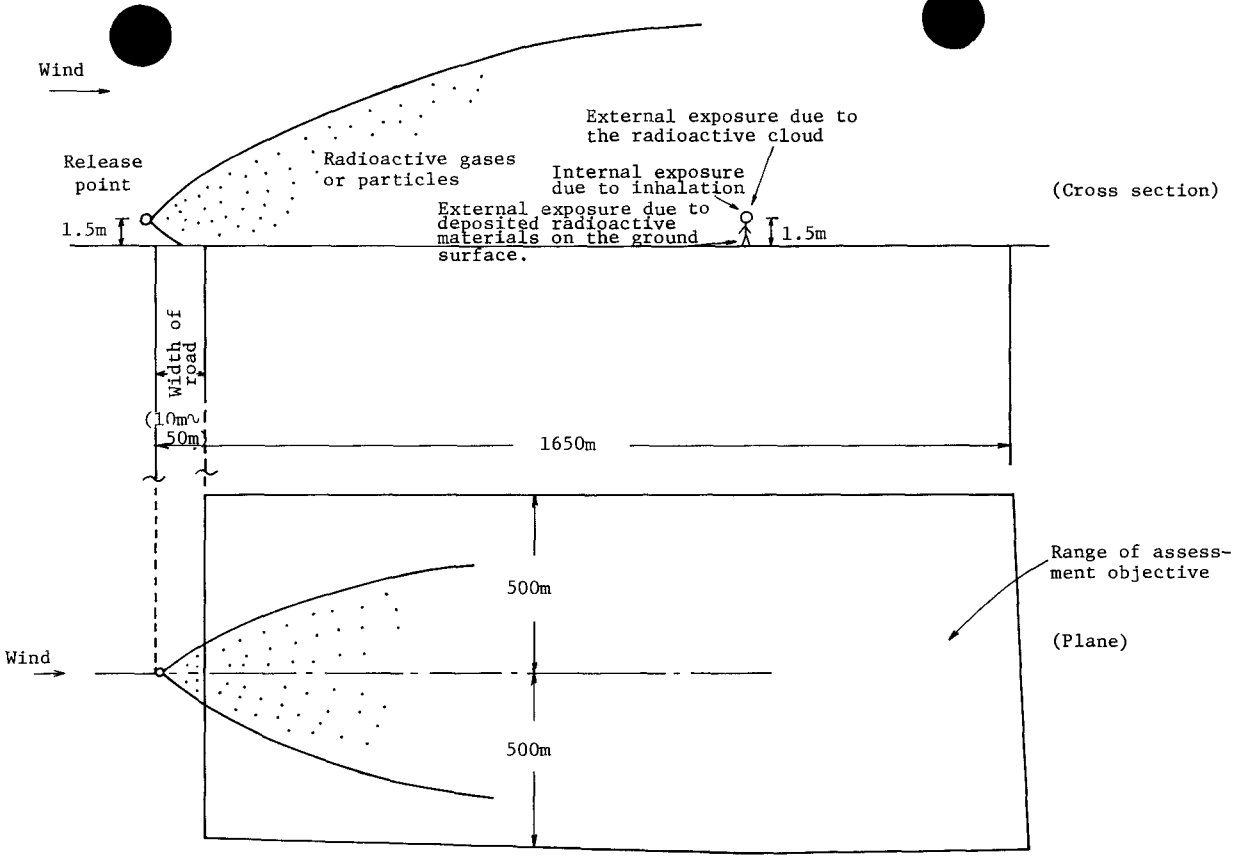

Fig. 2 Exposure model of the release accident

DISCUSSION

Question by M. Brown, United Kingdom: If I understand correctly, you have 5 escort cars for a spent fuel shipment. Are these intended purely for security, or are they meant to have an effect on the chance of an accident or the consequences (by intervention after the event)?

Answer: Your understanding is correct. These consist of cars which protect against traffic accidents and in the case of an accident ensure quick recovery.

Question by D. H. Lohmann, United Kingdom: The results presented are for large container movements; is it intended to extend the studies to include movements of smaller packages of, say, radiopharmaceuticals and radiochemicals?

Answer: Analyses were carried out using the drum type container for UO_2 powder (height $\sim$ 0.8 m, diameter $\sim$ 0.6 m). It is not intended to extend the studies concerning radiopharmaceutical and radiochemical containers at the present.

ENVIRONMENTAL IMPACT OF PLUTONIUM TRANSPORT IN THE EVENT OF ACCIDENTS WITHIN THE EUROPEAN COMMUNITY - ESTIMATION OF POTENTIAL RELEASES -

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W. Anspach
Transnuklear GmbH

1. Introduction

This study was carried out within the framework of a research and development program of the European Communities on plutonium recycling in light water reactors /1/.

The paper assesses the release of radioactive material due to accidents during transportation. In each case the limit of the system is the container surface. The quantity of released material is indicated as being appropriate for the transport by airborne pathway. Thus initial data for calculation the environmental burden under accident conditions are derived.

2. Reference Scenario

In the reference scenario it is assumed that 15 Mg fissionable plutonium - corresponding to about 24 Mg Pu tot - are recycled yearly in light water reactors. This mass would have to be transported within the stages of the outer fuel cycle several times. If one adds the plutonium masses which are transported between the stages of the fuel cycle, the result is a fictive transported plutonium mass of about 68 Mg. The total transport distance in the reference system amounts to about 500 000 km on public roads. The reference scenario comprises the following transport systems:

material	means of transport	share in transported Pu tot (%)	share in transport distance (%)
plutonium oxide	truck	18	6
plutonium nitrate	internal vehicle	18	0.05
fresh mixed oxide fuel elements	truck	32	43
spent mixed oxide fuel elements	train	32	40
alpha waste of low specific activity	truck	< 0.1	11

The weights, heat loads and radioactive inventories of the reference containers are given in the next table:

material	grossweight of container (daN)	heat source per container (kW)	radioactive inventory per container (Curies)
Pu-Oxide	200	0.136	1.30 E+5
Pu-Nitrate	150	0.030	2.95 E+4
Fresh MOX Fuel	1 700	0.740	6.76 E+5
Spent MOX Fuel	92 000	105	1.17 E+7
Low-level α - waste	160	3.21 E-4	2.93 E+2

The specific activity of the reference Pu-mixture amounts to 14.75 Ci/g Pu. In case of the transport of spent fuel according to another CEC-study /2/ a mixed loading of the cask was assumed. The mixture consists out of one fourth of MOX-Fuel and three fourth of Oxide-Fuel. The specific activity of the reference load amounts to 2 E+6 Ci/Mg heavy metal.

3. Method of analysis

The exspectancy values of release of radioactive material under accident conditions should be evaluated. The release risk was defined as product of the released amount per accident and the accident frequencies.

The possible release risk was assessed by the method of the probabilistic risk analysis using the fault-tree method. In each case one transport carrier and one reference container were investigated. The containers already exist today.

For the construction of fault-trees it has proven expedient to differentiate between several types of containers as follows:

- for small, light containers the structure of the fault-tree is related to the failure of barriers,
- for large, heavy containers, used for the transport of fuel elements, the fault-tree is constructed according to types of impact.

These different procedures have no effect on the results of the analysis.

When determining the failure rate for a system with small containers, it is at first assumed that in case of an accident only one single container in a truckload is damaged to such an extent, that radioactive material may be released. The failure of several containers due to an accident is the subject of a sensitivity investigation for any special case. When considering the reference system it is always presupposed that the outer surface of the container forms the limit of the system.

At NUKEM, the fault-trees are calculated by means of the computer program ALLCUTS. The fault-trees so found are traced by a plotter.

4. Results

The fault-tree analysis and its evaluation result in combinations of events which might lead to release of radioactive material. Mechanical, thermal causes and their possible combinations also with regard to the chronological

order of their occurence were taken into account. Material defects and human failure were also considered so far as it seemed necessary or as data were available. The following table gives a survey of the distribution which was related to the types of accidents.

type of load	number of accidents of the system (%)				
	Pu oxide	Pu nitrate	fresh MOX fuel	spent MOX fuel	low level α-waste
mechanical only	30	-	33	26	29
thermal only	6	-	-	2	6
mech.-therm. combined	64	8	67	35	65
material defects and others	-	87	-	37	-
handling only	-	25	-	-	-
handling and others	-	59	-	-	-

Examples for fault trees are given in Figure 1 and 2.

Pu-Oxide

The greatest part to the release risk is caused by mechanical loads. Then follow combined load cases such as impact and fire; fire and puncture. Contributions to the release risks which are in between are caused by fire.

The following can be said with regard to the amount of release fractions:

- Mechanical loads only and those in which the container was weakened by a fire earlier on, lead to high release fractions.
- Loads through fire, however - even those in which the container was damaged beforehand through mechanical stress - lead to lower release fractions.

In case of thermal action, only the gasket fails, a fact which does not allow the whole content of radioactive material to be released.

The Farmer diagram (Fig. 3) shows that the highest accident rate is at 1.0 E-5 a^{-1}; that means that every 100 000 years one must reckon with an accident in which a part of the container inventory is released.

The cut-sets which provide the highest contribution to the release risk occur about every one million years only.

The most probable accident with a maximum release bears an occurrence frequency of 1 E-8 a^{-1}, which means every 100 million years.

Pu-Nitrate

The significant deviation of the plutonium nitrate system from the general trend is due to the particular conditions of the internal transport. The percental distribution in the above table, however, does not permit conclusions on the extent of the release risk.

Almost half the accidents are caused by mechanical loads in connections with either handling errors or quality assurance errors. It also appears that about 25 % of the cut-sets leading to release of radioactive material are due to human errors.

From the Farmer diagram it can be seen that the highest accident rate lies at approximately 1.0 E-9 per year. This is tantamount to stating that an accident resulting in release of radioactive material occurs once in a billion years.

The frequency of the most probable accident with maximum release may be expected to be ten times lower.

Fresh MOX Fuel

The greatest parts in the release risk are caused by mechanical loads only. Then follow contributions of the risk which are caused by combined loads. It has to be noted that only combinations in the sequence thermal-mechanical lead to release. The reverse case does not lead to any consequences. This is essentially due to the definition of the system. The load fire (only) does not provide a constribution to the release risk either.

It can be gathered from the Farmer diagram that the highest accident rate is at 1.0 E-3. This means that every 1,000 years one has to reckon with an accident in which, for reasons of the assumptions made, a release of radioactive material has to be faced. The occurrence probability of the next accident regarding frequency is already lower by one order of magnitude, namely every 10,000 years.

Spent MOX Fuel

The greatest contributions to the release risk arouse from mechanical stress alone. The contributions to the risks following hereafter are caused by combined events. These events already range below the maximum release risk by 2 orders of magnitude.

The case of a release produced by thermical stress alone ranges under the greatest contribution by about 4 orders of magnitude and only happens once.

The Farmer diagram shows that the highest accident rate is of the order of about 1.0 E-3. This means, that about every 1,000 years an accident resulting in a release of radioactive material is to be expected.

Low Level Alpha Waste

The highest contribution to the release risk is caused by the accident fire. The following contributions to the risk are provided by the loads impact plus fire, crush plus fire, puncture, fire plus puncture or vice versa. Nearly 2/3 of all accidents are caused by combined events. This tendency was already observed regarding the transport of Pu oxide.

The following can be said with regard to the amount of the release fractions: Nearly all types of load are characterized by the release fraction one. The accidents puncture (only) as well as fire plus puncture to which a release fraction of 0.25 is related are excluded therefrom.

It can be gathered from the Farmer diagram that the highest accident rate amounts to 1.6 E-5. This means that, about every 60,000 years, one has to reckon with an accident which leads to release of radioactive material.

5. Conclusions

In all sub-systems few accidents determine the release risk, often in an interval of several orders of magnitude. Fig. 3 is a survey of the release risk of all systems in a Farmer diagram. It is an excerpt of the complete diagrams of the partial systems.

The greatest part of the release risk is formed by the transport of spent MOX fuel elements. The next contribution regarding frequency is provided by the fresh MOX fuel elements. Regarding these two subsystems the risk extent is largely determined by a relatively high accident frequency. Then follow the systems plutonium oxide and low-level alpha waste, whereby the release risk of the first is characterized by large release quantities. The low contribution to the release risk provided by liquid plutonium nitrate is due to the conditions of internal transport. These are, above all, low accident frequencies and low accident loads.

The individual systems were subject to sensitivity considerations whereby the failure rates as well as release fractions were varied. Apart from the system Pu-nitrate, with a factor of 30, the upward variation of the risk of release amounts one order of magnitude. Thus the upper limit of the range of the release risk can be considered as defined.

The data presented in the Farmer diagram form the basis for subsequent calculations of the effect on the population concerned.

/1/ Assessment of the Environmental Impact of Plutonium Transport Within the European Community
A. Cricchio (CEC-Belgium)
J. Hamard (CEA-France)
I. Lafontaine (Belgonucléaire-Belgium)
K. Schneider (Transnuklear-Germany)
TANSAO-31 1-666 (1979, p. 475 - 477

/2/ R. Christ
Suitability of existing casks for the transport of spent mixed oxide fuels
Final report (1978)
CEC-contract 025-77-4 RPU D
Transnuklear GmbH

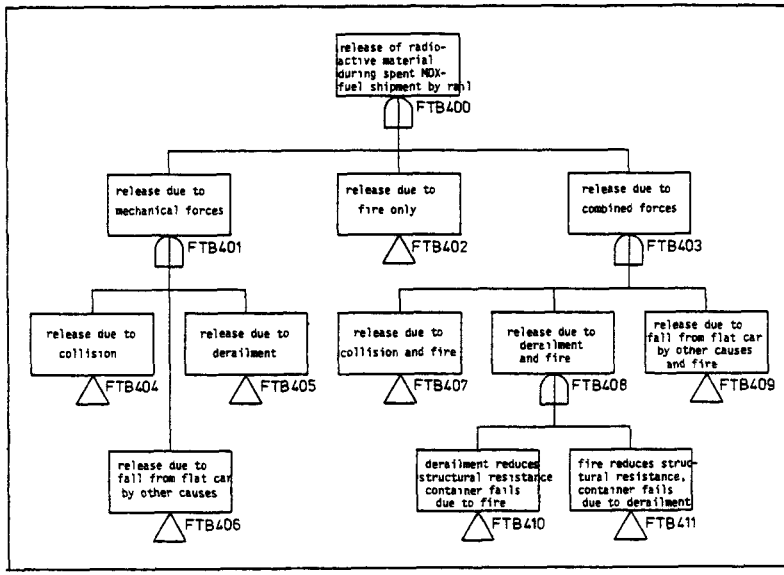

Fig. 1 Spent mox Fuel, head of fault tree

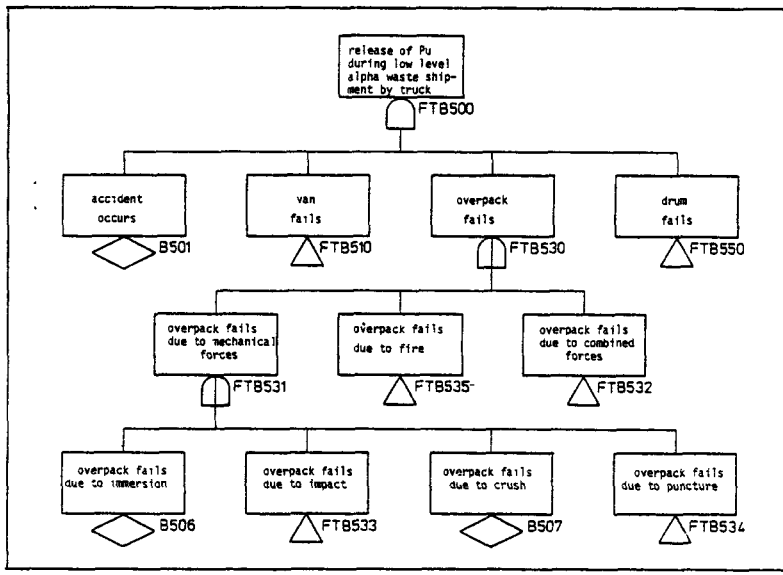

Fig. 2 Low-level-α-waste, head of fault tree

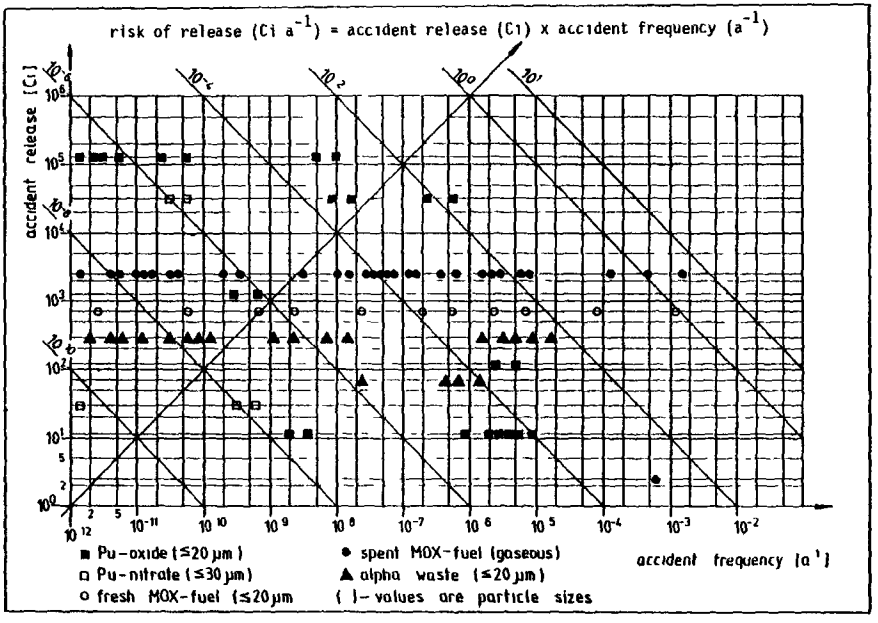

Fig 3 Accident Risk of Release during Transportation of Plutonium

DISCUSSION

Question by G. Wehner, FR Germany: Did you also consider in your study the possibility of a pressure failure of the vessel in the case of Pu-nitrate transport?

Answer: Yes, we did. But in the case of transport on-site, the fire alone does not lead to release. Additional failures resulting from handling etc. have to occur. It has been assumed that the plant's fire engine will be quickly available.

RISK ASSESSMENT FOR THE TRANSPORTATION OF PuO_2, UF_6 AND SPENT FUEL ON ROADS IN THE FEDERAL REPUBLIC OF GERMANY

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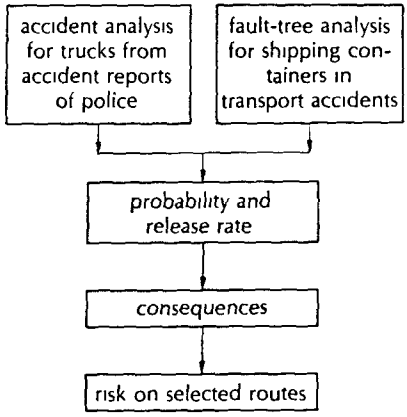
INTRODUCTION

This paper presents results of an investigation /1/ performed on behalf of the Bundesministerium des Innern, Referat RSI/2(1).

The purpose of this investigation was to evaluate the risk arising during transportation of spent fuel, PuO_2 and UF_6 on the roads of the Federal Republic. In order to assess this risk, the probability of an accident occurring in such a manner as to release radioactive material has to be ascertained (severity distribution of accident). The quantities and combinations of radioactive material that may be released during an accident then have to be estimated. Finally, the consequences (acute fatalities) of such a release have to be calculated taking into consideration the density of population and the actual meteorological situation. This procedure is outlined in the following flow chart:

Survey of the analysis



The sections of this paper are arranged according to the flow chart.

RESTRICTIONS

The investigation was performed with the following limitations:

- the investigation was restricted to road transport
- normal traffic frequencies were assumed
- no technical countermeasures after an accident were taken into account
- evacuation was not considered
- shielding effects of buildings were excluded
- sabotage was not taken into account
- radiological effects on the road users were disregarded
- no traffic jams were assumed.

EXAMINATION OF ACCIDENT SEVERITY

Accident research in the Federal Republic (in commercial road transport) is mainly concerned with personal injury; material damage in relation to accident severity plays a secondary role. This means that no accident statistics are available that can supply information on distribution of accident severity. A preliminary investigation therefore had to be carried out in which a sample of 3,304 reported accidents in 1977 (where lorries were involved) was evaluated. The quoted random samples refer mainly to Autobahn (motorway) accidents. The breakdown of the investigated cases is as follows:

Autobahn	92.5 %
Trunk-roads	1.1 %
Local roads	6.0 %
Others	0.4 %

In 1977, 12,835 lorry accidents occurred on motorways, which means that the random sample accounts for 26 % of the total number of annual accidents. In order to establish accident probabilities, the number of accidents has to be related to vehicle capacity and total number of accidents as follows:

Vehicle capacity

	Motorway	Other roads
Single trucks	24.4 %	75.5 %
Trucks with trailers	51.8 %	48.1 %
Trailer trucks	61.2 %	38.7 %

Number of accidents

	Motorway	Other roads	Accidents per vehicle km
Single trucks	4,568	14,104	$1.53 \cdot 10^{-6}$
Trucks with trailers	5,837	5,420	$1.90 \cdot 10^{-6}$
Trailer trucks	2,430	1,537	$1.39 \cdot 10^{-6}$
Total	12,835	21,091	$1.65 \cdot 10^{-6}$

The accidents are classified according to their severity into accidents involving slight, medium and severe damage. This classification is shown in the following figure:

Distribution of the severity of truck accidents

Accidents per 10^6 truck-km	Slight damage	Medium damage	Severe damage	Total
Single truck	1.47	0.028	0.026	1.52
Truck with trailer	1.76	0.114	0.048	1.93
Trailer truck	1.25	0.104	0.038	1.39

In addition, the accidents are divided into separate classes according to the physical causes of damage:

- impact/crush
- puncture
- fire

Impact and crush are the main processes of destruction in almost all accidents.

Because there is not enough information in the original accident reports, it is not possible to distinguish between these two processes. They are therefore regarded as a single process in the following. For medium and severe accidents the region of damage is given in the following figure (in %):

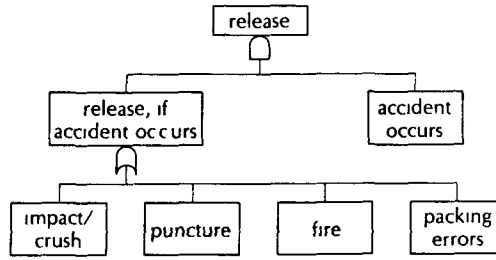
front	4.7
side	16.7
front + side	1.6
back	6.2
both sides	17.8
back and side	5.0
no damage to the trailer	48.1
	100.0

Puncture occurs in 6.5 % of accidents.

Fire occurs in 1.5 % of accidents, 20 % lasting longer than 0.5 h, 7 % of severe accidents involve fires.
For literature see also Statistisches Bundesamt /2/.

FAULT TREE ANALYSIS

A detailed fault tree analysis was performed to estimate the failure rate of selected transportation containers (6M for PuO_2, 30 B for UF_6 and NTL-14 for spent fuel transports). The simplified flow chart below shows the first few steps of the fault tree for release of radioactive materials:



For this analysis basic data and sequences were, for example, BNWL-1848/3/, PNL-2211/4/, PNL-2588/5/, NUREG-0034/6/.

RELEASE SEQUENCES

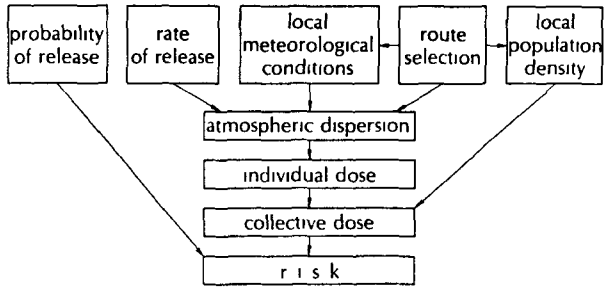
It was shown in the fault tree analysis that packing errors have no major influence on the overall risk. The dominant accident sequences for release are given in the following table:

Material transported	Accident probability	Release rate (fraction)
PuO_2	puncture: $3 \cdot 10^{-5}$	PuO_2 : 0.05
UF_6	fire: $7 \cdot 10^{-6}$	UF_6 : 1.0
spent fuel	impact: $1 \cdot 10^{-3}$	Cs-134 : $1.35 \cdot 10^{-3}$ Cs-137 : $1.35 \cdot 10^{-3}$ Kr-85 : $3.6 \cdot 10^{-1}$ I-129 : $1.25 \cdot 10^{-1}$ Te-127m: $3.0 \cdot 10^{-4}$ Se : $1.5 \cdot 10^{-2}$

Basic data were, for example, BNWL-1848/3/, PNL-2211/4/, PNL-2588/4/, S.R. Fields /7/, WASH 1238 Supplement II /8/, JJ. Hillary et al. /9/, H. Albrecht et al. (10).

CONSEQUENCES

The consequences of release were analysed according to the following flow chart:



The transportation routes have been subdivided in different segments. This was done according to meteorological, orographical and road technical aspects. This is in contrast to other investigations published so far. Population densities in segments of 3 km were taken into account.

Because of this type of evaluation the transportation risk fluctuates with road segment.

This is valid for the single values of the lethal effects, for the collective doses as well as mean values for the respective traffic sectors.

There are marked differences for sections leading through country-side and sparsely populated areas as against more densely populated areas.

The risk increases by one to two order of magnitude in areas of high population density where roads cut through cities and towns.

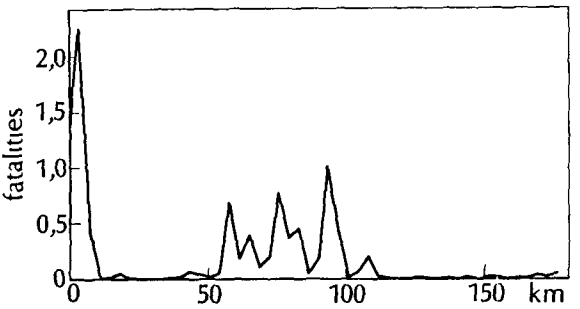
For all three materials i.e. plutoniumoxid, uranhexafluoride and spent fuel results are given below.

For literature see BMI-Richtlinie /11/, G. Marnier et al. /12/.

Plutonium Oxide

As mentioned earlier, the 6M was chosen as the transportation container for PuO_2 for this study. In an accident, one of the two cans is destroyed in the container, and the PuO_2 powder is such that a maximum of 20 % is sufficiently small to remain airborne over larger distances and be inhaled by people. Most of the dose is due to inhalation of Pu-241 and Pu-238.

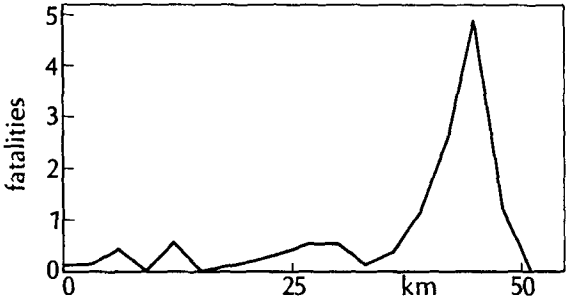
The following figure shows the number of acute fatalities along a transportation route at each point on a given Autobahn. The figure shows a weighted mean of the consequences. The population density and the frequencies of weather situations form the weighing factor. The peak values in the figure are due to highly populated areas in contrast to rural districts.



Uranium Hexafluoride

It has become obvious from various comparisons of release situations and the subsequent dose calculations that the incident where the container detaches from the overpack and undergoes a prolonged fire is the most serious one. It is assumed that under the influence of fire 80 % of the UF_6 will be released at higher altitudes and 20 % at ground level.

The danger caused by hydrofluoric acid developing during release exceeds the danger represented by radioactivity by two orders of magnitude. Only 100 mg of inhaled hydrofluoric acid is sufficient to be lethal in a short period of time. The following figure shows the number of acute fatalities along an Autobahn route different from the previous one. Again, the main cause of the peaks in the number of fatalities is a densely populated region of a city.

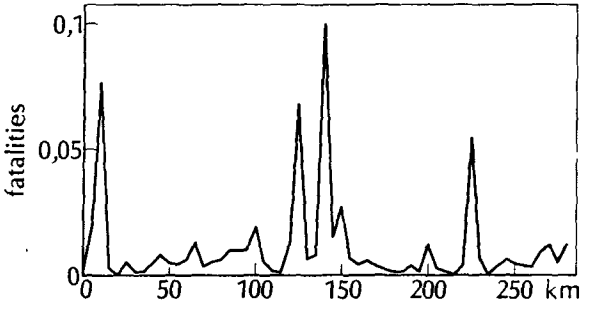


Spent Fuel

Spent fuel which has a burn-off of 29,000 Megawatt-days per ton of uranium and had already been stored 180 days was selected as typical for this investigation. The NTL-14 was assumed as transportation cask. In addition, the most severe accidents (impact) have been selected for the investigation.

The largest releases can be expected on side and end impacts. It may be assumed that the cooling water will escape in gaseous or aerosol form. Part of the remainder will burst later by decay heat. Most of the dose is accounted for by the inhalation of caesium (Cs-134, Cs-137). The release values used are those measured by Hillary /9/.

The subsequent figure shows the number of acute fatalities due to whole body dose along an Autobahn route.



CONCLUSIONS

Using the tools presented in this paper, the essential factors affecting the total risk may be recognised. It is then possible to introduce improvements and preventive measures. Above all the traffic route selected from all possible routes should be associated with the smallest possible risk. The detailed procedure is documented in the report /1/. From this investigation, the relationship between the risk connected with transportation, the other risks associated with the nuclear fuel cycle, and the general technical risk may be reduced. Nevertheless, the assessment of the relevance of this risk with a view to a general risk strategy remains a political matter which is beyond the scope of this investigation.

REFERENCES

/1/ Schwachstellen und Risikoabschatzung beim Transport radioaktiver Materialien. Bericht fur den BMI, SR 58 (1979)

/2/ Stat. Bundesamt, Fachserie 8, Reihe 3.3

/3/ Assessment of Transport PuO_2. BNWL-1846 (1975)

/4/ Assessment of Transport UF_6. PNL-2211 (1978)

/5/ Assessment of Transport Spent Fuel. PNL-2588 (1978)

/6/ Environmental Statement on Transportation NUREG-0034

/7/ Fields, S.R., Spent Fuel Shipping Cask Accident (Draft) Westinghouse (1975)

/8/ WASH-1288 Supplement II. Potential Release of Cesium. NRC (1976)

/9/ Hillary, J.J., UK. TRG Report 2433 (W) (1973)

/10/ Albrecht, H. et al, KFK-Bericht No. 2770 (1979)

/11/ BMI-Richtlinie. Allgemeine Berechnungsgrundlage fur Strahlenexposition durch Emission radioaktiver Stoffe mit der Abluft.

DISCUSSION

Question by P. Erlenwein, FR Germany: Did you calculate the consequences for the three cases shown (PuO_2, UF_6, spent fuel) by using the same model, or did you need a special model in the case of UF_6?

Answer: The dispersion was generally calculated by Gaussian models. Heavy gas models were not used because of the insufficient information about the release term.

Question by K. Schneider, FR Germany: 1. Would you please define the failure thresholds for the spent fuel cask in the case of mechanical and thermal loads. 2. Is it not so, that the long-term risk for plutonium transport will be higher than the short-term risk which Battelle has estimated?

Answer: 1. We assumed that 10 % of the severe accidents led to a spent fuel cask failure and therefore to release. 2. This may be possible, but our calculation was restricted to accute fatalities.

Question by P. E. Eggers, USA: Did your study consider the sequence of possible events involved in accidents. For example, often there are a combination of events which sometimes occur such as first impact, then fire, or first fire then drop and impact.

Answer: We took into account all kinds of events including also combinations of different events. Generally the probability of combined events is low compared to the single events and therefore negligible.

Question by L. M. Curtis, United Kingdom: I should like to ask you whether the model which you use for the dispersion of UF_6 takes into account the chemical interactions between the released gas and moisture in the atmosphere?

Answer: In fact the chemical reaction producing HF has been taken into account.

Question by B. Balmer, Switzerland: By various investigations it is known that the usual dispersion models produce inexact results for the release of UF_6, which do not correlate very well with experimental data. Which model did you use?

Answer: This is in fact true, but there is a lack of knowledge concerning the heavy gas properties of UF_6. We were therefore restricted to the use of Gaussian models.

Comment by J. Hamard, France: The application of the current dispersion models is very difficult in the case of heavy particles such as UF_6. French experience concerning accidents involving dispersion of UF_6 indicates that particles of UF_6 quickly fall to the ground near where the release occurs and only a small fraction of the potentially dispersable HF is actually transferred.

RISK ASSESSMENT APPLIED TO TRANSPORTATION OF TRITIATED HEAVY WATER

G.J. Dicke
Ontario Hydro

1. INTRODUCTION

Ontario Hydro owns and operates several CANDU-PHW (CANada Deuterium Uranium - Pressurized Heavy Water) type of nuclear generating stations. The units at these stations have a combined net capacity of about 5 200 MW e. One multiunit station is located at Pickering, just outside Toronto. Another multiunit station is located at the Bruce Nuclear Power Development site, about 150 miles northwest of Toronto. In 1979, the ten nuclear units at four stations supplied more than 33% of Ontario's electrical requirements. Twelve additional nuclear units with a combined net capacity of about 8 600 MW e are under construction at three Ontario sites.

The CANDU-PHW reactor uses natural uranium as a fuel, heavy water as a moderator, and heavy water as a coolant. The heavy water inventory in these reactors is about 1 Mg per rated MW e. Transportation of heavy water is needed, for example, when new reactors are commissioned and when make-up water is required for operating reactors.

Heavy water is not radioactive, but some of it is converted into tritiated water while it is inside an operating nuclear reactor. Tritiated water is chemically similar to heavy water and to ordinary water, but it is radioactive. The characteristic radiation from tritiated water is of such a low energy that it cannot penetrate the human skin or a container wall. Tritiated water will result in a radiation dose only if absorbed into the body. Thus, there is no external radiation from a drum containing clean tritiated water, and there is no radiation dose from spilled tritiated water unless it is absorbed into the body. Normal transport conditions will, therefore, not result in any radiation dose from tritiated water.

A transportation accident which is severe enough to result in a spill could result in a radiation dose. The probability of a spill is small if strong containers are used. Heavy duty steel drums, manufactured according to Specification 5B of the Canadian Transport Commission, are used by Ontario Hydro as containers for heavy water. When transportation is required, these drums are placed in a closed truck. This method of transportation is used for heavy water with low concentrations of tritiated water. A risk assessment was developed to investigate if this method of transportation could be recommended also for higher concentrations of tritiated water. This paper is intended to provide a summary of the risk assessment method and the major assumptions. Some of the results will also be given.

2. RISK ASSESSMENT

The general equation used for this risk assessment is:

RISK = FREQUENCY x MAGNITUDE

where:

RISK = Consequences/Unit Time

FREQUENCY = Events/Unit Time

MAGNITUDE = Consequences/Event

For the purpose of this risk assessment, these terms have been further defined as follows:

Consequence = Radiation Dose Received

Event = A Transportation Accident Involving a Specific Spill Scenario

Unit Time = One Year

The risk equation can then be written as:

Radiation Risk =	Frequency of Specific Spill Scenarios	x	Magnitude of Resulting Dose

The radiation risks are thus expressed in terms of the frequency of specific spill scenarios and the magnitude of the resulting dose. Before describing the FREQUENCY and MAGNITUDE terms, it is useful to consider how a tritiated water spill can result in a radiation dose.

As mentioned, tritiated water has to be absorbed into the body to result in a radiation dose. At the scene of a spill, this uptake into the body could occur by inhalation, by absorption through dry skin, and by absorption through wetted skin. The rate of uptake of spilled water through these three pathways is determined by the concentration of spilled water in the air and by the extent of skin wetting.

The concentration of tritiated water in the air at the scene of a spill depends to a great extent on the size of the spill and the weather. Thus, spill size and weather condition determine the dose-producing properties of a specific spill scenario. Both spill size and weather condition are therefore considered in some detail before defining the specific spill scenarios.

3. FREQUENCY OF SPECIFIC SPILL SCENARIOS

The size of a spill will generally be related to the severity of the accident. The US Nuclear Regulatory Commission has, on the basis of studies by SANDIA Laboratories, classified motor vehicle accidents in eight severity categories. These categories are numbered I to VIII, in order of increasing severity. Categories I, II, III, and IV represent minor and moderate accidents.

Based on container testing reported in the literature, it is expected that spills from the drums will occur in only 0.4% of the accidents classified in severity categories greater than Category IV. In order to be conservative, we have assumed that there will be a spill in the 9% of the accidents which are classified in severity categories greater than Category II. Thus the assumed frequency of spills is about 20 times greater than the expected frequency.

A different spill size was selected for each of the six accident severity categories exceeding Category II. The smallest selected spill size is 750 L, or 5% of a full truck load. The 5% spill is assumed to occur for all accidents in severity Category III. The other spill sizes represent 10, 20, 50, 90, and 100% of a full truck load. A 100% spill is assumed to occur for all accidents in the worst category, Category VIII.

The spill size is one factor in defining the specific spill scenarios according to their dose-producing properties. The other factor is the weather. The weather condition has a strong influence on the potential concentration of spilled water in the air. For example, the maximum moisture content of the air almost doubles with every 10 °C rise in temperature.

The wind has a similar effect. Doubling the wind speed halves the contact time with the spill. The rate of evaporation will increase when the wind speed increases, but the overall effect would be a lowering of the concentration of spilled water in the air at the spill site.

Other weather conditions, such as rain, fog, and high humidity, tend to keep the concentration of spilled water in the air at a low level. In order to be conservative, these influences have not been taken into account. Only the effects of temperature and wind speed have been considered when defining the five weather conditions shown in Table 1. Also shown are the approximate frequencies for these weather conditions. The frequency data are based on hourly readings of temperature and wind speed over a 14-year period at Toronto International Airport.

TABLE 1

Weather Condition Number	Temperature °C ± 5°C	Wind Speed mph	Frequency % of Time
1	40	all	1
	30	< 4	
2	30	> 4	10
	20	< 4	
3	20	> 4	25
	10	< 4	
4	10	> 4	30
	0	< 4	
5	0	> 4	34
	-10	all	
	-20	all	

High temperature and low wind speed allow the highest concentration of spilled water in the air. Weather Condition 1 is therefore the worst weather in the event of a spill. The concentration of spilled water in the air is reduced approximately by a factor of two each time when changing to a weather condition with the next higher number. The related radiation dose changes by the same factor.

The six spill sizes and five weather conditions have been combined to provide the thirty specific spill scenarios for the dose calculations. The frequencies for these scenarios can be calculated by combining the above mentioned weather statistics with accident statistics.

Accident rates, expressed as accidents per vehicle mile, change little from year to year. Accident rates are available for accidents in the various severity categories. The accident rates used in this study are based on accident statistics for large motor carriers in the US. The overall accident rate is 2.5 accidents per million vehicle miles.

The probabilities per vehicle mile are multiplied by an annual mileage to provide frequencies, or events per year. It has been estimated conservatively that to serve Ontario Hydro's operating and planned nuclear stations, the transportation of tritiated heavy water involves an annual vehicle mileage of about 14 000 miles. The calculated frequencies for accidents and accidental spills are shown in Table 2.

TABLE 2

Accident Severity Category	Frequency Accidents/ Year	Spill Size % of Load
I	1.9×10^{-2}	0
II	1.2×10^{-2}	0
III	2.4×10^{-3}	5
IV	5.5×10^{-4}	10
V	9.6×10^{-5}	20
VI	3.8×10^{-5}	50
VII	2.9×10^{-6}	90
VIII	5.2×10^{-7}	100

The frequencies for the six spill sizes are multiplied by the frequencies for the five weather conditions to produce the frequencies for the thirty specific spill scenarios.

We have now defined the FREQUENCY term of our RISK equation. The next step is the calculation of the MAGNITUDE term.

4. MAGNITUDE

The MAGNITUDE term defines the total radiation dose which will result from an uptake of tritiated water at a specific spill scenario. A radiation dose is being received as long as the tritiated water is inside the body. Half the tritiated water taken up at the time of a spill will have disappeared from the body after about 10 days because of normal body fluid exchange processes. The dose from any uptake of tritiated water is thus received over an extended period of time. The uptake of tritiated water is therefore expressed in terms of dose commitment rather than dose.

The dose commitment rates depend on the concentration of spilled water in the air and on the extent of skin wetting. Different rates will therefore be experienced by individuals wetted by the spill, by rescue personnel, and by members of the public at some distance from the spill. For individuals wetted by the spill,

it is assumed arbitrarily that the extent of skin wetting is the same as the fraction of the load which is spilled. Thus, 100% skin wetting is assumed in the event of a 100% spill, and 5% skin wetting is assumed with a 5% spill. It is further assumed that there is no skin wetting for rescue personnel, and there is, of course, no skin wetting for members of the public at some distance from the spill.

The definition of the specific spill scenarios, and of the extent of skin wetting, allows the calculation of the dose commitment rates. The dose commitment rates must be multiplied by the exposure times to result in the overall dose commitments.

Exposure time is the time from the occurrence of a spill until corrective action starts taking effect. Corrective action with regard to radiation is aimed at minimizing exposure. This can be achieved by simply moving upwind just outside the spill area and by removing wet clothing. A fire department can help by producing a mist over the spill area or by flooding the spilled water with natural water. Any of these corrective actions will affect the dose-producing conditions at a specific spill scenario. The dose commitment rates for a specific spill scenario apply, therefore, only until the scenario is changed by corrective actions.

The assumed exposure times are: 1 hour for individuals involved in the accident, 1/2 hour for rescue personnel, and 2 hours for members of the public at some distance from the spill. In order to calculate a potential benefit of escort vehicles, it is assumed that the exposure time for individuals involved in the accident will be 1/2 hour rather than 1 hour if there is an escort vehicle with each shipment.

Thus, the total dose commitment can be calculated by multiplying the dose commitment rates by the assumed exposure times. If exposed individuals drink more fluids, then the eventual dose could be only half as much as the dose commitment. In order to be conservative, it is assumed that no steps are taken to minimize the eventual dose.

Thus, the total dose can be calculated for an individual involved in the accident, for an individual rescuer, and for a member of the public at some distance from the spill. These total dose values have been calculated for exposure of these individuals to each of the thirty specific spill scenarios.

The dose calculations provide the MAGNITUDE terms of the RISK equation. The radiation risks can now be quantified by multiplying the frequency terms for each specific spill scenario by these magnitude terms.

5. RADIATION RISKS

The large number of radiation dose data have been grouped into ranges of dose. Figure 1 shows such dose ranges as a function of the frequency for any spill in each of these ranges. Figure 1 applies to the most exposed individual at the scene of the various spills. The dose data in Figure 1 have been calculated for a reference tritiated water concentration of 1 Ci/kg and an exposure time of 1 hour.

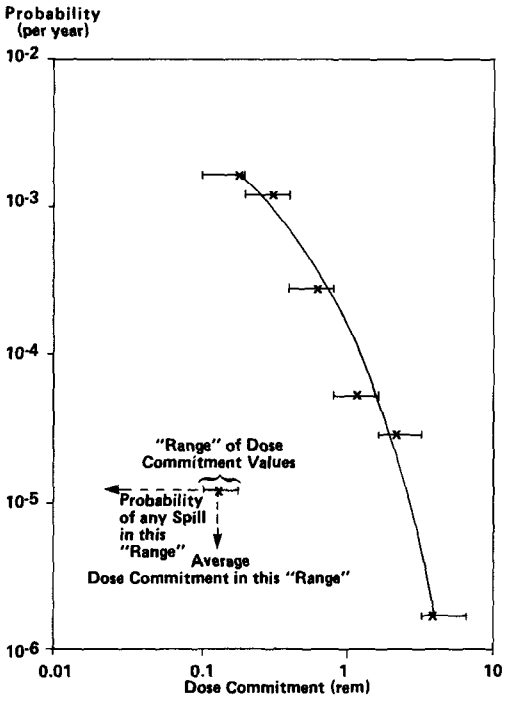


Figure 1

DOSE COMMITMENTS FOR
INDIVIDUAL WETTED BY SPILLS

The calculated frequencies and the calculated radiation dose values, as shown in this figure, are much higher than would be experienced with this method of transportation, because of all the conservative assumptions. As already mentioned, the assumed frequency for any spill is about 20 times higher than the expected frequency. It is also assumed that all the people involved in all these accidents will always survive to receive a radiation dose. The probability of a fatality from conventional aspects of a motor vehicle accident is 900 times greater than the assumed probability of a 100% spill. Thus, exposure to the worst specific spill scenarios will be considerably less frequent than indicated.

The calculated magnitude of the resulting dose is also much higher than would be experienced. Recent studies with regard to evaporation rates indicate that the assumed concentrations of spilled water in the air are about six times too high for most of the specific spill scenarios. The resulting dose, as calculated, will be too high by the same factor.

Another factor which makes the calculated dose values too high is the assumed exposure times. Experience in southern Ontario indicates that corrective action such as moving upwind or cleaning up the spill area is usually started in less than 15 minutes, rather than the assumed 1 hour.

Treatment of the few individuals exposed to any tritiated water spill is relatively simple and could reduce the calculated dose by a factor of two.

The combination of all the conservatisms could thus result in the calculated risk values being three orders of magnitude higher than the expected risk values. This is an important consideration in situations which involve both a radiation risk and a conventional risk. For example, when people are injured in an accident involving a tritiated water spill, there is a radiation dose involved for the person providing first aid. The probabilities of injury, disabling injury, and death from conventional aspects of motor vehicle accidents are respectively 7 800 times, 1 700 times, and 140 times greater than the probability of a rescuer receiving a potential dose greater than 1 rem. The dose value is calculated for a reference tritiated water concentration of 1 Ci/kg.

The calculated potential dose is higher than the expected dose because of all the conservative assumptions, but there is no uncertainty about the probabilities for injury from conventional aspects of motor vehicle transportation. First aid will minimize fatalities and injuries from conventional aspects of accidents. The calculated radiation consequence of providing first aid is small in comparison with these benefits. Thus there should be no hesitation about providing first aid in the event of an accidental spill of tritiated water.

Similarly, these data show that it is not desirable to provide an escort vehicle when transporting tritiated water, if the escort is intended to minimize radiation consequences of an accidental spill. The probabilities of injury, disabling injury, and death from conventional aspects of motor vehicle transportation are respectively 78 times, 17 times, and 1.4 times greater than the probability of a potential dose greater than 0.5 rem for the most exposed individual. This dose value has been calculated using all the conservative assumptions and a reference tritiated water concentration of 1 Ci/kg. Thus, if an escort vehicle were provided with all shipments of tritiated water, then a small potential reduction of radiation dose could be achieved at a cost of injuries and possibly fatalities from conventional aspects of motor vehicle transportation.

The examples show that a measure taken to reduce the radiation risk is likely to increase the conventional risk and vice versa. The radiation risk involved in providing first aid is small in comparison with the potential benefit of saved lives. The conventional risk involved in providing escort vehicles is large in comparison with the small potential reduction in radiation dose. In such cases the risk assessment can be used to recommend the most effective measures.

This risk assessment can be used also to recommend a maximum tritiated water concentration, or other limits for this method of transportation, if levels of acceptable risk are defined. It seems appropriate that the current regulations, the experience with conventional aspects of transportation, and the experience with other hazardous materials be used as a guideline for the defining of levels of acceptable risk in the transportation of radioactive materials.

RADIOACTIVE MATERIALS PACKAGES AND

THE BRITISH RAILWAY ACCIDENT ENVIRONMENT

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SUMMARY

The place of accident statistics analysis in the SRD programme of research into risks due to transport of radioactive materials is described. An outline is given of research into the British railway accident environment, and the parameters affecting accident severity and frequency are discussed. With the aid of accident data supplied by official UK bodies, quantitative information on the incidence of different categories of accident has been obtained. Some of the results describing incidence of derailments, collisions and fires during a sample period are presented, and further effort necessary to supply the required data for transport risk assessments is discussed.

1. INTRODUCTION

An analysis of transport accident environments provides essential inputs to the study of risk due to transport operations. A qualitative study enables an accident classification system to be developed; a quantitative study permits the definition of "per mile (or km)" frequencies of various categories. Package response studies define the damage done to a package by a particular category of accident, and hence the conditions to which radioactive materials might be exposed. A study of radionuclide behaviour then enables a radioactive release source term for each accident category to be defined. The resulting set of radioactive releases with associated frequencies may then be used as input to a consequence modelling program such as TIRION[1], which will provide estimates of numbers of casualties vs frequency - the well-known f/N lines. A schematic of the methodology is given in figure 1:

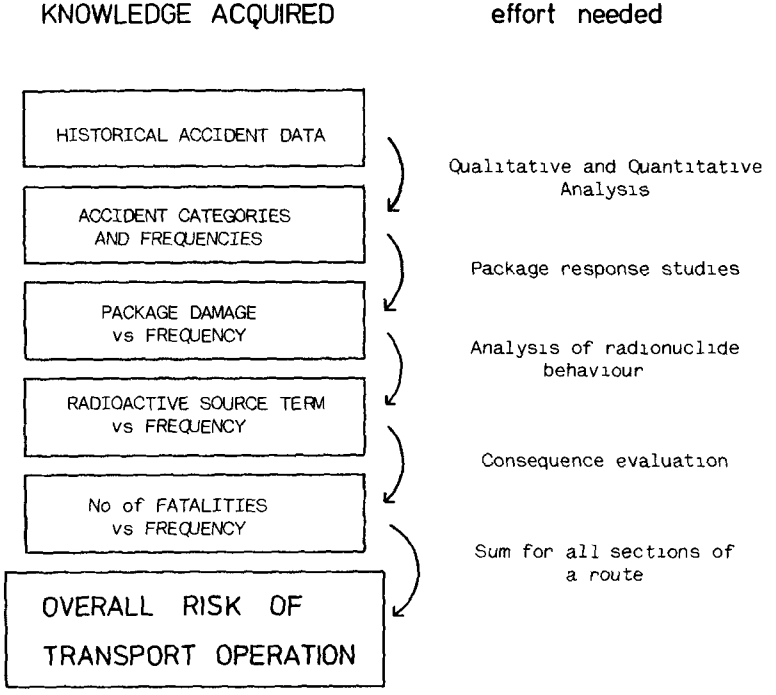


Figure 1: Methodology for Transport Risk Assessment

This paper outlines the course of a study of the British railway accident environment which aims to provide the first stage of the risk analysis procedure shown in figure 1 for rail transport operations in the UK. Differences between railway systems used in other countries and that in the UK mean that the quantitative results are of strictly limited application; the methodology, though, would be appropriate for general use.

2. PARAMETERS GOVERNING ACCIDENT SEVERITY AND FREQUENCY

It is difficult to assemble a comprehensive list of the parameters which affect accident severity. Those which have received consideration in this study are listed below. Factors are grouped into accident specification; railway system and 'external' parameters; parameters on which information is presented in this paper are marked with an asterisk:

Derailments and Collisions	* - speed
	* - nature of impact
	* - position in train of wagons involved
Fires	* - extent
	* - duration
	* - severity

Railway system	- effects of track on derailment probability - effects of associated systems on collision/derailment probability - effects of train or wagon type
External parameters	- population distribution - topography - weather

The eventual aim of this study, if pursued to its ultimate conclusion, would be to define categories of each parameter (eg speed categories, track categories, weather categories) and determine the relative frequency of occurrence of each.

3. ASSEMBLY OF A DATA BASE

It was decided, as a first step in this study, to investigate to what extent its aims could be realised with a survey of just two years' of relevant data, those for accidents occurring in 1976 and 1977. Accident report forms were obtained for accidents on or adjacent to passenger lines from the Railway Inspectorate (RI), and for dangerous goods accidents from the British Rail Dangerous Goods Branch (BRDG). The data contained in these reports on individual incidents was analysed and collated by the author in a rather different manner to that employed by the RI[2]. This implies no criticism of the RI, merely a difference in purpose of the two approaches. Ready-analysed data on freight derailments that occurred on plain line between 1970 and 1979 was obtained from British Rail's Derailment Investigation Service (DIS)[3] and collated at SRD. None of these data sources provides a subset of either of the others, and there is no guarantee that a fully comprehensive list of accidents for 1976 and 1977 has been obtained. Between them, though, the three sets of data should provide a record of the vast majority of significant accidents to have occurred during the years considered.

4. INCIDENCE OF DANGEROUS GOODS ACCIDENTS (BRDG DATA)

Some basic data on the incidence of various accident types is given below:

Total no. of incidents	202
No. derailments	76
No. collisions	6
No. fires	15
No. releases of material	85

These numbers appear appreciable, but serve only to illustrate the importance of an analysis of the frequency of the parameters affecting cargo damage, for there were in fact no incidents leading either to significant releases of hazardous materials or to any hazard to members of staff or the public during the period considered. Little direct information on accident severity was given on the accident report forms, but what was available all pointed towards the quite minor nature of the incidents in general.

5. INCIDENCE OF DERAILMENTS (RI & DIS DATA)

Total numbers of freight and passenger derailments were collected, and split into those occurring on plain line and those not occurring on plain line (eg on points or switch and crossing gear). Data were analysed both in terms of number of trains involved in a derailment, and in terms of total number of wagons or other

vehicles involved. Some data on the incidence of various important parameters associated with derailments is given below.

5.1 Effect of Speed at Time of Derailment

Figure 2, below, illustrates the proportion of 300 freight derailments in which the train speed at the time of derailment fell into various speed categories:

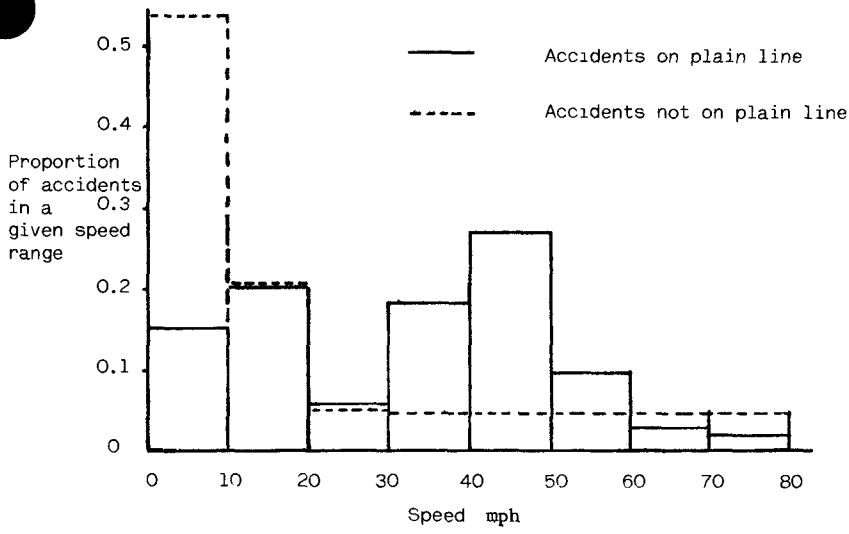


Figure 2: Freight Derailments Categorised by Speed

While derailments on plain line tend to occur either at low speed or around normal running speeds, those not on plain line tend to occur almost exclusively at low speeds. Figure 3, below, illustrates the difference between ordinary 4-wheel freight wagons and bogie vehicles (such as are used for the transport of spent fuel flasks):

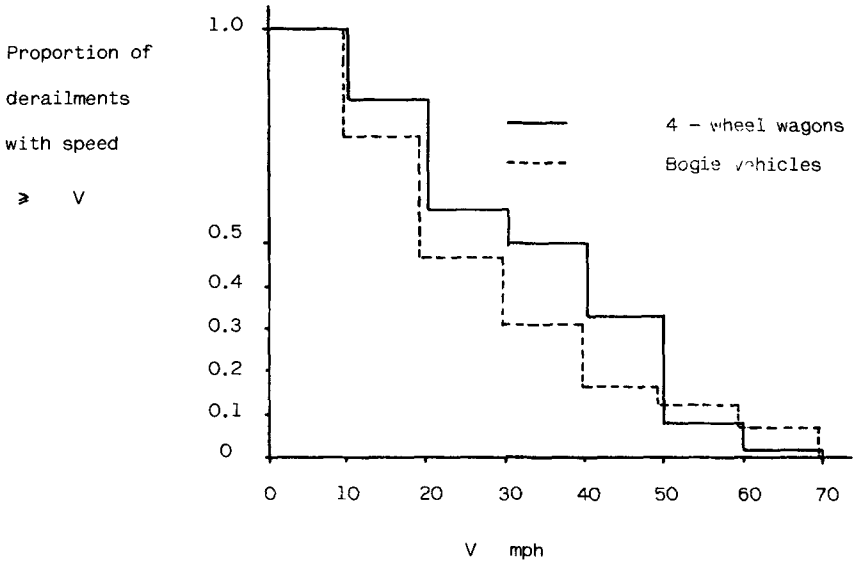


Figure 3: Cumulative Speed Categorisation of 4-wheel and Bogie Derailments

This figure is a cumulative curve - ie the ordinate represents the proportion of accidents with speed greater than or equal to the abscissa. Bogie vehicles are clearly less likely to derail in the middle speed range, though slightly more prone to high speed derailment, presumably because they normally operate at these speeds, while most other freight traffic was or is restricted to a 45 or 60 mph speed limit.

5.2 Effect of Position in Train of Wagons Derailed

The proportions of derailed wagons from the front ¼, middle ½ and rear ¼ of the trains was analysed in two independent ways. Firstly, DIS data was used to investigate the proportions of first wagons to derail from the various parts of the train. Secondly, the RI data was used to investigate the proportions of all wagons derailed which came from the corresponding sections of trains. The results corroborate each other, showing that at higher speeds, wagons to the rear of trains are much more prone to derailment than those at the front, which are also less susceptible than those in the middle. Normalised proportions are shown below for first wagons to derail:

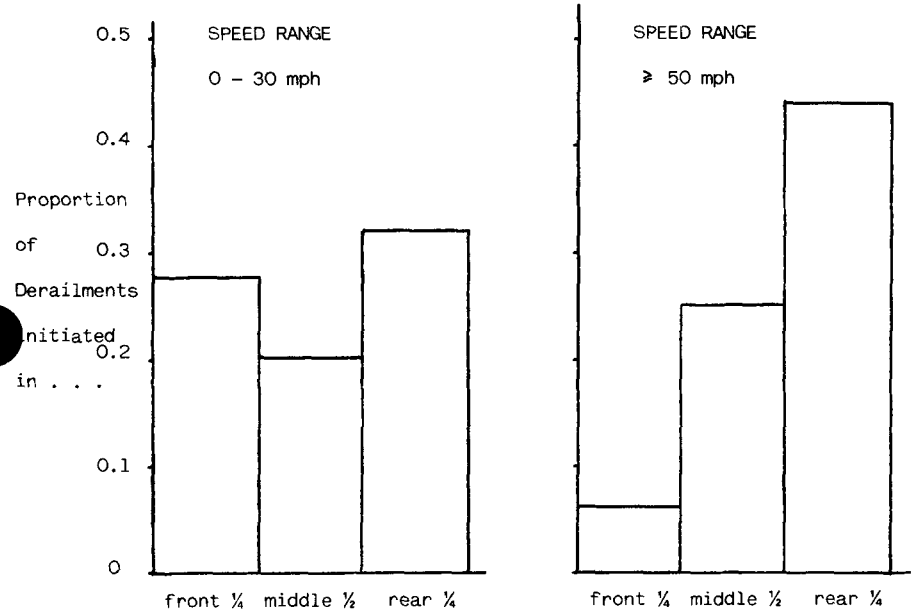


Figure 4: Position in Train of First Wagon to Derail

The effect is not apparent at lower speeds.

5.3 Incidence of Other Parameters

The incidence of other parameters was noted:

- All-wheels derailment occurs in about 50% of incidents; in other derailments, no significant travel off the track is anticipated.
- About 25% of derailed freight wagons turn on their sides.
- 0.64 unprotected lines are blocked per freight derailment, and 4% of line blockages lead to collision with another train.
- A further 4% of all freight derailments lead to a collision with objects off the track.

6. INCIDENCE OF COLLISIONS (RI DATA)

Of the numerous collisions reported over the two-year period, 47 incidents (84 train collisions) were selected as being relatively non-trivial. Some measure of their severity may be gained from the facts that about half led to derailment of one or more wagons and about a quarter to significant mechanical damage. Interestingly, 3 instances were reported where vehicles (but not packages carried) were punctured by broken rails or sharp projections from other trains. Little data on closing speeds was available. Information on closing angle is summarised in figure 5 below:

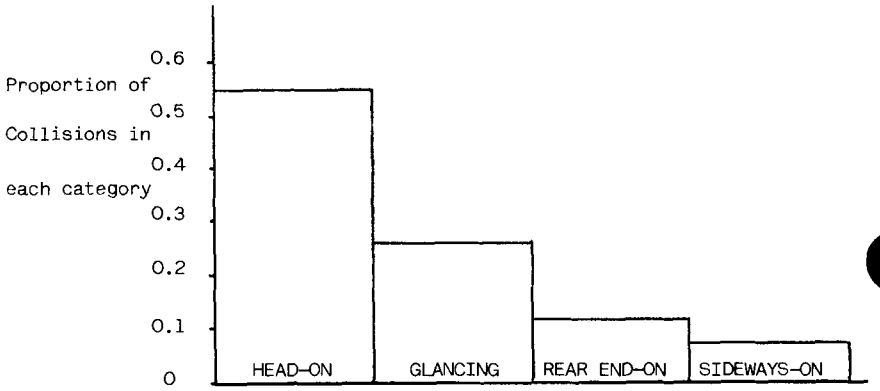


Figure 5: Collision Angle Data

It is interesting to note that less than 10% of collisions involve a direct side-on impact, which would expose the package carried to maximum impact. The nature of objects with which trains collided is demonstrated by the following percentages:

75% - other trains
12% - relatively light-weight objects on track*
5% - buffers or end-stops*
8% - relatively massive objects off the track, eg buildings or bridges.

*Many other incidents in these categories were discounted as being trivial.

7. INCIDENCE OF FIRES (RI DATA)

The main feature of the data on fires for 1976/77 was the lack of major incidents. No fires involving more than one wagon, no BLEVEs or other explosions, and no damage to property or persons were recorded. 4 instances of a wagon being burnt out were recorded; only one of these was a freight wagon. Information on extent, duration and severity is summarised below:

Extent - No fires covering an area greater than 1 wagon

Duration - Some data available, but of little relevance, since most long duration fires are those allowed to burn out by the emergency services when safety and economic factors are unaffected

Severity - No major fires involving highly flammable cargoes - only ordinary building materials were involved in major fires.

8. CONCLUSIONS

Descriptions of accident environments of various quality have been obtained. The quality of quantitative descriptions of accident incidence for Great Britain resulting from this study to date is as follows:

Derailments - good

Collisions - inadequate speed data

Fires - inadequate data base

External effects - unanalysed

9. FURTHER PROGRESS

Since the work described above was performed, mileage data has been obtained from British Rail, which will be combined with data on incidence of accidents to yield per mile (or kilometer!) accident probabilities. A comprehensive report covering this field will then be issued by SRD[4]. Possibilities for future studies include the following:

i) Extension of the data base on fires by looking at a longer historical period;

ii) Modifying the procedure for collecting information on collisions in order to provide better descriptions of future accidents;

iii) Examining the variation of accident frequency with population density.

10. ACKNOWLEDGEMENTS

The author wishes to thank the staff of British Rail and the Railway Inspectorate, both for the direct provision of information and for their extensive helpful discussion on and around this subject, and the Health and Safety Executive, who funded this project.

11. REFERENCES

1. Fryer L S and Kaiser G D; "TIRION 4 - a computer program for use in nuclear safety studies" SRD report R134.

2. Railway Accidents; HMSO, 1977 and annually.

Duncan I G T et al; Proc Inst Mech Eng, 191 323 (1977).

Taig A R; SRD report in preparation.

SESSION VII
LEAKAGE, LEAKRATE AND SEALS

IMPORTANCE OF LEAK FLOW MODE ASSUMPTIONS FOR INTERPRETATION OF PACKAGE LEAKTIGHTNESS TESTS AND INFLUENCE OF TEST CONDITIONS ON CONCLUSIONS

by B.P. KIRCHNER and P.T. BLUM
TRANSNUCLEAIRE

SUMMARY

When leakage from package occurs, leak geometry and flow mode through leaking path are unknown.

Therefore, even if we consider just leakages through apertures (excluding other phenomena such as permeation of materials, especially gaskets), the assessment of the leakage rate of packages should take into account all possible flow modes, in particular the adiabatic one as well as isothermal ones.

Furthermore, special attention should be paid to test conditions which may correspond to flow modes different from that corresponding to actual conditions during transport of a considered package. The test results should also be corrected to take into account the influence of parameters such as gasket temperature and gas viscosity.

This paper recalls the general equations giving the leakage rate for different flow mode assumptions and compares the results for typical practical instances. It shows that the conservative assumption depends upon the upstream-to-downstream pressure ratio in test and operation conditions.

1. INTRODUCTION

In order to determine the gas leakage rate of a package under normal transport or accident conditions from that which was observed during the acceptance tests of packaging at the end of fabrication, one commonly uses formulas which are based on a given flow mode assumption and permit to express the leakrate as a function of corresponding physical and chemical characteristics (upstream and downstream pressures, temperature, nature and viscosity of gas, etc).

In particular, one often refers to document ANS-N14-5 <1> in Appendix B of which the gas leakage for laminar, transition, or molecular flow modes is estimated by the following equation :

$$L = 3810 \frac{D^3}{a} \left[323 \frac{D}{\eta} (P_u^2 - P_d^2) + \left(\frac{\pi}{M}\right)^{1/2} (P_u - P_d) \right] \qquad (1)$$

where :

L = gas leakage rate (atm.cm³/s) at gas temperature T (K)

D = leakage path diameter (cm)

a = leakage path length (cm)

η = gas vicosity (centipoise) at gas temperature T (K)

P_u = gas upstream pressure (atm. abs.)

P_d = gas downstream pressure (atm. abs.)

M = gas molar mass (g)

Equation (1) represents the sum of gas flows in the laminar and molecular flow modes. The first term in the bracketed expression pertains to laminar flow; the second term pertains to molecular flow. Equation (1) can be used for pure laminar or pure molecular flow because the unwanted term becomes insignificant.

Nevertheless it should be paid attention not to exceed the application range of Equation (1) which may not cover the conditions of certain utilizations.

For instance, for an air leakage at 20°C (M = 29 g, η = 0.0183 cP) through a 1 cm long and 0.03 cm diameter leaking path, under an upstream pressure P_u = 2 atm and a downstream pressure P_d = 1 atm, the leakrate derived from Equation (1) is L = 164 atm.cm³/s. If one calculates the corresponding air velocity, one will observe that it reaches 2300 m/s, i.e. much more than the velocity of sound in air (approximately 330 m/s).

This shows that Equation (1) is not applicable in the case when a high value of P_u/P_d induces a too high velocity of gas.

As this may be the case under operation or test conditions, it is important to evaluate leakage rates not only by Equation (1), but also by other equations suitable for representing other flow modes, in particular adiabatic flow mode with or without friction.

As concerns friction, we know that it would just reduce leakrate. Therefore we will analyse the single case of adiabatic flow mode without friction. Then we will compare the results from basic flow modes (especially laminar and adiabatic) under various conditions, so as to conclude for the most conservative one.

2. GAS LEAKAGE ADIABATIC FLOW MODE WITHOUT FRICTION

2.1 General Equation for Adiabatic Flow Mode without Friction

The basic equations for adiabatic flow mode without friction are :

$$P/\rho = \frac{R}{M} T \qquad (2)$$

$$P/\rho^{\gamma} = \text{Constant} \qquad (3)$$

$$\rho S V = \text{Constant} \qquad (4)$$

$$d(V^2/2) + dP/\rho = 0 \qquad (5)$$

$$d(V^2/2) + dU + d(P/\rho) = 0 \qquad (6)$$

with :

S = variable cross section area of leaking path (cm^2)

P = gas pressure (atm)

ρ = gas density (g/cm^3)

V = gas velocity (cm/s) — at any given section of leaking path

U = gas internal energy (erg/g)

T = gas temperature (K)

= perfect gas constant (8.315×10^7 erg/K)

M = gas molar mass (g)

γ = gas adiabatic flow constant

The above equations permit to express the gas leakrate under adiabatic flow without friction at gas temperature T, as follows:

$$L = P_u \left(\frac{P_d}{P_u}\right)^{1/\gamma} \times \frac{\pi D^2}{4} \times \left(\frac{2\gamma}{\gamma-1}\frac{RT}{M}\right)^{1/2} \times \left[1 - \left(\frac{P_d}{P_u}\right)^{(\gamma-1)/\gamma}\right]^{1/2} . \qquad (7)$$

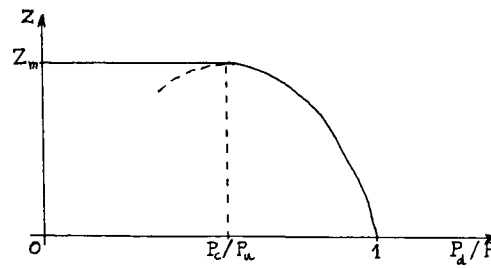
2.2 Adiabatic Leakage Equation in the Case of Air

For air leakage (M = 29 g and γ = 1.4) at room temperature (T = 293 K), we obtain :

$$L = 7.67 \times 10^4 \times P_u \times \frac{\pi D^2}{4} \times Z \text{ (atm.cm}^3\text{/s at 293 K)} \qquad (8)$$

with: $$Z = \left(\frac{P_d}{P_u}\right)^{1/\gamma} \left[1 - \left(\frac{P_d}{P_u}\right)^{(\gamma-1)/\gamma}\right]^{1/2} \qquad (9)$$

The curve representing the variations of Z as a function of $\frac{P_d}{P_u}$ is given on the diagram below :



Z becomes maximum, Z_m, when P_d/P_u equals the critical ratio P_c/P_u:

$$\frac{P_c}{P_u} = \left(\frac{2}{\gamma+1}\right)^{\gamma/(\gamma-1)} = 0.528$$

$$Z_m = \left(\frac{2}{\gamma+1}\right)^{1/(\gamma-1)} \left(\frac{\gamma-1}{\gamma+1}\right)^{1/2} = 0.259$$

At that time, the gas velocity $V_c = \left(\gamma \frac{P_c}{\rho_c}\right)^{1/2}$ is equal to the sound velocity in gas.

When P_d becomes less than P_c, leakrate L obviously cannot decrease. In fact in the smallest cross section of leaking path is establishing pressure P_c, and the flow remains drived according to adiabatic mode by the upstream part where pressure P is between P_u and P_c. The downstream part of leaking path ($P_c > P > P_d$) with sonic flow is no longer involved in the determination of the leakrate which remains constant and equal to:

$$L_m = 6.02 \times 10^4 \times P_u \times D^2 \times Z_m$$

$$= 1.56 \times 10^4 \times P_u \times D^2 \text{ (atm.cm}^3\text{/s at 293 K)} \qquad (10)$$

3. APPLICATION TO THE ACTUAL CASE OF A PACKAGING FOR SHIPPING RADIOACTIVE MATERIALS

As an example of application, we will take the case of the TN 12/1 packaging loaded with 12 irradiated fuel assemblies of type PWR 15 × 15.

3.1 Main Data concerning TN 12/1 Package

- Total activity of fission gases released in the cavity of TN 12/1 package by cladding rupture of all fuel pins (accident conditions):

$$A = 8760 \text{ Ci}$$

- Value of A_2 for the mixture of gaseous radionuclides inside the cavity of TN 12/1 package, according to Regulations <2> :

$$A_2 = 995.6 \text{ Ci}$$

- Volume available for gases inside the package cavity :

$$v = 2730 \text{ ℓ}$$

- Accident conditions :

  . pressure of the gaseous mixture inside cavity : 3.84 atm
  . temperature of the gaseous mixture inside cavity : 640 K
  . temperature in gasket region : 428 K

3.2 Leakrate Criterion for Fabrication Acceptance Test

According to Regulations, the leaktightness control performed at the end of fabrication should guarantee that the accumulated loss of radioactive contents will not exceed $A_2 \times 10^{-3}$ in a period of one week (see para. 233 of <2>): this is the most severe requirement specified by Regulations.

Therefore, the maximum allowable rate of activity release is :

$$L_A = \frac{A_2 \times 10^{-3}}{7 \times 24 \times 3600} = 1.65 \times 10^{-6} \text{ Ci/s}$$

In particular, this rate of activity release should not be exceeded under the accident conditions (such as defined in Section 3.1) where the upstream-to-downstream pressure ratio is maximum. Under those conditions, the activity per unit volume of gases released beyond gasket (428 K, 1 atm) is :

$$A_V = \frac{A}{V} \times \frac{640}{428} \times \frac{1}{3.84} = 1.25 \text{ Ci/}\ell \text{ (at 428 K and 1 atm)}$$

so that the rate of gas leakage measured at the same temperature (428 K) should be less than :

$$L = \frac{L_A}{A_V} = 1.32 \times 10^{-3} \text{ atm.cm}^3\text{/s at 428 K}$$

But, to guarantee that this requirement is met, the leaktightness test at the end of fabrication may be performed under various conditions. Hereafter, we will consider two of them :

The first conditions of leaktightness test are :

(I)
- Upstream pressure (outside) : $P_u = 1$ atm
- Downstream pressure (inside) : $P_d = 0$
- Temperature of air : $T = 293$ K (20°C)

The second conditions of leaktightness test are :

(II)
- Upstream pressure (inside) : $P_u = 5$ atm
- Downstream pressure (outside) : $P_d = 1$ atm
- Temperature of air : $T = 293$ K (20°C)

For both above conditions of leaktightness test, we will successively calculate the values of leakrate criterion such as derived from both the laminar and adiabatic assumptions.

(a) Leakrate Criteria derived from Laminar Assumption

For gases X and Y in the laminar flow mode (leakage rates greater than 10^{-6} atm.cm³/s) and using Equation (1) above, the correlation between gas-to-gas leakage rates is given by the following equation :

$$L^{\ell}_X = L^{\ell}_Y \frac{\eta_Y (P_u^2 - P_d^2)_X}{\eta_X (P_u^2 - P_d^2)_Y} \qquad (11)$$

where :

L^{ℓ}_X = atm.cm³/s of gas X at gas X temperature

L^{ℓ}_Y = atm.cm³/s of gas Y at gas Y temperature

Taking subscript Y as referring to accident conditions with a gaseous mixture of which viscosity at the gasket temperature of 428 K is $\eta_Y = 0.0255$ cP, and subscript X as referring to test conditions with air of which viscosity at 293 K (20°C) is $\eta_X = 0.0183$ cP, we respectively obtain for test (I) and (II) conditions:

$$L^{\ell}_{(I)} = 0.134 \times 10^{-3} \text{ atm.cm}^3/\text{s at 293 K (20°C)}$$

$$L^{\ell}_{(II)} = 3.21 \times 10^{-3} \text{ atm.cm}^3/\text{s at 293 K (20°C)}$$

(b) Leakrate Criteria derived from Adiabatic Assumption

For gases X and Y in the adiabatic flow mode, the correlation between gas-to-gas leakage rates is derived from Equation (7) as follows :

$$L^{a}_X = L^{a}_Y \frac{k_X P_{uX} Z_X}{k_Y P_{uY} Z_Y} \qquad \text{where} \qquad k = \left(\frac{2\gamma}{\gamma - 1} \frac{RT}{M}\right)^{1/2} \qquad (12)$$

and Z is given by Equation (9) or equal to Z_m, according to each gas X or Y considered.

For both test (I) and test (II) conditions, we have, from Section 2.2 :

$k_X = 7.67 \times 10^4$ cm/s and $Z_X = Z_{mX} = 0.259$

For accident conditions, the gaseous mixture involved in TN 12/1 package includes monoatomic and diatomic gases in such proportions that, on the average:

$\gamma_Y = 1.6$ and $M_Y = 25.5$ g

This leads to :

$k_Y = 8.63 \times 10^4$ cm/s and $Z_Y = Z_{mY} = 0.310$

Thus, for test (I) and (II) conditions, we respectively obtain from Eq. (12):

$$L^{a}_{(I)} = 0.255 \times 10^{-3} \text{ atm.cm}^3/\text{s at 293 K (20°C)}$$

$$L^{a}_{(II)} = 1.275 \times 10^{-3} \text{ atm.cm}^3/\text{s at 293 K (20°C)}$$

4. COMPARISON BETWEEN LAMINAR AND ADIABATIC FLOW MODES

4.1 Criterion for the Leaktightness Test at the End of fabrication

Thus it appears that when the leaktightness test at the end of fabrication is performed according to test (I) conditions, the leakrate criterion derived from laminar assumption, $L^{\ell}_{a(I)}$, is smaller than the leakrate criterion derived from adia-batic assumption, $L^{a}_{(I)}$. Therefore, in that case, laminar assumption is more conservative than adiabatic assumption.

On the contrary, when the leaktightness test is performed according to test (II) conditions, $L^{\ell}_{(II)}$ is much greater than $L^{a}_{(II)}$, so that, is that case, laminar assumption is much less conservative than adiabatic assumption.

Therefore we calculated, for the same value of maximum allowable leakrate under accident conditions ($L = 1.32 \times 10^{-3}$ atm.cm³/s at 428 K : see Section 3.2), the values of leakrate criterion as a function of the pressure inside cavity during leaktightness test (assuming a constant external pressure of 1 atm), such as derived from both the laminar and adiabatic assumptions. Both corresponding curves are shown on Figure 1 hereafter.

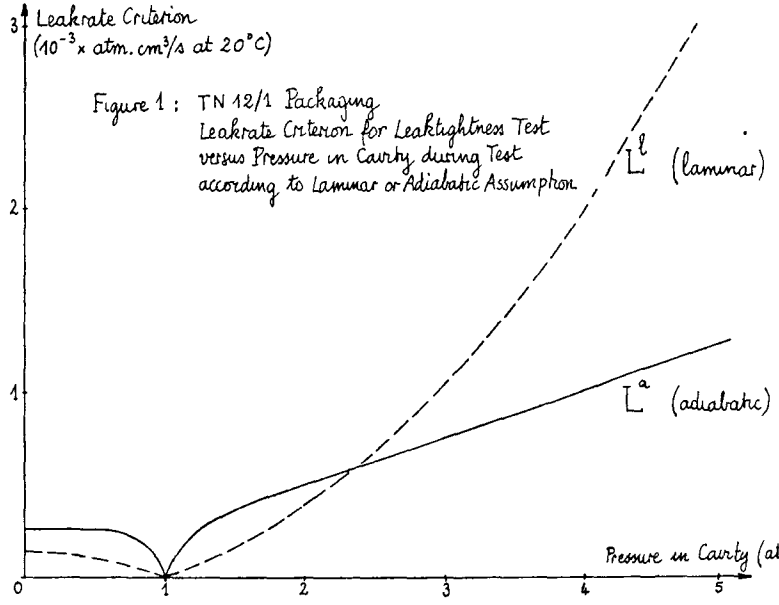


Figure 1 : TN 12/1 Packaging
Leakrate Criterion for Leaktightness Test versus Pressure in Cavity during Test according to Laminar or Adiabatic Assumption

Figure 1 shows that as long as the cavity pressure during test remains less than about 2.3 atm, laminar assumption is more conservative. But, as soon as the cavity pressure during test exeeds that value, adiabatic assumption becomes the most conservative one, and the difference may become very high when test pressure reaches 5 or 10 atm as frequently used.

In addition, it should be noted that diameter D of leaking path (assumed to be cylindrical) is about 10 times smaller when derived from adiabatic Equation (7) than when derived from laminar Equation (1).

4.2 Leaktightness Control before each Shipment

TN 12/1 packaging is designed to be shipped dry under partial vacuum (0.5 atm at thermal equilibrium). That characteristic, added to the possibility of easily performing, during the loading operation at reactor site, a leaktightness control of all gasket interspaces (by pressure rise method), permits to guarantee zero-activity release (except perhaps by permeation through gaskets, as Mr. KOWALEWSKY will explain) under normal transport conditions, during several months, and even one year (according to the criterion selected for this leaktightness control before dispatch), i.e. during a period which largely exceeds the duration of one transport.

A calculation similar to that of Section 3.2 shows that, under the operating conditions of TN 12/1 packaging, adiabatic assumption is then much more conservative than laminar assumption, namely:

- for a given criterion value of the leaktightness control before dispatch, the duration of zero-activity release guaranteed is much shorter according to adiabatic than laminar assumption.

- for a given minimum duration required for zero-activity release guaranteed, the maximum allowable leakrate for the leaktightness control before dispatch is significantly lower according to adiabatic than laminar assumption.

5. CONCLUSION

In the containment evaluation of packagings for transport of radioactive materials, a laminar flow mode assumption is commonly used to show that the regulatory requirements of activity release are met.

Nevertheless, as the geometric and physical characteristics of leakage are unknown, and as they may be different according to each shipment, all possible flow modes through leaking path should be considered so as to show the most conservative one. And this essentially depends upon the upstream-to-downstream pressure ratios under test and operation conditions.

In this paper, we analysed the particular case of TN 12/1 package and we have shown that, by reference to accident conditions, laminar assumption is well the most conservative one when the leaktightness test is performed with the whole cavity of packaging being under partial vacuum. But when the leaktightness test is performed with the packaging cavity being somewhat over-pressurized (for example 5 or 10 atm), then adiabatic assumption should be retained as the most conservative one, more especially as laminar flow mode assumption is rather doubtful when a high pressure ratio induces a too high velocity of gas, and as laminar equation is very sensitive to gas temperature.

6. REFERENCES

<1> Draft American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials ANS-N 14-5.

<2> I.A.E.A. : Regulations for the Safe Transport of Radioactive Materials - 1973 Revised Edition (As Amended) - Vienna, 1979.

DISCUSSION

Question by J. A. Andersen, USA: In your abstract you mentioned "... the results for typical practical instances." What methods do you utilise to measure actual leakrates?

Secondly, how do you correlate leakrate to the requirement, as you mentioned in your presentation, of containing an A2 x 10^{-3} limit of loss? (e.g. how do you relate leakrate to containment or loss of radioactive material?)

Answer: To measure actual leakrates, we generally utilise either the pressure rise method or the helium test method using a mass spectrometer.

The rate of activity release under normal transport or accident conditions is derived from the leakrate under test conditions in two steps:

- Firstly, from the leakrate under test conditions, we derive the leakrate under normal transport or accident conditions using the formula corresponding to the most conservative assumption, which may be either laminar or adiabatic assumption according to the considered case.
- Secondly, we calculate the corresponding rate of activity release from the isotopic composition of fission gaseous products in the package cavity under normal transport or accident conditions.

LEAK TESTING AND ACTIVITY LEAKAGE RATE EVALUATION - PRACTICAL EXPERIENCES, FIRST APPROACHES TO SOME SYSTEMIZATION AND OUTSTANDING PROBLEMS.

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1. INTRODUCTION

BAM is the competent authority in the Federal Republic of Germany for Type B packaging and special form radioactive material testing, and at present is performing a research program on the assessment and comparability of different leak test methods and on their appropriate utilizations.

For the different types of containments for radioactive materials certain activity leakage rate limits have been established. For practical reasons most leakage tests cannot be conducted with the actual radioactive contents, and then the problem of correlation between activity leakage rates and conventionally measured volumetric leakage rates arises. More detailed interpretations of test results taking into consideration the potential leak geometry and the physico-chemical nature of the leaking material may contribute to making test requirements more practicable and reasonable, e.g. by distinguishing between permeation leakage and leakage through any real openings.

At two IAEA organizings in 1976[1] and 1979[2] the author has outlined his view of leak testing assessment. In the meantime we gained quite more experience in practical testing and it seems useful to discuss some of our new results but also to point out some remaining difficulties.

2. LEAKTIGHTNESS CRITERIA

2.1 Type A Packagings

Concerning Type A packagings there is the very general and only qualitative requirement of preventing any release of radioactive contents. The great majority of transports of radioactive materials is carried out in Type A packages but most activity, if taken as multiples of A_2 values, is circulating within Type B packages. One example: Out of about 140,000 Type A transports 1975 in the Federal Republic of Germany about 80 % dealt with radiopharmaceuticals on an average and mainly with about 10^{-3} A_2 as radioactive contents.[3] By some tests done at Harwell the mainly used "fruit cans" have been shown to be helium-leaktight if obtained from a reputable manufacturer and if sealed properly[4]. This leaktightness generally will not be affected by normal conditions of transport. Hence the requirement of preventing any release of radioactive material is fulfilled at least for liquid and solid contents.

Type B packagings for spent nuclear fuel may contain activities up to between 10^6 and 10^7 A_2, theoretically each presenting a risk factor of up to 10^{10} times that of representative Type A packaging and therefore they give much more reason for quantitative leaktightness considerations.

2.2 Leakage Limits for Type B Packagings and Special Form Radioactive Material

a) Type B packaging after tests specified by IAEA Safety Series No. 6[5] paras 709-714, simulating the damage resulting from normal transport conditions and minor mishaps: $\leq A_2 \cdot 10^{-6}$ per hour (2 mCi $\leq A_2 \leq$ 1000 Ci).

b) Type B packaging after tests specified by IAEA Safety Series No. 6 paras 718-721, simulating the damage resulting from severe transport accidents: $\leq A_2 \cdot 10^{-3}$ per week for B(U); $\leq A_2$ per week for B(M).

c) Sealed source at normal usage and reasonable accidental risks: < 5 nCi within 4 hours immersion (ISO 2919[6]; ISO TR 4826[7]).

d) Special form radioactive material after tests specified by IAEA Safety Series No. 6 paras 732-735, simulating the damage from severe transport conditions: $\leq$ 50 nCi within 4 hours immersion.

Fig. 1 shows the range of limits for the activity leakage rates mentioned above; the limits for sealed sources are comparable to that for Type B packagings. On the right side some test gas (helium or air) leakage rate limits are shown which are considered to be equivalent to the activity leakage rate limits for sealed sources according to IAEA Safety Series No. 37[8] para. 749 and ISO TR 4826[7].

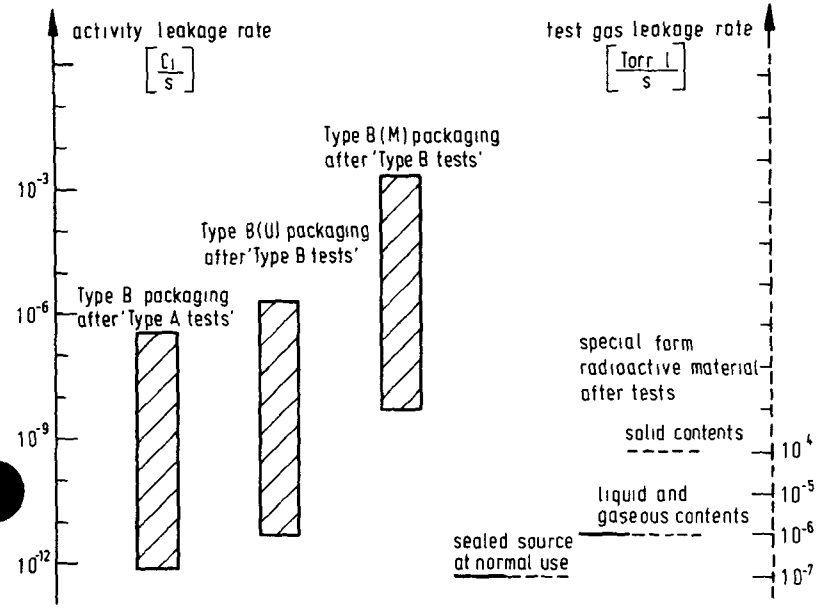

Fig. 1: Activity leakage rate limits and test gas leakage rate limits for Type B packagings and sealed radioactive sources

IAEA Safety Series No. 6 specifies A_2-values for about 260 radionuclides. Taking into account their specific activities mass leakage rate limits can be calculated. Fig. 2 shows the frequency distribution of those mass leakage rates, each square representing one radionuclide. On the left side one mainly finds radionuclides with high specific activities corresponding to short half lives. In Fig. 2 the positions of the main plutonium isotopes and the resulting position of typical reactor plutonium have been indicated. Another interesting selection might include e.g. all radionuclides to be considered for storage of spent nuclear fuel. For comparison the mass leakage rates corresponding to 10^{-7} mbar•l•s^{-1} of air (leaktight according to ANSI N 14.5)[9] and to 10^{-8}mbar•l•s^{-1} of helium are indicated.

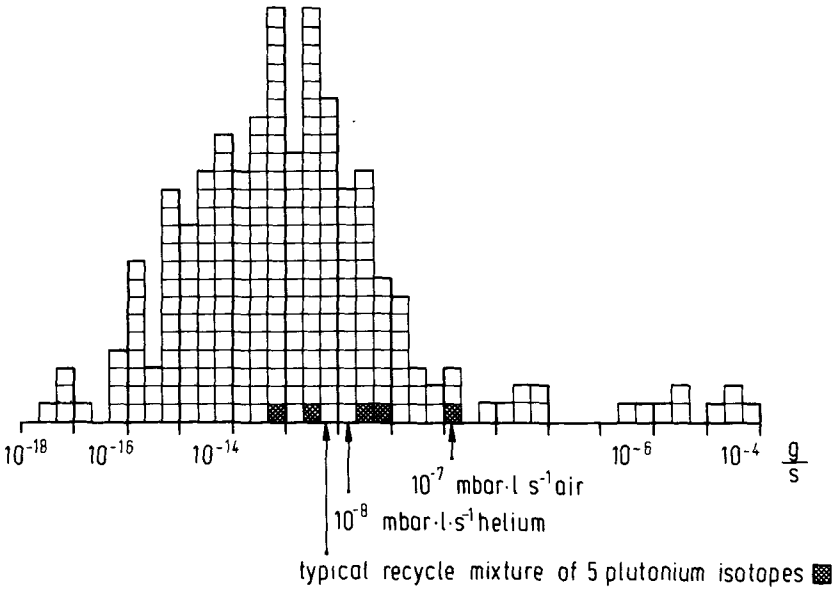

Fig. 2: Frequency distribution of mass leakage rates corresponding to $A_2 \cdot 10^{-6} \cdot h^{-1}$

3. EXPERIENCE FROM PRACTICAL TESTING

3.1 Leakage Through Openings

Radioactive material may escape from the containment system of the packag through different kinds of leaks and by different mechanisms of leakage. Mostl interesting is leakage through any unintended openings. Fig. 3 shows some correlations between leakage and geometrical leak dimensions. If we consider e.g. an opening of 1 µm diameter corresponding to a standard leakage rate of about 10^{-4}mbar•l•s^{-1} for apertures, or of about 10^{-8}mbar•l•s^{-1} for capillaries with length 1 cm one should bear in mind that such a small opening would be clogged by any liquid film if present or, in case of powders, by some individual particles. For formation of a stream of gas bubbles the driving gas pressure must be

higher than the blocking pressure of the liquid at the opening; in this example using water more than 2.5 bar would be required.

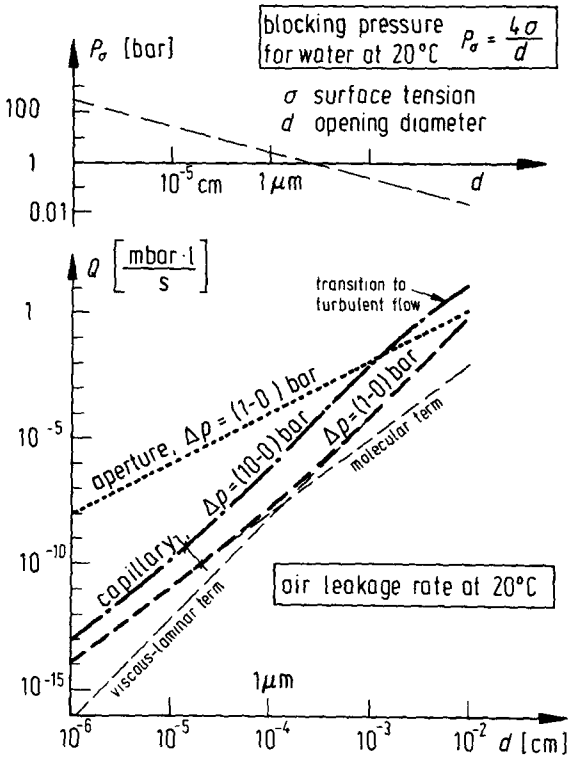

Fig. 3: Correlations between leakage rates and opening diameter

3.2 Permeation Leakage

Permeation of gases through elastomeric seals includes surface solubility and bulk diffusion. Fig. 4 illustrates the generalized rise of permeation with time. One practical example: Silicone rubber o-rings show a standard helium permeation rate of about 10^{-4} mbar·l·s^{-1} per m length at ambient temperature. Measuring the leakage rate immediately after having filled such a container with helium could exclude the existence of any minute openings corresponding to a standard helium leakage rate of more than 10^{-8} mbar·l·s^{-1}.

Our evaluation of PAT-1 (plutonium air transportable package developed in USA) quantitative leak test results[10] before and after NRC criteria testing suggested that all copper gaskets had failed and that the elastomeric gaskets had maintained their sealing function. All 5 leakage rate values measured after the tests were found relatively close together corresponding to the expected permeation rate through Viton-o-rings used (but see discussion remarks!).

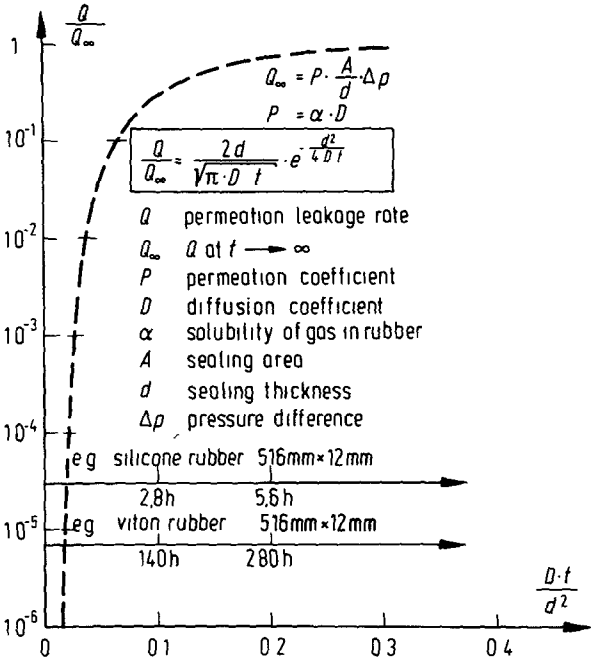

Fig. 4: Rise of permeation leakage with time

3.3 Temperature Limits for Elastomeric Gaskets

Some drop tests with packagings made of cast iron at -40 °C raised no problems with any bulk brittleness, but the Viton seals failed even before the drops. At about -20 °C standard Viton becomes stiff; with further decreasing temperature elastomeric material contracts more than metal resulting in a growing gap. Table 1 shows some first results from laboratory experiments using small flanges of 16 mm nominal width mounted on a copper bar dipping into liquid nitrogen. With a cooling rate of about -5 °C per minute the different elastomeric gaskets showed sharp increases of the helium leakage rates at the temperatures listed in this table.

IAEA Safety Series No. 6 para. 213 requires design of any transport pack[illegible] and selection of materials accordingly to a temperature range between -40 °C a[illegible] +70 °C. In case of sufficient internal heat generation by radioactive decay Viton could be used even at ambient temperatures of -40 °C, otherwise one might restrict to a Type B(M) approval.

material	lower temperature limit[1]	measured by BAM	upper temperature limit[1] [2]
standard Viton	-25 °C	-22 °C [3]	200 ... 260 °C
Viton GLT	-40 °C	--	200 ...
silicone rubber	-60 °C	about -65 °C	200 ... 260 °C
EPDM	-50 °C	about -55 °C	120 ... 170 °C

Table 1: Temperature limits for elastomeric gaskets

1) informations from manufacturers and literature
2) higher temperature limits correspond to short-time heating
3) 2 compounds from different suppliers

3.4 Additional Comments

It seems practicable to comply with the IAEA activity leakage limits applying appropriate engineering leakage tests. In most cases it is useful and sometimes even absolutely necessary to consider the different barriers against release of radioactive contents. For this some further generally accepted informations on activity release e.g. from the different kinds of spent fuel elements is needed.

Another group of leak test methods concerns the control of proper sealing after each loading. In most cases wrong handling would result in relatively large leaks and therefore simple and less sensitive leak test methods may be acceptable. But we have to take into account possible disturbing influences on test results as e.g. the presence of water or moisture. Some experimental results with hairs and thin wires within a sealing device showed the bubble test as a wet method to be about one order of magnitude less sensitive than pressure rise measurement as a dry method.

The assumption accepted by French authorities that maintenance of underpressure in a packaging during transport time would prevent escape of radioactivity[11] is right concerning solid and liquid contents but does not take into account that permeation of e.g. Krypton 85 depends on the relevant partial pressure difference and therefore can occur even against a higher total pressure. We found that helium at 100 mbar shows the same permeation rate through silicone rubber plates into air of 1 bar or into vacuum.

4. CONCLUSIONS

From our practical experience, experimental work, study of relevant literature and theoretical considerations we feel that the IAEA activity leakage limitation requirements are practicable and generally can be verified using conventional leak test methods. But it will never be possible to derive simple figures for volumetric leakage rate limits for all radionuclides of interest each with its different potential particular physico-chemical properties. It seems possible however to develop some general recommendations for the design criteria, for prototype and production leak testing and for leak testing before shipment appropriate to the most representative kinds of radioactive contents.

BIBLIOGRAPHY

(1) H. Kowalewsky "Correlation and Applicability of Different Leak Test Methods" in "Transport Packaging for Radioactive Materials" IAEA Vienna 1976

(2) H. Kowalewsky "Leakage Rate Limits; Correlations between Activity Leakage Rates and Volumetric Leakage Rates", Consultant Review Paper IAEA Vienna 1979

(3) S. Hartwig "Schwachstellen und Risikoabschatzung beim Transport radioaktiver Materialien" Battelle Report BF-R-62.891-1 Frankfurt 1978

(4) F.E. Dixon, L.R. Cohen "Testing Type A Packagings" in "Tests on Transport Packaging for Radioactive Materials" IAEA Vienna 1971 and personal communication at Harwell 1979

(5) "Regulations for the Safe Transport of Radioactive Materials 1973 Revised Edition (As Amended)" IAEA Vienna 1979

(6) International Standard ISO 2919 "Sealed Radioactive Sources - Classification" 1980

(7) ISO Technical Report 4826 "Sealed Radioactive Sources - Leak Test Methods" 1979

(8) "Advisory Material for the Application of the IAEA Transport Regulations" Vienna 1979

(9) "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials" ANSI N14.5-1977

(10) *Safety Analysis Report NUREG-0361 "Plutonium Air Transportable Package Model* PAT-1" 1978

(11) P. Rocco "Preuve de la conformité de château Essor aux limites de relâchement définies par l'AIEA en 1973" in "Transport Packaging for Radioactive Materials" IAEA Vienna 1976 and some Safety Reports prepared by Transnucléaire

DISCUSSION

Question by C. B. G. Taylor, United Kingdom: Would you please summarise the methods you use to assess the significance, in terms of radioactive leakage, of a measured gas flow rate.

Answer: Radioactive leakage rates as well as gas flow rates depend on the particular contents (physical, chemical nature, partial or absolute pressure, ...), on the leak geometry, on the specific activity ... In practice we decide each case on its own merits but we are preparing some more general rules for the most typical cases, this being one target of our current research program.

Question by S. F. Johnson, Sweden: 1. You mention an elastometric gasket material of type EPDM. What does this abreviation stand for? 2. Did you perform leakage tests with neoprene gaskets? *Neoprene contains the lubricant molybdenum sulphide which makes the* gasket, as far as I know, more resistant to low temperatures.

Answer: 1. Ethylene-propylene-diene rubber. M designates a saturated chain of the polymethylene type (ASTM 1418). 2. Further we tested only one type of nitride-butadiene used for vacuum connections which failed at -32°C; chloroprene gaskets should show similar behaviour.

Comment by J. A. Andersen, USA: In the compact of your paper, in the second paragraph of 3.2, you make the observation concerning the PAT-1 package (USA) that the copper seals had failed after drop tests and that the leak rates measured corresponded to the use of viton seals. This is not the case. The viton O-ring in the TB-1 containment vessel in the PAT-1 package is completely carbonised by the 582°C temperature that is attained by the containment vessel in the fire test, as reported in SAND 76-0587 (the PAT-1 document) and the leak rates reported, of 10^{-5} to 10^{-6} atm cc/s, are entirely a function of the copper seal and the martensitic stainless steel containment vessel. In fact, following drop tests only, we did perform leak rate tests on the containment vessel and they were N.D.L. (No Detectable Leak; L < 10^{-10} atm cm^3/s).

Answer: Neither in the safety analysis report NUREG-0361 nor in report SAND 76-0587 (= NUREG/CR-0030) "PARC (Plutonium Accident Resistant Container) Program Research, Design and Development" did I find any reference to leakage rates measured between the sequential tests. And only on page 217 of SAND 76-0587 did I find a reference to "decomposed viton O-ring material."

Viton gaskets can be used in air for a short time up to temperatures of more than 300°C, and in a helium atmosphere they should endure even higher temperatures. Therefore we had not excluded their function after the fire tests.

Because all inner containment vessels of PAT-1 showed leakage just after the fire tests (according to your statement) with very similar leakage rate values, this effect in any case should not have been a random one. An interpretation could be based on the formation of narrow gaps along the copper ring caused by the high inner pressure at high temperatures during the fire tests. Such an evaluation results in a gap width in the order of 10^{-3} to 10^{-2} mm preventing release of the large majority of the mostly much bigger powder particles.

Question by K. Goldmann, USA: Leakage rates through seals are a function of O-ring groove configuration and applied pressure as well as O-ring material characteristics. Is it possible that groove geometry and seal pressure might not have been appropriate for low temperature operation of the viton O-rings? Laboratory test data obtained for TN cask seals which utilise carefully shaped, dove-tail O-ring grooves and viton O-rings, show that leakage rates remain below allowable values over the complete range of anticipated temperatures from -40°C to the maximum temperatures expected after the regulatory fire.

Answer: One main parameter is the appropriate choice of the O-ring material whose characteristics may vary considerably even for one class of elastomers like viton. Appropriate groove geometry might be another important parameter. But my general statement remains valid: Referring to any arbitrary viton material one cannot take its sealing properties down to -40°C for granted.

AN ASSESSMENT OF THE CONTAINMENT OF THE UK 250 LITRE PLUTONIUM NITRATE PACKAGE

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UKAEA
Risley

Introduction

The UK programme for the development of Fast Reactors involves the shipment of plutonium enriched fuel elements from the production plant at Windscale to the Prototype Fast Reactor at Dounreay; with the return of the plutonium nitrate from Dounreay to Windscale for fuel element production the closing of the Fast Reactor Fuel Cycle is demonstrated.

A package to contain 250 litres of this product has therefore been designed and manufactured to meet the Type B requirements of the IAEA Safety Regulations which means that the containment standard of the package has to be at a level to ensure compliance with regulatory normal and accident conditions of transport.

This paper outlines the leak tightness requirements of the package and the testing and quality assurance procedures which have been carried out to ensure that the containment standards have been met.

Standard of Containment

The IAEA Regulations (Ref 1) stipulate an allowable activity leakage as the criterion for the containment standard of Type B package designs. For normal conditions of transport the requirement is that the loss of radioactive contents is to be restricted to $A_2 \times 10^{-6}$ per hour, where A_2 is the activity in curies of the particular radionuclide as listed in Table VII of the regulations. For accident conditions the loss of radioactive contents is to be restricted to $A_2 \times 10^{-3}$ over a period of one week for Unilateral approval or to A_2 over a week for Multilateral approval. In the case of a mixture of different radionuclides the permissible activity release of each radionuclide shall be in accordance with the "unity rule" as defined in para 408 of the regulations.

The regulations specify an "activity leakage" criterion, but for design and operational purposes, leak tightness expressed in "engineering units" must be specified. In the case of plutonium nitrate with known isotopic fractions this is calculated as follows:

The 'unity rule' specifies that where the identity and activity of each radionuclide is known, the permissible activity of each radionuclide R_1 R_2 ... R_n shall be such that $F_1 + F_2 + \ldots F_n$ is not greater than unity where:

$$F_1 = \frac{\text{Activity of } R_1}{A_1\ (R_1)} \qquad F_2 = \frac{\text{Activity of } R_2}{A_1\ (R_2)} \qquad F_n = \frac{\text{Activity of } R_n}{A_1\ (R_n)}$$

A_1 (R_1 R_2 ... R_n) is the value of A_1 and A_2 as appropriate for the nuclide R_1 R_2 ... R_n. Applying this rule to calculate the regulatory allowable release of activity:

$$\frac{\text{Release of } R_1}{\text{Allowable release } R_1} + \frac{\text{Release of } R_2}{\text{Allowable release } R_2} + \cdots \frac{\text{Release of } R_n}{\text{Allowable release } R_n} \leq 1$$

To convert allowable 'activity' release to a leak rate expressed in 'engineering' terms

Let: Fraction of radionuclide R_1 in package contents = F_1
Specific activity of R_1 = S_1 curies/gram
Concentration of radioactive element in contents = C gms/cm^3
Allowable leakage from containment = L cm^3/hour

Considering the regulatory allowable loss of radioactive material to be restricted to $A_2 \times 10^{-6}$ curies/hour for normal conditions of transport the 'unity rule' can be expressed as:

$$\frac{F_1 \, L \, C \, S_1}{A_2(R_1) \times 10^{-6}} + \frac{F_2 \, L \, C \, S_2}{A_2(R_2) \times 10^{-6}} + \cdots \frac{F_n \, L \, C \, S_n}{A_2(R_n) \times 10^{-6}} = 1$$

or alternatively: $$\sum \frac{F \, S}{A_2(R)} = \frac{10^{-6}}{L \, C}$$

For plutonium nitrate of the following assumed isotopic composition the allowable leakage expressed in 'engineering' units can therefore be calculated:

Radionuclide (R)	Fraction (F) %	Specific Activity (S) Curies/gm	A_2	$\frac{F. S.}{A_2}$
Pu 238	0.15	1.7×10	0.003	8.50
Pu 239	71.20	6.2×10^{-2}	0.002	22.07
Pu 240	24.50	2.3×10^{-1}	0.002	28.18
Pu 241	3.20	1.1×10^{2}	0.1	35.20
Pu 242	0.95	3.9×10^{-3}	0.003	0.01
				93.96

Assuming, for calculation purposes, the concentration of plutonium in the solution (C) to be maximum of 0.3 grams/cm^3

$$\sum \frac{F. S.}{A_2(R)} = 93.96 = \frac{10^{-6}}{L \cdot 0.3}$$

The allowable liquid leakage L is therefore 3.5×10^{-8} cm^3/hr.

$= 9.8 \times 10^{-12}$ cm^3/sec

Containment Test Standard

Testing a vessel requiring such a high standard of containment can only be satisfactorily carried out by helium mass spectrometry. The allowable liquid leakage has therefore to be converted to an equivalent helium gas leakage.

The 250 litre package has been designed for a maximum normal operating pressure of 100 psig to contain the radiolytic gas evolved from the plutonium nitrate during transport and this value is therefore taken as the maximum driving pressure causing leakage.

The size of pore which will allow the liquid leakage with a specified driving pressure can be calculated from Poiseuille's law for laminar flow in a long narrow tube with constant cross section of leak path.

$$L = \frac{\pi}{8} \frac{r^4}{b} \frac{(p_1 - p_2)}{\eta}$$

where:

r	= 'pore' radius. cm	
b	= 'pore' length. cm	= 0.53 (assumed for gasket)
$p_1 - p_2$	= pressure difference. dyne/cm^2	= 6.89×10^6 (100 psi)
η	= dynamic viscosity. poise	= 0.013
L	= liquid leak rate. cm^3/sec	= 9.8×10^{-12}

Assuming the leakage is past a single pore the radius r = 1.25×10^{-5} cm.

The gas leakage for laminar and molecular flow through small capillaries can be estimated from the following equation based on an equivalent hole size (Ref ?).

$$L_G = \frac{3810\ d^3}{b} \left[323 \frac{d}{\eta} (Pu^2 - Pd^2) + \sqrt{\frac{T}{M}} (Pu - Pd) \right]$$

where:

L_G	= Gas leakage. atmos cm^3/sec	
d	= Pore diameter. cm	= 0.53
b	= Pore length. cm	= 0.03
η	= Dynamic viscosity. centipoise	= 1
P_d	= Fluid downstream pressure. atmos abs	7.8
P_u	= Fluid upstream pressure. atmos abs	= 295
T	= Absolute temperature. °K	= 4
M	= Gas molecular weight. atomic mass unit	

For a pore of radius 1.25×10^{-5} cm

$$L_G = 9 \times 10^{-9} \text{ atmos cm}^3\text{/sec of helium}$$

This value is the basic standard to which the containment vessel must be tested in order to comply with the regulatory requirements for normal conditions of transport. It is within the measurement capacity of helium mass spectrometry, but would be extremely difficult to achieve by other methods. The above calculation assumes a single pore leakage path. In practice it is much more likely that there will be multiple pores. However, it can be shown that by assuming a single pore the most pessimistic result is obtained in determining the leak test standard. The effects of surface tension may also be taken into account in determining the allowable liquid leakage; it can vary from approximately doubling the calculated leak rate with perfect wetting to complete blockage of the pores with imperfect wetting. Taking the worst possible effects of surface tension the helium leak test specification reduces to a detection value of 4×10^{-9} atmos cm^3/sec.

The Containment System of the Package

The basic package design consists of a stainless steel containment vessel carried in a [illegible] steel framed outer structure which gives shielding, impact and thermal protection. This paper is concerned mainly with the inner containment vessel, the closures of which are shown diagrammatically on Fig 1. These are:

(i) Plug - Item 1. This component is screwed into the top of the filling neck of the vessel against an elastomer seal. Its prime purpose is to prevent any possible surging of the liquid into the area of the main seal. It is inaccessible for leak testing.

(ii) Main seal. The flange (item 2), seal assembly (item 3) and valve (item 4) constitute the innermost containment barrier which can be tested in service and is nominated as the main seal. For test purposes the space between this seal and the plug is pressurised to 100 psig with a metered quantity of helium through the valve. This condition is maintained during transit and if the plug seal is effective a positive pressure difference ensures no out leakage from the vessel.

(iii) Top cover - Item 5. The top cover is the final closure and is connected after removal of the package from the plant filling facility. It is sealed by double elastomer 'O' rings with the facility for interspace leak testing.

(iv) In addition to the above seals there is a la Calhène flange below the top cover which forms an alpha seal when transferring the vessel from the filling and emptying facilities. This is not an essential part of the sealing system of the package. Although not leak tested, it provides a supplement to the closure system.

It can be seen from the diagram that all the above seals act in series to form an effective barrier against leakage. The basic problem is testing the system to ensure that the required standards are achieved and maintained through all stages of the transport operation.

Containment System Leak Testing

The leak testing procedures to be carried out may be considered as follows:

(i) Demonstration of the capability of the design to meet the required standards.

(ii) Tests carried out during manufacture and prior to operational use.

(iii)Routine testing at the filling plant prior to each shipment.

(iv) Tests to be carried out during planned periodic maintenance.

Design Capability

To demonstrate that the design as proposed was capable of meeting the regulatory containment requirements under normal and accident conditions, a full scale prototype package was subjected to drop tests with associated leak tests. Initially a series of model tests was carried out, from which the most damaging attitudes of drop were determined. Two series of drop tests were then carried out on the full scale prototype.

The first series consisted of a one metre drop on to a steel punch followed by a nine metre drop on to a face of the package.

The second series consisted of a one metre drop on to a steel punch, on the region immediately adjacent to the vessel filling neck followed by a one metre drop on to a steel punch, on the face opposite to the face that was the target of the nine metre drop in the first series.

Helium mass spectrometer leak tests were carried out prior and subsequent to the drop tests; one to establish the leakage rate of the main seal alone, the other to establish the rate when the main seal was backed up by the top cover, thus forming a double seal acting in series. Prior to each of the test series the interspace between the plug and the main seal was pressurised to 100 psig helium by using the valve on the mass seal flange. When leak testing the main seal singly a special top cover was used which enabled connections to be made to the vacuum pump and mass spectrometer. To leak test the main seal and top cover acting in series an additional test cover was located over the top cover, again to enable a working vacuum to be established. The test results are given in Table 1.

To complete the final testing of the prototype vessel a complete leak test on the whole body and seals was performed. The vessel was pressurised to 100 psi helium and then sealed inside a polythene bag which was subsequently evacuated to reduce the volume of air between the vessel and the bag. Over a test period of 24 hours no leakage was detected with the mass spectrometer calibrated to 1×10^{-9} atmos cm^3/sec.

The results of the tests carried out on the combined main flange and top cover seals (tests 2, 3, 6 and 7 in Table 1) and the whole body test described above showed no detectable leakage. As the instrument sensitivity for these tests was better than 4×10^{-9} atmos cm^3/sec the ability of the design to meet the IAEA prescriptions for a Type B package was confirmed.

Testing of Production Vessels Prior to Operational Use

The production vessels were leak tested at the manufacturer's works to ensure that they complied with the manufacturing specification before delivery. The objectives of the tests were to prove the integrity of the vessel body welds and the performance of individual and multiple seals. For the whole body weld test the vessel was pressurised to 100 psig helium and for the seal tests the plug/main seal interspace pressurised to 100 psig helium. Again a special test cover was mounted over the top cover to enable a working vacuum to be obtained in the multiple seal tests. The results of the tests were given in Table 2. No leakage could again be detected with the two seals acting in series and even when considering the single main seal the leak tightness was appreciably better than that found on the prototype vessel and in itself almost met the required regulatory standard.

Part of the quality compliance checks to be carried out during planned periodic maintenance of the package is to ensure that there is no deterioration in leak tightness from the original standard. Meaningful results can only be obtained from these tests if the test conditions and techniques as applied to new vessels are exactly reproduced. To establish these conditions a further series of leak tightness checks will be carried out at the dispatch site prior to filling with active liquid. The results of these tests will be used as a basis for comparison with the results from planned maintenance.

Routine Tests at Each Shipment

In an ideal situation each vessel when filled and fully closed should be checked against the regulatory standard of leak tightness for normal conditions of transport prior to each shipment. Because of operational limitations at the filling plant an overall test to this extremely stringent requirement cannot be carried out on the combined seals when fully assembled. Tests which can be more readily carried out are helium mass spectrometer on the main seal followed by interspace pressure drop on the top cover. However the prototype and production

new vessel leak tests indicated that the required standard of 4×10^{-9} atmos cm^3/sec could not readily be achieved on the single main seal. To overcome this difficulty the following has been agreed with the UK Competent Authority:

As each new vessel has been fully leak tested the ability of the design to meet the regulatory standards has been demonstrated. Therefore if new or fully inspected seals are used and the closure procedure used on the new vessels prior to leak testing exactly reproduced, a less stringent test on the closures is considered acceptable for routine shipments. A detailed closure procedure, together with a main seal mass spectrometer test to better than 10^{-5} atmos cm^3/sec and an interspace pressure drop test on the top cover has therefore been specified to conform to this requirement.

Planned Periodic Maintenance

To ensure that the vessel continues to meet the regulatory requirements during service a comprehensive maintenance programme is to be carried out at regular intervals. Included in this programme will be a full leakage test on the vessel body and closures.

The vessel will be decontaminated, dried out and subjected to leak tests identical to those carried out on the new vessel as outlined above and using the same test equipment. This will demonstrate that there has been no deterioration of the vessel or closures during service.

Conclusions

Since the issue of the 1973 IAEA Regulations there has been much discussion on the relative merits of specifying an allowable leakage based on activity as compared with one expressed in engineering units. Basically the use of an activity criterion, which is related to the potential hazard to transport workers or members of the public, is objective and avoids the setting of arbitary standards which may result in either over or under design of package containment systems. This is particularly relevant when the radioactive material being carried is in liquid or gaseous form.

Irrespective of the criterion used it is apparent that a major problem in all packages designed for an extremely high standard of leak tightness is proving that they comply with that standard during the whole period of development and operation. However, the detailed testing procedures carried out on the UK 250 litre plutonium nitrate package as outlined in this paper show that compliance with the IAEA regulations for containment under normal and accident conditions of transport has been achieved and Type B(U) certificate of approval has been granted by the UK Competent Authority.

References

1. IAEA Safety Series 6 - Regulations for the Safe Transport of Radioactive Materials - 1973 Revised.

ANSI N 14.5 (1977) - American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials.

TABLE 1

Test No	Description	Test Duration	Ref leak Atmos cm^3/s	Instrument Sensitivity Atmos cc/sec	Leakage Atmos cc/sec
	Prior to drop				
1	Main Seal (Item 1)	24 hours	4×10^{-7}	1×10^{-9}	5.3×10^{-5}
2	Main Seal backed up by Cover (Item 2)	48 hours	4×10^{-7}	1×10^{-9}	Nil detected
	After drop				
3	Repeat of test 2	2 hours (See Note 1)		1.75×10^{-10}	Nil detected
4	Repeat of test 1	9 hours		4.2×10^{-6}	2.5×10^{-4}
	Prior to drop				
5	Repeat of test 1	24 hours	4.37×10^{-10}	$4-6 \times 10^{-10}$	3.2×10^{-5}
6	Repeat of test 2	10 hours	4.3×10^{-9}	3.1×10^{-10} to 4.3×10^{-9}	Nil detected
	After drop				
7	Repeat of test 6	9 hours		2.3×10^{-11}	Nil detected
8	Repeat of test 5	- (See Note 2)		1.9×10^{-11}	(See Note 2)

TABLE 2

Test No	Description	Duration Hours	Ref leak Atmos cc/s	Instrument Sensitivity	Results Atmos cc/sec
1	Body welds	24	4.7×10^{-7}	7.5×10^{-9}	Nil detected
2	Main seal	24	4.1×10^{-7}	4×10^{-9}	2.5×10^{-8} to 5×10^{-9}
3	Main seal backed up by cover	72	4.1×10^{-7}	4×10^{-9}	Nil detected

Note 1: Instrument error detected after 2 hours.

Note 2: The leak was greater than the instrument sensitivity setting. An isolation pressure rise test was therefore carried out and the leakage calculated to be 1.15×10^{-5} atmos cc/sec.

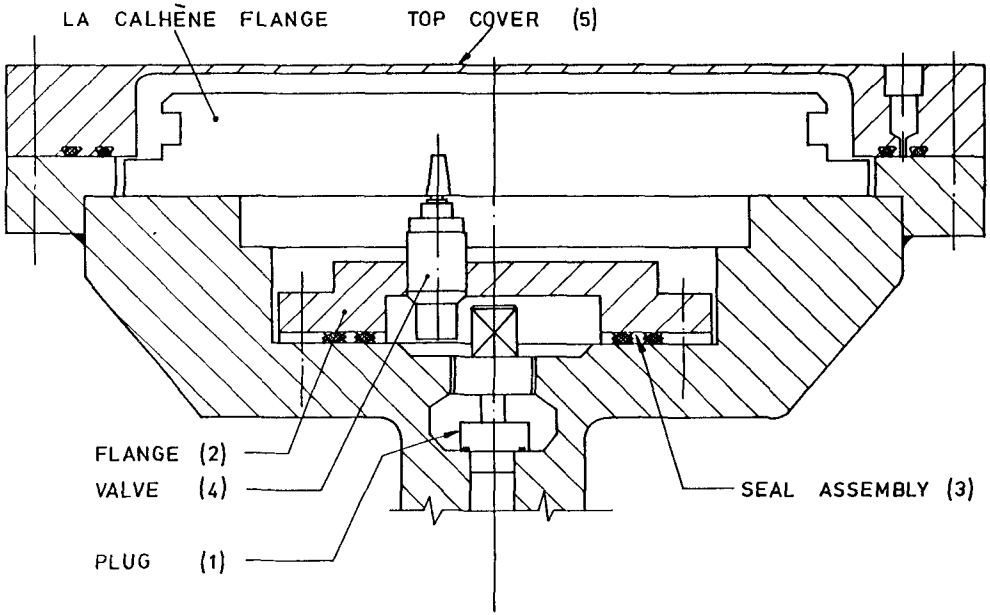
LA CALHÈNE FLANGE
TOP COVER (5)
FLANGE (2)
VALVE (4)
PLUG (1)
SEAL ASSEMBLY (3)

FIGURE 1

Question by G. Cohendy, France: If I have understood correctly, it is the cask with real dimensions which was tested using the 9 meter drop test. If so, what is its weight?

Answer: Yes, the package weight is just under nineteen tonnes. A full size prototype was drop tested.

Question by M. S. T. Price, United Kingdom: Are there any interpretational problems which arise from radiolytic gas generation under operational test conditions? For example, does it obscure leakage or is it a variable quantity?

Answer: The interspace between the main seal and the plug is pressurised to 100 pounds per square inch helium for test purposes. This helium pressure is maintained during transport. Therefore there should be no problem with radiolytic gas evolved affecting the leak test interpretation.

Question by J. D. Hart, United Kingdom: To achieve such high standards of testing have you found that the outgassing of the test system takes a considerable time? What is the time taken to conduct the routine tests?

Answer: Routine shipments have not yet been carried out so that I cannot give the time for seal testing. I agree that outgassing could be a problem in obtaining the necessary vacuum prior to helium mass-spectrometer testing.

LEAKAGE OF RADIOACTIVE POWDERS FROM CONTAINERS

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

1. INTRODUCTION

This study has been directed at determining an appropriate standard of leak tightness for containment of plutonium oxide powder during shipment. The IAEA regulations specify an allowable leakrate in terms of mass of PuO_2 escaping in unit time; the experimental work described in this paper has been carried out in order to interpret the IAEA requirement in terms of a flowrate of gas, which is more easily measured in practice.

2. REGULATORY REQUIREMENTS

The allowable leakrate for plutonium oxide under normal transport conditions is defined by the IAEA Regulations (Safety Series No 6) as 10^{-6} x A2 value per hour. Each isotope of plutonium has its individual A2 value. In the UK the plutonium transported originates mainly from reprocessing of fuel from MAGNOX reactors, and a typical isotopic analysis is given in table 1.

Table 1: Composition of Magnox Plutonium

Pu 238	Pu 239	Pu 240	Pu 241	Pu 242
0.1%	76.9%	19.3%	3.2%	0.5%

The specific activity of this material is 3.63 Ci/g, dominated by the contribution from Pu 241. Using the prescribed A2 values, calculation shows that the allowed leakrate for PuO_2 of this composition is 2.8×10^{-8} g/hr, equivalent to 7.8×10^{-12} g/sec.

3. PACKAGING SYSTEM

In the UK, PuO_2 is packed for transport into a sealed metal can of about 2 litres capacity. Several varieties of can are in use, ranging from tin-plate with crimped lids to stainless steel cans sealed by welding. Inside the can there is usually a PVC bag, enclosing an inner screw-cap container into which PuO_2 is loaded in a glove-box. The safety case for the package depends on the leak tightness of the outer metal can.

4. ENVIRONMENTAL CONDITIONS

Cans are sealed containing air at room temperature and standard pressure. After sealing, self-heating by PuO_2 can cause a modest temperature rise of say 50°C, with an associated pressure rise of about 0.2 atmospheres. The pressure developed is the driving force for leakage of PuO_2 from the can.

Other factors, which can arise as a result of accident conditions, will in-

crease this driving force. Metal cans containing PuO_2 are transported in the UK in a variety of wood cadmium containers, and fire tests of specimen containers at 800°C for 30 minutes have resulted in an increase of temperature in the can of a further 70°C. The pressure outside the can is normally 1 atmosphere; this can fall to 0.5 atmospheres in an aircraft depressurisation accident, increasing the differential pressure by 0.5 atmospheres.

It is clear that the maximum differential pressure across a leak-path in a can will not exceed 1 atmosphere in normal or accident conditions; the experimental work reported in this paper has been restricted therefore to this pressure range.

5. MECHANISM OF POWDER LEAKAGE

It is postulated that PuO_2 can be carried through an aperture by a gas stream. For a given leak-path and differential pressure, the mass transfer of powder should be proportional to the air flowrate, and the constant of proportionality will be the density of the aerosol.

6. GASEOUS LEAK-RATES

If the pressure of a gas on one side of an orifice is P1, and the pressure on the other, P2, is progressively reduced, the flowrate through the orifice increases up to an equilibrium value, at which the linear flowrate reaches the velocity of sound. It can be shown theoretically that the maximum gas flow through a circular orifice can be related simply to the orifice diameter. In practice, measured gas flows reach equilibrium at about 60-80% of the theoretical maximum.

Flowrates of air through the orifice plates used in the experimental programme have been measured for a range of differential pressures up to 1 atmosphere, and the results are shown in figure 1.

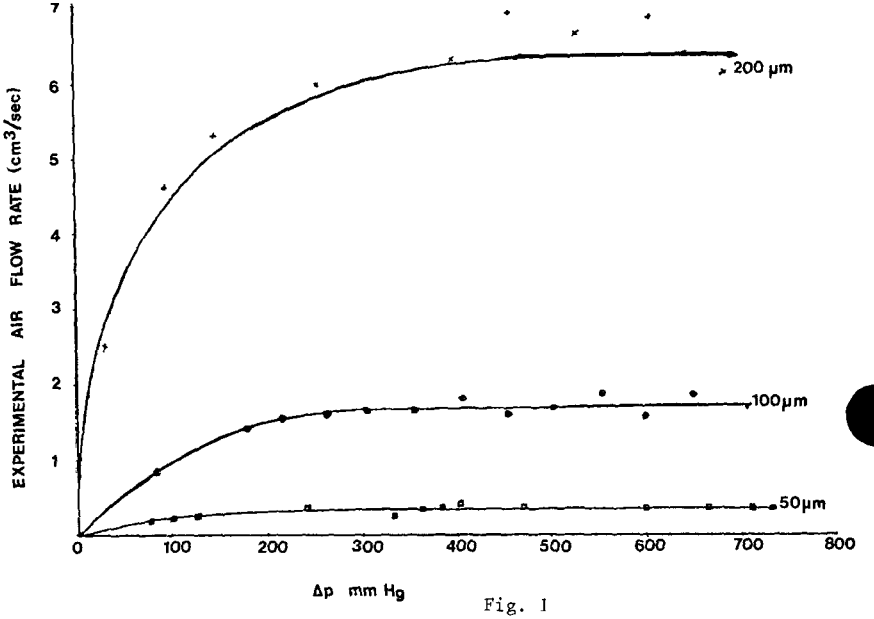

Fig. 1

7. EXPERIMENTAL PROGRAMME

An experimental programme has been carried out to establish a correlation between mass transfer and gas flowrate for a range of orifice sizes. In the experiments reported here, UO_2 powder with a particle size range of 1-20 μm has been used to simulate PuO_2. The apparatus used is shown in figure 2.

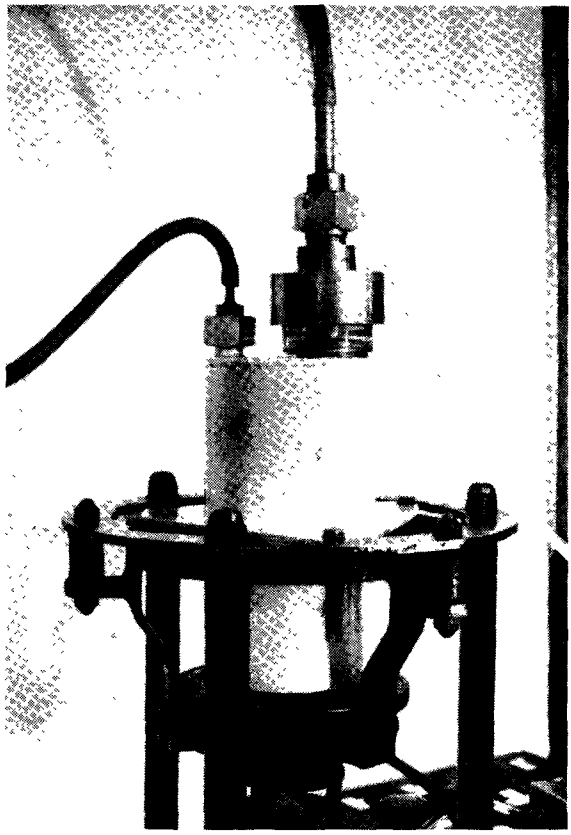

fig. 2

It consists of a metal container 100 mm diameter and 200 mm high, mounted on a vibrating table. At the top of the container is a pipe through which nitrogen or air can be admitted to pressurise the can. There is also an adaptor to which a thin metal orifice plate 10 mm diameter can be fitted, and a mixing chamber.

These items are shown in more detail in figure 3.

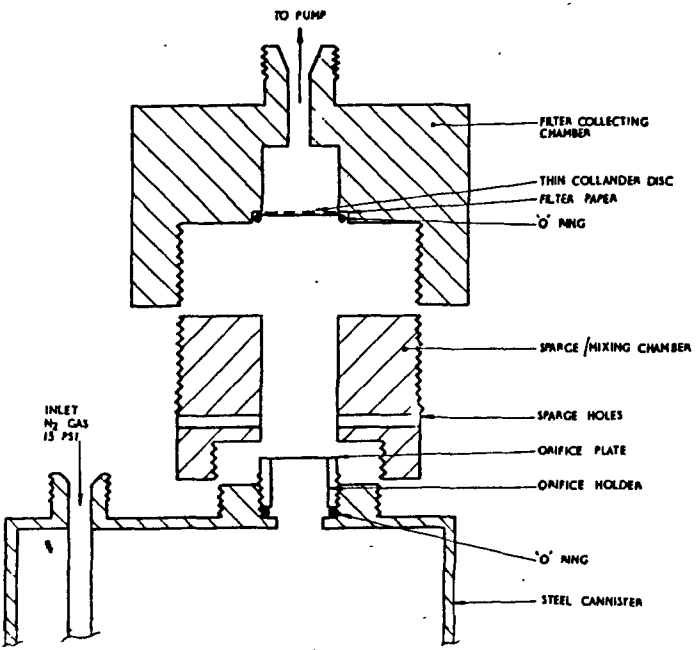

fig. 3

UO_2 particles emerging from the orifice plate enter a mixing chamber connected to a vacuum pump. Air, drawn into the mixing chamber through radial holes, sweeps the UO_2 particles onto a 0.2 um filter, which retains them. The mixing chamber has undergone a series of modifications, and in its present form it has been shown that 75-80% of the UO_2 emerging from the orifice is collected by the filter. Orifice plates are available in the range 200 μm to 5 μm diameter.

To carry out an experiment, about 50 g UO_2 powder is loaded into the can, and the pressure inside the can is adjusted to an appropriate value. The can is attached to the vibrating table, and this is energised. At the end of the run, the filter paper is removed, and the amount of UO_2 collected is assayed by alpha-counting.

An electron microscope is used to assess particle sizes.

The amount of UO_2 transferred through orifices of 200 μm and 100 μm diameter as a function of differential pressure is shown in figure 4.

The curves obtained are similar in form to those shown in figure 1, which showed variation of air flowrate with pressure. Once choked flow conditions are established, both air flowrate and UO_2 flowrate become constant.

Using a constant differential pressure of 1 atmosphere across the orifice, the amount of UO_2 collected is proportional to time. This is illustrated in figure 5 for orifices of 200 μm and 100 μm diameter.

Similar information for smaller orifices of 50 μm and 25 μm diameter is shown in figure 6.

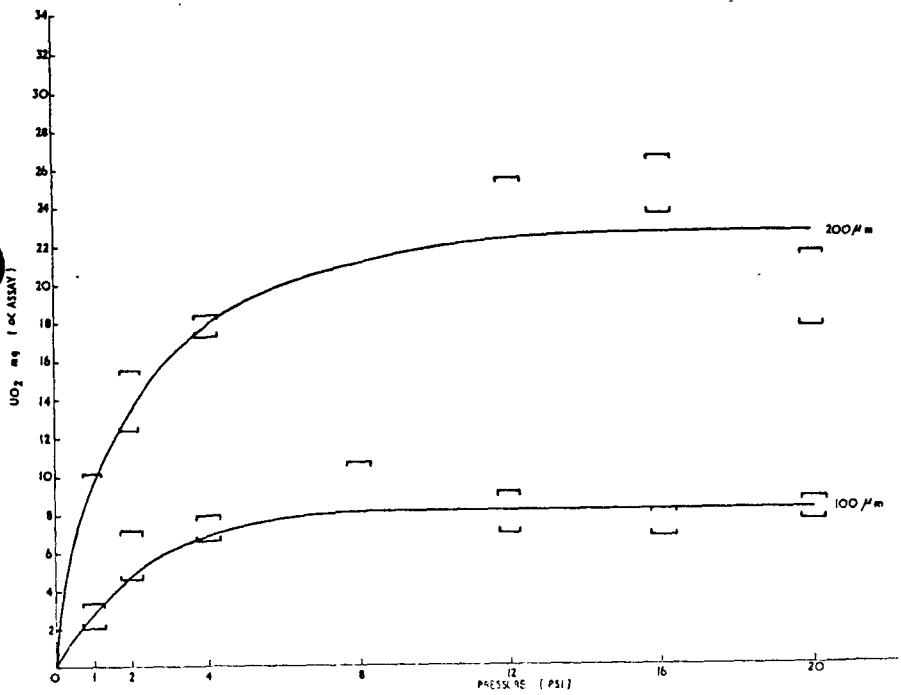

fig. 4

These results were obtained with the test container in the upright position, and the orifice 190 mm over the UO_2 layer. Under these conditions the aerosol density, for a range of orifice diameters and differential pressures, can be calculated to be between 1.7×10^{-7} g/ml and 4.5×10^{-7} g/ml. If the container is tilted, to bring the orifice closer to the UO_2 surface, the aerosol density increases by a factor of about 5. Under these conditions, blockage of the orifice with UO_2 frequently occurs during the experiment. With the orifice below the UO_2 layer, blockage always occurred before any escape of UO_2 was detected.

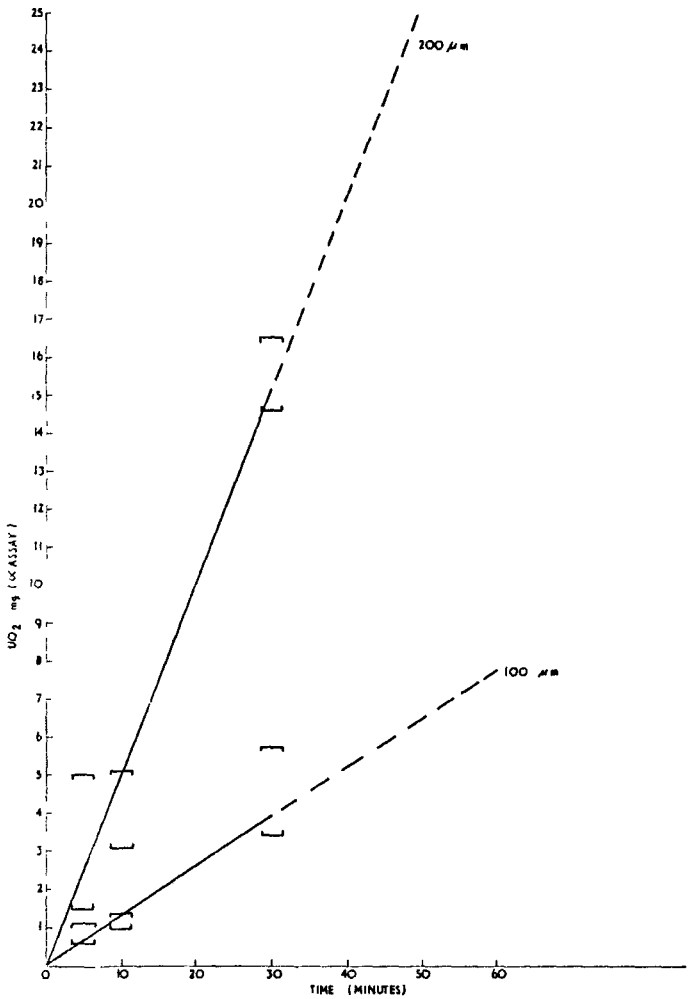

fig. 5

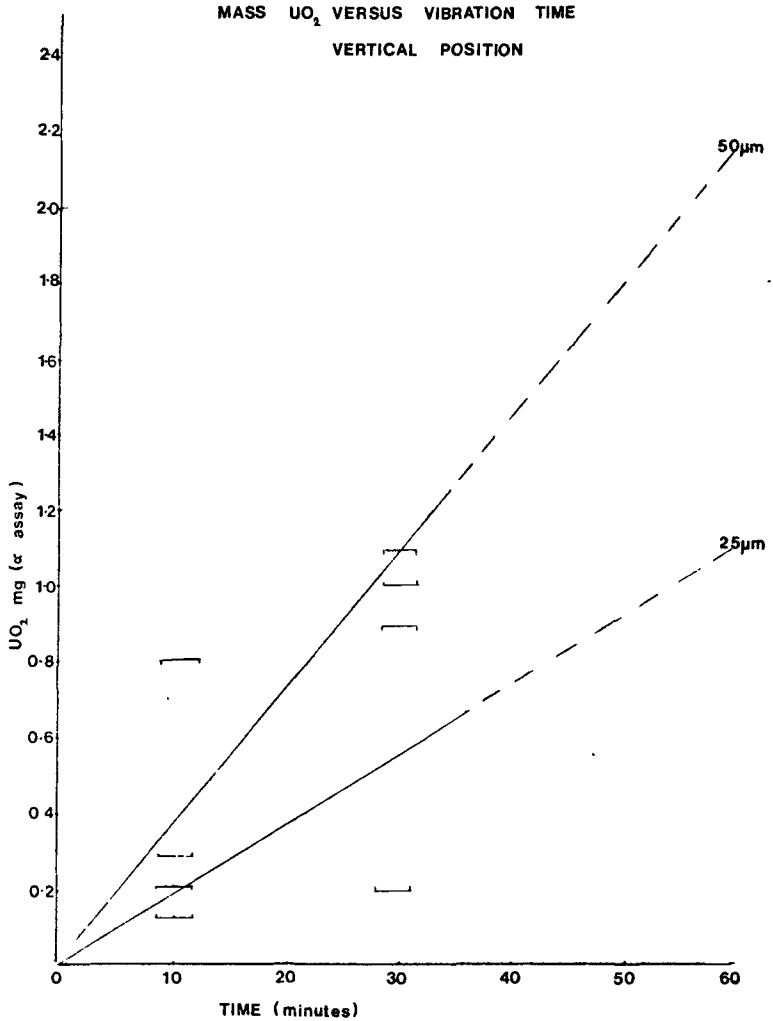
MASS UO2 VERSUS VIBRATION TIME
VERTICAL POSITION
50µm
25µm
UO2 mg (α assay)
TIME (minutes)
2·4
2·2
2·0
1·8
1·6
1·4
1·2
1·0
0·8
0·6
0·4
0·2
0
10
20
30
40
50
60

fig. 6

8. DISCUSSION

A series of experiments has been carried out which has demonstrated that, for differential pressures up to 1 atmosphere, the rate of transmission of UO_2 through an aperture is directly proportional to the rate of air flow.

The density of the UO_2 aerosol transmitted through the orifice was found to be constant, within an order of magnitude, for orifice diameters in the range 10 - 200 um, for differential pressure up to 1 atmosphere, and for experiments lasting up to 1 hour. In order to promote maximum flow of UO_2, the UO_2 container was vibrated far more energetically than would be conceivable in normal transport. The maximum value of aerosol density found, with the orifice 20 mm above the UO_2 surface, was 9×10^{-6} g/ml.

Assuming that similar aerosol densities can be achieved using PuO_2, the IAEA regulations will be satisfied by a leak tightness standard of 10^{-6} ml/sec at 1 atmosphere; a figure which is readily achievable using conventional engineered closures.

Clearly, this calculation contains several factors of pessimism. A leakrate of 10^{-6} ml/sec corresponds to a circular orifice less than 0.1 um in diameter; only the smallest particles could be transmitted, and blockage would occur quickly.

It is planned that experiments with PuO_2 will be carried out and correlated with the results of UO_2 experiments reported here.

DISCUSSION

Question by J. A. Andersen, USA: 1. To what condition do you assign the 10^{-6} ml/s leak rate - the normal conditions of transport, which are required to limit the loss of radioactive material to A2 x 10^{-6}, or the B(U) (A2 x 10^{-3}) or B(M) (A" x 1) post-accident condition? 2. What do you believe to be a practical or engineered leakage rate for an actual cask or containment vessel for plutonium, post-accident? (e.g. what are your findings regarding an allowable post-accident leak rate for a practical plutonium containment vessel?).

Answer: 1. To the normal conditions of transport. 2. The experimental work described in this paper provides results measured under steady state conditions; it is not easy therefore to calculate a target figure for post-accident leakrate. I would expect, however, that the target figure, post-accident, should be more relaxed than 10^{-6} ml/s.

Question by T. Tamberg, FR Germany: Could the intense vibrati[on] you use prevent plugging of the pores to a certain extent?

Answer: I think that this is very likely. In spite of the intense vibration used, blockage often occurs during experiments using 25 μm and 10 μm diameter holes.

Comment by J. Draulans, Belgium: Our experience indicates that there is a quick decomposition of P.V.C. at a temperature even below 120°C. Such a decomposition produces highly corrosive gases, able to attack metal containers (aluminium and stainless steel) and silicon gaskets.

Answer: Under normal, i.e. accident-free conditions of transport, the temperature of the can does not exceed 70°C. At this temperature, P.V.C. decomposition is not a problem. I agree that P.V.C. can produce corrosive breakdown products at 120°C.

Question by C. B. G. Taylor, United Kingdom: You said the aerosol density is higher by an order of magnitude, 20 mm above the powder surface, compared to that at the top of the can. Did you establish the details of this density gradient; and which figures did you use for your final statements of radioactive leak rate?

Answer: The aerosol density increased by a factor of 5 in reducing the distance between orifice and powder bed from 200 mm to 20 mm. In making the final statement of permissible leak rate, I used the highest value of aerosol density found in a single experiment, viz: 9×10^{-6} g/ml.

INVESTIGATION OF GAS LEAKAGE FROM SEALING CONSTRUCTIONS AT CONTAINMENTS FOR RADIOACTIVE MATERIALS

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1. INTRODUCTION

In designing containments for radioactive materials special attention must be paid to the construction of the sealing system and the selection of appropriate gasket materials. The properties of the sealing material determine the release rate of gaseous radioactive substances from the containment under normal storage and transport as well as test conditions. During container prototype test programs the leakage rate of the construction is usually measured with He-leak-detectors. The measured leakage rate is caused by "real leaks" and by the permeation of the test gas through the gasket material. The contribution of "real leaks" can be estimated if the permeation of the test gas as a function of time is known with sufficient accuracy. In order to provide the necessary data for assessing the leak-tightness of containment systems permeation constants for several combinations of gases and elastomer gasket materials were measured within the framework of a research program sponsored by the German Ministry of the Interior.

2. PERMEATION DATA

2.1 Theoretical Description of Gas Permeation

From the first and the second Fick diffusion equations the following expression for the leakage rate can be derived:[1)]

$$Q(t) = \frac{A \cdot P_k \cdot \Delta p}{d} \left(1 + 2 \sum_{n=1}^{\infty} (-1)^n \exp\left[-\left(\frac{n \cdot \pi}{d}\right)^2 D t\right]\right) \qquad (1)$$

$Q(t)$ leakage rate as a function of time
A surface area of the gasket
P_k permeation constant
Δp difference of the partial pressures of the test gas on either sides of the gasket
d thickness of the gasket
D diffusion constant of the gasket material

The coefficient

$$Q_\infty = \frac{A \cdot P_k \cdot \Delta P}{d} \qquad (2)$$

describes the asymptotic leakage rate for $t \to \infty$ (equilibrium value). Using equ. (2) the permeation constant can be calculated from the measured equilibrium leakage

rate. The permeation constant and the diffusion constant D of the sealing material are coupled by equ. (3)

$$P_k = \alpha \cdot D \qquad (3)$$

α solubility of the gas in the sealing material

The diffusion constant can be evaluated from the rising part of the permeation function. Since the series in equ. (1) converges slowly for small times, an equivalent expression must be used for the rising part of the permeation curve [1]

$$Q(t) = \frac{2A \cdot P_k \cdot \Delta p}{\sqrt{\pi D t}} \exp\left(-\frac{d^2}{4Dt}\right); \qquad \frac{Dt}{d^2} < 0{,}4 \qquad (4)$$

From this so-called Holstein approximation a scaled representation of the time dependence of gas permeation at small times can be derived.[2]

2.2 Experimental

Fig. 1 shows a block diagram of the apparatus used for measuring permeation rates of various test gases through commonly used sealing materials. A circular plate of the sealing material mounted in a vessel (EPG3) is pressurized with the test gas. The partial pressure of the permeated gas is measured with a quadrupole mass spectrometer. Fig. 2 shows a cross section of the base plate of the test container with the sample of sealing material. Elastomer plates with an area of 1300 mm^2 and a thickness of about 5 mm were investigated. The test gas pressure was varied between 50 mbar and 2000 mbar. With the aid of a heating system inside the test vessel the temperature dependence of the permeation rate between room temperature and 400 °C can be measured. Calibration of the mass spectrometer was performed using capillary leaks with known He-leakage rates (Fig. 3). The leakage rates of the capillaries for other gases than He were calculated from their geometrical dimensions using the Knudsen's capillary equation[3] From a series of measurements with varying test gases and pressures the relationship between the spectrometer reading and the permeation rate was determined.

2.3 Results and Discussion

Permeation data were measured for He, Ar and Kr in silicone-rubber (S), Viton (V) and ethylene-propylene-diene rubber (EPDM). Some typical permeation functions are shown in Figs. 4, 5, 6 and 7. From the asymptotic values permeation constants were calculated using equ. (2). In table I the results are compiled together with some data from the literature. The differences between the values might be attributed to different experimental conditions, e.g. different chemical composition of sealing material and different temperature. Some general statements can be derived from the measured permeation curves:

1. The permeation constant increases with increasing temperature. Generally permeation constants are exponentially related to temperature and depend on the activation energy.
2. For silicone rubber the permeation constant increases with increasing atomic number of the noble gas.
3. In the case of Viton and EPDM the permeation of He is favoured compared with the heavy noble gases.

3. CONCLUSIONS

The measured permeation data are useful for the selection of a suitable sealing material for a container. In general the material with the lowest permeation constant (e.g. Viton) should be applied. On the other hand further properties may become important for special applications, e.g. the sensitivity against radiation, mechanical stability and the high and low temperature behaviour. The construction of a sealing system always requires a detailed assessment of the relative importance of the various material properties.

The shape of the permeation curve of the gasket material is a limiting factor for the sensitivity of a He-leak test. If the sealing has a high permeation constant, the partial pressure of the test gas will rapidly increase with time. After a short time a high permeation leakage rate is measured which can completely ma[illegible] the effect of a "real leak". Therefore only a short time interval at the very b[illegible] ginning of the permeation process can be used for a high sensitivity measurement. In the case of a silicone rubber gasket only a few minutes after the admission of He are available for effective leak testing whereas with Viton gaskets the useful time interval is very long.

4. EXAMINATION OF COMPLETE SEALING CONSTRUCTIONS

The apparatus shown in Fig. 1 allows leak testing of complete sealing constructions under working conditions. As a typical example Fig. 9 shows a blind plug which was used during a prototype test of a type B package. In this construction the gasket material is not only compressed but also subject to shearing, so it was supposed that the sealing would fail after a few opening and closing cycles. Silicone, Viton and EPDM O-rings were tested. Even after 50 cycles no increase of the leakage rate of the sealing construction was observed.

In order to study the dynamic behaviour of sealing constructions (especially with metal gaskets) simulating transport conditions an apparatus is under construction which is coupled to an electrodynamic shaker. The quadrupole mass spectrometer will be used to measure the leakage rate of the vibrating sealing as a function of acceleration, frequency and compression.

REFERENCES

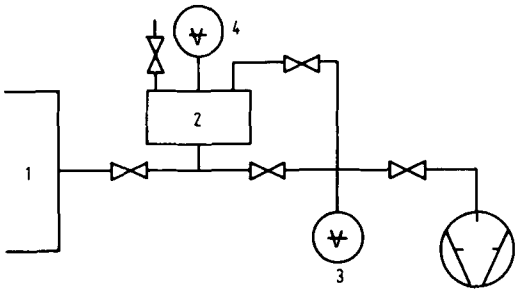
1) Beckmann, W. und M.H. Seider, Kolloid-Zeitschrift und Zeitschrift für Polymere 220, 2 (1967), 97 - 107
2) Kowalewsky, H., Proceedings of PATRAM '80
3) Roth, A., Vacuum Technology, North-Holland Publishing Company, Amsterdam, New York, Oxford (1976)
4) Vacuum Manual, L. Holland, W. Steckelmacher and Y. Yarwood Ed., E. and F.N. Spon, London (1974)
5) Private Communication, C. Freudenberg, Weinheim
6) Barrer, R.M. and H.T. Chio, Journal of Polymer Science, Part C, 10 (1965), 111 - 138

Table 1 Permeation Constants (in $m \cdot s^{-1}$)

Gas	Material	1	2	3	4	5	6
		T = 23 °C	50 °C	100 °C	23 °C	23 °C	0 °C
He	S	$4.1 \cdot 10^{-10}$	$8.7 \cdot 10^{-10}$	$1.7 \cdot 10^{-9}$	$2.9 \cdot 10^{-10}$	$0.4\text{-}2.7 \cdot 10^{-10}$	$1.8 \cdot 10^{-10}$
	E	$2.9 \cdot 10^{-11}$				$1.1 \cdot 10^{-11}$	$4.2 \cdot 10^{-10}$
	V	$1.9 \cdot 10^{-11}$			$7.5 \cdot 10^{-12}$	$0.4\text{-}1.1 \cdot 10^{-11}$	$7.8 \cdot 10^{-10}$
Ar	S	$7.9 \cdot 10^{-10}$	$1.1 \cdot 10^{-9}$	$1.8 \cdot 10^{-9}$	$5.3 \cdot 10^{-9}$		
	E	$2.0 \cdot 10^{-11}$					
	V	$1.8 \cdot 10^{-12}$					
Kr	S	$1.5 \cdot 10^{-9}$	$1.9 \cdot 10^{-9}$	$2.3 \cdot 10^{-9}$			
	E	$2.8 \cdot 10^{-11}$					
	V	$6.3 \cdot 10^{-12}$					

S = Silicone rubber
E = EPDM
V = Viton

1, 2, 3 this work
4 data from "Vacuum Manual" [4]
5 data from firm Freudenberg [5]
6 Barrer, R.M. and H.T. Chio [6]



1 = Q 200
2 = EPG 3
3 = thermovac (range 10^{-3} 1000 mbar)
4 = precision manometer (range 0 - 2000 mbar, 10 mbar per scale)

Fig. 1

Block diagram of the experimental set-up for measuring gas permeation through sealing materials

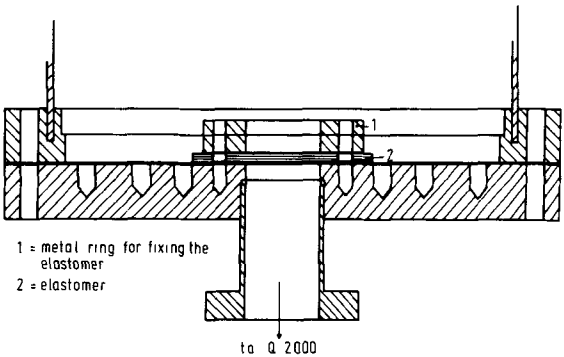
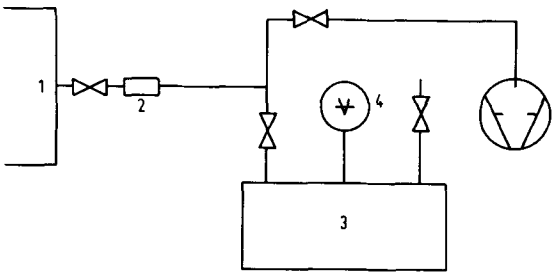


Fig. 2

Bottom plate of the test vessel with the elastomer sample



1 = Q 200
2 = capillary leak
3 = test gas
4 = precision manometer (range 0-2000 mbar, 10 mbar per scale

Fig. 3

Experimental set-up for calibration of the mass spectrometer

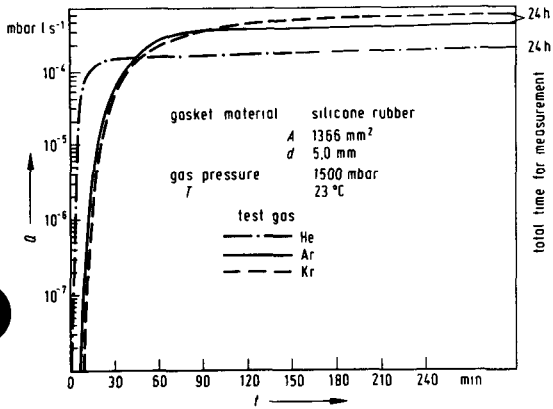


Fig. 4

Permeation of He, Ar and Kr through Silicone rubber

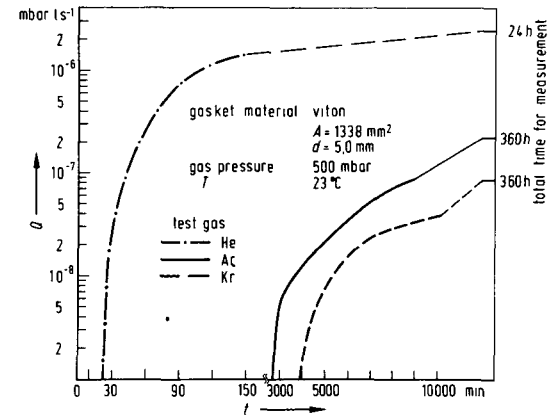


Fig. 5

Permeation of He, Ar and Kr throuth Viton

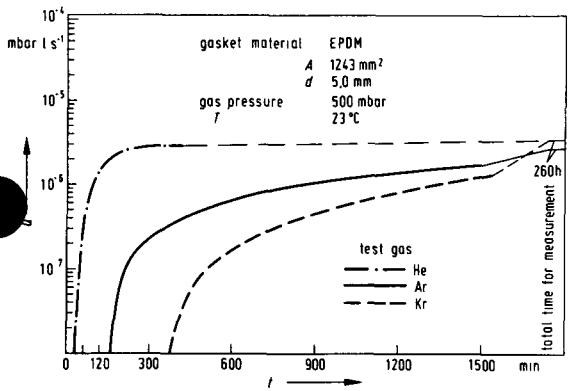


Fig. 6

Permeation of He, Ar and Kr through EPDM

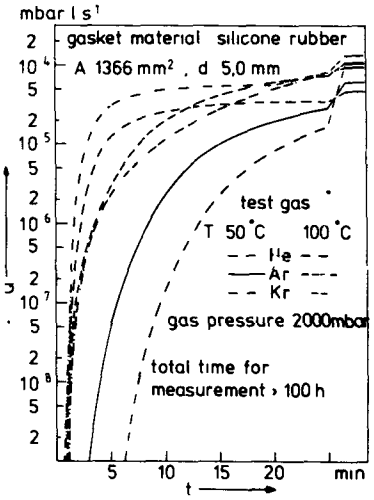


Fig. 7

Permeation of He, Ar and Kr through Silicone rubber at T = 50 °C and T = 100 °C

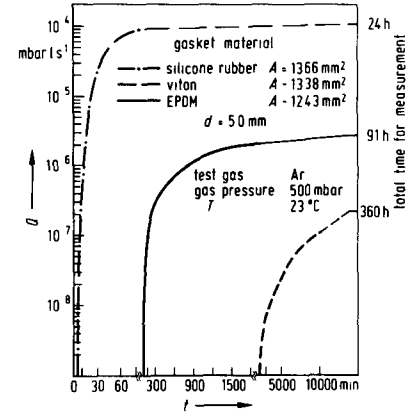


Fig. 8

Comparison between permeation curves for Ar through Silicone rubber, Viton and EPDM

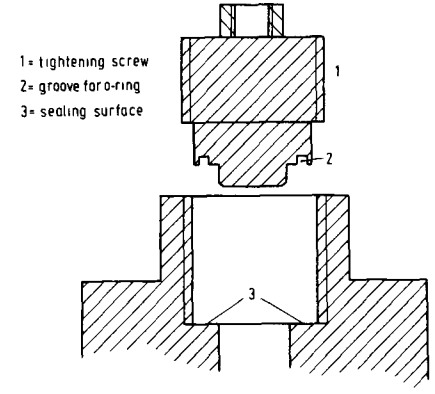


Fig. 9

Cross section of a tested sealing construction

VACUUM DRYING AND LEAKAGE TESTING OF IRRADIATED FUEL PACKAGINGS

Kurt Goldmann
Transnuclear, Inc.

Henri Bernard
Transnucleaire, S.A.

1. INTRODUCTION

Irradiated fuel assemblies are normally loaded into the cavity of a transport packaging in a water filled pool. If the fuel is to be transported dry, the water has to be removed from the cavity after the packaging has been taken out of the pool. While most of the water can be removed by gravity draining, some liquid water will remain in stagnant pockets and on cavity surfaces.

It is desirable to remove this residual liquid to avoid evaporation and buildup of steam pressure during transport. This paper describes a vacuum drying method which may be utilized for the removal of residual liquid. It also describes how this method can be extended to include leakage testing.

The contents of this paper are based on experience gained during the performance of drying and assembly and fabrication verification leakage testing of TN-9 packagings.

2. VACUUM DRYING

2.1 Test System

The packaging with its longitudinal axis in a vertical position, is drained by gravity. A vacuum drying system is then connected, as shown schematically in Fig. 1. The connection is made through a lid opening. The basic components of a vacuum drying system are shown by solid lines. They include a vacuum pump, an isolation valve (V_i), a vacuum break valve (V_c), and a vacuum gage (G_3).

Additional components, which were used for the TN-9 tests to provide further insight into the drying process, are shown by dashed lines. They included a liquid nitrogen cold trap, a vacuum gage at the cold trap (G_2), a cold trap isolation valve (V_a) and another vacuum gage (G_1) which was installed in a second opening of the lid. Sensor outputs from all pressure gages were recorded continuously during these tests. Also, a transparent receptacle was installed at the outlet of the drain line and a thermocouple was located adjacent to the cavity wall near the center of the packaging.

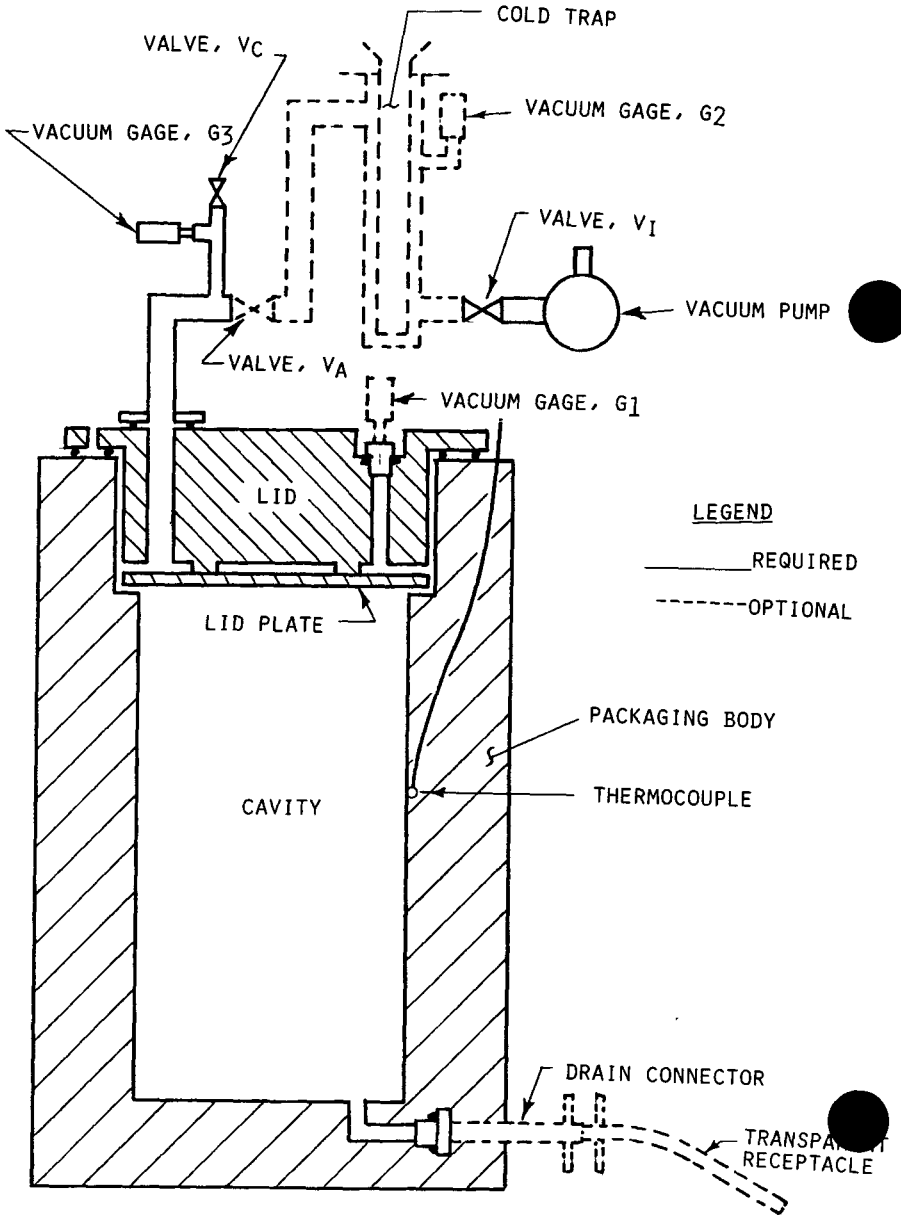


FIG. 1 - VACUUM DRYING AND ASSEMBLY VERIFICATION LEAKAGE TEST SYSTEM

2.2 Operations and Observations

With the vacuum pump operating and valves set in appropriate positions, the cavity is evacuated to a pressure which is somewhat lower than the saturation pressure of water corresponding to the cavity wall temperature. At that time, valve V_i is closed. Upon breaking the vacuum by opening valve V_c, one can observe a flow of liquid water into the drain receptacle. For the TN-9 tests, the quantity of liquid varied from 100-200cm^3. After a second cycle of evacuation and vacuum break, additional liquid flowed into the receptacle, but the quantity was less than 40 cm^3. Further cycling produced no additional liquid. An explanation for this phenomenon is that liquid water which remains on cavity surfaces after gravity draining, is swept down the cavity and into the drain line by the in-rushing air.

This evacuation/vacuum break cycling represents an effective method for removing most of the free liquid which remains in a cavity after gravity draining.

Complete drying can be accomplished by evaporating the remaining liquid under a vacuum and removing the water vapor as it is being formed, as shown by the upper curve in Fig. 2. With a constant speed pump, system pressures will decrease at a decreasing rate as a function of time. When a pressure is reached which corresponds to the saturation pressure of water at the temperature of the cavity surfaces, the pressure will tend to remain constant while the remaining water in the cavity is evaporating. With a relatively slow pumping speed, all of the liquid will evaporate before the pressure is reduced to 4.6 torr, which corresponds to the vapor pressure over ice at a temperature of 0^{o}C. If the cavity is completely dry, the rate of pressure change will be constant at the ice point.

On the other hand, if the pumping speed is too high, and liquid remains when the pressure reaches 4.6 torr, the liquid will freeze and there will be a discontinuity in slope as shown by the lower curve in Fig. 2. With the relatively high capacity pump and the use of a cold trap it was difficult to prevent ice formation during the TN-9 tests. A block of ice formed in the drain line after an initial test in which excess water had not been removed by vacuum break cycling. During another test in which the transparent receptacle was left connected to the drain after vacuum break cycling, a transformation of fog into ice frost on the receptacle surfaces was observed.

Attempts at further vacuum drying after ice formation are nearly self defeating. Sublimation rates are very low and the volumes of vapor are very large because of the low density of the vapor at low pressures. Sonic choking in the discharge connections limits the rate of vapor removal from the cavity. When ice formed, the pressure at the cold trap decreased typically to 10^{-2} torr or less within a few minutes while it took several hours for pressures in the cavity, as measured by the vacuum gage G_1, to decrease to less than 10^{-1} torr. It seemed more practical to stop pumping, let the ice melt and start again under controlled conditions which

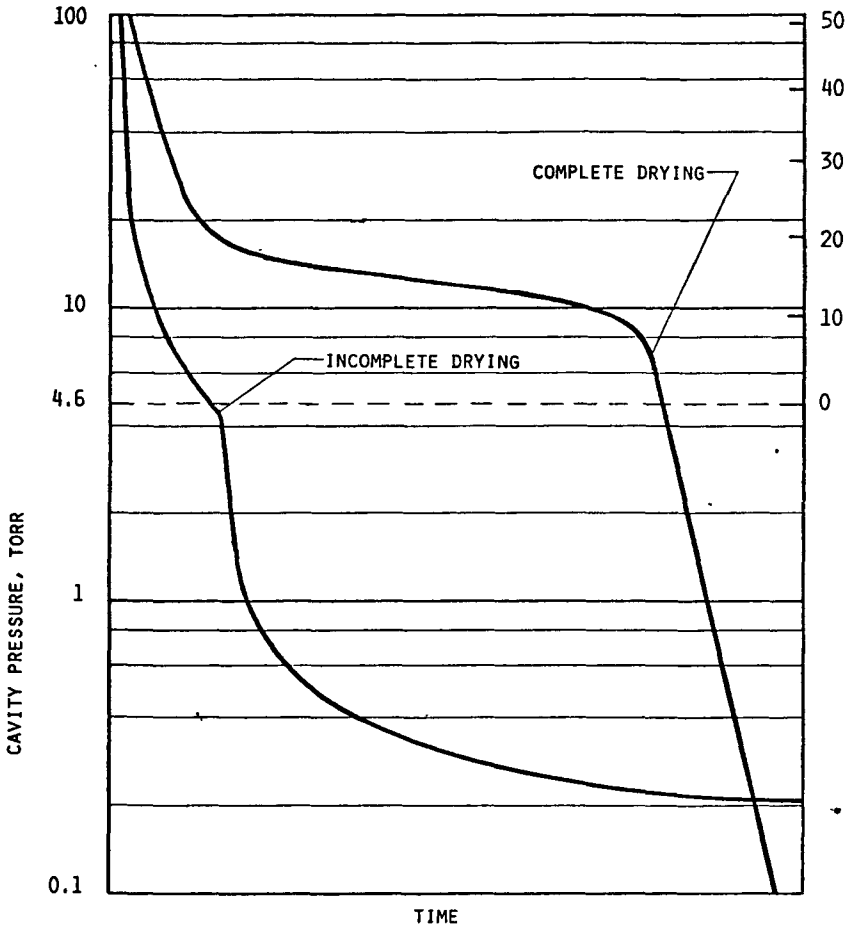


FIG. 2 - CAVITY PRESSURE VS. TIME DURING DRYING

assured that cavity pressures remained above 5 torr until all liquid had evaporated.

It should be noted that the tests described herein were performed without fuel in the cavity. One would expect that the decay heat from irradiated fuel assemblies would greatly enhance evaporation rates and thereby reduce the probability of ice formation.

3. LEAKAGE TESTS

3.1 Leakage Test Method

For the fabrication and assembly verification leakage tests, the cavity is evacuated to an appropriate pressure level and then isolated by closing all valves. The pressure on vacuum gage G_3 (or G_1) is noted at the time at which the cavity is isolated and again some period of time afterwards. An observed pressure rise during this period could be due to surface outgassing, evaporation of liquid, or air in-leakage. Since it is not possible to differentiate between these sources, it is conservatively assumed that all of the observed pressure rise is due to in-leakage of air.

Based on this assumption, the leakage rate can be determined from the following equation:

$$L = \frac{\Delta p \times V}{760 \times t} \qquad (1)$$

where:

L = leakage rate, atm cm^3/sec
Δp = vacuum pressure rise, torr
V = isolated volume, cm^3
t = isolation time, sec.

3.2 Containment System Assembly Verification

Prior to each transport, leakage tests must be performed to verify the integrity of the containment system of a packaging. The same test system, Fig. 1, which is used for vacuum drying may be used for these tests. The maximum allowable leakage rate for containment system assembly verification of the TN-9 has been specified as 9.2×10^{-2} atm cm^3/sec.

The assembly verification test may be performed immediately following the cavity drying by reducing the cavity pressure to the 10^{-2} to 10^{-1} torr range and observing the pressure rise after isolating the cavity for 10 minutes. If the pressure rise during that period is less than 6×10^{-2} torr, the leakage rate is less than the maximum allowable value, as determined from equation 1.

3.3 Containment System Fabrication Verification

Acceptance tests of a packaging at the end of fabrication include fabrication verification leakage tests. For the TN-9

containment system, the acceptance criterion is a maximum allowable leakage rate of 2.8×10^{-4} atm cc/sec which is almost 3 orders of magnitude lower than the maximum allowable leakage rate for assembly verification.

The vacuum pressure rise method and the test equipment used for assembly verification are also suitable for performing the fabrication verification tests, except that a cold trap is required to reduce vapor pressures below 10^{-3} torr. The test must be performed at this low pressure to achieve the required sensitivity of 1.4×10^{-4} atm cm^3/sec within a reasonable test period.

The system was outgassed by operating the vacuum pump and cold trap for several hours prior to performing the TN-9 tests. Because of the absence of outgassing and vapor formation and the apparent tightness of the cavity, no pressure rise was observed during extended periods of isolation. Under such condtions, the criterion of acceptability is the minimum time that the cavity must be isolated. This minimum time can be calculated from the following equation:

$$t_{min} = \frac{\Delta Ps \times V}{45{,}600 \times S} \qquad (2)$$

where:

t_{min} = minimum isolation time of cavity, min
ΔPs = sensitivity of the pressure gage, torr
V = isolated volume, cm^3
S = required test sensitivity, atm cm^3/sec

The vacuum gage which was used for the TN-9 tests has a sensitivity of 0.2×10^{-3} torr at the 10^{-3} torr pressure level. Using this value and equation 2 the cavity must be isolated for a minimum time of 23 minutes to assure that the leakage rate is less than the allowable value, if no pressure rise is observed.

3.4 Leakage Testing of Closures

Blind flanges serve as closures of the lid openings which were used for the cavity drying and leakage testing. The blind flanges are equipped with viton O-ring seals.

A vacuum bell leakage test system was utilized to verify the proper sealing of the closures after the cavity tests. The system which was used for this purpose is shown schematically in Fig. 3. It included a vacuum bell, cold trap, valves V_i and V_a, and vacuum gages G_1 and G_2. The system was small and light enough to be carried and set in place by a technician.

The areas immediately surrounding the TN-9 closures are flat machined surfaces. The vacuum bell is placed over the closures so that its O-ring can seal against these flat surfaces. The sealing force is provided by the pressure differential between atmospheric pressure on the outside and vacuum on the inside of the system.

The leakage test method is identical to that used for the cavity. Because of the much smaller test volume, the test sensitivity

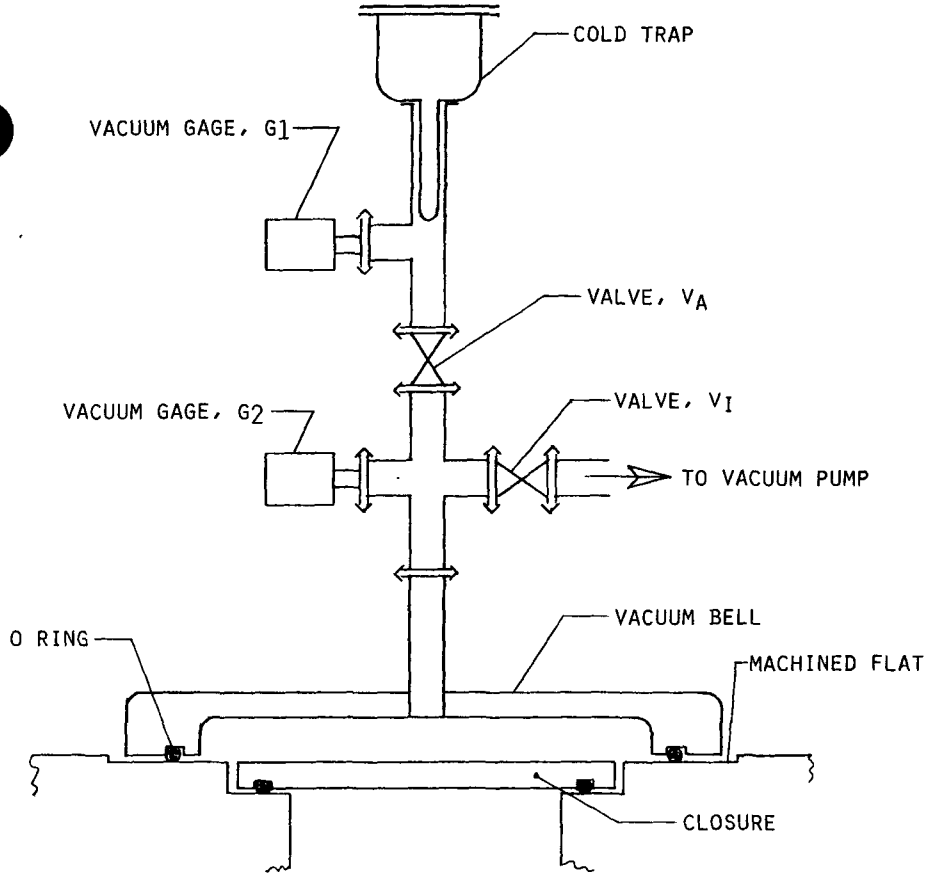


FIG. 3 - VACUUM BELL LEAKAGE TEST SYSTEM

is greatly increased. The vacuum bell system for the TN-9 tests had a volume of approximately 300 cm^3. Pressure rises of approximately 2 x 10^{-2} torr in 3 minutes were typically observed, corresponding to leakage rates of 4 x 10^{-5} atm cm^3/sec from equation 1. It should be noted that this value included not only the leakage across the cavity closure seal but also air in-leakage from the surrounding atmosphere past the bell O-ring seal.

4. CONCLUSIONS

Experience with vacuum drying and leakage testing of TN-9 packagings has lead to the following conclusions:

1. Evacuation/vacuum break cycling represents an effective means for removing most of the free liquid which remains in the cavity after gravity draining.

2. Residual moisture can be removed efficiently from the cavity by evaporation under a vacuum.

3. To prevent freezing of remaining liquid, vacuum pumping rates must be controlled to assure that the cavity is completely dry before pressures reach 5 torr or less.

4. If ice forms inadvertently, it is recommended that the vacuum be broken and evacuation be resumed only after waiting for a sufficient length of time to allow the ice to melt.

5. The vacuum pressure rise method is a convenient and effective technique for determining leakage rates after vacuum drying.

6. A vacuum bell system can be utilized for leakage verification testing of closures.

7. With the vacuum pressure rise method, the rate of air leakage into an isolated cavity of known volume can be readily determined from an observed pressure rise within a selected time interval.

8. If no pressure changes are observed, one can establish the minimum period of time during which a cavity of known volume must be isolated to assure that air leakage into the cavity is less than the maximum allowable value.

Question by C. B. G. Taylor, United Kingdom: In your paper you state 9.2×10^{-2} atmospheric cc/s as the maximum allowable leakage rate for verification tests on a TN-9 cask. What was the basis for that criterion?

Answer: The acceptance criterion of 9.2×10^{-2} atm cm^3/s as the maximum allowable leakage rate for assembly verification was based on providing assurance that the cavity remains below atmospheric pressure during transport. After fuel loading, the cavity is sealed at a pressure of approximately 0.5 atm absolute to assure that the pressure at the end of transport is still below atmospheric pressure to prevent outleakage of gases.

The maximum leakage rate of 2.8×10^{-4} atm cm^3/s for fabrication verification was based on the regulatory requirement which restricts the accumulated loss of radioactive contents to not more than $A_2 \times 10^{-3}$ in a period of one week following a regulatory fire accident. 100 % fuel clad failure was assumed leading to a higher than ambient pressure in the cavity.

The methods described in ANSI Standard N 14.5 were used to relate leakage rates to pressure increases for assembly verification and radioactivity releases for fabrication verification tests. Further details are given in the Safety Analysis Report for the TN-8 and TN-9 packagings.

Question by K. Tagder, FR Germany: Vapour is also found in defect fuel elements. I presume that you have not carried out any investigations to find out how far the vapour from defect fuel elements can be eliminated by means of the vacuum drying system presented. May I ask your opinion, whether this vapour is also eliminated?

Answer: We have not performed any tests to determine the extent to which steam can be eliminated from defective fuel elements, using the vacuum drying technique. I expect that the heat generation in the fuel and attendant higher pin temperatures will greatly enhance the evaporation rate of moisture in defective fuel over that which remains on the relatively colder cavity walls after drainage.

PRESSURIZED POWDER RELEASE THROUGH MICRO-OPENINGS IN FAULTED CONTAINERS

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Estimates of the quantity of a powder potentially released from a faulted shipping container can be highly conservative under the practices currently in use. The integrity of accident-resistant shipping containers is often determined by measuring the leak rate of a gas that can be measured with great sensitivity (such as helium) after a series of rigorous tests. The gas is assumed to contain some mass concentration of powder and the quantity released to the air is the volume released times the mass per volume. A method of more accurately estimating releases of powder through micro-openings has been developed.

Under the sponsorship of the Systems Performance Branch of the U.S. Nuclear Regulatory Commission, the Pacific Northwest Laboratory conducted a series of experiments to determine the transmission of a stable powder, such as plutonium dioxide, through micropathways. Dimensions of these pathways were consistent with theoretical openings and measured leak rates determined while testing the inner container of the Plutonium Air Transportable (PAT) package (NRC 1978, Table 4.2). The driving force required to obtain a positive leak outward is supplied by heat (radioactive decay and potential fires). In a real situation, the number, size, and shape of the pathways and their orientation with respect to the powder contained (above or below) can vary greatly. To make the problem more tractable, the study was limited to a single, round opening (orifice or capillary). The assumption is still conservative since it results in the largest cross-sectional area for any given gas flow.

The study was divided into three phases: the prediction of gas flows through micro-orifices and -capillaries; the transmissions of powder through these pathways; and a methodology to apply the finding to assess the potential airborne release of contained powders from faulted containers.

GAS FLOWS

A means of relating gas leak rates to orifice and capillary diameters was required. A review of the literature indicated that the standard calculational formulae would be applicable (Bombelberg 1977). The validity of the formulae was untested in the region of interest (20 to 200 μm diameter) and a series of tests was performed to corroborate their applicability.

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

Orifices (Sutter et al. 1978)

Seventeen thin-edge orifices for the gas and subsequent powder flow experiments were fabricated by drilling 20 to 200 μm holes in the centers of stainless steel disks, 1.27 cm in diameter by 0.16 cm thick. The holes were mechanically or laser drilled and subsequently counter sunk on the rear face to produce the thin edges. The mechanically drilled orifices had V-shaped exits while the laser-drilled exits were concave. Bore diameters were measured to an accuracy of ±1 μm using a scanning electron microscope. Bore length:diameter ratios were found to range to 10.

Volumetric air flow (cc/min at 22°C, 101.3 kPa) through the orifices was measured using calibrated variable-area flowmeters as a function of varying temperature (ambient to 650°C) and upstream pressure (up to 10^4 kPa).

The relationship is:

$$Q = \alpha A \psi \sqrt{2g\, P_o\, \rho_o} \qquad (1)$$

where

Q = mass flow rate, g/s
α = orifice coefficient
A = geometric cross section of orifice, m^2
ψ = dependence of flow on ratio of specific heats and pressure ratio
g = gravitational acceleration, 9.8 m/s^2
P_o = upstream pressure, g/m^2
ρ_o = air density at upstream pressure, g/m^3

and ψ, an exit function, can be defined as:

$$\psi = \left(\frac{P}{P_o}\right)^{\frac{1}{\gamma}} \sqrt{\frac{\gamma}{\gamma - 1}\left[1 - \left(\frac{P}{P_o}\right)^{\frac{\gamma - 1}{\gamma}}\right]} \qquad (2)$$

where P = ambient pressure
P_o = pressure in vessel
γ = isentropic ratio (ratio of specific heats C_p/C_v), which depends on gas.

At ambient temperature, air flow rates above "choked flow" (ratio of downstream to upstream pressure less than 0.53) were found to be a linear function of the upstream pressure. At less than "choked flow", the increases in air flow rates are rapid and nonlinear. Air flow rates were found to decrease with increasing temperature (to 650°C) with a ΔP of 103 kPa. Based upon these experimental data, orifice coefficients of 0.88 for P_d/P_u less than 0.025 (an upstream pressure of 4140 kPa exhausting into the ambient atmosphere), 0.80 + 2 (0.065-P_d/P_u) in the P_d/P_u range of 0.065 to 0.025, and 0.80 for a P_d/P_u range of 0.53 to 0.07 were determined. Existing predictive formulae appear to be applicable to air flow rates in the orifice size range of interest for this study.

Capillaries (Owzarski et al. 1979)

Early in the study of capillary flow, it was found that the accepted gas leak correlations held only for isothermal, subsonic, laminar flow with no entrance or exit losses in hydraulically smooth tubes. Since the anticipated flow in this study would likely be turbulent and sonic or choked sonic at the exit, be through

rough channels, be nonisothermal, and be along short capillaries where entrance and exit losses are at least 50% of the total friction losses, a rigorous mathematical model (CAPIL) was developed as a predictive tool.

Experimental, standard condition flows were measured in circular cross-section metal capillaries in six diameters ranging from 48 μm and 0.087 relative roughness to 275 μm and 0.0044 relative roughness. Lengths were either or both 0.762 cm (0.3 in.) to 2.54 cm (1 in.) at each diameter. Input conditions included a series of runs at constant upstream temperature (22°C) and downstream pressure (103 kPa) with the upstream pressure being varied from 6.9 to 10^4 kPa (gauge). Also included was a series of runs at constant upstream and downstream pressures (103 and 0 kPa (gauge) respectively) and varying upstream temperatures from 22 to 650°C.

The following conclusions were made from the comparison of CAPIL-predicted flows and experimental flows: 1) experimental flows were in the continuum fluid flow regime, 2) laminar flow could be stable to Reynolds numbers up to 5000, 3) turbulent friction factors were lower than expected as a result of longitudinal groove roughness of the capillaries or thicker-than-expected laminar boundary layers, 4) CAPIL flows were usually within 85 to 105% of the measured flows, 5) attenuation of standard flow as the temperature was raised was caused by higher viscosity and lower fluid density at elevated temperatures, and 6) entrance and exit losses were a significant part of the total flow resistance.

POWDER FLOW (Sutter et al. 1980)

The transmission of depleted uranium dioxide powder (mass median diameter 3.5 μm Aerodynamic Equivalent Diameter) was measured through the micro-orifices and -capillaries used in the airflow study as a function of upstream pressure (103 to 10^4 kPa), pathway diameter, and leak orientation (i.e., whether the leak was above or below the powder level).

Under the range of conditions used in the study, the quantity of powder transmitted ranged from below the detection level to 8×10^4 μg.

A statistical correlation suggested a two-stage decision rule based upon $\ln(A\sqrt{P})$, where A = area in μm^2 and P = pressure in psig, leak location and aperture type. When information on leak type and location are missing, statistics on all data can be used.

The statistical correlation for data in two regions is shown below:

Statistical Data for $\ln(A\sqrt{P})<10.5$ (Random Region) of DUO Test Results

	Under Powder Leaks (UPL), μg	Agitated Powder Leak Apparatus (APLA), μg)
Expected Average	33	5
90% Confidence Intervals, on the avg.		
• Lower Limit	24	4
• Upper Limit	46	6

DUO Correlation Summary for $\ell n(A\sqrt{P})>10.5$

$$\ell n\ DUO\ (est) = a + b_1(\ell n\ A) + b_2(\sqrt{P})$$

	UPL Orifices	APLA Orifices	Capillaries	Total
Coefficient a	-10.2848	-14.1959	-17.9875	-14.2790
Coefficient b_1	1.6080	1.7906	2.1658	1.8280
Coefficient b_2	0.0449	0.1095	0.1170	0.1052

Other pertinent observations from the study were: 1) diameter appears to be the single most important parameter in powder leakage; 2) the duration of the run had no statistically discernible effect on powder transmission for times up to 24 hours; 3) agitation did not influence the flow from an opening below the bulk powder level; 4) plugging was a frequent occurrence for openings above the bulk powder level (6% of the orifices and 17% of the capillaries plugged immediately upon pressurization); and 5) efforts to increase powder transmission by other techniques were unsuccessful.

POWDER TRANSMISSION CALCULATIONAL METHODOLOGY (Owzarski et al. 1980)

The powder transmission data from pressurized microleaks using depleted uranium dioxide (Sutter et al. 1980) and plutonium dioxide (Yesso et al. 1980) powder were combined and incorporated in a computer code, PAGAN (Particulate and Gas Accident Numerics). PAGAN calculates transient gas leak rates from the inner container of the PAT, and instantaneous conditions in the container. Input conditions of temperature versus time (e.g., in the event of a fire) can cause internal pressurization and leakage through defined pathways. The plutonium and uranium correlations are programmed to indicate the potential powder loss at any given instant. The correlation used is shown in Figure 1.

The potential loss of powder from the inner container if it fails during heating is shown in Figure 2. The potential loss of powder that is anticipated if the container is subjected to a maximum fire (a container temperature of 582°C for 4 days) is shown in Figure 3.

Calculations of the plutonium oxide potential losses following an airplane crash use probable leak-path diameters predicted from impact-test data for a plutonium oxide transport package, and reveal the following: 1) the mean leak path can be represented by a capillary 0.716 cm long and 5 μm in diameter, 2) the predicted mean particle loss from a leak path 5 μm in diameter is 0.09 μg if a fire occurs, 3) the probability of a leak path larger than 10 μm in diameter is 10%, 4) the probability of a particle loss greater than 0.3 μg from an opening 10 μm in diameter is 10%, 5) the maximum particle loss supported by the range of experimental data is 10 μg, and 6) the loss of an amount of plutonium oxide exceeding the allowable standards (interpreted as 2550 μg) is extremely unlikely.

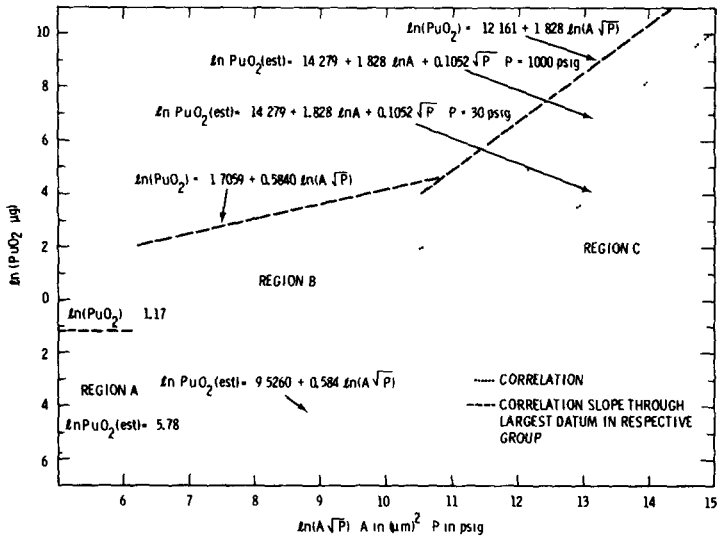

FIGURE 1. Data Correlations for Estimating PuO_2 Loss

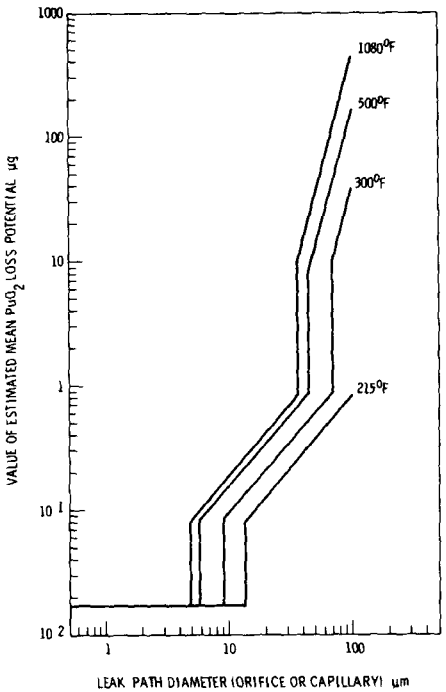

FIGURE 2. Potential Loss of PuO_2 if Container Fails at Given Temperature

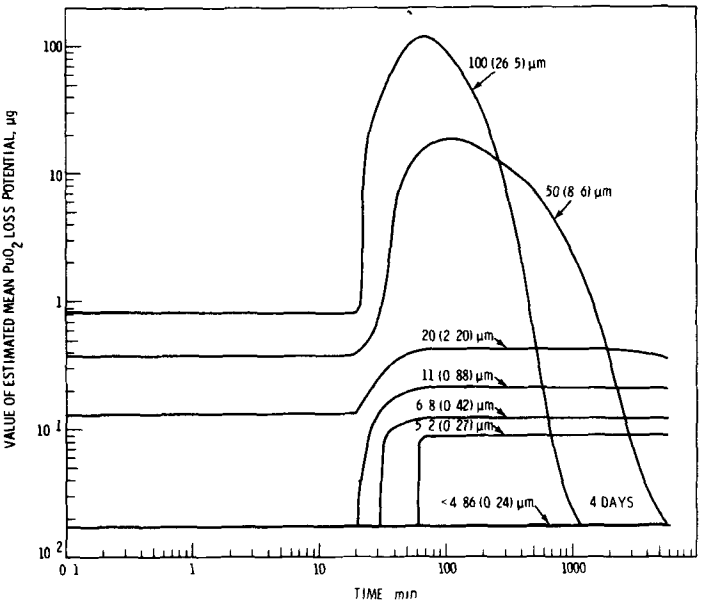

FIGURE 3 Potential Loss of PuO_2 During a Maximum Fire[a]

(a) Particle path is a 0.716-cm-long capillary. Equivalent orifice diameters are given in parentheses, next to the capillary diameters.

REFERENCES

Bombelburg, H. J. 1977. "Estimation of Gas Leak Rates Through Very Small Orifices and Channels." BNWL-SA-2223. Pacific Northwest Laboratory, Richland, Washington.

Nuclear Regulatory Commission. 1978. Plutonium Air Transport Package Model PAT-1, Safety Analysis Report. NUREG-0361, U.S. Nuclear Regulatory Commission, Washington, DC

Owzarski, P. C., S. L. Sutter, J. Mishima, L. C. Schwendiman and T. J Bander. 1979. Measured and Predicted Gas Flow Rates Through Rough Capillaries. NUREG/CR-0745, U.S. Nuclear Regulatory Commission, Washington, DC.

Owzarski, P. C., J. Mishima, L. C Schwendiman and C M. Devary (Ed). 1980 Analysis of Particulate Transmission Through Small Openings Resulting From Container Stresses. NUREG/CR-0958, U.S. Nuclear Regulatory Commission, Washington, DC

[illegible]r, S. L., T. J. Bander, J. Mishima and L. C. Schwendiman. 1978. Measured [illegible]Flow Rates Through Micro-Orifices and Flow Predication Capability NUREG/CR-0066, U.S. Nuclear Regulatory Commission, Washington, DC.

Sutter, S. L., J W. Johnston, J. Mishima, and P. C. Owzarski and L. C. Schwendiman. 1980. Depleted Uranium Dioxide Powder Flow Through Very Small Openings NUREG/CR-1099, U.S. Nuclear Regulatory Commission, Washington, DC.

Yesso, J. D., W. J. Madia, G. H. Beatty, E. W. Schmidt and L. C. Schwendiman/ J. Mishima, Project Managers. 1980. Study of Plutonium Oxide Powder Emissions From Simulated Shipping Containers. NUREG/CR-1302, U.S. Nuclear Regulatory Commission, Washington, DC.

DISCUSSION

Question by J. A. Andersen, USA: Please discuss the similitude or dissimilarity between a capillary, as used in the Battelle experiments, and the crack-like escape path or pores in an actual transport vessel. What was the nature of, or consistency of, the data on which the various scaling zones of ln A $\sqrt{P}$ are based? Please discuss the Battelle Columbus results of plutonium particulate transport through cracks, rather than capillaries. What was the smallest leak rate of helium at which any plutonium escaped?

Answer: There is probably very little similarity between the capillaries used in these experiments and the actual opening formed by accident stresses. Capillaries and orifices were used experimentally since the size could be defined by measured leak rates and their use would result in higher powder losses than from the "cracks" formed in accidents.

There are up to 3 orders of magnitude variation for some of the data points for mass vs. ln A $\sqrt{P}$.

BCL (Battelle Columbus Laboratory) used some machined pathways to represent cracks in some experiments. They found no mass releases for flows less than 2 cc/s but we must keep in mind the variability of these results mentioned above.

POSTER SESSION A

CONCEPTUAL DESIGN OF A ^{85}KR TRANSPORTATION SYSTEM*

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INTRODUCTION

In addition to tritium, carbon-14, and iodine-129, the radioactive noble gas krypton-85 (^{85}Kr) is one of the important volatile radionuclides arising in light-water reactors (LWR). Formed exclusively by fission, ^{85}Kr is produced at an annual rate of 16 PBq/GW of electrical power in a typical LWR. Only a very small fraction of the gas is released during the actual operation of the power plants themselves, the bulk of the ^{85}Kr being transferred to the fuel reprocessing plants. Based upon the handling of 5 tonnes of irradiated fuel per day, which corresponds to 40 GW of electrical power installed, about 630 PBq (17 MCi) ^{85}Kr per year may be expected.

^{85}Kr is a beta-emitting fission product with a half-life of 10.7 years; the stable decay product is rubidium, which is a very reactive metal. Dependent upon the projected growth of the nuclear power industry, the release of such large amounts of ^{85}Kr to the atmosphere - as it is common practice now - may pose global as well as local problems in the future. Therefore, separation and storage/disposal of most of the ^{85}Kr produced by nuclear fuel reprocessing will be required.

As an example, the Environmental Protection Agency regulations will limit the amount of ^{85}Kr released to the atmosphere in the U.S. for commercial nuclear fuel irradiated after January 1, 1983. These regulations [1] state that for the entire fuel cycle operations must be conducted to assure that less than 1.85 PBq of ^{85}Kr are released to the general environment per gigawatt-year of electrical energy produced.

After trapping, there are several options for the management of ^{85}Kr, which include long-term storage in engineered structures, disposal in sub-surface vaults, and disposal in the deep ocean, with the Kr being in gaseous or in solidified form. Transportation of the krypton (on-site or off-site) is an important feature of all concepts.

TRANSPORTATION SYSTEM

Composition of the Off-gas to be Handled

During reprocessing, the noble gases xenon and krypton are liberated from the fuel in the dissolution step and appear in the dissolver off-gas, from which they are separated by cryogenic distillation. The resulting gas mixture to be transported is given in table I.

*Work performed under the European Community's R and D programme on Management and Storage of Radioactive Waste

Table I: Gas components to be transported

krypton (total)	:	$\geq$	90	vol-%
xenon (stable)	:	$\leq$	10	vol-%
nitrogen	:	$\approx$	0.41	vol-%
argon	:	$\approx$	0.08	vol-%
O_2, H_2O, CO_2	:	$<$	1	ppm

Due to the relatively short half-life of the noble gas nuclides, the radioactivity of the gas mixture is due to the radioactive isotope ^{85}Kr, which represents about 7 vol-% of the total amount of krypton, with its specific activity A_{sp} (^{85}Kr) = 389 Ci/g.

Design of the Transport Cask

For off-site transportation of the gas mixture a transport cask has been designed, the conceptual design of which is shown in figure 1. The cask has been designed similar to a cask being licensed already in the Federal Republic of Germany for the transport of irradiated fuel elements from high temperature gas-cooled reactors to the Asse salt mine [2].

The outer shell of the cask is a lead-in-steel composite structure with an asbestos intermediate layer which provides gamma-shielding, a cushion against impact, and thermal insulation. With a diameter of 1.3 m and a height of 1.6 m the cask weighs approximately 9 t.

Inside the cask there are 4 gas cylinders with a nominal volume of 28 ltr each, which contain the compressed gas mixture. 85 cylinders per year may be expected which contain 88 mol of gas mixture or 7.4 PBq (0.2 MCi) ^{85}Kr each at a normal operating pressure of about 10 MPa. This corresponds to a heat-generation of 314 W for each cylinder or 1.3 kW for the cask.

The cylinders are fixed in tubes which provide lateral support in the event of the cask being dropped, and act as a guide for loading. On the basis of a detailed study of literature on the subject, a special stainless steel 1.4439 is proposed as the cylinder material; it is characterized by a high chromium and molybdenum content and possesses outstanding resistance to corrosion by brine.

The compressed gas cylinders are fitted with a special valve, the principle design of which is shown in figure 2. It is characterized by 2 inverted all-metal valves mounted in series and acts as a doublebarrier against the release of ^{85}Kr.

The cask is partially filled with water. The water surrounding the guide tubes acts as a heat-transfer medium and possesses sufficient heat capacity to absorb energy in the event of an accidental fire. The resultant steam escapes through a tube leading to the top of the cask, on the end of which is fitted a rupture disc. The top of the cask interior is designed to form a buffer volume to contain the gas content of one cylinder in the remote event of a leakage. The buffer volume and water bath have gastight separation from each another.

A cost estimate was conducted at the exchange rate for US dollars in early 1980. The resulting costs amount to about $ 128,000 including the estimated costs for four cylinders/valves of $ 16,000.

IAEA Test Specifications

The designed transport cask is a type B(M)-package in accordance with IAEA Transport Regulations [3], which among other things state that it must withstand certain test conditions without the shielding and the integrity of the cask being seriously impaired. Mainly the fire test, where the cask has to withstand a fire of 800°C during 30 minutes and natural cooling thereafter for 3 hours, is a severe test for a cask containing a self-heating gas.

By computation of the steady and non-steady state temperatures it has been demonstrated that the maximum permissible temperature in the cask structure is not exceeded, and that the temperatures inside do not reach a level which will produce an unacceptably high pressure build-up in the cylinders. Figure 3 shows the resulting temperatures during and after the fire test.

Preliminary Safety Analysis

An analysis of postulated abnormal occurrences has been performed. Frequencies have been estimated to be between 10^{-3} and 10^{-5} per year for severe transportation accidents. Taking into account the characteristic properties of a type B-Package, one may conclude that, even under severe accident conditions, there would be no release of ^{85}Kr at all.

Still assuming an hypothetical release of the contents of one cylinder as a result of an accident during transportation or dumping, and using the Gaussian model for atmospheric dispersion [4], this would result in a maximum total-body dose of 1.7 Sv (170 rem) to a person at a distance of 11 m. This is of course an unrealistically high dose, as none person is likely to remain in the immediate vicinity of the accident for longer than neccessary, as it has been assumed in the dispersion model, because of the extreme severity of such an hypothetical accident.

Hence, it is concluded, that the cask meets the IAEA requirements and that ^{85}Kr shipments can be done in a safe and efficient manner.

REFERENCES

[1] Environmental radiation protection standards for nuclear power operation, Federal Register, 42, No. 9, Part VII, 2858 (1977).

[2] WOLF, J., Endlagerung verbrauchter Brennelemente aus dem AVR-Versuchskernkraftwerk im Salzbergwerk Asse, Kernforschungsanlage Jülich Rep. JÜL-1163 (1975).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, IAEA, Vienna (1973).

[4] VOGT, K.J., GEISS, H., Kurzzeit- und Langzeitausbreitungsfaktoren zur Berechnung der Umweltbelastung durch Abluftfahnen, Kernforschungsanlage Jülich Rep. KFA-ZST-IB-198 (1974).

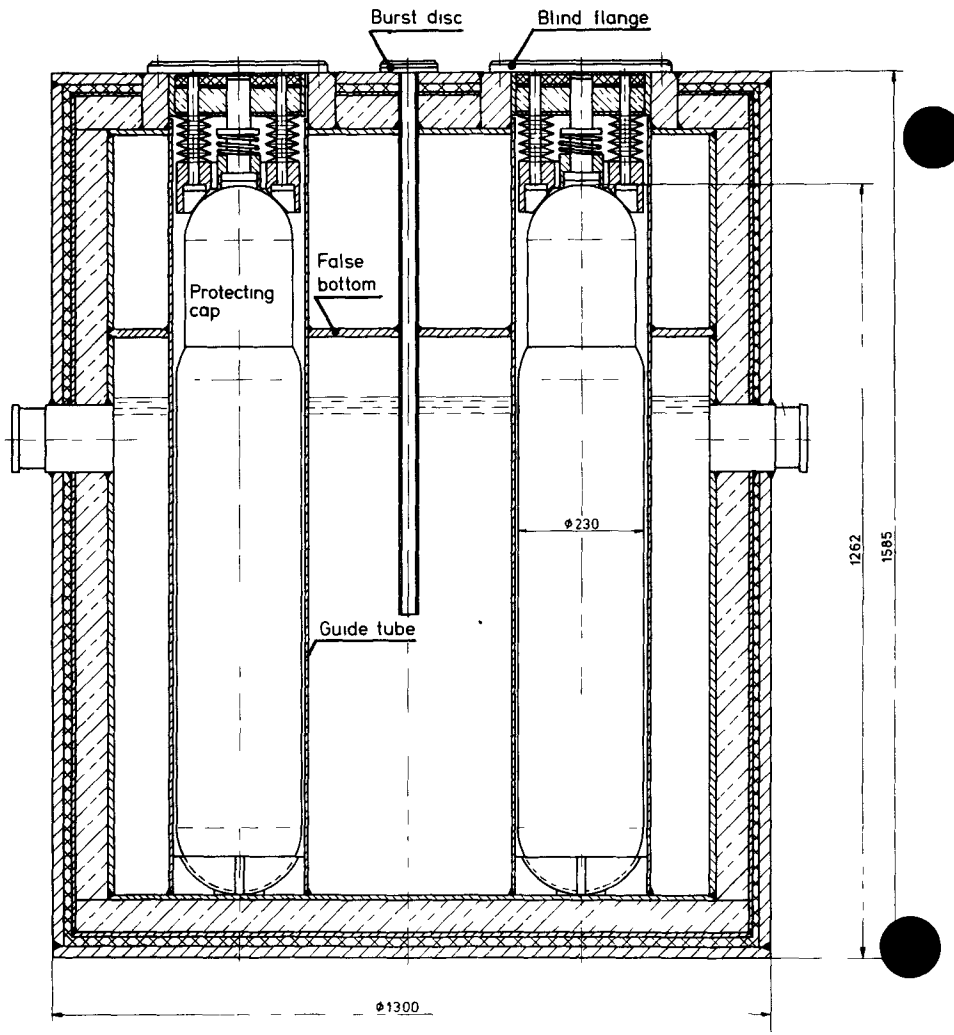

Fig. 1: Schematic view of the ^{85}Kr transport cask

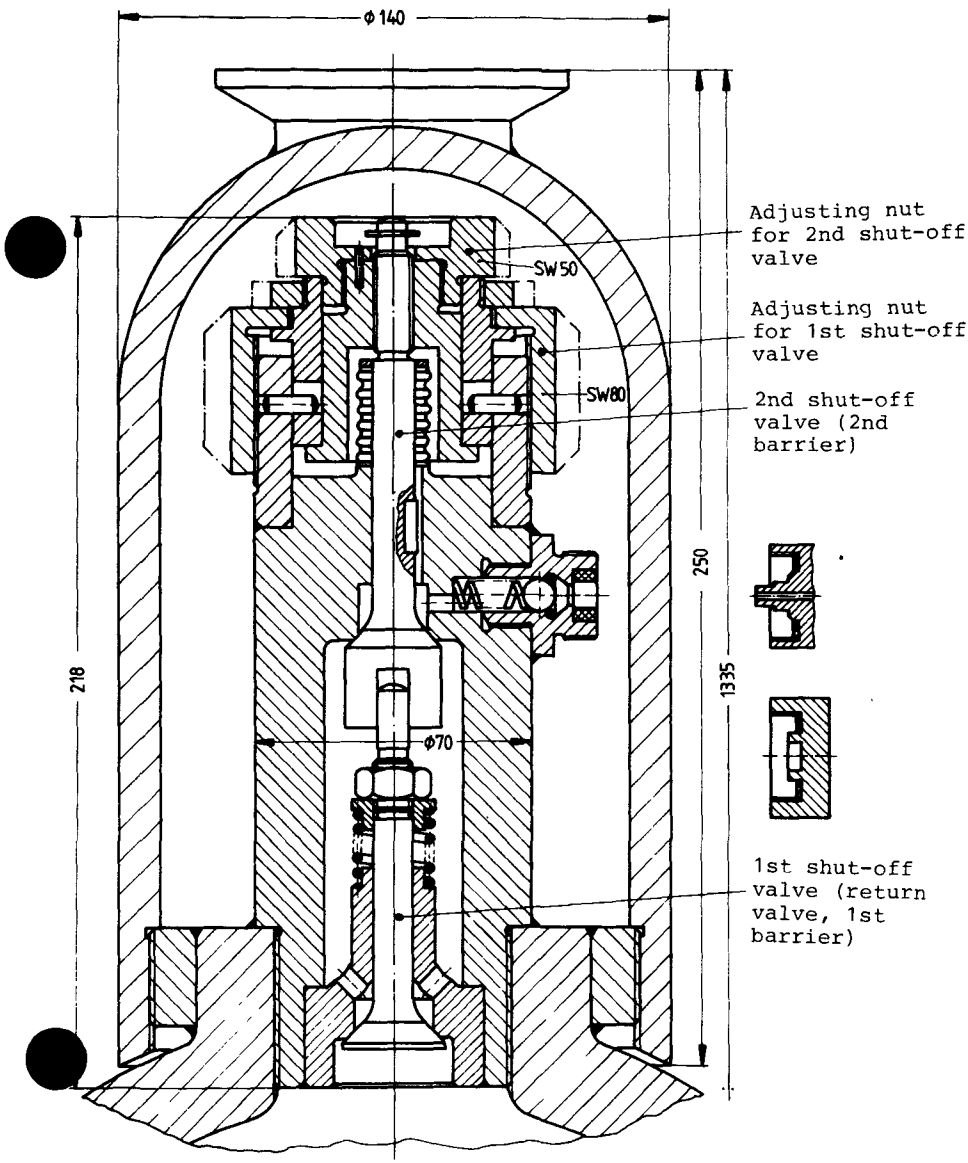

Fig. 2: Schematic view of the valve

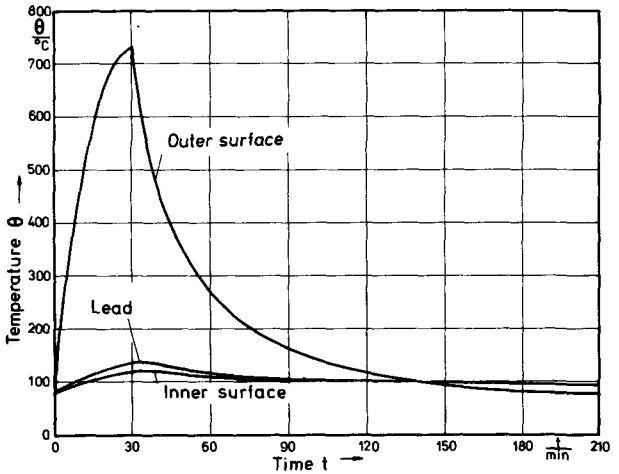

Fig. 3a: Non steady temperature rise in the cask structure during fire test

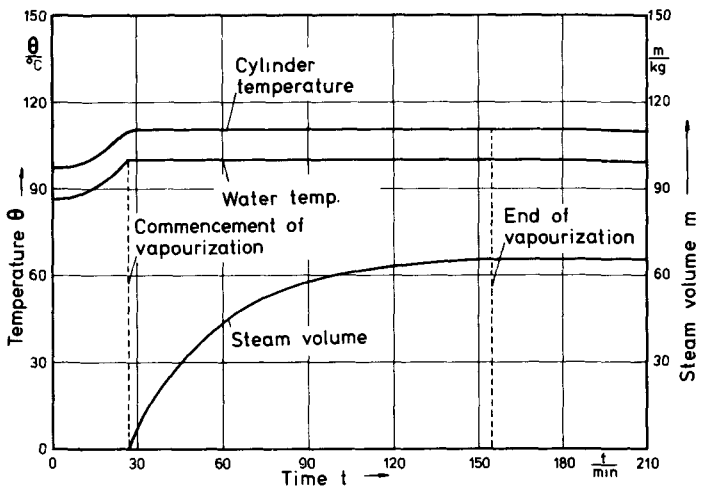

Fig. 3b: Non steady temperature rise inside the cask and volume of steam during fire test

A HANDY CALCULATION METHOD OF TRANSPORT INDEX AND RADIATION SPECTRUM OUTSIDE OF CASK

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I Introduction

Estimation of transport index is required on planning a transportation of spent fuels by shipping casks. Here, the transport index is the value of radiation dose rate in unit of mrem/hr at 1m distance from the middle of the cask surface. Therefore, the estimation of transport index sometimes means a full repetition of the radiation shielding calculation already made for the cask, especially when energy spectrum of source neutrons from spent fuels is different from ones ever experienced. Incidentally, the shape of neutron energy spectrum from spent fuels is dependent on the relative strength of neutron source due to the (alpha,n) reactions of oxigen and other light nuclei in comparison with the source strength of fission neutrons, i.e. the shape of the source neutron spectrum is dependent on the burnup of the given spent fuels.

In order to avoid such repetitions, a simple and effective method is proposed in this report for estimation of transport index as well as estimation of radiation shielding effect.

II Outline of the Method

PRINCIPLE

The point of this method is to introduce a matrix for the cask named " Radiation Response Matrix of Cask ", (RRMC). The matrix describes contributions of neutrons and gamma rays from an energy group in the radiation energy spectrum at a point " IN " just inside of the cask cavity to the otner energy groups in the radiation energy spectrum at the point " P ", where the transport index is defined. In Fig. 1, a schematic relation is illustrated for the point " P " and the point " IN ".

Following relation holds between radiation energy spectrum at the point " P " and the one at the point " IN ".

$$\psi^{P} = \tilde{M} \cdot \bar{J}_{+}^{IN} \qquad (1)$$

where ψ^{P} and $\bar{J}_{+}^{IN}$ are column vectors. The former stands for radiation flux, the latter stands for current of radiations directing towards outside of the cask. M is the RRMC transposed for matrix operation. They consist of neutron and gamma-ray components.

The Eq. (1) can be rewritten as follows in terms of neutron and gamma-ray components in the expressions for ψ^{P} and $\bar{J}_{+}^{IN}$.

$$\begin{pmatrix} \Phi_n^P \\ \Phi_\gamma^P \end{pmatrix} = \begin{pmatrix} \widetilde{M}_n & 0 \\ \widetilde{M}_{n,\gamma} & \widetilde{M}_\gamma \end{pmatrix} \begin{pmatrix} J_n^{IN}(+) \\ J_\gamma^{IN}(+) \end{pmatrix} \qquad (1')$$

where Φ_n^P represents neutron component of Ψ at the point " P ", while $J_n^{IN}(+)$ stands for neutron component of J_+ at the point " IN ". Submatrix M_n describes contribution of neutrons from a neutron energy group at " P " to other neutron energy group at " IN ". The same matter holds for M for gamma rays. On the other hand, the submatrix M_n describes contribution of neutrons in the cask cavity to the gamma-ray spectrum at " P " through secondary gamma rays produced in the cask wall due to neutron capture and neutron inelastic scattering reactions.

Radiation dose rate D_0 at the point P is obtained by inner product of column vectors Ψ^P and C as shown in Eq. (2), where C is a column vector consisting of flux to dose conversion factors for neutrons and gamma rays.

$$D_0 = \widetilde{C} \cdot \Psi^P \qquad (2)$$

also consists of neutron component C_n and gamma-ray component C_γ. As a consequence, Eq. (2) can be rewritten as follows,

$$D_0 = (C_n, C_\gamma) \begin{pmatrix} \Phi_n^P \\ \Phi_\gamma^P \end{pmatrix} \qquad (2')$$

CALCULATION OF MATRIX ELEMENTS

Group Structures

Matrix elements for RRMC can be obtained by a popular radiation shielding calculation code such as ANISN or MORSE, once energy group structures are given for both neutron and gamma-ray spectra.

As the aim of the present method is to economize the calculation of the transport index, fewer group structure seems to be more preferable. However, excessively few numbers of energy groups may lead poor accuracy of calculated result.

For a typical case of cask wall as shown in Table 1, suitable number of energy groups is found to be 7 for both neutron and gamma-ray components. In the determination of the radiation group structure, energy dependence of C_n and C_γ is taken into account. Group constants for 14 energy group structute are obtained by collapsing group constants in the DLC-data library with ANISN code.

Figures 2 through 5 show energy spectra of neutrons and gamma-rays calculated for DLC-23E type 22 group structure and final 7 group structure for neutrons, and for DLC-23E type 18 group structure and final 7 group structure for gamma rays. In Fig. 2 and 3, spatial dependence of energy spectra are given for neutron and gamma rays along with energy dependence of flux to dose conversion factors for neutron and gamma rays. Final group structures for neutron and gamma rays are also shown in these figures.

Matrix Element

For calculation of matrix element $M_{i,j}$, one dimensional Sn calculation code ANISN is applied to a system of cask geometry which consists of cask wall surrounded by air void of 1m thick, and of vacant cask cavity. A radiation source is assumed in the cavity. Source strenght is chosen to be unity. For convenience of calculation, source energy spectrum is chosen to be zero except an energy group i, which is chosen one after another from the first energy group to the last energy group. Energy spectra for radiation flux at the point P are calculated for all cases of source energy spectrum as well as energy spectra for radiation current towards outside way at the point IN. Matrix element $M_{i,j}$ is defined by the ratio;

$$M_{i,j} = \psi_j^P / J_{+\,i}^{IN}$$

where, ψ_j^P is the j-th energy component of the radiation flux. $J_{+\,i}^{IN}$ is the i-th energy component of radiation current.

Adopted cask geometry is shown in Table 1 as a typical case of casks. Obtained matrix is shown in Table 2. The matrix should be transposed when it is operated to the column vector of effective current $\bar{J}_+^{IN}$ for fuel loaded state.

III Reproduction of Spectrum

ESTIMATION OF EFFECTIVE CURRENT $\bar{J}_+^{IN}$

For calculation of radiation energy spectrum for flux at the point P, one needs estimation of a quantity $\bar{J}^{IN}$. The quantity is named effective current and is different from the current J_+^{IN+} calculated at the surface of naked cavity zone filled with spent fuels, water, structural materials and so forth. The difference is due to reflection and energy degradation effects of those radiations which impinge again on the cask wall from the cavity zone after back-scattering from the cask wall. J_+^{IN} is corrected into the effective current by taking account of these effects.

The correction is done by the following procedure;

$$\bar{J}_{+\,j}^{IN} = \sum_{k=1}^{\delta-1} \bar{J}_{+\,k}^{IN} f_{k,j} + J_{+\,j}^{IN} (1 + f_{j,j}) \qquad (3)$$

where, the quantity $f_{k,j}$ is a sort of reflection matrix of radiations in the coupled system of cask cavity and the cask wall. $f_{i,j}$ is obtained from reflection matrixes for cask wall $R_{l,m}$(cask wall) and for fuel loaded cask cavity $R_{l,m}$(cavity);

$$f_{i,j} = \sum_{m} R_{i,m}(\text{ cask wall }) R_{m,j}(\text{ cavity }) - R_{i,j}(\text{ cask wall }) \delta_{i,j} \qquad (4)$$

Reflection matrixes are shown in Tble 3 for the case of radiation source condition shown above. Matrix elements for $f_{l,m}$ are shown in Table 4. In Table 3, diagonal elements are put zero on the assumption that radiations entering into the cask wall directly from the radiation source are either absorbed in the cask wall or scattered into other energy groups or escape the cask wall, and never scattered back into the cavity.

APPLICATION OF RRMC

As an example of application of the RRMC, neutron energy flux spectrum at the point P is calculated for the case of radiation source energy spectrum being found

in the DLC-23 data library[1]. Cask goemetry and material descriptions are shown in Table 1. Obtained result is compared with exact one in Table 5. In the table, effective current obtained by the present method is also compared with exact one, where exact calculation is performed by the code ANISN.

Another example calculation is performed for a case of fictitious source energy spectrum, i.e. neutrons in energy group around 1 MeV is half the value of normal fission type source energy distributions. Obtained results for neutron and gamma-ray fluxes are shown in Figures 4 and 5.

IV Discussion

Table 5 shows that effective current and gamma-ray flux reproduced by the present method are twice as large as exact ones. Reflection data in matrixes R(cask wall) and R(cavity) are thought to be responsible for this overestimation. In obtaining reflection data, detailed information is not available yet for energy dependence of albedo of radiations impinged on cask materials. Detailed albedo data can be expected from Monte Carlo calculation.

Overestimation of gamma-ray flux in Tble 5 is due to pile-up of neutrons in 6-th energy group. One can see an evidence for this pile-up in the 6-th component of effective current. This pile-up is expected to be decreased by improving the reflection data through Eq.(3).

Matrix elements for gamma rays in RRMC become comparable order of values if one multiplies a factor 10^6. This fact is consistent with our experience. Relative magnitude of secondary gamma-ray component in the matrix against the other gamma-ray part seems to be in consistent with our experience,too.

V Conclusion

(1) Present method is effective to save machine time for calculation of transport index.
(2) Improved agreement can be expected between results obtained by the present method and exact one, if reflection data is improved by taking acount of detailed information for albedo data concerning with cask materials.
(3) RRMC seems to be more suitable quantity than the ordinary transport index in discussion of radiation dose in ships because the RRMC informs much about secondary gamma-rays produced in the cask wall in detail in addition to shielding ability of the cask against neutrons and gamma rays.

VI reference

1) Roussin R. W., DLC-23, RSIC Data Library Colection, (ORNL), (1973)

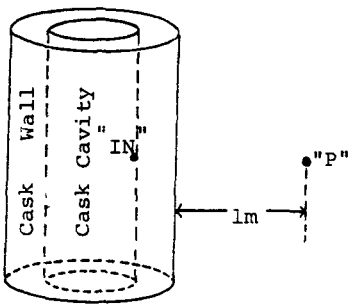


Fig.1 Schematic illustration of points "P" and "IN"

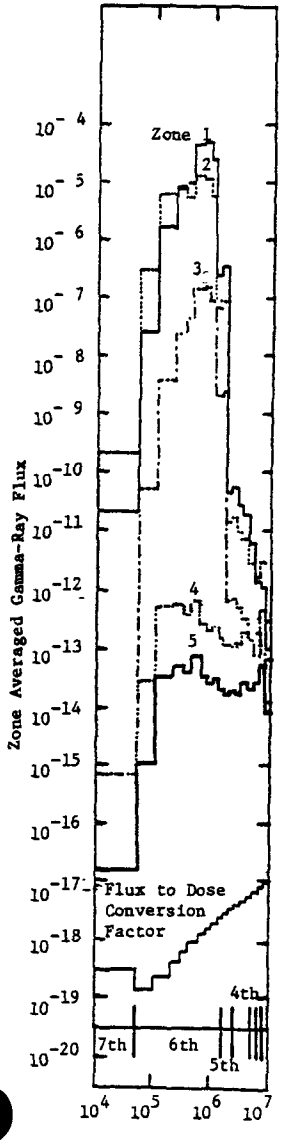


Fig. 3 Zone Averaged gamma-ray flux and flux to dose conversion factor.

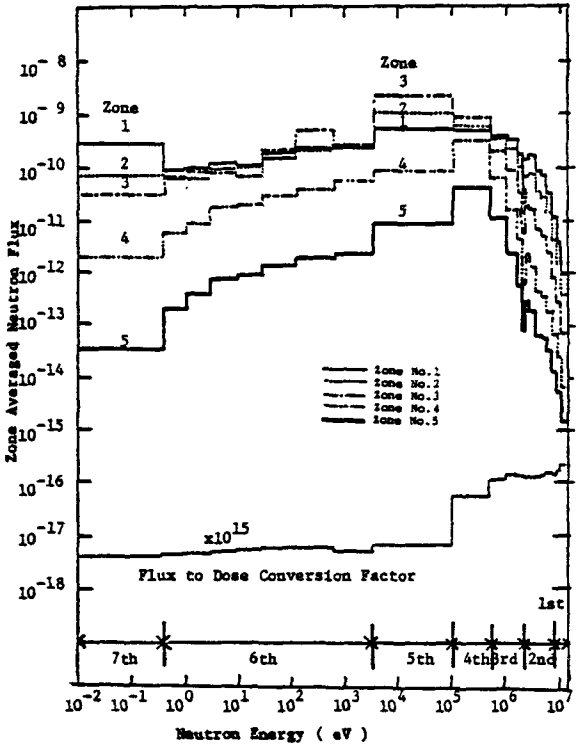


Fig. 2 Zone averaged neutron flux and flux to dose conversion factor for 22 energy groups. Energy group structure for 7 energy groups is also shown. Spatial behaviour of energy spectrum and energy dependence of the conversion factor are taken into account in the determination of group structure for the 7 energy groups.

Table 1 Description of Each Zone

Zone Number	Outer Radius	Inner Radius	Material *
1	0.0 cm	40.0 cm	Mixture
2	40.0	42.0	Iron
3	42.0	62.0	Lead
4	62.0	72.0	Iron
5	72.0	172.0	Air

*	Material	Concentration
	U-235	2.662-4
	U-238	7.701-3
	Zr	3.768-3
	H	3.540-2
	O	3.282-2
	Fe	4.347-3
	B	9.597-4

Neutrons	2.760-3	1.927-3	5.570-2	1.680-1	4.021-2	8.677-3	4.204-5	2.049-4	1.028-3	6.487-4	5.511-4	2.086-3	2.523-3	6.635-3
	0.0	2.921-3	3.371-2	1.173-1	2.734-2	5.508-3	2.639-5	1.188-4	6.055-4	3.176-4	2.746-4	1.064-3	1.270-3	3.671-3
	0.0	0.0	2.329-2	7.173-2	1.771-2	3.139-3	1.485-5	6.981-5	3.402-4	1.749-4	1.480-4	4.120-4	4.729-4	1.715-3
	0.0	0.0	0.0	2.651-3	5.168-3	3.335-3	1.597-5	3.357-5	2.302-4	1.093-4	8.473-5	2.264-4	1.745-4	5.624-4
	0.0	0.0	0.0	0.0	4.703-2	1.508-2	7.023-5	1.501-4	1.011-3	4.768-4	3.742-4	1.002-3	7.520-4	2.545-3
	0.0	0.0	0.0	0.0	0.0	1.594-2	8.505-5	1.859-4	1.369-3	6.650-4	5.028-4	1.342-3	1.086-3	3.130-3
	0.0	0.0	0.0	0.0	0.0	0.0	1.726-6	3.237-6	4.524-5	2.328-5	1.809-5	5.046-5	4.280-5	8.832-5
Gamma Rays	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.133-8	3.406-8	5.773-8	6.312-8	2.736-7	2.170-7	2.684-7
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.256-7	9.383-8	9.848-8	4.065-7	3.206-7	3.911-7
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.017-7	1.332-7	5.376-8	4.210-7	4.862-7
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.308-7	6.274-7	4.878-7	5.202-7
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.192-7	3.357-7	3.182-7
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.671-8	1.150-8
	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.24-20

Table 2 Radiation Response Matrix of Cask

0.0	9.725-2	8.259-1	8.080-1	3.736+0	3.257-1	6.548-3	4.205-3	2.010-2	2.321-2	2.037-2	1.060-1	1.562-1	7.811-1
0.0	0.0	5.640-1	4.596-1	2.169+0	1.968-1	4.000-3	1.075-3	7.391-3	4.185-3	5.010-3	5.708-2	8.051-2	4.461-1
0.0	0.0	0.0	2.399-1	1.270+0	1.135-1	2.428-3	6.639-4	4.214-3	1.667-3	1.373-3	3.471-3	1.013-2	1.140-1
0.0	0.0	0.0	0.0	8.030-2	3.212-2	2.677-4	2.420-4	9.915-4	4.393-4	3.931-4	1.011-3	6.108-4	7.727-3
0.0	0.0	0.0	0.0	0.0	2.558-1	2.257-3	6.116-4	4.532-3	1.791-3	1.351-3	3.260-3	2.351-3	2.839-2
0.0	0.0	0.0	0.0	0.0	0.0	7.280-3	1.308-3	1.054-2	4.113-3	2.910-3	6.744-3	5.318-3	6.253-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	7.390-3	9.707-2	3.682-2	2.576-2	5.889-2	4.796-2	5.534-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.917-4	4.890-4	5.430-4	2.762-3	6.009-3	4.442-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.112-4	6.032-4	3.180-3	6.859-3	4.033-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	6.049-4	3.823-2	8.229-3	3.706-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.726-3	9.389-3	3.411-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.052-2	3.121-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.493-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

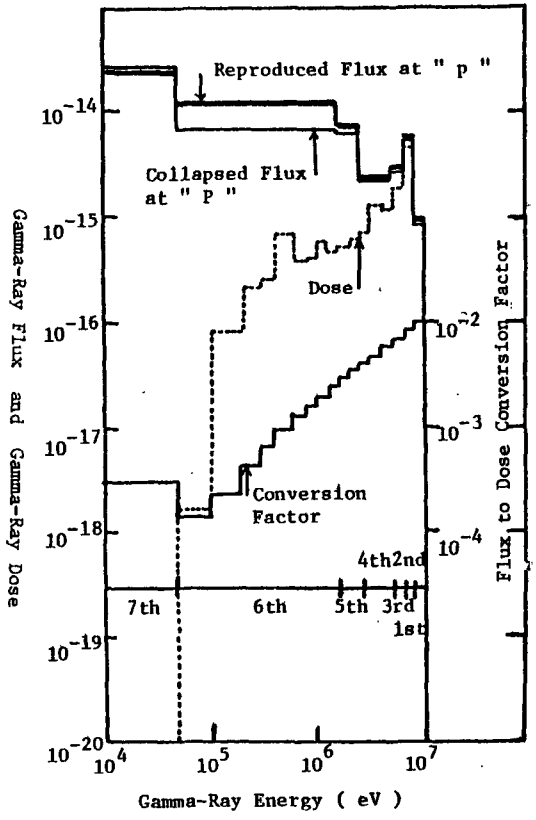
Table 3 Reflection matrix for cask wall
diagonal part is put zero

0.0	6.858-2	7.458-2	4.278-2	4.700-2	8.250-2	2.135-2	1.255-4	3.853-3	1.608-2	5.456-3	7.940-2	1.144-1	4.360-1
0.0	0.0	9.214-2	5.927-2	6.724-2	1.156-1	2.990-2	2.734-5	3.218-4	1.129-3	4.704-3	7.983-2	1.232-1	4.826-1
0.0	0.0	0.0	8.800-2	1.072-1	1.747-1	4.510-2	3.261-5	3.645-4	7.513-4	5.401-3	9.256-2	1.290-1	5.410-1
0.0	0.0	0.0	0.0	1.695-1	2.671-1	6.900-2	3.874-5	4.376-4	8.708-4	6.456-3	1.119-1	1.528-1	6.577-1
0.0	0.0	0.0	0.0	0.0	3.334-1	8.891-2	3.983-5	4.544-4	9.017-4	6.741-3	1.173-1	1.604-1	7.114-1
0.0	0.0	0.0	0.0	0.0	0.0	1.132-1	3.664-5	4.221-4	8.323-4	6.294-3	1.102-1	1.507-1	6.958-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.524-5	5.778-4	1.130-3	3.422-3	5.028-2	9.297-2	7.291-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.358-4	2.839-4	3.532-4	1.988-3	4.318-3	2.031-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.625-4	3.758-4	2.278-3	5.099-3	1.855-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.384-4	2.523-3	5.953-3	1.708-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	2.603-3	6.837-3	1.068-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	7.930-3	1.338-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.325-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 3 (Continued) Reflection matrix for cask cavity

0.0	8.889-3	1.012-1	1.679-1	7.954-1	1.716+0	4.678-1	2.270-4	2.675-3	5.709-3	3.763-2	6.526-1	8.973-1	4.074
0.0	0.0	7.496-2	1.103-1	4.788-1	1.020+0	2.779-1	1.342-4	1.527-3	3.122-3	2.260-2	3.921-1	5.404-1	2.440
0.0	0.0	0.0	6.818-2	2.841-1	6.065-1	1.651-1	8.016-5	9.104-4	1.821-3	1.348-2	2.339-1	3.210-1	1.424
0.0	0.0	0.0	0.0	1.667-1	2.800-1	7.473-2	3.995-3	4.496-4	8.944-4	6.636-3	1.151-1	1.572-1	1.681
0.0	0.0	0.0	0.0	0.0	2.953-1	8.828-2	3.577-5	4.092-4	8.082-4	6.060-3	1.057-1	1.446-1	6.548-1
0.0	0.0	0.0	0.0	0.0	0.0	7.856-2	2.514-5	2.853-4	5.738-4	4.278-3	7.483-2	1.025-1	4.758-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.783-5	2.341-4	4.708-4	1.400-3	1.985-2	3.804-2	3.981-1
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.362-8	1.648-7	3.862-7	6.203-6	5.046-5	5.598-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.383-7	3.028-7	6.211-6	5.589-5	5.108-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.619-7	3.838-5	3.361-4	5.186-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	5.251-6	6.258-5	4.370-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.712-5	3.996-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.351-2
0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Table 4 Reflection matrix $f_{i,j}$ for coupled system



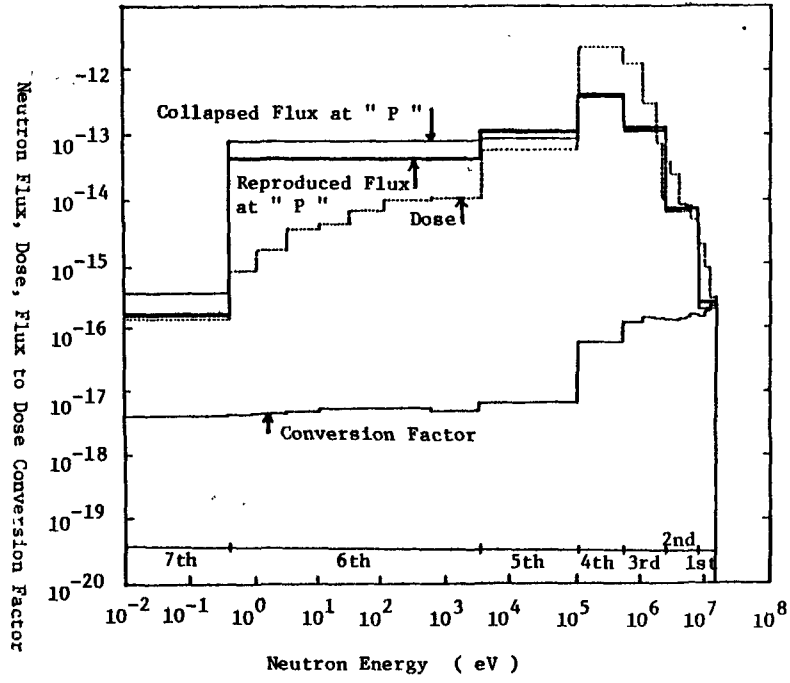


Fig. 4 Collapsed neutron flux (neutrons/cm^2/sec) and reproduced neutron flux for 7 energy groups, neutron dose (mrem/hr) for 22 groups and neutron flux to dose conversion factor (mrem/hr/neutrons/cm^2/sec) for 22 groups. Neutron fluxes are multiplied by 10^{-2}.

Fig. 5 Collapsed gamma-ray flux (gamma rays/cm^2/sec) and reproduced gamma ray flux for 7 energy groups, gamma-ray dose (mrem/hr) for 18 groups and gamma-ray flux to dose conversion factor (mrem/hr/gamma rays/cm^2/sec) for 22 groups. Gamma-ray fluxes are multiplied by 10^{-2}.

	Current at " IN "			Flux at " P "	
	J_+^{IN} (Uncorrected)	$\bar{J}_+^{IN}$ (Corrected)	$\bar{J}_+^{IN}$ (Exact)	ψ^P (Predicted)	ψ^P (Exact)
Neutrons	2.389-6	2.389-6	2.389-6	6.952-9	6.987-9
	6.994-5	6.996-5	7.110-5	2.090-7	2.197-7
	1.134-4	1.189-4	1.252-4	5.260-6	4.709-6
	4.015-5	5.638-5	5.662-5	1.728-5	1.242-5
	3.380-5	1.117-4	6.510-5	9.661-6	2.787-6
	4.535-5	2.417-4	1.187-4	6.505-6	2.537-6
	5.487-6	7.874-5	2.172-5	3.315-8	1.280-8
Gamma Rays	3.630-8	2.562-7	7.518-8	8.095-8	3.447-8
	4.436-7	8.235-7	8.323-7	5.457-7	2.251-7
	1.202-6	1.965-6	1.579-6	2.666-7	1.116-7
	5.822-6	1.129-5	7.717-6	2.029-7	8.783-8
	9.426-5	1.890-4	1.256-4	5.812-7	2.615-7
	1.317-4	2.627-4	1.744-4	5.109-7	2.578-7
	4.452-4	1.109-3	6.383-4	1.549-6	7.689-7

Table 5 Effective current and flux spectrum

IMPACT TEST OF TRUCK WITH NUCLEAR FUEL SHIPPING CASK

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INTRODUCTION

At present,the land transportation of nuclear fuel materials is mainly by trucks loaded with new fuel assemblies,of which the transportation distance is considered to be relatively long extending from 150 to 800 Km. Careful elaboration,therefore, is necessary in the safety evaluation of truck transportation against the risk of accidents which might arise enroute.

We carried out a series of preliminary tests and a main test in order to find ways and means to cope with the anticipated increase of nuclear fuel materials transportation in the future.

The objects of these test were:

i) To clarify the corresponding relations between the experimental conditions provided for by the present regulatory safety standards and the experimental conditions for the test of the transport system consisting of the package (hereinafter called the "cask") combined transport vehicle.

ii) To examine and evaluate the integrity of the cask under the impact of a vehicular system crash.

iii) To gather and obtain data for further simulation analyses.

Experiments

The experiments were undertaken in 1978 at the automotive crash test facility of the Japan Automobile Research Institute Incorporation (JARI) , using a vehicle system consisting of a large-size truck,a shock absorber , a cask with a fresh fuel assembly (PuO2-UO2 dummy fuel),and other auxilliary equipment.

The cask is called "TN 9180 Type Packaging". (Type BU) The total weight of the vehicular systems was approximately 10 metric tons. The vehicular systems were crashed head-on against a fixed flatfaced barrier at a velocity of 50 Km/h, impact load,acceleration,velocity, displacement,deflection,etc. at each section of the transport systems were measured,and the data were analysed with the above listed purposes in mind.

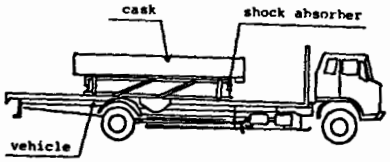


Fig.1 Side view of the test vehicle

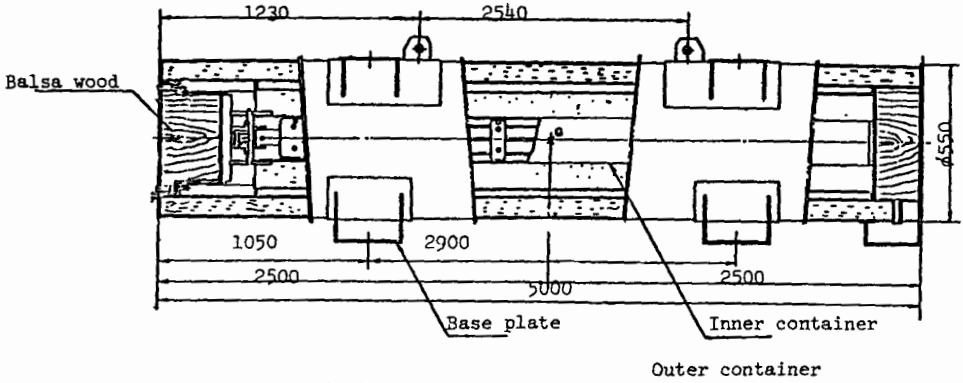


Fig.2 Cask

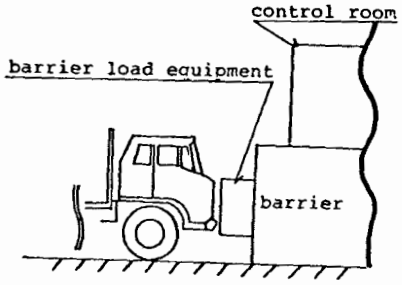


Fig. 3 Impact test configuration

Conclusion

The impact load,acceleration,velocity,displacement rate,deformation rate,etc. produced by frontal collision against a fixed barrier of a transport vehicle systems of gross weight of about 10 metric tons were measured and the obtained data were analysed.

The results of the experiment are summed up as follows:

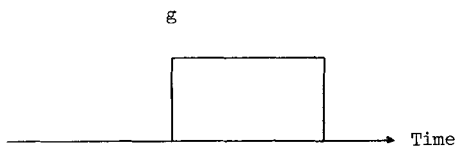
(1) the experimental results differed considerably in comparison with those of the 9-meter free drop test.

(2) The integrity and crashworthiness of the trans port cask and the shock absorber have been confirmed.

(3) The shock absorber has a vertical buffering effect. But its horizontal buffering effect has not been made clear.

(4) The shock continuing time was in the range of 200 msec-270 msec.

(5) The cask's acceleration was about 11 g and in square wave form.



(6) The peak value of the barrier load was about 200 metric tons. This result was mainly because of the mass and collapsing effect of the vehicle itself,excepting the platform.

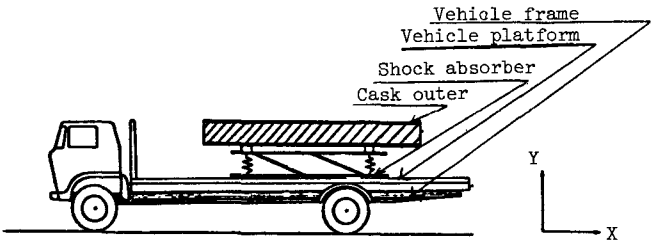
(7) Atotal of 9 % of the kinetic energy of the platform, damper, the cask was absorbed in the test. In other word, about 5 % of the kinetic energy of the transport system was absorbed by the platform displacement. About 90 % of the kinetic energy when the transport vehicle crashed was absorbed by the cabin.

(8) The diplacement of each part of the transport systems was shown in Fig.4.

(9) The velocity of each part of the transport systems was shown in Fig.5.

(10) The comarison of the acceleration data of 9-meter drop test and 50 Km/h truck impact test was shown in Fig.6.

(We performed 9-meter drop tests when made a safety analysis report.)



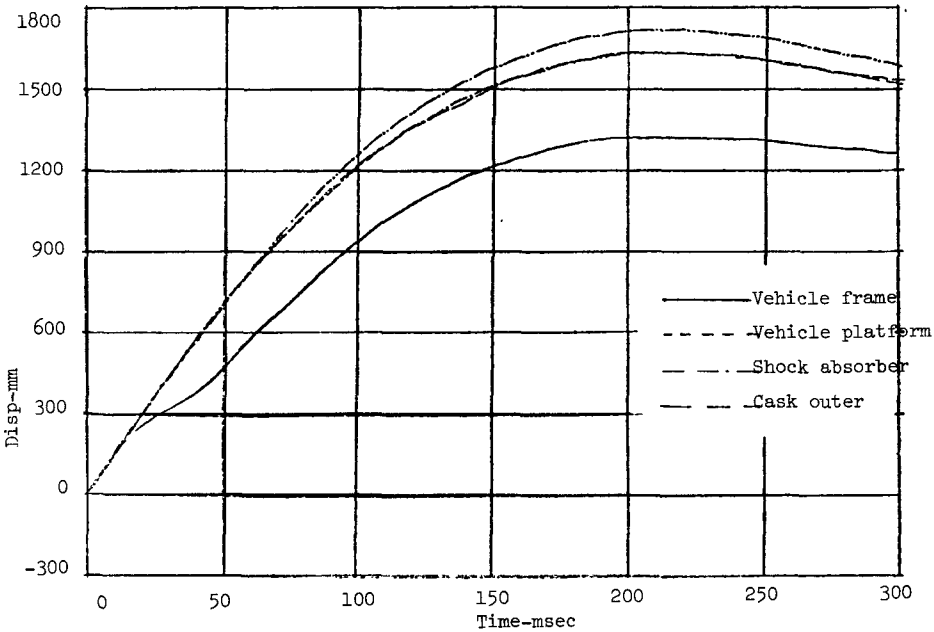
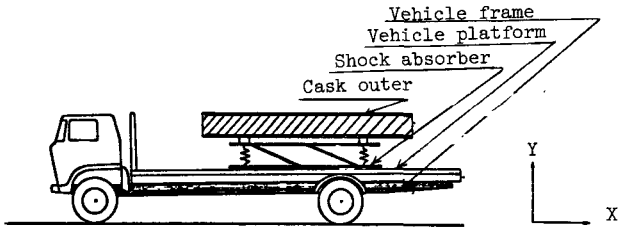


Fig.4 Displacement of each part of transport systems(X)



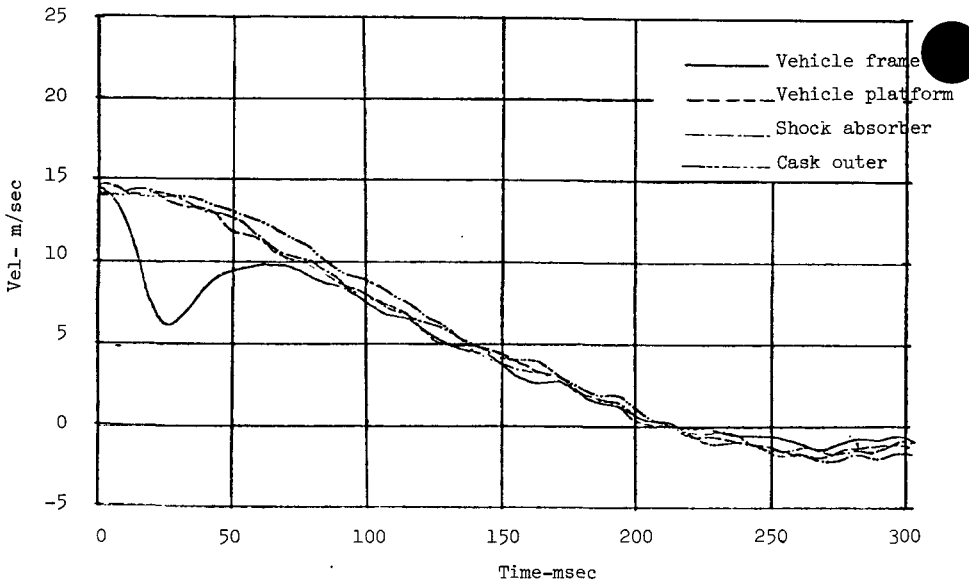


Fig.5 Velocity of each part of transport systems (X)

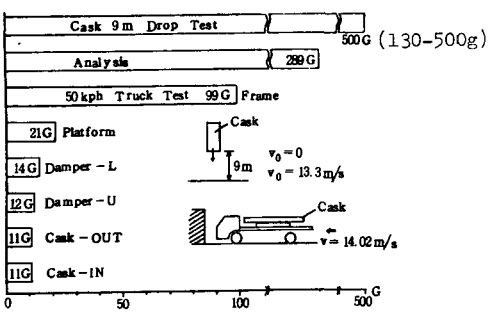


Fig. 6 Comparison of cask 9m drop test and 50 kph truck test

STUDIES OF PHYSICAL PROPERTIES OF PLUTONIUM DIOXIDE POWDERS

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I - INTRODUCTION

Quantities of plutonium that have to be transported or stored increase with the development of nuclear reactors and fuel reprocessing and it appears necessary to study the design of large containers ($\simeq$ 30 kg PuO_2) allowing to carry safely plutonium dioxide powders.

The aim of this research is the improvement of the current knowledge about physical properties of PuO_2 required for the design of these containers.

Data that are necessary to know for this development are :

- the evolution of the pressure in a closed container as a function of time
- the evolution of the gas composition filling the container, which is composed of initial air, water vapor desorberd and radiolysis products generated from moisture absorbed on the powder.
- the temperature inside the powder and at the surface of the container

In this paper we shall describe the studies we have made on thermal conductivity, specific power and radiolysis phenomena of plutonium dioxide powders.

A review of the informations available in the literature has been achieved and it has shown that the influence of moisture is not very well known.

Experiments have been undertaken to elucidate this aspect and its influence on other parameters.

II - EXPERIMENTAL

II.1 - Plutonium Dioxide Samples

Two kinds of oxides have been used. Oxides issued from the French reprocessing plants of Marcoule and La Hague, which were low specific emissivity oxides. One oxide prepared in our laboratory to simulate oxides obtained from LWR reactors fuel reprocessing (2 % ^{238}Pu).

Their main properties are given in Table I.

n° sample	Origin	Isotopic composition weight % 238	239	240	241	242	Specific area $m^2.g^{-1}$	Specific power $W.Kg^{-1}$	Use
1	La Hague	0.13	72.28	22.80	4.04	0.75	-	3.482	Conductivity measurements calorimetry
2	"	0.08	76.79	19.52	3.15	0.46	-	3.103	
3	Marcoule	0.16	73.58	21.49	3.84	0.93	20.03	3.430	
4 and 6	La Hague	0.08	76.35	19.57	3.46	0.53	14.35	3.520	
5	our laboratory	2.07	55.75	29.83	6.01	6.30	2.7	≈13,50	Radiolysis studies (5 and 6)

Table I : Properties of the PuO_2 used in our study

II.2 - Thermal Conductivity Measurements

They have been achieved using two different measurements :

- a large container (9 kg of PuO_2) was used to measure thermal conductivity as a function of mean temperature and density. The method is based upon the measure of the radial heat flow in the cylindrical container, without complementary heating, by temperature measurements on the axis and on the surface of the container.

- others experiments have been achieved with smaller quantities of PuO_2 (0.2 kg) using a container fitted with a complementary electric heating. The cell is represented in fig. (1).

The general formula giving the thermal conductivity λ is

$$\lambda_{W.cm^{-1}C^{-1}} = \frac{1}{T_1 - T_0}\left[\frac{P}{4}\left(R_0^2 - R_1^2 - 2R_1^2 \ln\frac{R_0}{R_1}\right)\right] + \frac{\phi}{2\pi}\ln\frac{R_0}{R_1}$$

T_1 and T_0 are respectively internal and external temperatures (°C)
R_1 and R_0 are respectively internal and external radii (cm)
P specific power of the powder $W.cm^{-3}$
ϕ additive heat flow $W.cm^{-1}$

II.3 - Specific Power of the Powders

It has been measured by calorimetry using a Calvet Calorimeter.

II.3 - Radiolysis Studies

The rate of production and the composition of the evolved radiolytic gases has been determined using pression measurements in the closed container and a gas chromatographic method to analyse the gaseous products. The general view of the system is given in fig. (2) and the detailed view of the container for PuO_2 is given in fig. (3)

Radiolysis yields G defined as :

$$G = \frac{\Delta N}{I} \times 100$$

with - ΔN number of molecules of gas formed or consumed for one unit of time and one unit of mass of PuO_2
- I energy dissiped by the α particles by units of time and mass in eV.

$$I = \frac{N \text{ Log } 2}{d} \sum \frac{Em}{\gamma M} \times 10^6 \text{ eV}$$

with E = α energy in MeV
γ = half life in year
m = mass (in the unit of mass)
M = number of mass
N = Avogadro number
d = number of days in one year

III - RESULTS AND DISCUSSION

III.1 - Thermal conductivity has been measured as function of moisture of PuO_2 powders ; different moistures have been obtained by passing air of variable moisture through the powder till the equilibrium was reached. Moisture content of the powder is estimated from the loss of weight of the powder heated at 900°C.

The results of Table II show that no significant effect of moisture is observed.

Sample n°	Moisture % in weight	T °C	ρ g.cm^{-3}	λ W.cm^{-1} °C^{-1}	$\Delta\lambda/\lambda$ ±
3	1.2	90	1.97	$9.9.10^{-4}$	10 %
3	1.4	90	1.97	$11.0.10^{-4}$	10 %
3	1.67	90	1.97	$8.9.10^{-4}$	10 %
3	1.2	120	1.97	$10.1.10^{-4}$	10 %
3	1.4	120	1.97	$11.2.10^{-4}$	10 %
3	1.67	120	1.97	$9.1.10^{-4}$	10 %
4	1.2	90	1.69	$7.7.10^{-4}$	10 %
4	1.4	90	1.69	$7.9.10^{-4}$	10 %
4	1.67	90	1.69	$7.5.10^{-4}$	10 %

Table II - Thermal conductivity of PuO_2 powders

From tables II and III, it is observed that thermal conductivity increases slightly with temperature and rapidly when the density of the powder increases.

Sample n°	ρ $g.cm^{-3}$	T °C	λ $W.cm^{-1}.°C^{-1}$	$\Delta\lambda/\lambda$ ±
1 and 2	1.76	90	$8.2.10^{-4}$	3 %
	2.01	90	$10.1.10^{-4}$	8 %
	2.28	90	$11.2.10^{-4}$	8 %
	1.76	120	$9.0.10^{-4}$	3 %
	2.01	120	$11.2.10^{-4}$	8 %
	2.28	120	$12.5.10^{-4}$	8 %

Table III - Thermal conductivity of PuO_2 powders

Increasing densities of the powders were obtained by vibrating the powders.

The main sources of incertitude result of variations of temperature during the day which change the thermal equilibrium, a thermal flow which is not perfectly radial, incertitudes on temperature and estimation of the complementary heating.

From these experiments, it appears that the powder density is the most important factor which governs thermal conductivity, while moisture and temperature have little effect within the range (1.2 to 1.67 % in weight of moisture and 90°-120°C). Thermal conductivity increases with the density of the powder.

III.2 - Calculated and measured values of the specific power for several oxides of different isotopic compositions are compared in Table IV.

Sample n°	Calculated power $mW.g^{-1}$	Measured power	Difference %
1	3.424	3.482	2
2	3.017	3.104	3
4	3.330	3.520	5

Table IV - Specific power of PuO_2 powders

The errors are generally caused by a bad estimation of ^{241}Am.

III.3 - Radiolysis Studies

In our experimental conditions and over period of time of one to three months, we have never observed significant increase of pressure which could be attributed to radiolysis phenomenon. Radiolysis of moisture of PuO_2 powders is however observed by the appearance of hydrogen in the gas above the powder.

The effect of temperature on radiolytic yields of total and individual gase is seen in Table V.

T°C	G_T	G_{H_2}	G_{O_2}	Others
20	$1.7.10^{-5}$	$1.7.10^{-5}$	$\simeq 0$ limit of detection	-
100	$< 0^{*}$	$1.7.10^{-5}$	$- 1.2.10^{-3}$*	$GCO_2 < 10^{-5}$
200	$8.3.10^{-4}$	$2.0.10^{-5}$	$8.1.10^{-4}$	-
350	$1.6.10^{-3}$	0	$1.6.10^{-3}$	-

Table V - Effect of temperature on radiolytic yields (moisture $\leq$ 0,44 %, 2 % ^{238}Pu)

* These results seem in contradiction with the others that may be interpreted by an error in the analysis.

We observed an increase of the total and oxygen yields with temperature while G_{H_2} remains rather low.

The effect of moisture is clearly seen in Table VI, where it is seen that G_T, G_{H_2} and G_{O_2} increase with higher moisture (T = 20°C 2 % ^{238}Pu).

H_2O % weight	G_T	G_{H_2}	G_{O_2}	Others
0.18	$1.7.10^{-5}$	$1.7.10^{-5}$	$\simeq 0$ (limit)	-
1.11	$2.1.10^{-4}$	$3.8.10^{-5}$	$1.8.10^{-4}$	$GCO_2 < 10^{-5}$
1.36	$2.6.10^{-3}$	$1.3.10^{-3}$	$1.3.10^{-3}$	$GCO_2 \simeq 10^{-5}$

Total yield increase with higher ^{238}Pu content of the oxide and results mainly of an increase of G_{O_2}.

This study has shown that the gaseous yield remains low in the conditions of storage of PuO_2 powders and that it increases with temperature, moisture of the powder and ^{238}Pu content.

IV - CONCLUSION

This study has brought numerous data on behaviour of plutonium dioxide powder in the conditions of transport. Among the parameters, the moisture of the powder appears to be the most important factor conditioning the transport.

For instance in a large container of 30 kg PuO_2, containing 1 % moisture (300 g) the release of hydrogen could be 5,2 STP liter in one month. This example shows that hydrogen cannot be neglected in the conception of large containers for plutonium dioxide powders and that a special attention will be paid to the water content of the powder. The drier the powder will be, the surer will be the transport.

ITEMS OF THE FOLLOWING FIGURES

Fig. (1) General view of the cell used for thermal conductivity measurement

1 - Altuglass box
2 - Thermoregulated oil for circulation
3 - Thermocouples "thermocoax"
4 - Heating wires "thermocoax"
5 - Pump for air circulation
6 - Isolating gate for the cell

Fig. (2) General view of the system

Fig. (3) Detailed view of the container for PuO_2 powder

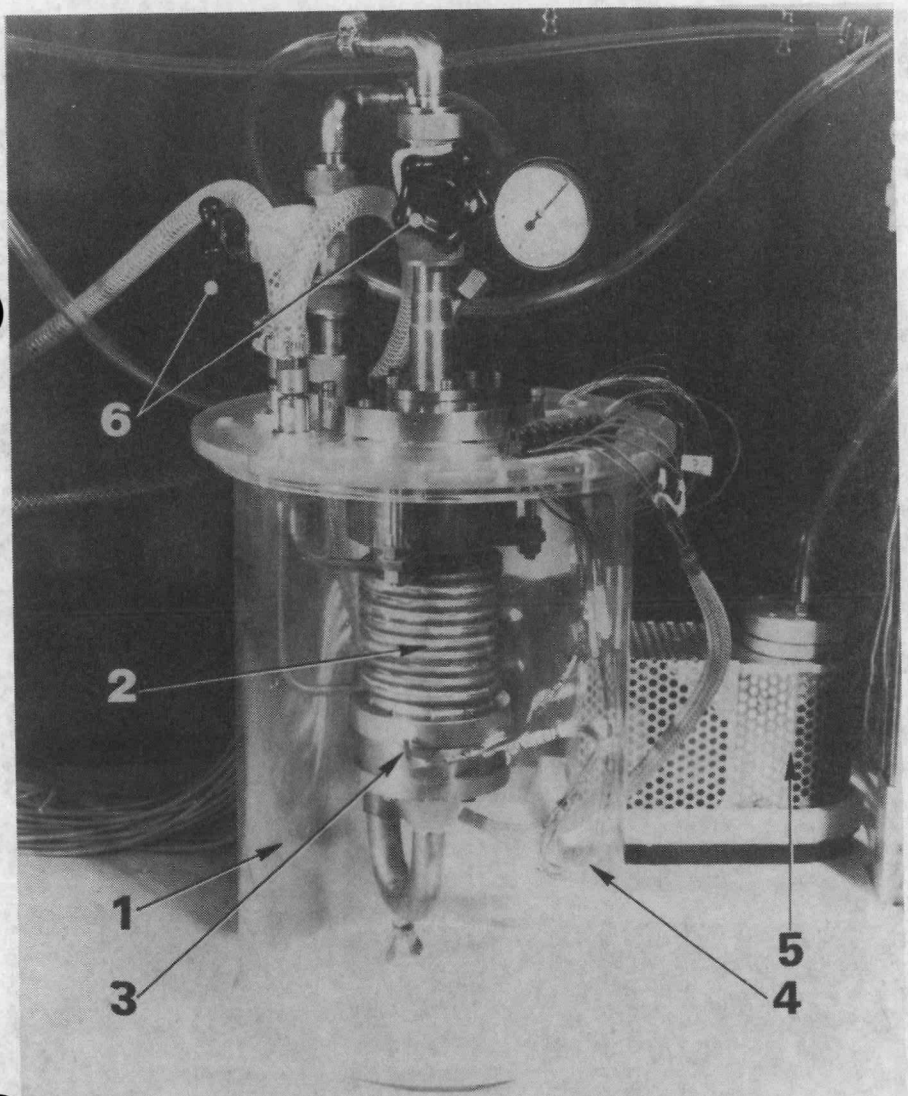

Figure 1

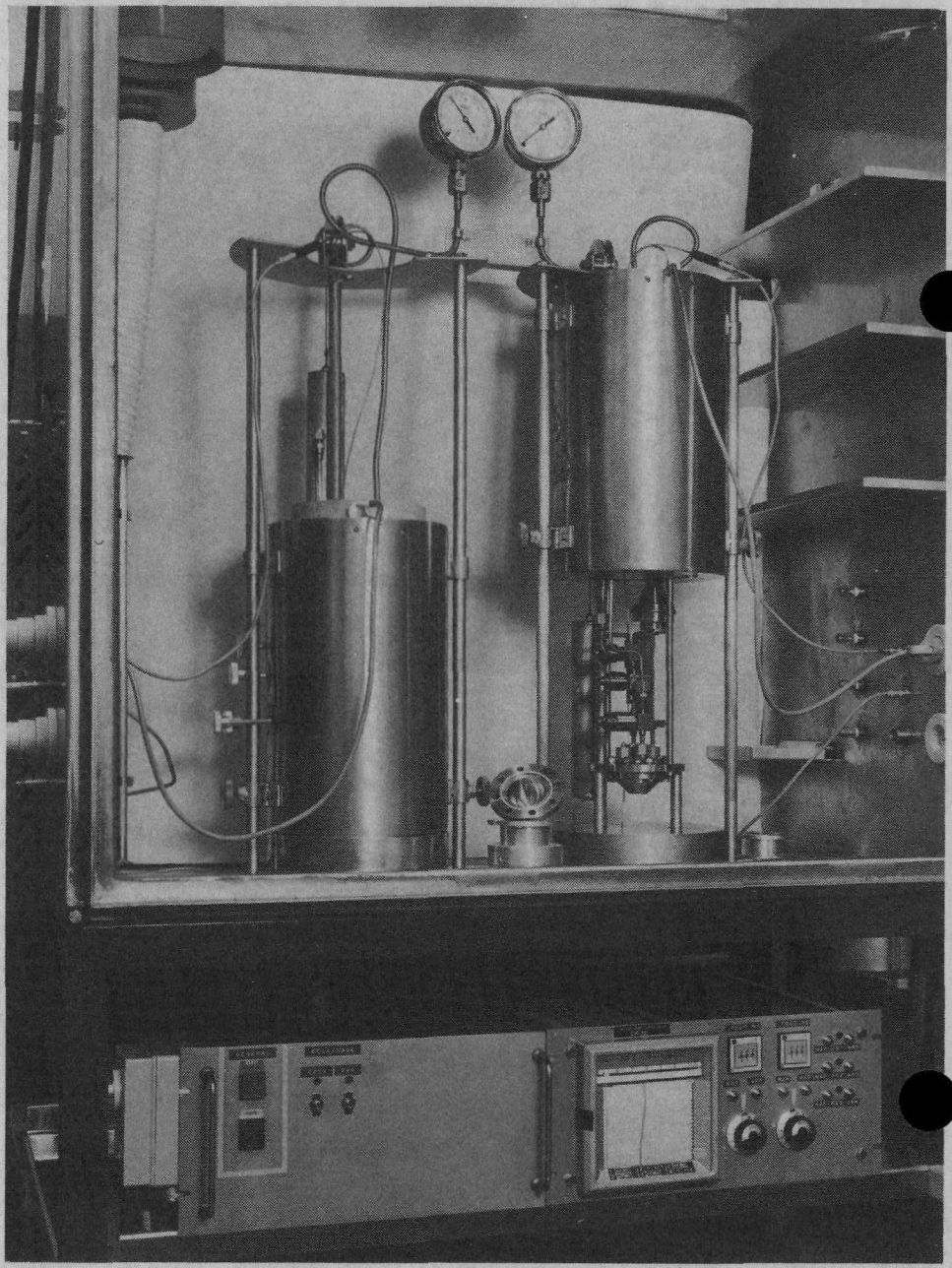

Figure 2

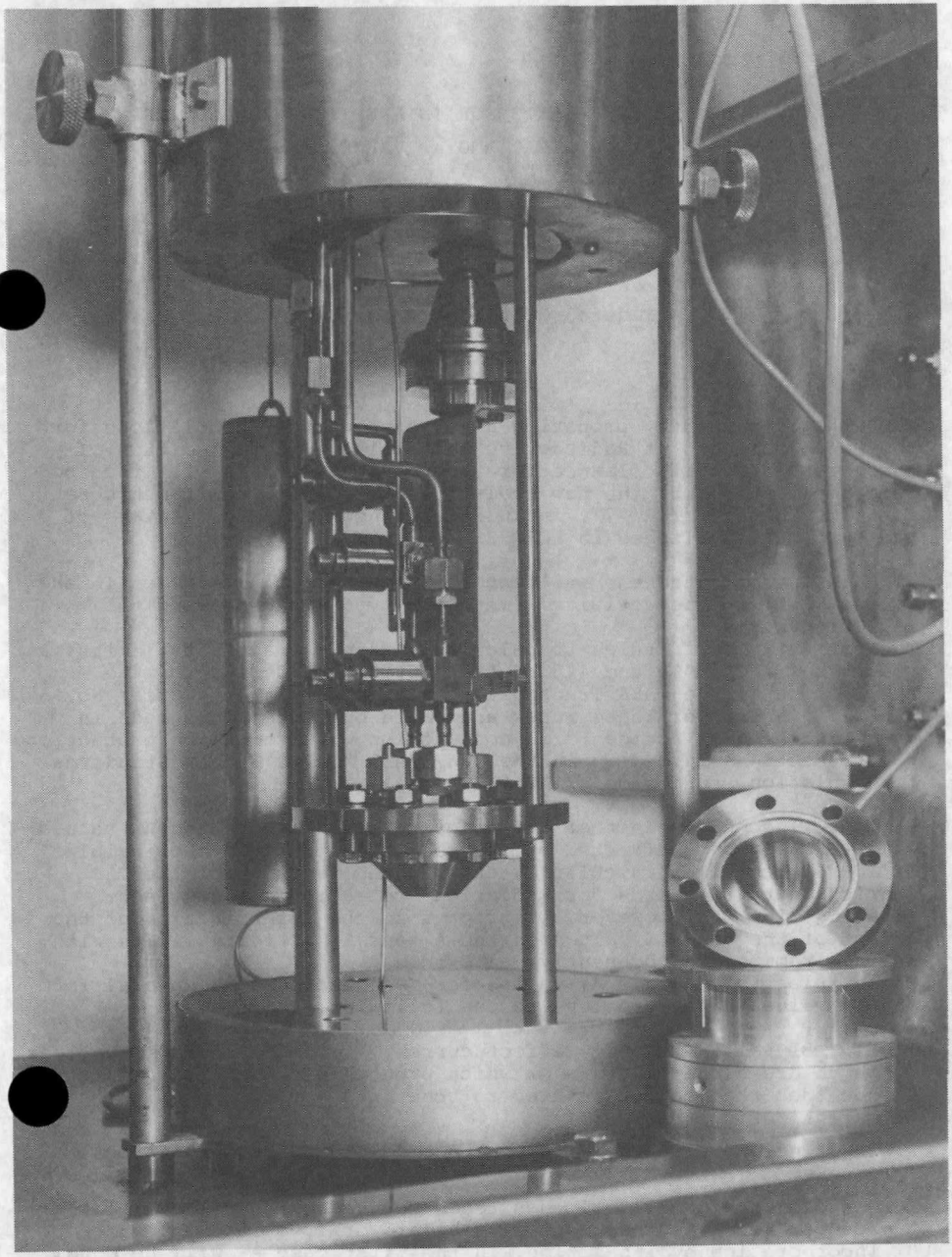

Figure 3

IMPLICATIONS OF THE NEW LIMITS FOR INTAKE (ICRP 26 AND ICRP 30) ON THE A_1 AND A_2 VALUES

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The IAEA is presently preparing a revision of the "Regulations for the Safe Transport of Radioactive Materials". The new edition of these regulations is planned for the year 1983 and it will incorporate implications of the new radiation protection requirements set forth in ICRP 26 /ICR 77/, including the new limits for intake of radionuclides published in ICRP 30 /ICR 79/.

We have investigated the influence of these annual limit for intake values (ALI) on the activity limits for type A packages on the following basis:
The currently valid rules to calculate A_1 and A_2 activity limits, as summarized in §§ 412 and 413 of the "Advisory Material for the Application of the IAEA Transport Regulations" (Safety Series No. 37) /IAE 73/, are used. These rules are based on assumptions made in case of a "median accident" in conjunction with radiation protection requirements as laid down in the 1967 IAEA "Basic Safety Standards for Radiation Protection" /IAE 67/.

It is well conceivable that the rules to determine A_1 and A_2 values will themselves be effected by the radiation protection principles of ICRP 26 and in particular by the planned revision of the IAEA "Basic Safety Standards for Radiation Protection". The present investigation was, however, restricted to the determination of the effect of the new ALI values on the A_1 and A_2 activity limits within the frame of the presently valid rules.
These rules to evaluate A_1 values for radionuclides in special form and A_2 values otherwise lead to the following findings:

1. New ALI values may only effect current A_1 limits of radionuclides which are alpha-emitters or which produce alpha-emitting daughter nuclides that can be assumed to come into equilibrium with the parent during transport.
2. In principle all A_2 limits can be effected by new limits for intake.
3. For each radionuclide it has to be decided whether the A_2 limit results from external radiation hazard or from internal after incorporation by inhalation. In the first case A_2 will be equal to A_1 and will not be effected by the new ALI for this radionu-

clide. But if the greater risk is to be attributed to possible incorporation, A_2 will be determined by the annual limit for intake by inhalation.

Table VII of the 1979 edition of the IAEA Safety Series No. 6 /IAE 79/ comprises 251 radionuclides. New ALI values are presently available for 143 of these nuclides, either from ICRP 30 Part 1 or preliminarily from NRPB Report R 82 /NRP 78/. It has to be noted, that up to three ALI values can exist for a radionuclide depending on the lung retention class (D, W or Y) attributed to different chemical compounds. Since such a distinction is not foreseen in Table VII of the Safety Series No. 6 new A_1 and A_2 values are derived from the most restrictive ALI value.

The results can be summarized as follows:

1. For 74 of those 143 radionuclides of Table VII of the Safety Series No. 6 for which new ALI values are presently available one finds that $A_1 = A_2$. This case can either result because external radiation hazard limits the allowed activity or because A_1 and A_2 take the upper limit of 1000 Curie or if the radionuclide is classified as "low specific activity". With two exceptions new ALI values do not change the A_1 and A_2 values in this case.
2. For the remaining 69 of 143 radionuclides for which $A_1 \neq A_2$ the new ALI values lead to new A_1 or A_2 values in 58 cases. The changes are quite balanced:
   for 27 radionuclides the A_1 and/or A_2 values are more restrictive for 33 less restrictive.
   These 58 radionuclides and the changes induced by new ALI values are summarized in a table.

References

/IAE 67/ IAEA, Vienna:
Basic Safety Standards for Radiation Protection,
Safety Series No. 9, 1967 Edition, Vienna, 1967

/IAE 73/ IAEA, Vienna:
Advisory Material for the Application of the IAEA
Transport Regulations, Safety Series No. 37,
Vienna, 1973

/IAE 79/ IAEA, Vienna:
Regulations for the Safe Transport of Radioactive
Materials, 1973 Revised Edition (As Amended),
Safety Series No. 6, Vienna, 1979

/ICR 77/ Recommendation of the International Commission on
Radiological Protection, ICRP Publication 26,
Pergamon Press, 1977

/ICR 79/ Limits for Intake of Radionuclides by Workers,
ICRP Publication 30, Part 1,
Annals of the ICRP 2, No. 3/4, 1979

/NRP 78/ Annual Limits of Intake of Radionuclides for Workers,
National Radiological Protection Board, NRPB-R 82 (1978)

Table of A_1 and A_2 values which change due to new ALI values

Nuclide	A_1 (OLD)	A_1 (NEW)	OLD⋛NEW	A_2 (OLD)	A_2 (NEW)	OLD⋛NEW
AM241	8	3	> 2.7	0.008	0.003	> 2.7
AM243	8	3	> 2.7	0.008	0.003	> 2.7
BI210	100			4	10	< 2.5
BR 77	70			40	70	< 1.8
CA 45	1000			40	400	< 10.0
CD109	1000			70	10	> 7.0
CE141	300			200	300	< 1.5
CF249	2	3	< 1.5	0.002	0.003	< 1.5
CF252	2	2		0.009	0.01	< 1.1
CL 36	300			30	100	< 3.3
CM242	200	100	> 2.0	0.2	0.1	> 2.0
CM243	9	4	> 2.3	0.009	0.003	> 2.3
CM244	10	5	> 2.0	0.01	0.005	> 2.0
CM245	6	3	> 2.0	0.006	0.003	> 2.0
CM246	6	3	> 2.0	0.006	0.003	> 2.0
CS135	1000			100	500	< 5.0
CS137	30			20	30	< 1.5
EU152	20			20	9	> 2.2
EU154	10			5	7	< 1.4
EU155	400			90	80	> 1.1
FE 55	1000			1000	900	> 1.1

Nuclide	A_1 (OLD)	A_1 (NEW)	OLD><NEW	A_2 (OLD)	A_2 (NEW)	OLD><NEW
I125	1000			70	30	> 2.3
I129	1000			2	4	< 2.0
I131	40			10	30	< 3.0
NB93M	1000			200	80	> 2.5
NI 59	1000			900	1000	< 1.1
NI 63	1000			100	1000	< 10.0
NP237	5	3	> 1.7	0.005	0.003	> 1.7
PA231	2	0.7	> 2.9	0.002	0.0007	> 2.9
PB210	100			0.2	0.09	> 2.2
PB212	6			5	6	< 1.2
PM147	1000			80	70	> 1.1
PO210	200	300		0.2	0.3	< 1.5
PR143	300			200	300	< 1.5
PU239	2	3	< 1.5	0.002	0.003	< 1.5
PU240	2	3	< 1.5	0.002	0.003	< 1.5
RA223	50	50		0.2	0.4	< 2.0
RA224	6	6		0.5	0.8	< 1.6
RA226	10	10		0.05	0.3	< 6.0
RA228	10			0.05	0.5	< 10.0
RU106	10			7	5	> 1.4
S 35	1000			300	900	< 3.0
SB125	40			30	40	< 1.3
SM151	1000			90	50	> 1.8
SR 89	100			40	70	< 1.8
SR 90	10			0.4	1	< 2.5

Nuclide	A_1 (OLD)	A_1 (NEW)	OLD>< NEW	A_2 (OLD)	A_2 (NEW)	OLD>< NEW
TC 99	1000			80	300	< 3.8
TE125M	1000			100	300	< 3.0
TE127M	300			40	100	< 2.5
TH227	200	100	> 2.0	0.2	0.1	> 2.0
TH228	6	5	> 1.2	0.008	0.005	> 1.6
TL204	300			30	300	< 10.0
U232	30	4	> 7.5	0.03	0.004	> 7.5
U233	100	10	> 10.0	0.1	0.01	> 10.0
U234	100	10	> 10.0	0.1	0.01	> 10.0
U235	100	30	> 3.3	0.2	0.03	> 6.7
U236	200	10	> 20.0	0.2	0.01	> 20.0
ZR 93	1000			200	3	> 66.7

TRIP - A COMPUTER PROGRAM USING DETAILED ROUTE MODELLING TO ASSESS COMPARATIVE RISKS FROM OVERLAND TRANSPORT OPERATIONS

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1. INTRODUCTION

The computer code TRIP is being developed as part of a programme of work aimed at making an objective estimate of the risks to the public arising from the overland transportation of hazardous cargoes. In this context risk is taken to depend jointly on the expected frequency of occurrence of various types of accidental releases leading to a public health hazard together with the magnitude of the corresponding consequences. The programme considers both short-range hazards, for which damaging effects are confined to within a small distance of the transporting vehicle and are independent of the prevailing wind direction, and dispersive hazards for which details of dilution via atmospheric turbulence and wind transport of the released toxic cloud are decisive in determining the magnitude of the consequences.

For any route between two points there are in practice a host of parameters determining both the accident rate and the magnitude of consequences. For example, in road transportation the expected accident frequency (per mile travelled) will vary along the route depending on road quality, maximum speed, the surrounding population density etc. It may also vary with time of day of the journey (via traffic density) and with the month of the year (via weather conditions). Similarly the magnitude of the consequences will depend on the damage suffered by the cargo package in the accident (which will again be a function of road and vehicle parameters), on the weather condition and on the population distribution relative to the route. A schematic of various components of the calculation is shown in figure 1.

In general, the quantification of risk requires detailed accident frequency data for the roads, vehicles and weather conditions of interest, and also population data It also requires physical modelling for the various phases of the consequence calculation. Currently the project and the computer programme TRIP are at an intermediate stage: the calculation of consequences for certain types of hazard is well advanced whereas the accident frequency and damage phases of the programme are less developed.

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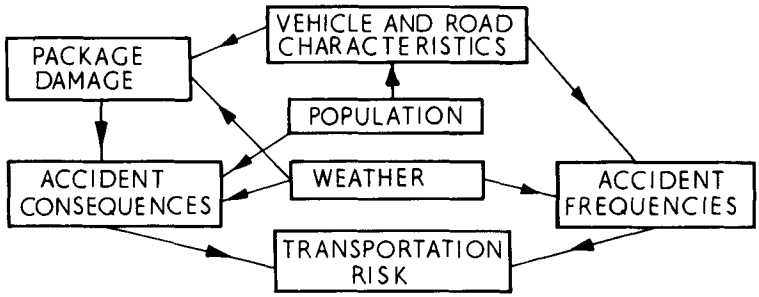


Fig. 1 Schematic of transportation risk assessment

2. CALCULATION OF CONSEQUENCES

2.1 Procedure

Consider an accident leading to a dispersive-type hazard - for example a given release of a neutrally-buoyant toxic cloud at an arbitrary point $\underline{r}_a$ on a route. Treating the ground-level surface in planar geometry, the total number of people suffering a particular consequence (called the number of consequences, N, below) in weather category w can be written

$$N(w,\underline{r}_a) = \int dS_p \; J(\underline{r}_p)\; C(\underline{r},w)$$

where $\underline{r}_p$ is an arbitrary point on the plane,

$J(\underline{r}_p)\, dS_p$ the population contained in surface element dS_p, $\underline{r} = \underline{r}_p - \underline{r}_a$ and $C(\underline{r},w)$ is the individual probability per given release of suffering the particular consequence at $\underline{r}_p$ in weather w. For the purposes of dispersion calculations the weather is specified in terms of atmospheric stability (s), windspeed (u) and wind direction (θ_w).

$$\therefore\; N(w,\underline{r}_a) = \int dS_p \; J(\underline{r}_p)\; C(s,\; u,\; \underline{r},\; \theta_w)$$

The surface integral is handled by first introducing a polar system of co-ordinates (r, θ) based on the accident point.

$$\therefore\; N(w,\; \underline{r}_a) = \iint r\; dr\; d\theta\;\; J(\underline{r}_a,\; r,\; \theta\;)\; C(s,\; u,\; r,\; \theta - \theta_w)$$

Then the integral is evaluated numerically by discretizing r into a series of annuli with edge radii r_i and θ into a number of equal sectors labelled by j

$$\therefore N(w,\; \underline{r}_a) = \sum_i \sum_j J_{ij}(\underline{r}_a)\; C_{ij}(s,\; u,\; \theta_w)$$

where J_{ij} is the total population within the segment defined by the ith annulus and jth sector and $C_{ij}(s, u, \theta_w)$ is averaged over the ijth segment.

2.2 Mesh Sizes

An adequate radial averaging procedure is simply to take the value of C at the mid-radius of the annulus. Since the radial dependence of C is known for weather conditions of interest, the radial mesh can be chosen such that the maximum potential error due to the discretization is less than a chosen limit irrespective of population distribution. An appropriate angular mesh size is determined jointly by the rate at which the population density varies with angle at given radius and the rate at which $C(s, u, r, \theta - \theta_w)$ varies with θ . Often the variation of C with θ is approximately Gaussian in shape (centred on θ_w) with a width small compared to the distance over which the population density varies appreciably. In this case the angular mesh size is chosen such that the population density can be considered constant across it. A significant contribution to the sum then arises from only one sector, the one including θ_w (j')

$$\text{ie } C_{ij}(s, u, \theta_w) \longrightarrow C_i(s, u, \theta_w)\, \delta_{jj'}$$

$$\therefore N(w, \underline{r}_a) = \sum_i J_{ij'}(\underline{r}_a)\, C_i(s, u, \theta_w)$$

The programme TRIP has 2 options: for certain simple types of consequences (eg where the individual's chance of suffering the consequence is simply a function of the ground-level concentration of a non-depositing toxic cloud) the calculation of the C_i can be performed by TRIP internally. For more complex consequences based on dispersive hazards, eg latent cancers due to a ^{137}Cs release - involving irradiation via inhalation, cloudshine and groundshine - the C_i can be calculated externally using the SRD code TIRION4[1] and simply input to TRIP. It should be noted that the $C_i(s, u, \theta_w)$ for a particular release are independent of $\underline{r}_a$ for fixed (r, θ) mesh size so only need to be calculated once for a particular consequence. Thus TRIP's major work in calculating consequences is the calculation of the $J_{ij'}(\underline{r}_a)$ for each relevant point along the route.

3. FREQUENCY - CONSEQUENCE PLOTS

3.1 Procedure

To obtain a measure of the risk involved in transporting the cargo, the expected frequencies associated with various numbers of consequences, N, are required. As outlined above, N varies with position along the route (denoted by parameter, L, say) and weather condition, so the association of frequency with number is achieved in the following manner: the full range of N is split into a discrete set of sub-ranges (bins) denoted by integer k. The aim is then to subdivide the total parameter space spanned by L, s, u and θ_w into sub-regions each yielding N within a particular bin. If $F(L, s, u, \theta_w)$ is known - where $F(L, s, u, \theta_w)$ dL ds du $d\theta_w$ is the expected frequency (per trip) of having an accident at a position along the route between L and L+dL, with weather defined by stability between s and s+ds, windspeed between u and u+du, wind direction between θ_w and $\theta_w + d\theta_w$ - then the frequency associated with each bin can be calculated, F_k. (The above treatment assumes, for simplicity, that package damage is determined uniquely by L and the weather.)

In practice this procedure is implemented by splitting the full ranges of L, s, u and θ_w each into a discrete set of sub-ranges labelled by l, m, n, j. The number of consequences is assumed to be constant over the subspace denoted by (l, m, n, j) and calculated at a representative point within it. Suppose the number falls within the kth bin; the contribution F_{lmnj} (the integral of F over the subspace) is allocated to bin k. The accuracy of this prescription depends

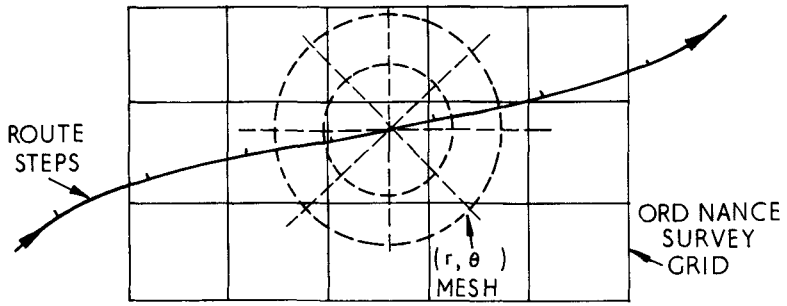


Fig. 2 Discretization meshes for risk calculation

on choosing a fine-enough discretization of L, s, u and θ_w that the number of consequences from all parts of the subspace do not in fact span more than one N bin in the main. The resolution of the overall procedure is determined by the size chosen for the bins of N.

3.2 Mesh Sizes

In practice the discretization of stability is carried out by introducing the 6 Pasquill categories [2]. The Gaussian plume model is used to relate the dispersion characteristics of neutrally-buoyant plumes to stability category with parameters obtained by Smith [3]: a convenient parametrization of the plume standard deviations has been given by Hosker [4]. It is usually sufficient to consider only a few - often just one - characteristic windspeeds for each category.

Concerning the θ_w discretization, if the widths of the dispersion Gaussians are small compared to the widths of the θ bins (as described in Section 2) these same bins can be used for the discretization of θ_w since N will be virtually the same for the wind blowing at any angle within a θ bin.

For the parameter L it is difficult to determine in advance the ranges over which N (for any weather condition) will span a single bin: the optimum step length may not be uniform along the route. In general step lengths must be decreased until adequate convergence of the results is attained. A lower limit the angle and route-step mesh sizes is set by the spatial resolution of the population data, currently 1 km, but this may be too small to be of practical use.

The various meshes are shown schematically in figure 2.

4. DATA

Production of numerical estimates of risk via the above procedure requires population data covering the region of interest and accident frequency data broken down according to factors such as route position, weather and package damage. Ideally the frequency data should be available with a spatial resolution determined by the considerations of Sections 2 and 3. If it is only available at coarser resolution, this will introduce uncertainties into the frequency-number distributions or, alternatively, force the resolution of the number bins to be coarsened.

Currently the modelling assumes no explicit correlation between accident rate and weather condition: the weather data is obtained from UK Met. Office statistics. It also assumes that the frequency distribution of weather conditions is invariant along the route. These are not intrinsic limitations of TRIP but rather represent the current state of the programme and the sparsity of readily-available data. The second of the above approximations is unlikely to lead to major error within the accuracy of the rest of the modelling, at least for not-too-lengthy routes.

The population information was derived from data compiled by the UK Office of Population Censuses and Surveys. Data was extracted from the master magnetic tape in the form of total number of persons within 1 km x 1 km squares referenced using the Ordnance Survey grid and was stored in direct-access form on magnetic disc. Similarly, for efficiency, summary statistics for 10 km x 10 km squares were also stored on disc.

5. POPULATION - HANDLING ROUTINES

As outlined in Sections 2 and 3 a major component in the calculation of consequences for any point along the route is the transformation of the population data from the rectangular grid to the (r, θ) mesh based on each accident point. From figure 2 it is clear that a variety of complex shapes are created by the intersection of the two meshes and careful attention has been paid to creating efficient computer code for handling the geometrical aspects. The population in parts of a square is apportioned pro-rata with the area. Clearly these sub-routines have a practical value beyond their immediate use in TRIP and have already found application in reactor-siting calculations.

6. ILLUSTRATIVE EXAMPLE

6.1 Problem Background

To demonstrate the current capability of TRIP and indicate its future potential, a simplified example will be presented. It concerns the difference in risk arising from transporting by road a particular hazardous cargo between the same end points via two different routes. In general risk differences arise via differences in accident frequency and differences in numbers of consequences for representative points along the route. The present idealized example considers the difference arising solely from the variations in the numbers of consequences as a result of different population-route relationships (in conjunction with differences in wind-direction frequencies). This is not necessarily just an academic exercise since the public often conceive the comparative risks of transporting a hazardous cargo via different routes solely in terms of how close the routes pass to major cities. It should be borne in mind, however, that

this contribution to the risk difference is only one of many, and may in some situations be a minor one compared to the difference in expected accident frequency.

6.2 Nature of Hazard

Clearly these risk differences will depend intimately on the manner in which the individual probability (given an accident) of suffering the consequence varies with distance from the point of release. Thus, for short-range hazards risk differences will arise entirely from differences in population densities close to the route but for long-range hazards this is not generally true.

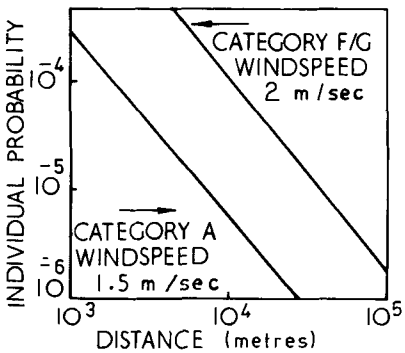


Fig. 3 Individual probability of death as a function of distance

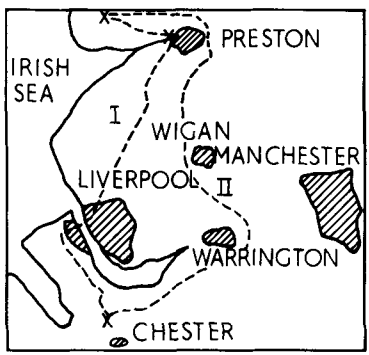


Fig. 4 The two routes used in the illustrative example

In this example two very different types of hazard are considered: first a notional short-range one which kills all people within a 200 m radius of the accident point (for every accident) and, secondly, the latent cancer deaths arising from a notional release of ^{137}Cs, taken to be a fixed 1000 Ci (emitted at constant rate over 3 hrs) for any accident. For the latter hazard the individual probabilities (given as accident) of contracting a fatal cancer as a function of distance for each of 7 weather categories were calculated using the TIRION programme, assuming a dry deposition velocity of 3 mm/sec and adding a 50-year dose from ground contamination to the dose from inhalation and cloudshine. No-threshold linear dose-risk relationships were used for latent-cancer induction in a variety of organs [4].

Figure 3 shows the individual probabilities as a function of distance for the two extreme weather stabilities (A and F+G). The values are averages over th angular bins of the (r, θ) mesh (30^o bins were used) assuming the wind is blowi along the mid-line of the sector. The radial mesh was chosen so as to limit th error in number of consequences in class D weather to < 20%.

A well known complication in the calculation of latent cancers arises from the fact that the population x individual probability product increases with increasing distance initially (for a uniform population density) until the cloud is eventually depleted by deposition. Thus large contributions to the number of latent cancers can arise from extremely small individual doses. In addition, the

contributions may not begin to decrease until much beyond 100 km, the distance normally taken as the limit of the Pasquill-Smith model. In the present calculation an arbitrary cut-off has been applied such that contributions arising from an individual probability of $<10^{-6}$ per accident (about 0.04% of the UK all-ages average individual risk per annum of contracting a fatal neoplasm) have been ignored. For a 1000 Ci release this cut-off is reached at <120 km for all weather categories. The TRIP programme organization facilitates investigation of the influence of cut-off on results.

6.3 Routes and Weather

The chosen routes connect two arbitrary points in the north west of England, one north of Chester and one north west of Preston (see figure 4). These points were chosen for illustrative purposes only since it is possible to connect them by 2 reasonable routes having quite different characteristics. There is no implication

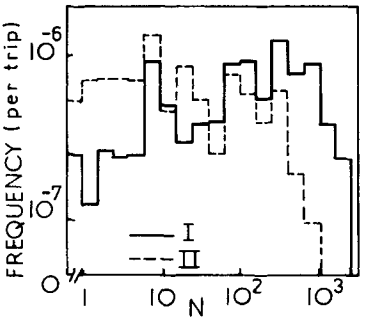
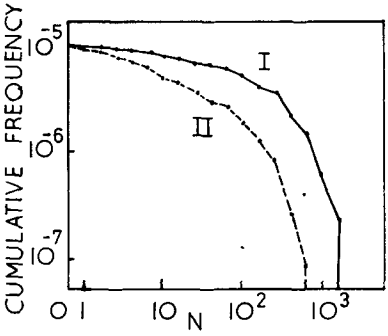


Fig. 5(a) Differential frequency -N histogram for short-range hazard



(b) Cumulative frequency -N line

that hazardous cargoes of the type considered here are transported between these locations, nor that the routes considered would satisfy the numerous criteria applicable to transportation of such cargoes. The first route (I) passes through major centres of population including Liverpool and Preston and lies fairly close to the coast. The other (II) uses the Motorway network and bypasses major population centres (passing equidistant from Manchester and Liverpool); it lies generally further inland than the first.

Weather stations lying close to the first route are Liverpool airport and Blackpool airport; data for the former were used but sensitivity to a change in weather was checked by substituting Blackpool data. For the second route, data from Manchester airport was taken as representative. Journeys were assumed to take place with equal likelihood in all months of the year and at all times of the day so the appropriate average weather frequencies were utilized.

For the sake of this exercise the expected accident frequency per unit route length was taken to be a constant along each route; furthermore a normalization by length rendered the total frequency the same (10^{-5}/trip) for the 2 routes thus highlighting risk differences arising solely from number-of-consequences differences.

6.4 Results

The histogram in figure 5(a) shows the frequency with which the number of consequences, N, lies within each bin for the short-range hazard; a 1 km route step was used. Since route I passes through major centres of population, the distribution spreads to greater N; nevertheless, route II does not have lower frequency for all N since the sums are normalized (ie it does not have zeroth degree stochastic dominance in the terminology of ref 6). However, one naturally gives greater weight to the high-N end of the distribution: intuitively route II is safer. Formally, $\bar{N}$ may be used as a risk-index - albeit not a very sensitive one - on which to base a comparison when the total frequencies are the same. More generally the weighting of higher N can be taken into account by comparing the cumulative f-N lines (frequency of having N or more) - figure 5(b). Since the route II line is never higher than that of route I, the former route is to be preferred (it has first order stochastic dominance). More sophisticated risk indices are described in ref 6.

For the ^{137}Cs case, consequences in population zones far from the accident point are important. Fig 6 shows the contributions from each radial annulus in sector 3 (60^{o} - 90^{o} clockwise from north) for a release point near Liverpool in F/G stability weather; the Greater Manchester conurbation gives contributions

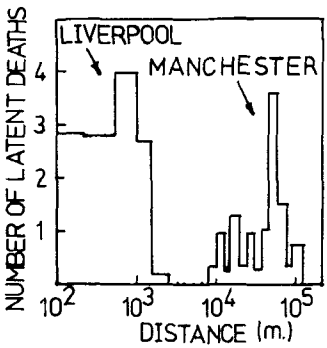


Fig. 6 Number of latent deaths from each annulus for sector 3, F/G stability, for a release near Liverpool

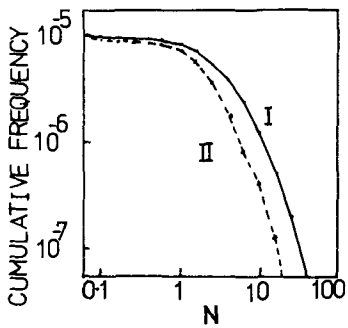


Fig. 7 Cumulative f-N lines for dispersive hazard (5 km route step length)

comparable to those from Liverpool itself. Fig 7 shows the f-N lines for the two routes: they are much closer together than those of fig 5(b) due to the longer range of the hazard. The route I line is still above that of route II but in this case as a result (in part) of its being consistently further from the Manchester population concentration. Removing the cut-off of 10^{-6} on individual probability causes a maximum of 20% difference in the f-N line. Using Blackpool weather statistics for route I gives up to a factor of 2 difference in frequency at the high -N end compared to results using Liverpool weather.

6.5 Comments

The above example has illustrated the influence on risk of the relative positions of route and population centres, showing how the spatial range of the hazard influences results. Results of the above type, however, do not necessarily imply

that one route should definitely be used rather than another. Both may be adequately safe by wide margins when judged against agreed safety standards. In this case other factors such as economics and convenience should be brought into the comparison.

It has to be stressed that the example has only dealt with one component in the variation of risk from one route to another, namely that due to number-of-consequences variations. In reality this may be a small component compared to the variation in accident frequencies. Thus the ^{137}Cs results do not imply that latent cancer risks are little dependent on whether or not a route passes through major centres of population but indicate that the influence of this factor on consequence variations is small, thereby directing attention towards its influence on accident frequencies.

7. CONCLUSIONS

Although the programme TRIP is at an intermediate stage of development, it has already demonstrated useful capabilities which indicate that it could prove a valuable tool in the assessment of risk from ground transportation of hazardous cargoes. However, to achieve the project aim of making realistic risk assessments and comparisons will require detailed accident frequency data, broken down according to factors such as route position and weather. Such data are not readily available at present, and the level of detail which can reasonably be attained may prove to be the limiting factor in the quantification of risk.

8. ACKNOWLEDGEMENTS

The authors are grateful to colleagues at SRD for advice and help, and particularly wish to thank Sue Kennett for running the computer programme.

REFERENCES

1. L S Fryer and G D Kaiser, 'TIRION 4 - A Computer Program for use in Nuclear Safety Studies', UKAEA Report SRD R134 (1979).

2. F Pasquill, Met. Mag. 90 (1961) 33.

3. F B Smith, 'A Scheme for Estimating the Vertical Dispersion of a Plume from a Source near Ground Level' Ch. XVII, Proc. of the Third Meeting of the Expert Panel on Air Pollution Modelling, N.A.T.O., Paris, France (1972).

4. R P Hosker, Jr, "Estimates of Dry Deposition and Plume Depletion Over Forests and Grassland" in 'Physical Behaviour of Radioactive Contaminants in the Atmosphere', IAEA/STI/PUB/354 (1974) p 291.

5. H Smith and J W Stather, "Human Exposure to Radiation following the Release of Radioactivity from a Reactor Accident: A Quantitative Assessment of the Biological Consequences", National Radiological Protection Board, NRPB R-52 (1976).

6. H A Munera and G Yadigaroglu, Nuclear Science and Engineering 75 p 221 (1980).

EXPERIMENTAL AND CALCULATIONAL ANALYSIS OF RADIATION DOSE IN A SPENT FUEL SHIPPING CASK

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To evaluate the accuracy of computational methods used for calculation of an external radiation dose of a spent fuel shipping cask, comparisons have been made between biological dose-rate calculations and measurements at the surface of a cask carrying five pressurized water reactor fuel assemblies. The radiation transport calculation was made by a two-dimensional transport code PALLAS-2DCY with the source intensities calculated by ORIGEN code, and the final neutron dose was determined by taking into account an effective multiplication factor k_{eff} estimated by CITATION code.

A fairly good agreement is obtained for both neutron and photon dose rates outside the cask except for photon dose rates at the position of the cooling fins, in which the calculation is about 2.2 times the measurement because of neglecting these fins in the calculation. It is also confirmed that the calculation can adequately estimate a localized peak of gamma-ray dose in the vicinity to the cask bottom, and that the shielding effect of the fins is found experimentally to be a factor 2 for gamma-ray dose attenuation.

INTRODUCTION

For the purpose of an evaluation of radiation dose calculation in a spent fuel shipping cask, an experiment was made for measuring gamma-ray and neutron dose rates around a shipping cask. As a result of comparisons made between dose-rate calculations and measurements at the surface of the cask carrying five pressurized water reactor (PWR) spent-fuel assemblies, the accuracy of computational methods used for analyzing the experiment was verified as adequate for evaluation of the external radiation dose leaked from

the cask.

The shipping cask used for this study is an EXCELLOX type cask shown in Fig. 1. A cross section of the cask containing five PWR fuel assemblies is depicted in Fig. 2. The detailed specification for the spent-fuel assemblies is given in Table 1.

For the analysis of the dose-rate measurement a series of calculations were made: the calculation of source intensities for neutron and photon was performed by ORIGEN code[1], the estimation of effective multiplication factor k_{eff} was carried out using CITATION[2], and radiation transport calculation was made by two-dimensional transport code PALLAS-2DCY[3].

EXPERIMENT

The experiment consists of two series of radiation dose-rate measurements for neutron and photon performed around the cask shown in Fig. 1. The detectors used in the measurements are listed in Table 2. In addition to dose-rate measurements, an energy spectrum measurement was also made for photon using a scintillation counter with NaI (Tℓ) scintillator at two positions, 1 meter distant from both cylindrical and bottom surface of the cask. The dose conversion from detector response was made by considering of the energy dependent detector efficiency and the calculated and/or measured spectrum of radiation.

The experimental investigation into the shielding effect of a series of circumferential cooling fins was made by using TLD's, in which they were set on directly on the cylindrical surface and at the position of these fins (14 cm distant from the cask surface). As a result it has been found that the presence of these fins reduces the gamma-ray dose rate by a factor 0.5. A rather large difference has been found between the values of gamma-ray dose rate measured with the TLD and ionization chamber, that is TLD/Ion-chamber=1.45 at the side surface and 1.30 at the bottom surface. This may be due to the difference of the size and measuring position of the two detectors. In both dose-rate measurements for neutron and photon, a significant azimuthal variation was observed in the circumferential dose-rate distribution on the cask side surface. This is attributed partly to an asymmetrical source intensity distribution with respect to the cask axis and partly to the ullage space between the cask steel wall and the lead liner for the horizontal cask. In addition, a peak gamma-ray dose rate was also measured in vicinity to the cask bottom, but the leakage photons were collimated and the dose-rate enhancement was of very limited lateral extent.

CALCULATION

For the analysis of this experiment, the ORIGEN code was used for estimating the source intensities for neutron and photon based on the parameters of the enrichment, burn-up history and cooling period for the fuel elements described in Table 1. As a result the gamma-ray and neutron source intensities were determined as given in Tables 3 and 4. Both values of these gamma-ray and neutron sources were used for those of PALLAS transport calculation.

For the evaluation of the effective multiplication factor k_{eff}, LASER diffusion code[4] was used for estimating nuclear densities within the spent fuels, which were used for calculation of the lattice constants in the fuel region by THERMOS[5] and GGC-4[6] diffusion codes. The effective multiplication factor was estimated to be 0.70 by CITATION diffusion code using the calculated lattice constants for the fuel loading arrangement.

The radiation transport calculation was made by two-dimensional direct integration code PALLAS-2DCY. The cask was approximated by (r, z) geometry, in which the spatial distribution of the source intensity was assumed to be symmetry with respect to the z axis, though the actual source intensity was significantly asymmetrical. Thirty-nine energy meshes were used for neutron transport calculation, while 16 energy meshes were used in the energy range of 3.5 ∿ 0.08 MeV for photon transport calculation. The source density was homogenized over the whole fuel region, and the cooling fins were disregard in the PALLAS calculation. The nuclear data used for the calculation were taken from ENDF/B-IV data library for neutron and from Hubbell's compilation for photon. The neutron dose rates outside the cask calculated by PALLAS were multiplied by a factor 3.3 determined from the k_{eff}=0.7.

For a more accurate evaluation of the neutron leakage from the cask surface it was necessary to treat the presence of the ullage space for the horizontal cask. Then two models were adopted: the first mode had the ullage space represented by an annular void for evaluating the neutron leakage from the upper surface of the cask, and the other had a water layer concentrically within the steel shell for evaluating it from the side and lower surfaces. On the other hand, for evaluation of the gamma-ray leakage from the cask surface, only the first model was applied because of negligible effect of the presence of the 2.6 cm thick water layer on the external gamma-ray dose.

COMPARISON

Comparisons of calculations with experiments were made for both the neutron and photon dose rates on the side surface and also on the bottom surface of the cask. In addition, comparison was also performed for the azimuthal variation of the circumferential dose-rate distribution at mid length of the cask. A further comparison was made for energy spectrum of gamma rays at 1 m from the cask surface.

Figure 3 illustrates the comparison of PALLAS calculation with measurement on gamma-ray dose rate distribution along the cask side surface, which results in a fairly good agreement even in the localized peak observed in vicinity to its bottom side. The PALLAS calculation, however, overestimates the experiment by a factor of 2.2 at the fin position as depicted in Fig. 4, which is due to the neglection of the fins with 14 cm long shown in Fig. 1. The dose rate calculated by PALLAS in the air at the fin position is almost the same as that on the cask surface as seen in Figs. 3 and 4. A fairly good agreement is also shown in Fig. 4 between calculation and experiment on gamma-ray dose rate distribution on the bottom surface. The measured values plotted in Figs. 3 and 4 for

comparison were averaged over the circumferential dose-rate distribution illustrated with the solid line in Fig. 5, in which the PALLAS calculation is drawn symmetrically with the dotted line.

On the other hand, for neutron dose rate distribution, agreement between calculation and experiment is fairly good at both external positions on the cask side and bottom surface except for the regions near the top and bottom sides as shown in Fig. 6 (A). This rather large difference can be attributed to the assumption of the constant source intensity distribution along the fuel elements adopted in the calculation. Another comparison is given in Fig. 7, in which the experiment represented by the dotted line shows a significant azimuthal variation, while the calculations are drawn symmetrically with the dot-dash line for the first model and with the solid line for the other model. The calculation with the dot-dash line is compared with the experiment above the cask, which results in agreement within about 25 %. The other calculation is also in fairly good agreement with the experiment at the side and the lower surface of the cask.

The result of comparison of the calculated with the measured gamma-ray energy spectra indicated that the calculation overestimated the experiment by a factor of 3 $\sim$ 7 in the energy range of 0.2 $\sim$ 2.6 MeV at 1 m from the fin position. This overestimation is too large to be attributed to the neglection of the fins. In contrast, a fairly good agreement is obtained between them at 1 m from the bottom surface.

CONCLUSION

It has been found that the experiment presented in this paper is very useful as a standard for evaluation of calculation of the external radiation dose leaked from a cask. Further it has been confirmed by making comparison with experiment that the computational methods consisted of a series calculations for estimation of the total source intensity and of the effective multiplication factor and for radiation transport are adequate for calculation of the external radiation dose around the spent fuel shipping cask. As a whole it is found that the calculation agrees with the experiment within a factor of two.

(REFERENCES)

1) M. J. Bell, ORNL-4628 (1973)
2) T. D. Fowler, et al., ORNL-TM-2496 (1971)
3) K. Takeuchi and N. Sasamoto, JAERI-M 9014 (1980)
4) C. G. Proncelet, WCAP-6073 (1966)
5) B. J. Toppel, et al., ANL-7023 (1965)
6) J. Adir and K. D. Lathrop, JA-9021 (1968)

Table 1 Specification of 5 P.W.R. spent fuel elements

Fuel parameters	3 fuel elements	2 fuel elements
Enrichment	2.28 w/o	3.04 w/o
Burn-up (MWd/MTU)	21328	28644
Cooling period (days)	1220	501
Weight	1.2 MTU	0.8 MTU

Table 2 Detectors used for the experiment

Radiations	Detectors
Gamma ray	Ionization chamber TLD GM survey meter NaI (Tℓ) Scintillation survey meter
Neutron	Neutron Survey Meter (4.2 cm or 7.2 cm thick paraffin covered BF_3 counter) Neutron Rem Counter (Studsvik Type 2202 D)

Table 3 Gamma-ray source calculated by ORIGEN code

Group No.	Emax (MeV)	γ/sec
1	3.5	3.569×10^{10}
2	3.0	1.132×10^{12}
3	2.6	1.472×10^{13}
4	2.2	1.153×10^{14}
5	1.8	3.151×10^{14}
6	1.35	1.006×10^{15}
7	0.9	2.684×10^{16}
8	0.4	3.697×10^{15}

Table 4 Neutron source calculated by ORIGEN code

Sources	n/sec
Spontaneous fission	3.924×10^{8}
(α,n)	3.696×10^{7}

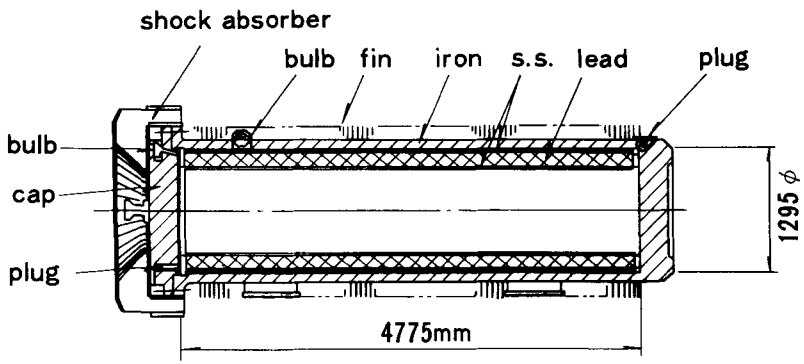


Fig. 1 Spent fuel shipping cask

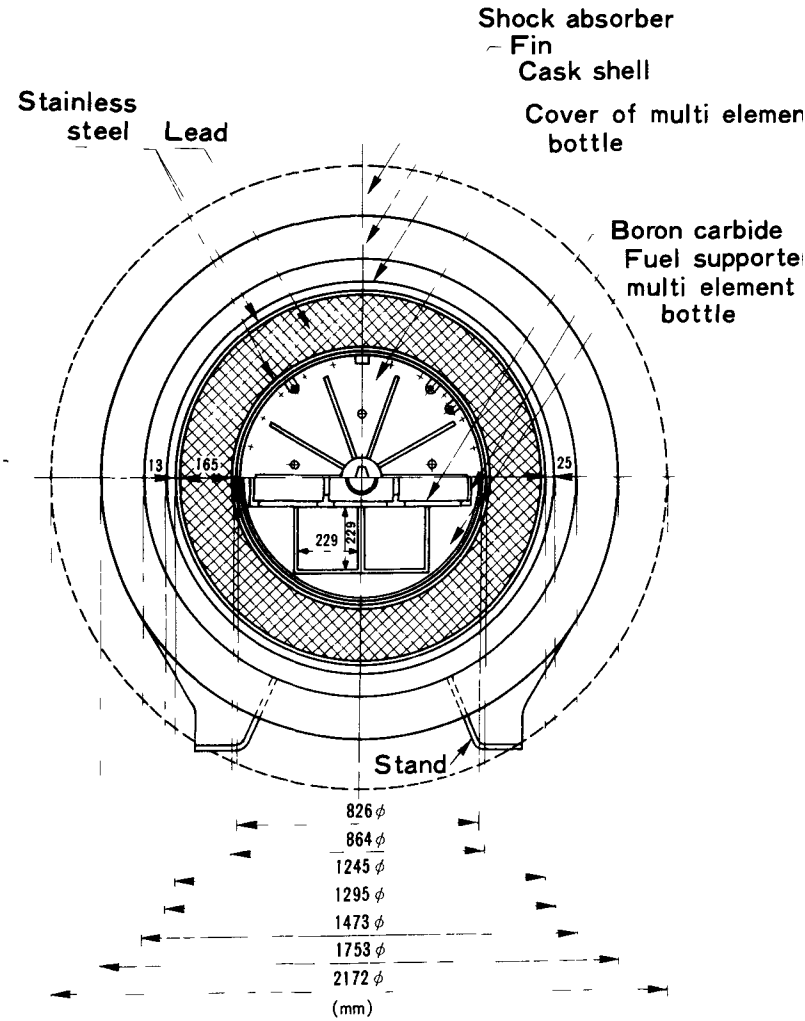


Fig. 2 Cross section of the cask

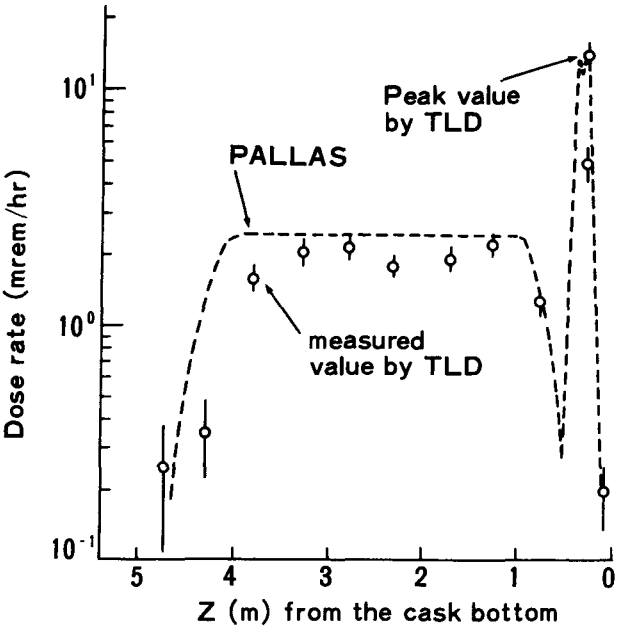


Fig. 3 Comparison of PALLAS calculated with measured gamma-ray dose rate distribution on the cask surface

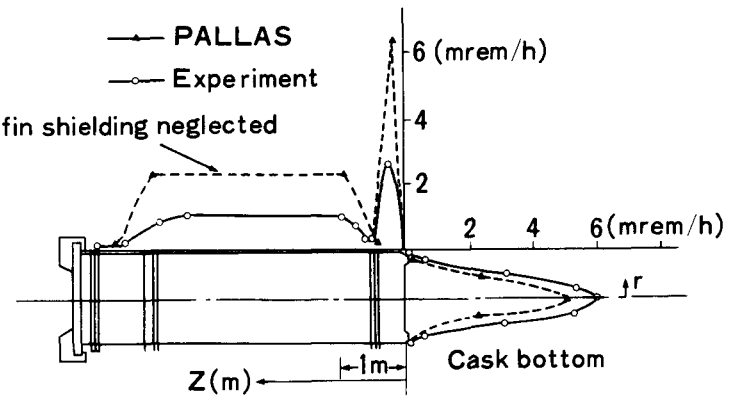
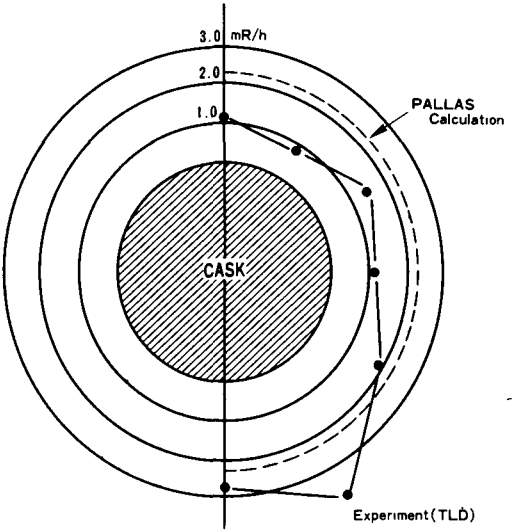


Fig. 4 Comparison of PALLAS calculated with measured gamma-ray dose rate distribution at the fin position of the cask surface and on the bottom surface

Fig. 5 Comparison of PALLAS calculated with experimental gamma dose distribution on the cylindrical surface of the cask (about the center for z distance)



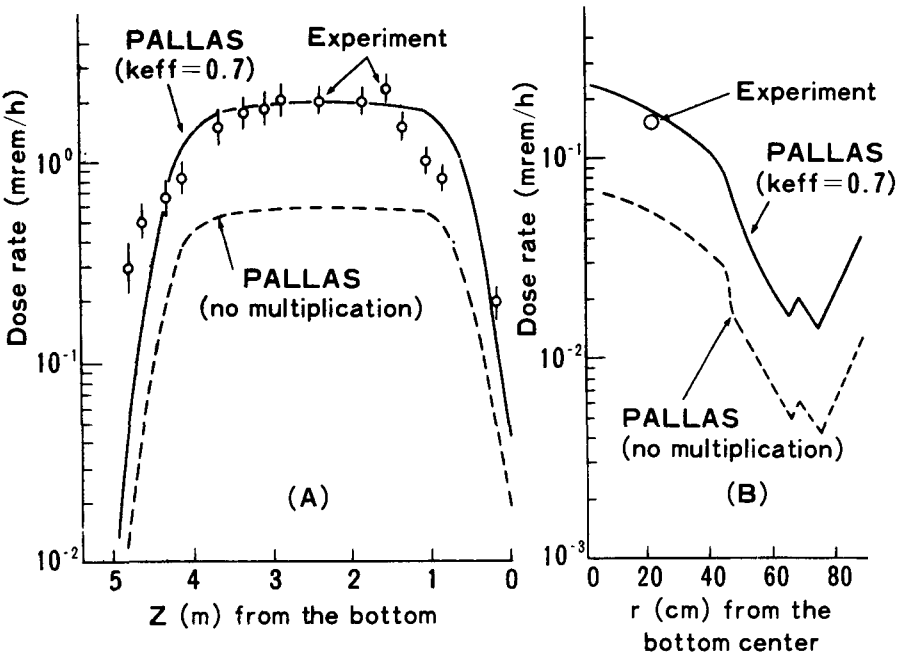


Fig. 6 Comparison of PALLAS calculated with measured neutron dose rate distribution at the fin position of the cask surface (A) and on the bottom surface (B)

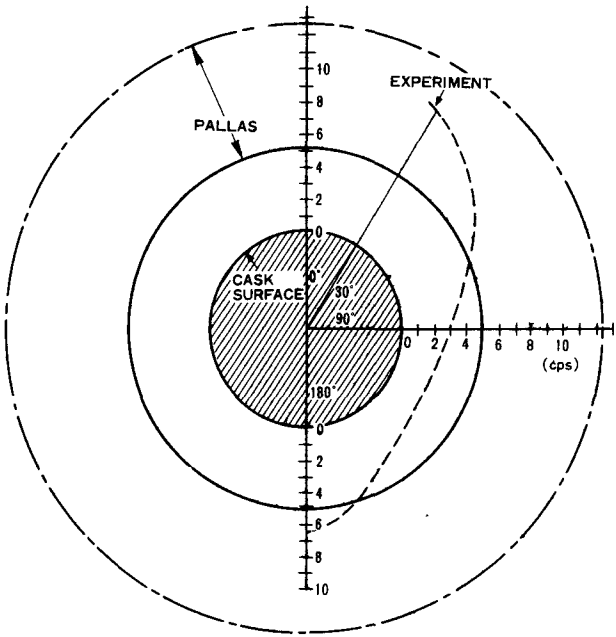


Fig. 7 Comparison of PALLAS calculated with experimental neutron dose distribution at the circumferential position (14 cm from the side surface).

CONFIRMATION TESTS OF PWR SURVEILLANCE CAPSULE SHIPPING CONTAINER (COMPARATIVE STUDY OF ANALYSES AND TESTS BY IAEA REGULATIONS)

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Mitsubishi Heavy Industries, Ltd., carried out the confirmation tests of the PWR surveillance capsule shipping container using a 10-ton fullscale model at our Kobe Shipyard and Engine Works in the early Spring of 1979. This report presents an outline of these tests and analyses.

Mitsubishi Heavy Industries, Ltd., has had a surveillance capsule shipping container as a supplier of the PWR plant in Japan. The Japanese Government has modified the safety standard for transporting radioactive materials in January 1978. Thus we needed to prepare a new container classified in type "B" of the new regulations, which are similar to IAEA regulations.

1. OUTLINE OF TEST

The test model was of the same construction, materials, and size as the actual container and was 1.080 meters in diameter, 3.289 meters long, and weighed about 10 tons, as shown in figure 1. This model consisted of a main body with a thick lead shield, heat insulation and a basket and shock absorbers.

The series of tests consisted of four 9-m drop tests, such as horizontal, vertical, corner and horizontal no 2, two 1-m puncture tests, such as horizontal and vertical, a fire test and an immersion test.

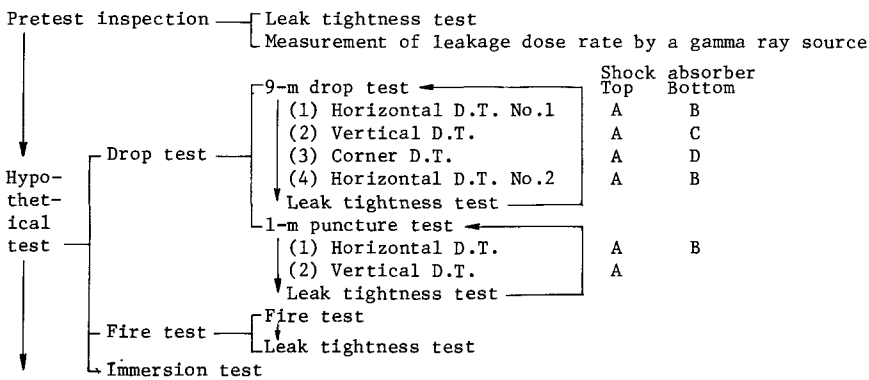


Fig. 2 Flow chart of the confirmation test using the fullscale model

The flow chart of these tests is shown in figure 2. These tests were carried out using one shock absorber "A" for the top and three shock absorbers "B", "C" and "D" for the bottom as shown in figure 2.

The test facilities for the 9-m drop and 1-m puncture test are shown in (a) and (b) of figure 3.

2. DROP TESTS

2.1 Test Facility and Others

The drop test facility consisted of a dropping tower, a test bed, a release mechanism, and a puncture test stand which were built temporarily and a 100 ton movable jib crane aside a shipbuilding birth. The test bed was set up with five 100 ton concrete blocks and a steel plate of 50 mm thickness and 5 m square. This total weight was about 500 tons or 50 times the model weight.

2.2 Instrumentation

The main body's decelerations were measured with piezoelectric accelermeters and resistance strain type accelermeters. Strain gauges, high-speed camera, video camera, and phototubes were also used.

2.3 Outline of the Test Conditions and the Test Results

Photographs of the test facility and test conditions are shown in figures (7), (8) and (9). Test condition of these drop tests are shown in (1) of table 1. The outline of these test results such as maximum deceleration and maximum strain of the main body, maximum deformation of shock absorber, and drop speed before impact, is shown in (2) of table 1.

(1) 9-m horizontal drop test no.1; The model rebounded about 360 mm. The main body's acceleration was 150G.
(2) 9-m vertical drop test; The model rebounded about 200 mm and stood vertically. The maximum deceleration of the model was about 200G.
(3) 9-m corner drop test; The model was suspended at on 18 degree angle from the vertical. The model rebounded slantwise about a height of 0.6 m. Acceleration measured 49G and the deformation of 18 degree angle was occured on the lower shock absorber.
(4) 9-m horizontal test no.2; The dropping condition was about the same as that of the no.1 test and the impact force was measured 140G.
(5) 1-m horizontal puncture test; The dent on the outer shell was about evenly 13 mm in depth. The outer shell was not penetrated through.
(6) 1-m vertical puncture test; The model without the lower shock absorber was dropped in a vertical position. The impact force was 79G.

2.4 Leak Tightness Test

The leaks by the gas pressure drop tests were carried out to confirm the leak tightness after each dropping test and they were proved the tightness of the model.

2.5 Comparison of the Analytical Results with the Test Results

The analyses of the 9-m drop tests and 1-m puncture tests were made and these results were compared with the test results as shown in (3) of table 1.

(1) Analyses of the 9-m drop test; Compression tests of the shock absorber materials were performed to get stress-strain curves. The maximum deformations of the shock absorbers needed to absorb the model's kinetic energy were calculated by the volumetric displacement method. The analytical value were about the same as the test values except the corner drop test.

(2) 1-m puncture test; The empirical equations by puncture test for the cylindrical shell and the flat plate were used for the analyses of the 1-m horizontal and vertical puncture test respectively. The maximum energy absorbed by body without penetration was enough large comparison with the kinetic energy of the model on both case. No penetration was occurred on the body.

3. FIRE TEST

3.1 Fire Test Facility, Procedure and Result

The furnace test no.2 of three fire test method proposed by IAEA was adopted using our furnace 8 m wide, 8 m high, and about 16 m long. This test condition of the furnace and model is shown in (c) of figure 3.

Fire test procedure and result: Heating the furnace began the night before the test day, and the furnace wall temperatures were settled at 940°C after about ten hours of heating. This condition was maintained for about 4.5 hours. When the model was settled inside the furnace using a dolly the furnace temperature dropped to 540°C because the furnace door was opened 6 minutes.

After the furnace door was shut, the furnace was heated again to 800°C. It took 12 minutes. The fire test was carried out at 800°C±10°C continuously for 30 minutes. After this, the model was pulled from the furnace and settled on the steel frame prepared on the floor.

During the test as shown in figure 4, temperatures of 27 points on the three sections (A), (D) and (B) of the model were measured with C-A type thermocouples. The temperatures of the main body center were highest. The temperature-time data not only of the parts of the model but also inside the furnace of this part are shown in the figure. The highest temperature of the outer shell was 645°C and that of the outer cylinder just outside of the lead shield was 161°C. This temperature is low enough not to melt the lead.

The leak test after the fire test showed that the tightness of the model had been kept and there was nothing wrong with the "O" ring gaskets and their vicinity. Two photographs of the fire test are shown in figures 10 and 11.

3.2 A Comparison of Analytical Result with the Test Result

The thermal analyses of the model were performed using the "TRUMP heat conduction computer program". This container was approximated to the symmetrical and this analytical model is shown schematically in figure 6.

Analytical result of several points is shown in a figure 5 and this is very similar to the test result.

4. WATER IMMERSION TEST

The inside of "the O ring gasket" of the model was painted with a coloring agent and the model was compressed under more than 2 kg/cm^2 for eight hours using our 100 kg/cm^2 water pressure test tank of 2 m in diameter. The water immersion test facility is shown in (d) of figure 3. There was nothing wrong with the coloring agent after the test, and there was no detachable leakage of water into the model.

5. RADIOACTIVE DOSE RATE

The radioactive dose rates on the surface of the model and one meter point from the surface were measured by scintillation counter before and after inserting a gamma ray source, about 10 curies of 60 Co into the cavity. The dose rates before and after these series of tests were 0.45 mrem/h at the outer surface of the model and 0.055 mrem/h at 1 meter from the outer surface, and there was no change between before and after each series of tests.

6. CONCLUSION

This series of tests using a fullscale 10-ton model showed that our designed container was confirmed to have good safety even under the hypothetical conditions. The analytical method was confirmed to be adequate practically in comparison with the test results.

Maximum outer diameter	1.080 meters
Total length	3.289 meters
Total weight	9.71 tons

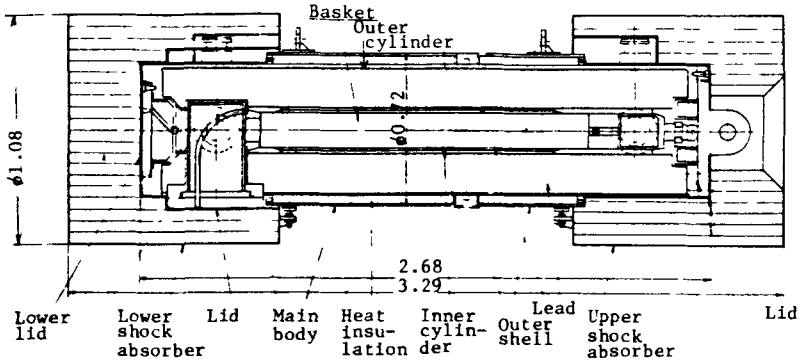


Fig. 1 Cross section of the test model Length: in meter

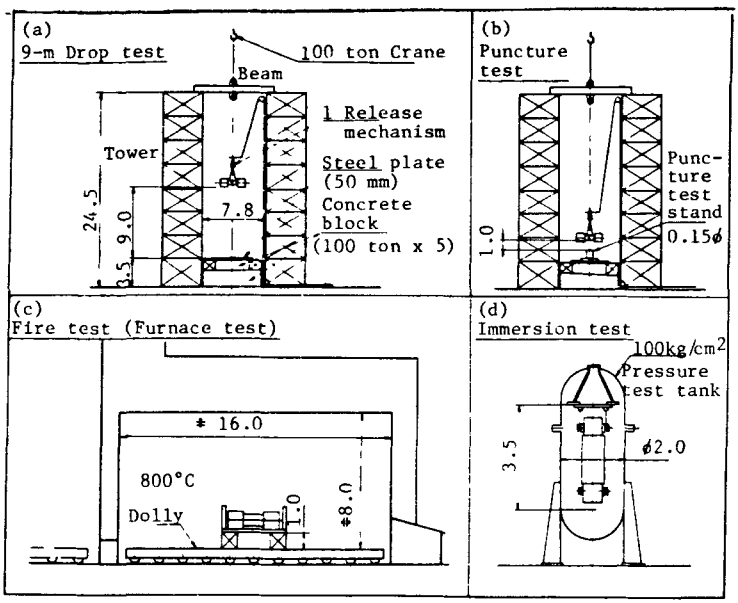
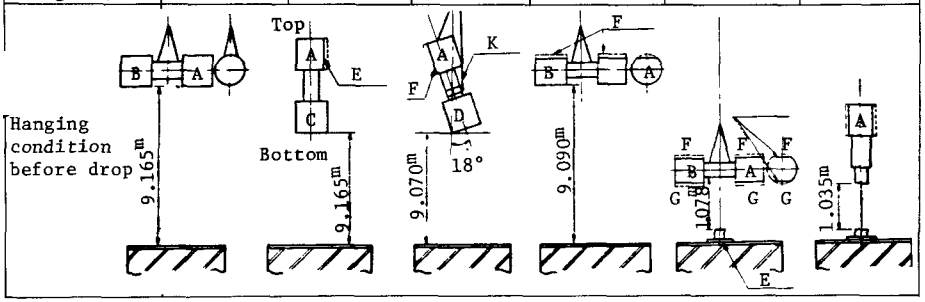


Fig. 3 Hypothetical test facilities Length: in meter

Table 1 Drop test

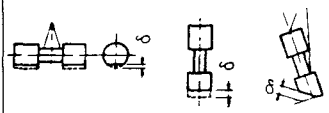
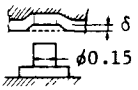
(1) Test conditions

Item	9-m Drop tests				1-m Puncture tests	
Dropping condition	Horizontal drop NO.1	Vertical drop	Corner drop	Horizontal drop NO.2	Horizontal drop	Vertical drop
Drop height(m)	9.165	9.165	9.070	9.090	1.078	1.035
Drop weight (ton)	9.711	9.709	9.839	9.711	9.711	9.170



A : Upper shock absorber
B,C,D: Lower shock absorber
E : Puncture test stand
F : Deformation after horizontal drop test No.1
G: Deformation after horizontal drop test No.2
K: Special band to hang the model

(2) Test results

Main body Max. deceler-ation (g)	150	200	49	140	47	80
Main body Max.strain(μ)	1,470	-780	-285	-1,090	710	-240
Max. defor-mation of shock A (mm)	94	50	236	—	19	13
Drop speed before inpact (m/s)	13.8	14.1	—	13.2		
Deformation of shock absorber						

(3) Test analysis

Main body Max. deceler-ation (g)	208	226	139	208		
Deformation of shock A(mm)	101	50	193	101		

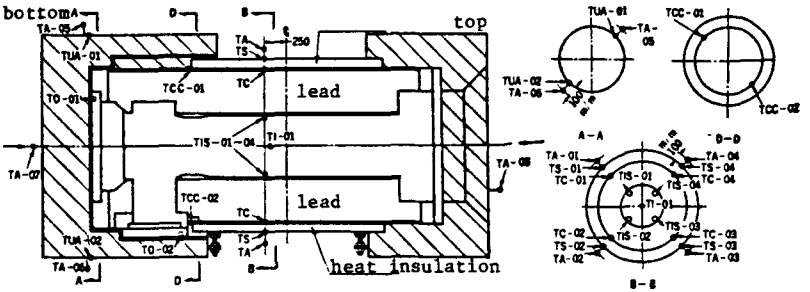


Fig. 4 Temperature measuring points

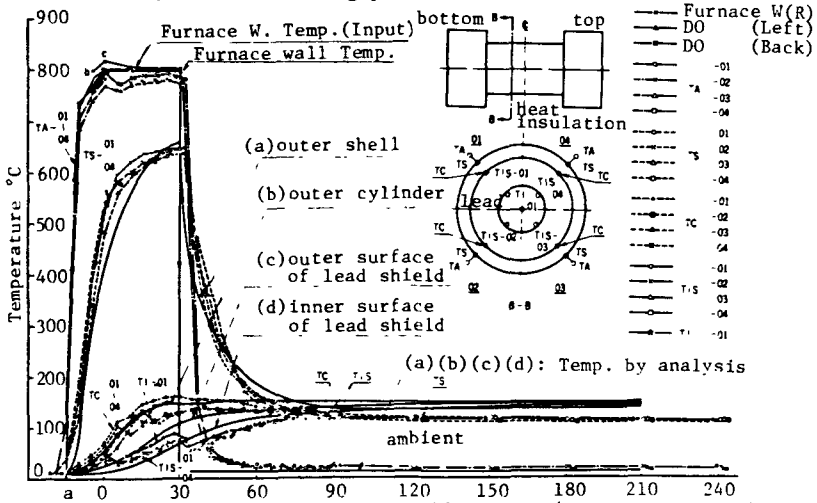


Fig. 5 Main body center Temp.-Time data measured and analyzed at the fire test condition

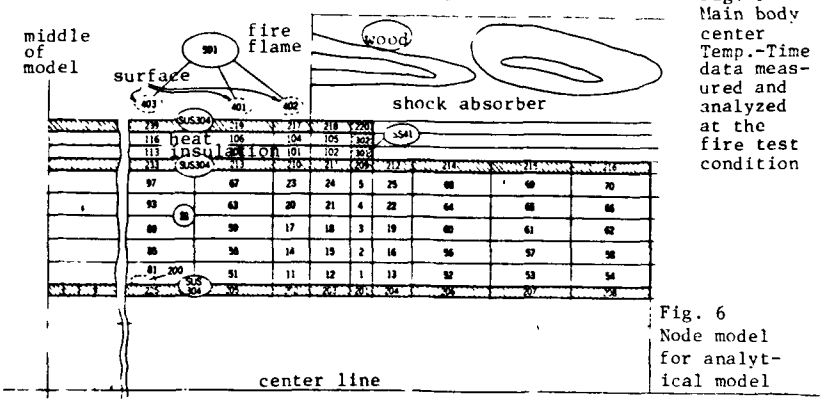


Fig. 6 Node model for analytical model

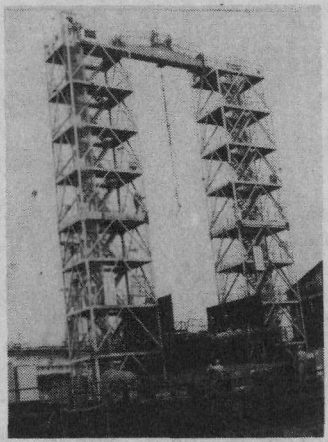


Fig.7 Overview of the drop test facilities

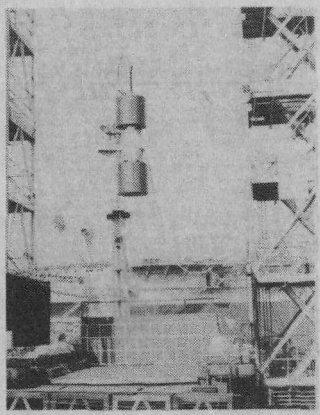


Fig.8 9-m vertical drop test condition before test

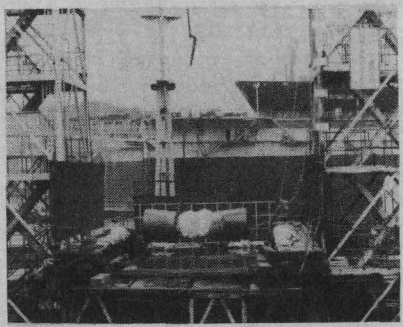


Fig.9 1-m horizontal puncture test condition before test

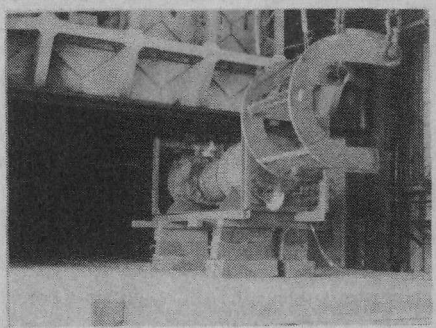


Fig.10 Model condition just after the furnace door was opened after fire test

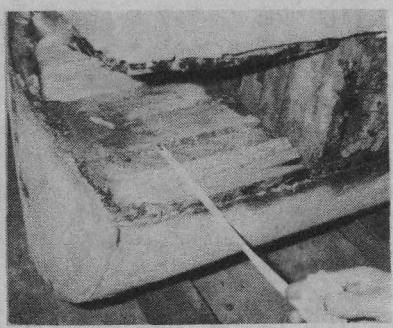


Fig.11 Charred condition of the wood of the shock absorber after the test

RADIATION DOSES FROM AIR TRANSPORT OF RADIOACTIVE MATERIALS IN ITALY

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It can be estimated that roughly 1/3 of the 150,000 radioactive packages transported in Italy in 1979 were shipped by air. Almost all these packages were radiopharmaceuticals and were carried in passenger aircrafts. The aim of this investigation was twofold: to evaluate directly the radiation exposure of passengers and crew members in these aircrafts; to derive a formula to predict the exposure of a generic passenger due to a package wherever located in a cargo compartment.

For this purpose two series of experiments have been planned: a) direct measurements on board of passenger aircrafts in normal service; b) experimental evaluation through the aircraft floor for different radiation energies and incidence angles.

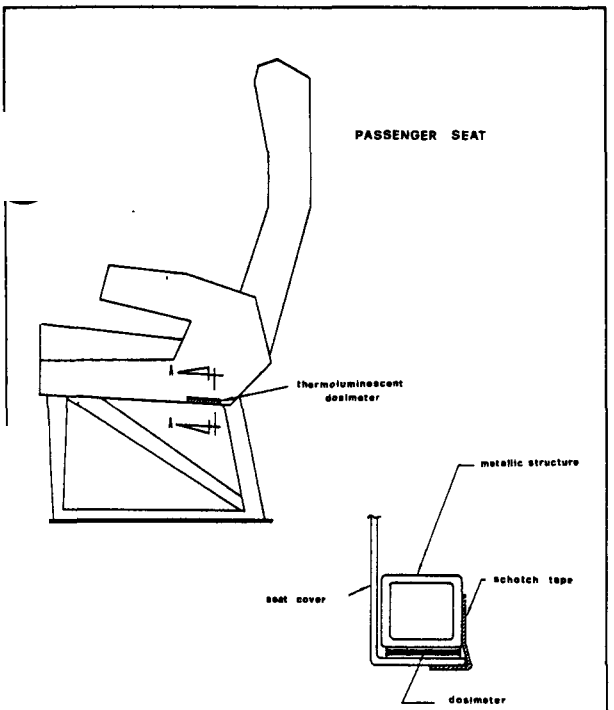
a) Direct measurements

The dosimeters used in this part of the study were CA F_2:Dy (TLD 200) as thermoluminescent sintered chips 1/4" x 1/4" x 0.03". The choice of this material was based upon a number of suitable characteristics such as sensitivity and reproducibility.

CaF_2:Dy is about sixty times more sensitive to X-rays than the ordinary LiF chips so that an absorbed dose of 0.01 mGy is easily detectable with a reproducibility of ± 10% standard deviation.

On the other side CaF_2:Dy is not a tissue-equivalent material: in fact its effective atomic number is 16.3 as compared to 7.4 of soft tissue. The response of the dosimeter is therefore strongly affected by the energy of the radiation used. This difficulty was overcome, within certain

limits,by sandwinching the chips between two lead sheets of 0.4 mm thickness. By this method the energy response of the flattened to within + 20% in the range 0.05 to 1.25 Mev (1).



Measurements were carried out on 4 aircrafts,representing around 10% of the DC-9 Alitalia fleet.

Groups of 4 dosimeters were located,as shown in Fig. 1, on the bottom frame of around 20 passenger and pilot-seats in each aircraft.

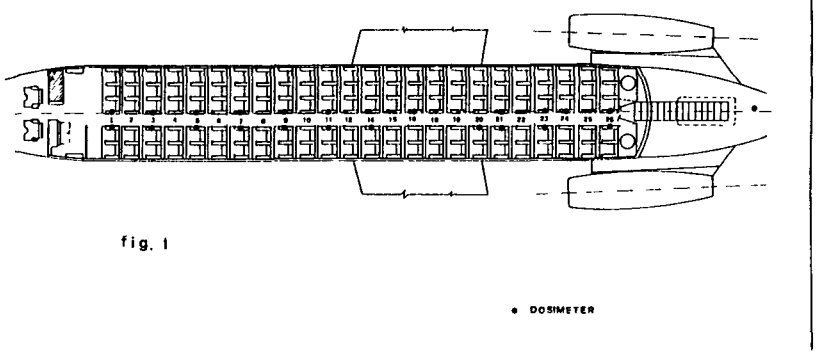


fig. 1

This particular location of the detectors is a good approximation for evaluating the whole body dose received by the persons involved. The dosimeters were left for about 2 months in the aircrafts, their calibration was performed by using a ^{60}Co gamma-ray source and readings by a Harshaw reader model 2000.

The distribution of dose in one of the aircrafts is illustrated in fig. 2.

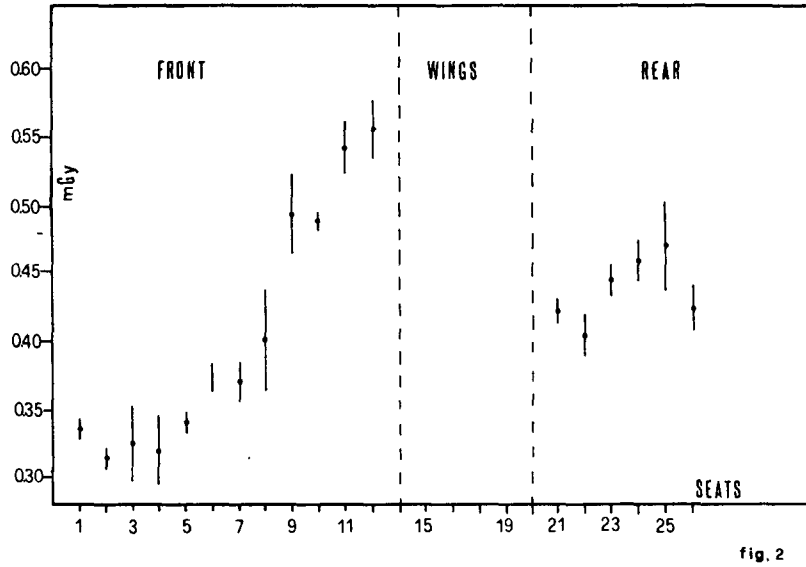


fig. 2

Most of the experimental points are distributed in a regular pattern showing two maxima. The higher one is located in the central part of the aircraft; the lower one in the rear part of the cabin. These maxima correspond to the position of the packages in the two cargo compartments.

Tabel 1 is a summary of all the results in the four aircrafts. In column 2 the average absorbed dose in the passenger cabin for each aircraft is followed by its standard error. Column 3 shows the dose measured on the pilot-seat. Owing to the distance between the pilot seat and the

usual location of the radioactive packages in the cargo compartments the figures of column 3 can be assumed as background doses and will be used to calculate net doses (see column 8). Columns 4 and 5 give the level flight time and cosmic radiation dose rate, respectively. The last one was calculated as the ratio of background dose and level flight time: it would correspond, therefore, to the maximum estimate of the real values. Column 6 shows the total number of flights during the observation period; the flights carrying radioactive packages are reported in column 7. The average net dose in the passenger cabin (column 8) was obtained by subtracting column 3 from column 2.

Table 1 Summary of results

1	2	3	4	5(3:4)	6	7	8(2-3)	9(8:6)	10(8:7)
aircraft	average measured dose in the passenger cabin	dose measured on the pilot seat	level time flight	cosmic radiation dose rate	total number of flights	number of flights with (radioactive) packages	average net dose in the passenger cabin	average dose per flight	average net dose per flight with radioactive packages
	/uGy	/uGy	h	/uGy/h			/uGy	/uGy	/uGy
DIKA	417±15	113±7	153	2.0	520	6	104	0.20	17.3
DIKP	541±24	402±16	129	3.1	364	35	139	0.38	4.0
DIKY	503±10	454±10	173	2.6	317	10	49	0.15	4.9
DIRO	554±26	409±11	149	2.7	263	16	145	0.55	9.1

The data given in the latter column divided by the total number of flights give an estimate of the average dose received by a passenger during a randomly selected flight. Most of the flights considered in this investigation are domestic flights and their mean duration is of the order of one hour. If only the flights carrying radioactive materials are considered the figures are, of course, much

higher as shown in column 10.

It is worthwhile noting that in the present statistics the flights carrying radioactive packages represent on the average about 5% of the total number of flights.

b) Experimental evaluation of the trasmission through the aircraft floor

The experimental set up used to carry out the transmission measurements is shown in fig. 3, the radiations impinged the aircraft floor under different incidence angles (0°, 30°, 60°); heavily filtered x-rays spectra, as reported in table 2, were used as radiation sources.

Table II

KV	filter	Effective Energy (KeV)
30	0.1 Cu 1 Al	24
60	0.6 Cu 1 Al	48
120	1 Sn 5 Cu 1 Al	100
250	3 Pb 2 Sn 0.5 Cu 1 Al	205

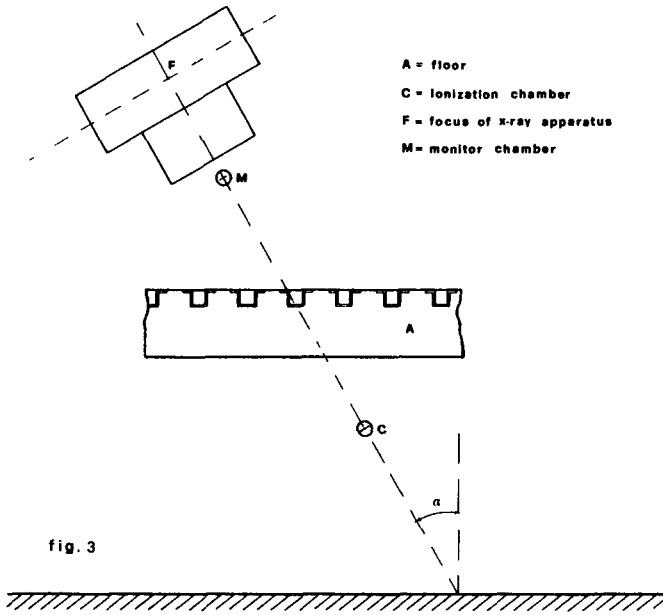


fig. 3

In fig. 4 we plot the transmission factor versus the incidence angle for each energy;

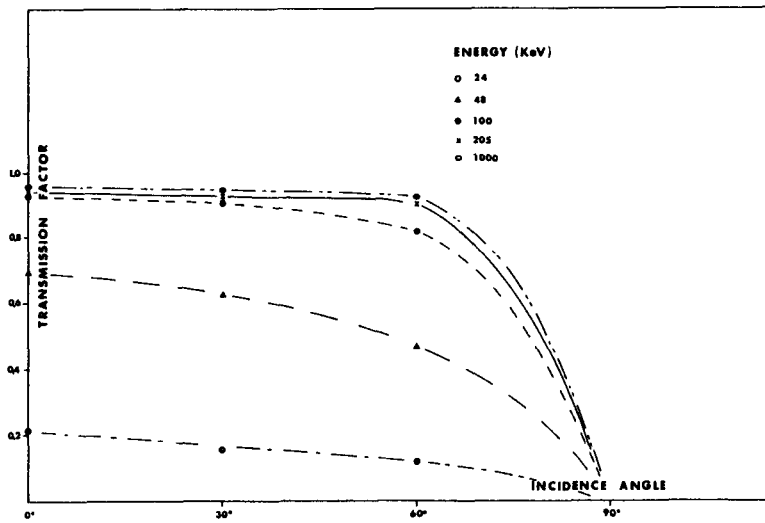


fig. 4

in fig. 5 the transmission factor versus energy is plotted at different angles of incidence.

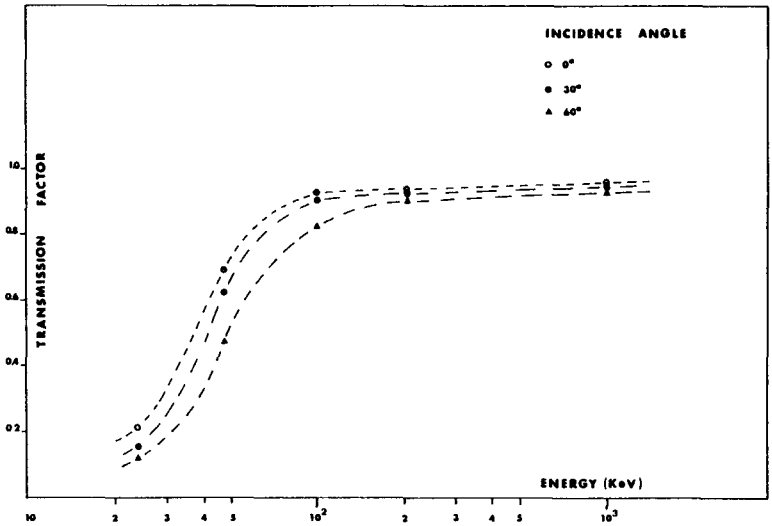
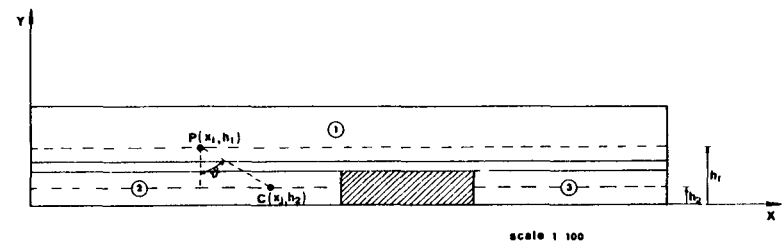


fig 5

These experimental curves were fitted by a polinomial regression and used to derive the transmission factor $\bar{T}(E_k)$, which represents the trasmission in passenger seat wherever located in the cabin, due to a package carrying a radioisotope of energy E_k, wherever located in the cargo compartment.

The details of the calculation are shown in fig. 6.



① passenger cabin
② front compartment
③ rear compartment

fig 6

$P_i(x_i, h_i)$ passenger

$C_j(x_j, h_2)$ package

$\vartheta_{ij} = \text{arctg}\,(x_j - x_i)/h$ $\qquad h = h_1 - h_2 = 120$ m

$T_{ij}(E_k) = f\left(\text{arctg}\,\frac{x_j - x_i}{h}\right)$

this function is obtained by the fit of the experimental points of fig. 4.

$$\bar{T}_i(E_k) = \frac{1}{N} \sum_{1}^{N}{}_J \; f\left(\text{arctg}\, \frac{x_J - x_i}{h}\right)$$

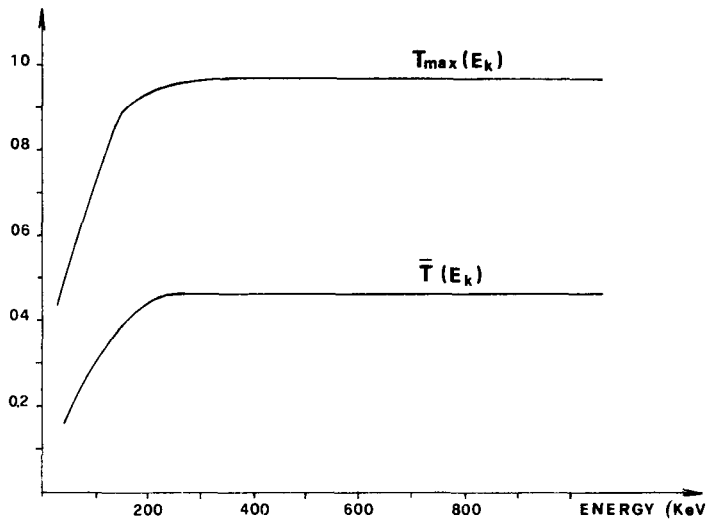
represents the transmission factor to the passenger i, from a package wherever located in the cargo compartment.

$$\bar{T}(E_k) = \frac{1}{n} \sum_{1}^{n}{}_i \; \bar{T}_i(E_k)$$

gives the transmission factor to the passenger wherever located. The transmission factor T max (E_k) for the seat immediately above the package was also evaluated. In table III the final results are summarized. The curve of fig. 7 is obtained by fitting the data of table III.

TABLE III

Energy (KeV)	24	48	100	205	1000
$\bar{T}(E_k)$	0.065	0.264	0.310	0.463	0.472
T max (E_k)	0.213	0.692	0.927	0.935	0.960



If the types of radioisotopes and the total amount of of transport indexes transported is known the average dose as well as the maximum dose to a generic passenger can be obtained from fig. 7.

Conclusions

From the direct measurements we derived a cosmic radiation dose rate at level flight altitude of 2÷3 μGy/h in fairly good agreement with the data available in the literature ranging around 3 μGy/h at an altitude of 9,000 meters (2). As regards passengers flying in aircrafts carrying radioactive materials, the measured cumulated dose in the cabin (0.4 ÷ 0.6 mGy) gives an average dose per flight of 4÷7 μGy. These values become 0.2÷0.6 μGy if a randomly selected flight is considered.

In other words a passenger flying for one hour is likely to receive an extra dose corresponding to about 0.02÷ ÷0.06% of the average yearly background dose (1 mGy/y at sea level). The measured cumulated doses in the passenger cabin can also be used to derive the yearly doses to cabin personnel (hostesses and stewards) these doses range between 2.5 and 3.5 mGy/y. The dose measured at the pilot seat during the whole observation period ranges between 0.3 ÷ 0.5 mGy. These dose values give an annual dose equivalent of around 1.8 ÷ 3 mGy.

Basing on these data we assume that both pilots and crew members can be assign to "B" condition of work requiring only periodic assessments of the actual working conditions.

From the transmission measurements we derived an average transmission factor for the mostly common radioisotopes transported in Italy for the seat directly above the package of 0.9, which is higher than 0.7 given by Barker at al. (3), this discrepancy could be probably ascribed to the presence of cables which pass through the floor structure.

References

1. Scarpa G.et al, Further Studies on the Mixed Field of Neutrons and Gammas in a Fast Reactor using solid State Dosimeters. Proc. 3rd Int. Symp. Neutron Dosimetry in Biology and Medicidine Munich 23-27 May, 1977.

2. Cowan F.P. et al Direct Measurements of Dose equivalent in aircraft BNL 17060.

3. Barker R et al. Population Dose Evaluation and Standards for Man and his Environment. Proc. of a Seminar on Portorose, May 1976 IAEA Vienna 1974.

PUNCTURE RESISTANCE OF TYPE B TRANSPORT SYSTEMS*

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United States certification of low level and transuranic (Type B) transportation systems requires that the systems survive the various hypothetical accident conditions described by U. S. Reg. 10CFR71. This regulation attempts to ensure that a shipping package provide sufficient protection for the contents from external heat (fire), general physical damage (crushing) and puncture. The puncture qualification test described therein dictates that the containment envelope of a completed Type B package equilibrated at 233 K not be breached by a 1 m free-fall drop onto a 0.15 m diameter bar. Since localized response at the point of impact will depend on both the container design and material mechanical properties, selection of materials for Type B containers must of necessity consider this accident condition. Unfortunately, no standard test procedure is presently available for screening materials for this application. This report describes a recent attempt to develop a test method for use in screening materials and for evaluating the effects of certain parameters, for example section stiffness, on container penetration resistance. In addition, it illustrates the application of this procedure to the selection of a sheet steel for a transuranic waste (TRUPACT) container.[1]

The test consists of penetrating a specimen, normally 0.6 m square, with a punch (tup) attached to a falling weight and recording and analyzing the force-time history to determine the energy absorption during the impact event. The test as developed simulates certain aspects of the 10CFR71 drop test in order to provide a means of comparing, for example, the penetrating resistance of various steels, this resistance being defined as the energy required to initiate fracture in the specimen.

The test method makes use of a 13,500 J capacity drop weight tower, Fig. 1. This tower consists of a 1050 kg weight (crosshead) guided by columns and raised to a height of 1 m by an electric ho[illegible] A steel indenter (tup) is mounted directly beneath the crosshead. The tup, which transfers the available energy to the specimen, is fabricated from hardened tool steel, R_C55, and is dimensionally identical to that called forth by 10CFR71, a 0.15 m diameter cylinder,

*This work sponsored by the U. S. Department of Energy under Contract DE-AC04-76-DP00789.
**A U. S. Department of Energy facility.

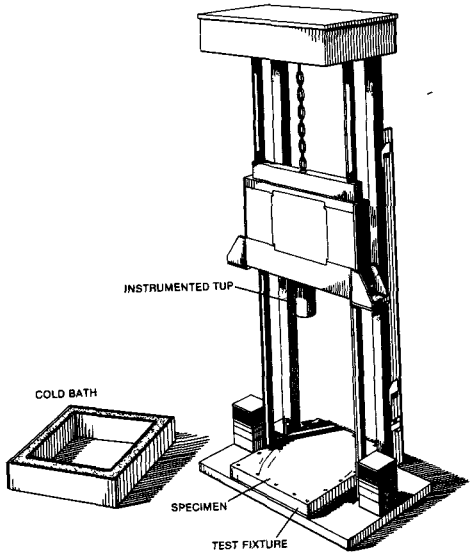
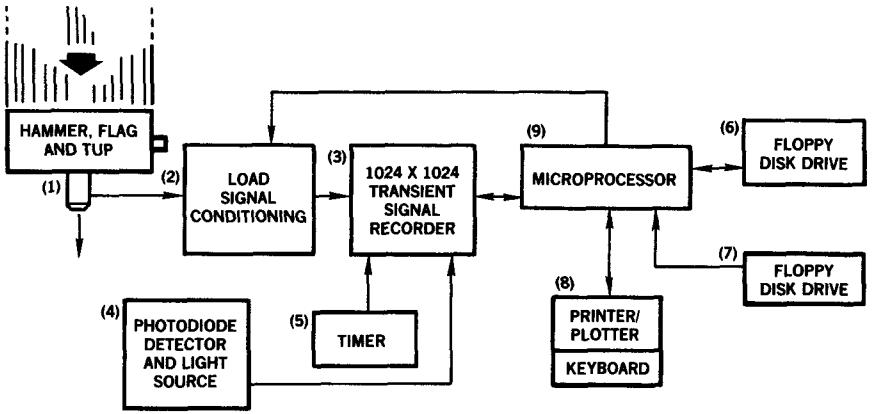


Fig. 1. Drop Tower/Sample/Tup Configuration Used to Evaluate Puncture Resistance of Sheet Steel.

0.2 m in length and having a 1.6 mm edge radius. The tup contains a reduced section onto which semiconductor strain gauges are bonded to sense compressive loading. An excitation voltage applied to the strain gauges produces a voltage output linear with compressive load. After fabrication, the tup is calibrated by applying various known loads to the tup. The tup output is amplified and measured at each load. The amplifier gain is then adjusted to establish a 1:1 correspondence between applied load and tup output.

An opaque flag 1 cm wide attached to the crosshead momentarily interrupts a collimated light beam just prior to impact. Measurement of the time of light beam interruption allows calculation of the initial impact velocity. The load and velocity signals are transmitted to a microprocessor-based data acquisition/analysis system. This system (block diagram, Fig. 2) digitizes the load signal and records the data against a time base. The data are then numerically integrated to calculate energy and deflection as a function of time. A hard copy record of load/deflection and energy/deflection is printed for each test, Fig. 3.

Initial evaluations utilizing the aforementioned drop tower arrangement examined the importance of differing tup/sample temperature and varying sample constraint on the fracture initiation process. Consideration of the former showed that there was no effect of indenter (tup) temperature on the amount of energy absorbed up to crack initiation, i.e., to peak load, Table 1. All tests were therefore conducted with the tup at ambient temperature. Test results



FUNCTIONS

1. Strain gauges provide load signal.
2. Amplification and digitization of load signal.
3. Recording of load versus time-data.
4. Provides signal for timing velocity of crosshead and trigger for data collection.
5. Supplies time base.
6. Stores and reads digitized test data and test parameters.
7. Reads programs into microprocessor memory (RAM).
8. Prints test data and analysis; allows user control of microprocessor.
9. Executes programs in RAM and ROM.

Fig. 2. Block Diagram for Automated Instrumented Impact System.

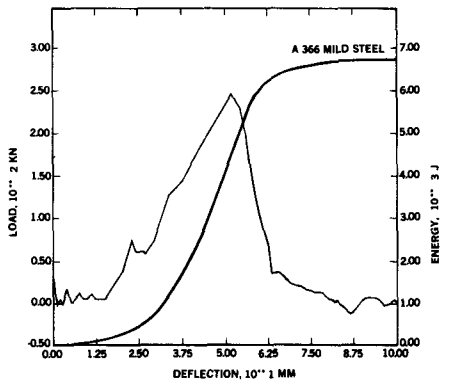


Fig. 3. Typical Load/Deflection and Absorbed Energy/Deflection Curves for A-7 Mild Steel Tested at 233 K.

were, however, found to be extremely sensitive to the degree of sample constraint. A maximum degree of constraint produced the most uniform and reproducible results and was therefore used for all subsequent tests. This constraint was achieved by welding a mild steel "skirt" to the plate (specimen) edges and bolting the skirt to the test fixture, Fig. 4.

Table 1

Influence of Indenter/Sample Temperature on Energy Absorbed to Crack Initiation*

Sample Temperature (K)	Indenter Temperature (K)	Initiation Energy (J)
233	298	6700
233	385	6660
368	298	8310
366	385	8015

*HF-80, 3.27 mm thick, indenter leading edge perpendicular to rolling direction.

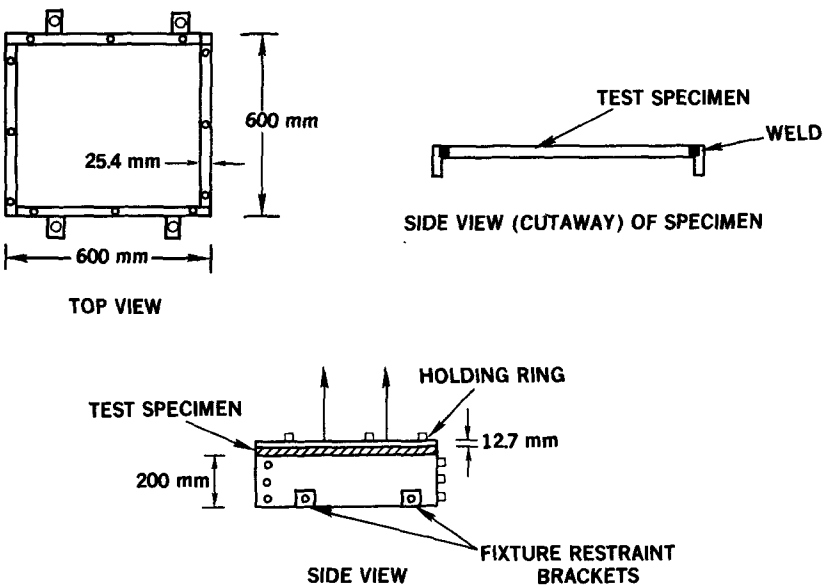


Fig. 4. Sheet Test Sample Hold-down Configuration

Failure in all constrained tests was believed to have initiated at a single point along the tup perimeter and propagated in either direction along the tup edge. Indeed the fracture path was never seen to leave the tup perimeter. During the screening tests, significant variations in the energy to initiate failure were noted in several steels due to the relative orientation of the specimen rolling direction and tup. This effect, evidenced by fracture always initiating in the same location, was traced to a slight lack of parallelism ($\sim$0.6 degree) between the tup surface and specimen causing one edge of the tup to consistently make initial contact. Since actual drop testing cannot be expected to achieve a more uniform container/punch contact, this geometry was maintained for the program duration and all drop tests were conducted by controlling the relative orientati of the indenter leading edge and the rolling direction.

The drop tower/tup and sample constraint arrangements described above were used to examine the influence of alloy chemistry, test temperature, and sheet thickness on the energy necessary to initiate failure, E_a. The penetration resistance of a series of high-strength, low-alloy steels, identified in Table 2, have been compared to that of ASTM A366 mild steel at 233 K.

Table 2

Candidate High-Strength, Low-Alloy (HSLA) Steels

Steel Designation	Manufacturer	Metallurgical Condition
YS-T70	Youngstown Steel	Ferritic
YS-T80	Youngstown Steel	Ferritic
NAPAC-F50	National Steel	Ferritic
NAPAC-F80	National Steel	Ferritic
NAX-80	National Steel	Ferritic
HF-50	Inland Steel	Ferritic
HF-80	Inland Steel	Ferritic
CORTEN-A	U.S. Steel	Ferritic/Pearlitic
EXTEN-50	U.S. Steel	Ferritic/Pearlitic
LO-TEMP	ARMCO Steel	Ferritic/Pearlitic

The results, shown in Fig. 5, indicate that the high-strength, low-alloy steels offer distinct advantages over mild steel. Not only are their tensile strengths typically twice that of mild steel, their ability to absorb energy during punch penetration can be as much as 60 percent higher than mild steel. Indeed, Fig. 5 suggests that a possible correlation may exist between the ultimate tensile strength and the initiation energy, i.e., increasing the ultimate tensile strength increases the energy necessary to initiate cracking duri puncture. There are, however, exceptions to this rule. For examp Fig. 5 shows that the results for three high-strength, low-alloy steels, i.e., CORTEN-A, NAX-80 and LO-TEMP, lie considerably above the general trend line. The reason(s) for this possible correlation and for the exceptions to same are presently unknown and are under continuing investigation.

Figure 6 shows that decreasing test temperature between 366 and 200 K results in a linear decrease in initiation energy. There was no evidence observed for a sharp ductile-to-brittle energy transition with decreasing temperature. Fractographic examination also indicated that the fracture surface was always characterized by shear dimples, independent of test temperature.

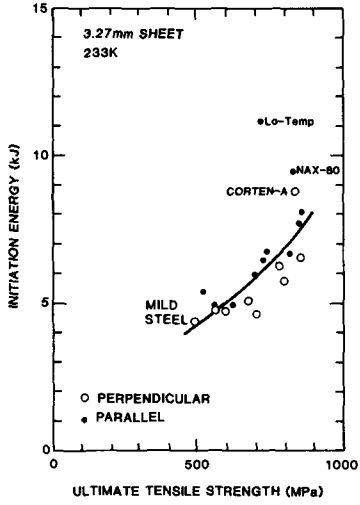


Fig. 5. Influence of Ultimate Tensile Strength on Energy to Initiate Failure in High-Strength, Low Alloy Sheet Steel. Indenter Leading Edge-Rolling Direction Orientation as Noted.

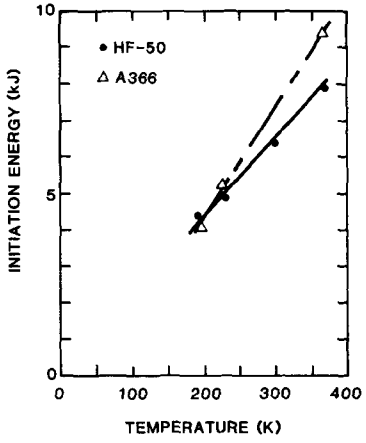


Fig. 6. Influence of Test Temperature on Energy Required to Initiate a Crack. Indenter Leading Edge Perpendicular to Rolling Direction.

Finally, the influence of sheet thickness on the energy necessary to initiate a crack, E_a, was examined by blanchard grinding CORTEN-A steel to three different thicknesses and testing at 233 K. These results, Fig. 7, show that, at least to a first approximation, E_a is a linear function of thickness, i.e.,

$$E_a = k_1 t$$

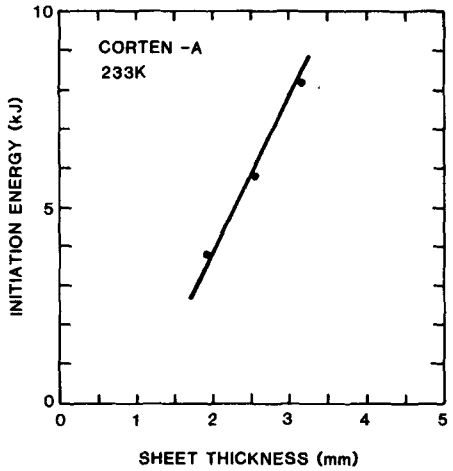


Fig. 7. Influence of Sheet Thickness on Fracture Initiation Energy, E_a.

In summary, this examination suggests that it should be possible to develop a laboratory test to rank and select materials for maximum puncture resistance. Although the initial results appear promising, more effort will be required before this procedure can be routinely applied to examining the various factors which control the puncture resistance of these materials. These results do, nonetheless, show that high-strength, low-alloy steels do offer significant advantages over mild steel for container penetration protection. Indeed, one of these steels, NAX-80, is presently considered as a prime candidate for the TRUPACT container being developed at Sandia National Laboratories.

REFERENCE

1. R. G. Eakes, G. H. Lamoreaux, L. E. Romesberg, S. H. Sutherla and T. A. Duffey, "TRU Waste Transportation Package Developmen Paper Number 140, PATRAM '80, Berlin, FRG, November 1980.

PERFORMANCE TESTING OF LARGE METALLIC SEALS*

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I. INTRODUCTION

Background

Containment of radioactive material in spent fuel shipping casks can be achieved if the internal cavity pressure boundary is kept intact under both normal and accident conditions. The major potential boundary weakness is the seal used for cask closure.

In an effort to evaluate parameters which influence closure seal performance, an experimental program has been undertaken at the Sandia National Laboratories' Transportation Technology Center (SNL/TTC). This paper describes the program, early results, and test hardware.

Work Objectives

This performance testing program for candidate metallic seals for shipping casks was undertaken with two primary objectives:

1. To supply practical information concerning large cask seals to SNL/TTC's Licensing and Data Base Methodology efforts.

2. To provide data and experience on several types of seals to aid in seal selection for the Liquid Metal Fast Breeder Reactor (LMFBR) spent fuel assembly shipping canisters and casks.

The data gained from the testing program will provide evaluations of seal reusability, expected leakage, and sensitivity to dirt and misalignment. Emphasis to date has been placed on metallic seals.

II. PREVIOUS WORK†

als

The first candidate seal selected for the LMFBR cask was the GRAYLOC, a patented, proprietary design of Gray Tool Co. (GTC), Houston, TX. A cross section of the seal and mating surfaces is shown in Figure 1. The GRAYLOC is commonly used in the chemical and petroleum industries and has been used in one model of a light-water reactor spent fuel cask, the General Electric IF-300. It is reusable,

*Work performed for U. S. Department of Energy under contract DE-AC04-76DP00789.
†This work will be described in detail in a report to be published by SNL in 1981.

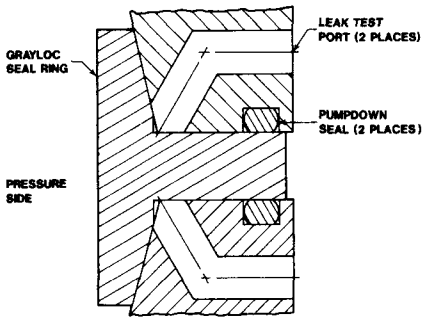

Figure 1. Cross Section, GRAYLOC Seal Ring

self-aligning, and self- and pressure-energized. Sealing action is obtained from a small amount of plastic deformation at the contact points.

Two types of GRAYLOC seals were purchased for evaluation. The first type was a standard version and the second was designed specifically for helium service. In addition, one of the standard seals was returned to GTC for modification into the helium sealing configuration. All of these seals were Size 292 (743 mm, 29.25 in diameter), AISI 630 (17-4 PH) stainless steel with a dry-film molybedenum disulfide lubricating coating.

Test Fixture

A fixture was designed as described in Reference 1. Figure 2 shows an overall view. Basically the fixture consisted of two hubs of AISI/SAE 1050 carbon steel which mate with the seal. Two domes formed a pressure chamber and three jackscrews, 120° apart, provided the seating force.

A Veeco model MS 90-AB helium leak detector was used to measure the helium leakage through the seal. Strain gage load cells, Interface, Inc., model 1240MP, measured the force applied by the jack screws, and dial indicators measured the relative axial motion between hubs. Some seals were instrumented with strain gages to provide data (not included here) for comparison with analytical efforts.

Test Results

Due to design deficiencies and rework efforts, no successful leak tests were completed on the standard type seals. The modified standard (MS1) and two helium seals (HS1 and HS2) were successfully tested, and the results are summarized in Table 1. The seals were repositioned after each seating and were occasionally removed, washed, and replaced.

MS1 showed a relatively constant leak rate throughout the testing except for two readings which were roughly an order of magnitude higher. Both times washing and cleaning restored the leak rate to its former level. HS1 also showed a relatively constant leakage until the leak rate jumped roughly two orders of magnitude. Washing and cleaning failed to help. The HS2 leak increased after each seating. Washing and cleaning did not reduce the leak rate.

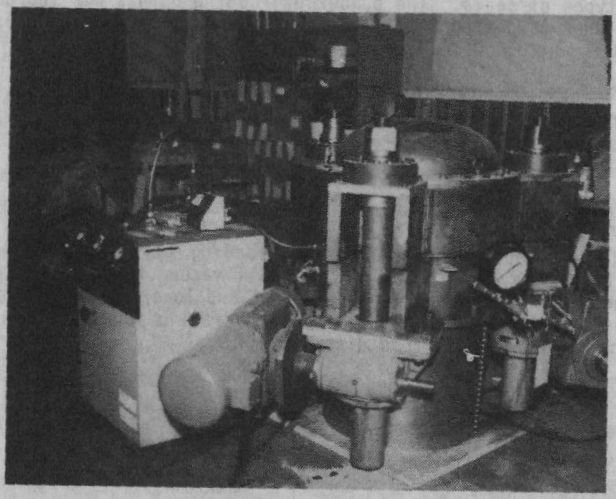

Figure 2. GRAYLOC Seal Test Fixture

Table 1. Seal Test Results

	Modified Standard Seal MS1	Helium Seal HS1	Helium Seal HS2
Number of Good Seatings	20	16	5
Average Seating Force, kN	352	462	390
Average Axial Motion, mm	5.50	6.69	6.18
Average Leak Rate, atm cc/s	5x10-5	5x10-6	8x10-5
Leak Rate Range, atm cc/s	9x10-6 to 2.5x10-4	3x10-6 to 8x10-6	<1x10-6 to 2x10-4

It was suspected that hub deflection between the jackscrew locations was excessive during testing. An analysis[2] and measurements showed the deflection to be approximately 0.15 mm (0.006 in). It is not currently known what, if any, effect the deflection had on the sealing results. Additional tests on the GRAYLOC seal will be conducted using the new test fixture, which is described next.

III. CURRENT WORK

New Test Fixture

Since the seal testing program is a continuing effort at SNL/TTC it was decided to design and build a new test fixture. This fixture, unlike the first one which could test only GRAYLOC seals, has replaceable adaptors to allow testing

of different types of seals. The number of loading points was increased from 3 to 12 to reduce the hub deflection. Calculated maximum hub deflection for the new fixture is 0.005 mm (0.0002 in).

A cross section through the new fixture is shown in Figure 3. This fixture uses a set of adaptors specifically designed for the type of seal to be tested. Any typical seat material can be used for the adaptors, but those built to date have used AISI 304 annealed stainless steel.

The fixture can circumferentially load a seal having an envelope of 25x25 mm (1x1 in) by 79 cm (31 in) mean diameter to 1.2 MN/m (7000 lb/in) using 12 equally spaced, manually pumped hydraulic jacks and loading bars. The seal cavity can be pressurized internally or externally from full vacuum to 700 kPa (100 psi). Concentricity and parallelism between the upper and lower hubs and adaptors can be adjusted using set screws and shims. The 12 hydraulic jacks can provide a uniform or non-uniform circumfrential load distribution.

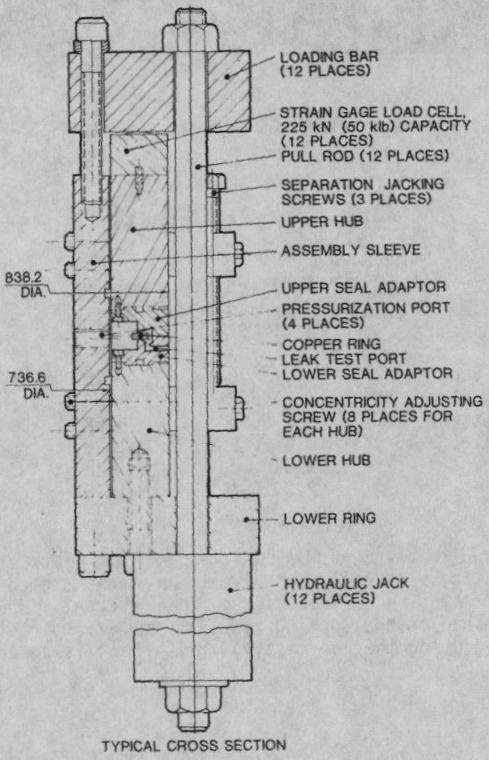

Figure 3. Typical Cross Section of New Test Fixture

Instrumentation

Each loading bar has a load cell and a linear variable differential transformer (LVDT) displacement gage. Celesco model 200-S-3-50K-1 strain gage load cells and Trans-Tek model 243-000 LVDTs are used. Readout is by an Acurex Autodata 9 data scanner which transmits the data either to a Tektronix 4923 cassette recorder or to a Tektronix 4051 minicomputer for on-line data reduction. A Veeco model MS 170 helium leak detector is used to measure leak rate. The fixture and instrumentation are shown in Figures 4 and 5.

IV. FUTURE WORK

It had been planned to include test results from the new fixture in this paper, but the hardware was not received in time. The present schedule includes tests on four types of seals. Each will be described briefly below and a cross sectional sketch of the seal as seated in the fixture is shown:

GRAYLOC: As mentioned previously, tests will be repeated on the GRAYLOC seal (Figure 1).

Copper Ring: This seal (Figure 6) was developed for high vacuum service by Varian Associates, Palo Alto, CA, under the registered tradename of ConFlat[3]. It was used quite successfully on an air-transportable plutonium shipping container[4]. It consists of a pair of concentric, hard teeth, one tooth in each mating part, that bite into a flat, soft copper ring. The teeth are directly opposed, and the mating parts must incorporate an aligning feature to hold the teeth in opposition. The seal is not self- or pressure-energized but depends completely on the clamping force. The copper ring appears to be reusable if the clamping force is increased.

Hollow Metal O-ring: Metal O-rings (Figure 7) are simple and relatively inexpensive and are frequently used in adverse environments[5]. However, they are sensitive to foreign material and require a very smooth surface finish. They are self- and pressure-energized and reusable.

GAMAH: This proprietary seal (Figure 8), designed and built by GAMAH Division of Stanley Aviation, Denver, CO, has been used successfully on several small special purpose radioactive material shipping containers[6]. It is self-aligning, self- and pressure-energized, and reusable.

Test procedures have not been completely developed but will include monitoring leak rate and axial motion while performing one of the following operations:

1. Seating seals with uniform or non-uniform circumferential loading
2. Releasing circumferential load uniformly or non-uniformly
3. Seating non-self-aligning seals with various amounts of eccentricity
4. Seating seals with various amounts of non-parallelism

Dirt sensitivity tests will consist of seating the seal with foreign material such as thread on the contact faces.

The new fixture will allow large scale testing of other types of seals as they become available. As such tests are completed, the performance data and evaluations will be published. Tests on the four types of seals described earlier should begin in late 1980 and be completed in 1981.

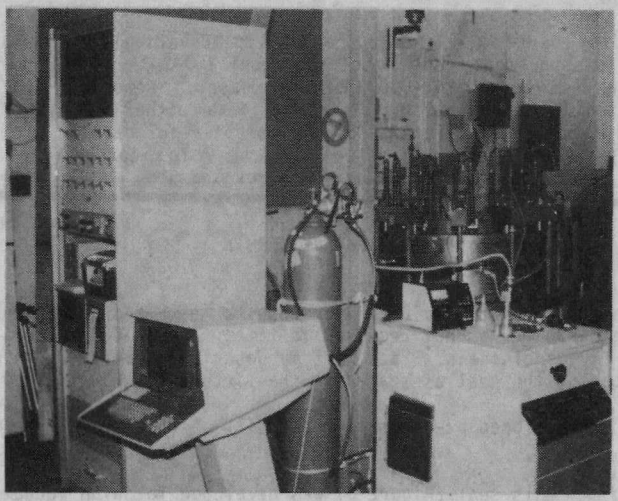

Figure 4. New Fixture and Instrumentation

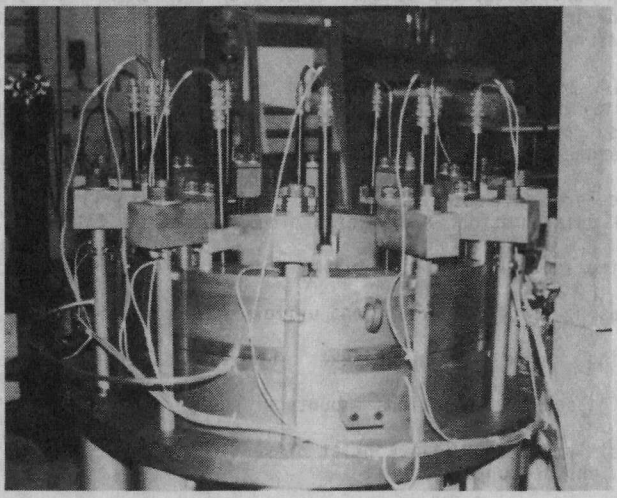

Figure 5. New Test Fixture

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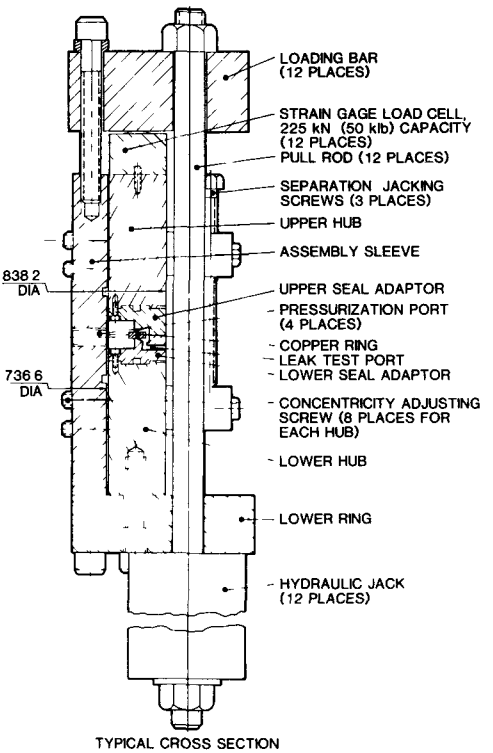

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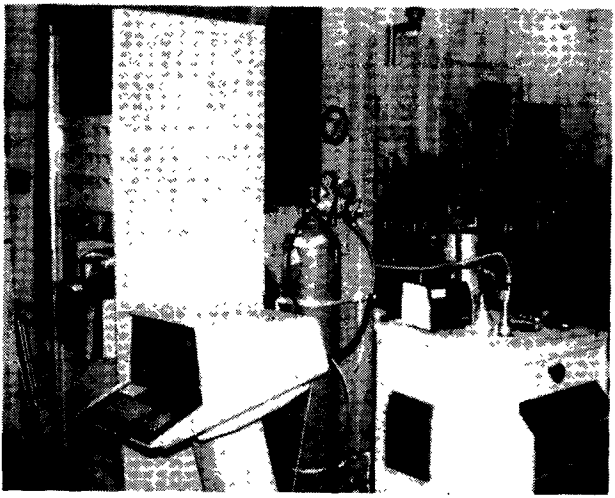
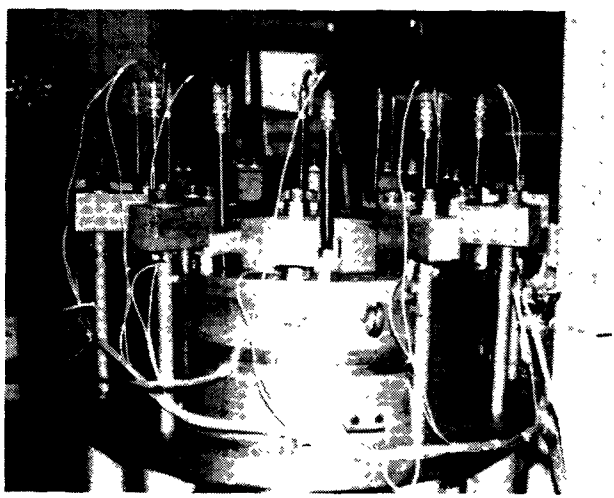


Figure 4. New Fixture and Instrumentation



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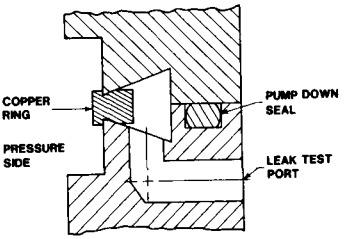

Figure 6. Cross Section, Copper Ring Seal

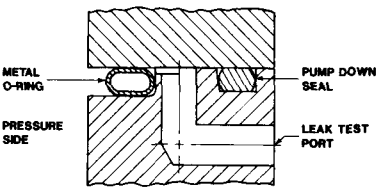

Figure 7. Cross Section, Hollow Metal O-Rings

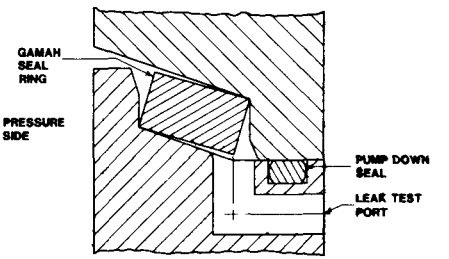

Figure 8. Cross Section, GAMAH Seal Ring

V REFERENCES

1. O. L. Burchett and J. H. Biffle, A Program for Static Testing Cask Seals, SAND76-0287, Sandia National Laboratories, Albuquerque, NM, June 1976.

2. M. J. Sagartz, Deformation of the Sandia Metallic Ring Seal Test Fixture, SAND78-2192, Sandia National Laboratories, Albuquerque, NM, January 1979.

3. W. R. Wheeler and M. Carlson, Ultra High Vacuum Flanges, Transactions of the Eighth National Vacuum Symposium, American Vacuum Society, p. 1309, Pergamon Press, Oxford, 1962.

J. A. Anderson, et al, PARC (Plutonium Accident Resistant Container) Program Research, Design and Development, SAND76-0587, Sandia National Laboratories, Albuquerque, NM, July 1978.

5. United Metallic O-Rings, Bulletin 101A, UAP Components, Inc., Columbia, SC, June 1978.

6. GAMAH Division, GAMAH Design Handbook, Stanley Aviation Corp., Denver, CO, No Date.

SESSION VIII
OPERATIONS AND SYSTEMS EXPERIENCE I

EXPERIENCE OF EUROPEAN LWR IRRADIATED FUEL TRANSPORT

H. W. Curtis
Managing Director
Nuclear Transport Limited

1. INTRODUCTION

At the Fifth International Symposium on the Packaging and Transport of Radioactive Materials in 1978, I presented a paper on the experience of the transport of the first five hundred tonnes of European LWR fuel. Since the last symposium, irradiated fuel transports have progressed steadily and the total is now close to one thousand tonnes of uranium. The operation is a continuous, year-round activity and as I speak, the total moves a little higher.

At the last symposium, one of the American speakers commented a little ruefully on the comparison between the success of the European movements and the almost total lack of U.S. activity. He postulated the view, which I contested at the time, that opposition to the European transports might increase and gain sufficient success that they would seriously restrict future European activity. I am happy to report that this has not been the case and the European and American scenarios are now even further apart. Indeed, the greater the background of safe transports successfully performed, the greater is the confidence of all those concerned and the greater the public acceptance of irradiated fuel transport as something which, if not particularly welcomed, is at least not particularly feared. In the United States on the other hand, the longer that no significant programme of irradiated fuel transport exists, the more difficult will it become to obtain acceptance by the public and those state and local agencies with power to impede the transports. It seems that this inertia can only be overcome by government action and a major publicity campaign to secure public acceptance.

2. QUANTITIES OF FUEL TRANSPORTED

The first transport of European LWR irradiated fuel transport took place in 1968 when Garigliano BWR fuel was transported by road to the port of Anzio in Italy, by sea to the port of Barrow in England and thence by rail to Windscale. During the next few years, transports of irradiated fuel averaged around 20 tonnes a year and slowly began to increase from 1973. From a uranium tonnage of 84 in 1973, seven years have passed before we have seen this figure doubled in 1980. Apart from a small tonnage in the early years, this fuel has been transported by Nuclear Transport Limited and the total number of LWR fuel elements transported is now in excess of 4,000.

The use of high density pond storage racks has not yet been generally authorised in Europe and continued operation of the reactors has been dependent upon the ability of the transporter to remove each discharge of fuel elements from the reactor pond before the shut-down for despatch of the following batch in order to maintain the regulatory requirement for a full core discharge to the reactor pond.

Completion of the new storage pond at La Hague in 1981 coupled with the transport of the fuel from reactors from which fuel has not been previously transported are expected to more than double the tonnage transported in 1981 in comparison with the previous year.

3. TYPES OF FLASK

The diversity of the physical limitations at the early reactors was almost as great as that of their geographical circumstances. These factors led to the development of a varied and flexible pool of flasks which could cover the needs of all the European reactors and be transported by combinations of road, rail and sea. Flask capacity covered the range between 1 and 3 tonnes of irradiated uranium and it has always been possible to provide a flask which offered the best solution to the problems of each reactor. As experience has been gained in flask handling at the reprocessing plants, it has become obvious that the radiation uptake of the operators of the flask unloading facility is proportional to the total number of flasks handled rather than the quantity of fuel contained in any particular type or size of flask and the reprocessors have developed a preference for the use of large flasks with a capacity which is greater than 2 tonnes and can be as high as 5 tonnes. The largest sizes of flask with an all-up weight up to 120 tonnes can carry 12 PWR elements or approximately 32 BWR elements. The size of such flasks has been determined by the limiting dimensions of the profile of flask and wagon which can be accepted by the European railways system.

The objective of a standard flask has always been regarded as desirable by all the parties concerned but it is now appreciated that allowances must be made for at least two different lengths of PWR fuel and for the handling limitations of BWR reactors. There is thus a minimum requirement for three sizes of flasks, without counting the many early reactors subject to numerous other limitations.

The following slides will illustrate the development of NTL flasks over the past ten years.

NTL 2

This one is shown for historical reasons only and is a dry flask designed by Transnucléaire with a capacity of 4 PWR or 9 BWR elements. They weigh 32 tonnes and were designed to travel exclusively by road on their own purpose-built semi-trailers. Heat is dissipated by the 16000 stainless steel pins, which gave them the name of "hedgehogs". They have transported over 800 fuel elements

and are now approaching the end of their useful life due to the absence of fixed neutron shielding and enter honourable retirement. They will still provide me with useful employment on my retirement, when I shall carry on as curator of the NTL Flask Museum.

NTL 8

This dry flask is the development from the NTL 2 and has a capacity of 3 elements in its PWR form or 7 in the BWR form. We have a pool of 6 of this type and they have transported over 1300 elements.

NTL 11

This flask is a development by NTL of BNFL's EXCELLOX flask. It is a wet flask of 75 tonnes with a capacity of 7 PWR or 17 BWR elements. We own 5 of these and 4 of a smaller, earlier type. Together they have transported over 1700 elements. Because of their weight they travel mainly by rail or sea.

NTL 10

This is the jumbo of the NTL flask pool and is known as "Big Daddy". It is a 105 tonnes dry flask designed by Transnucléaire and has a capacity of 12 standard Westinghouse elements or 11 elements of the 1200 MW reactor type. This flask is a version of the TN 12 with an extension to accommodate the longer fuel.

4. TYPES OF TRANSPORT

Geographical locations impose the need for flexibility in the means of transport of irradiated fuel. Reactors are frequently located in relatively remote areas without rail or sea connection. Where transport can be performed entirely by land, there was therefore a substantial incentive in the early days to perform the complete journey by road. This method provides maximum flexibility and speed but the pay-load is limited to approximately 1 tonne of uranium. The system has worked satisfactorily however for a number of years and to quote one example, more than 110 tonnes of fuel have been moved from the Stade reactor in the most northern part of Germany to La Hague over the past 8 years. Road transport will continue to play an important part in the future where its special benefits can best be used but it is unlikely that further quantities of light flasks of 38 tonnes weight will be purchased.

The transport of heavy flasks weighing from 70 tonnes upwards requires special high capacity road trailers and the availability of a suitable crane if the reactor is not connected to the railway system. It is preferable to avoid any change in the mode of transport between the reactor site and the reprocessing plant and it is now most inadvisable to consider the construction of a reactor without rail connection.

Transport to Windscale necessitates sea transport and the emphasis

has therefore from the beginning been on the heavier flasks, although a number of movements were made with light flasks using roll-on/roll-off ferry systems. Windscale plant has a rail connection so that flasks arriving at Barrow in charter ships can be transferred directly to the rail wagons which are transported into the plant. Use can also be made of the rail ferry system for the transport of heavy flasks direct from reactor to reprocessing plant. The La Hague reprocessing plant as yet has no rail connection and all heavy flasks are transferred from rail to road transport at a special terminal some 20 kms from the reprocessing plant.

A special rail wagon capable of carrying the heaviest flasks has been developed and this can be attached to normal freight trains. The European railway organisations have adapted themselves well to the transport of irradiated fuel and an excellent level of performance has been achieved.

5. PROBLEMS OF IRRADIATED FUEL TRANSPORT

At the 1978 symposium a number of current problems were discussed but few of these now present serious inconvenience. The release of crud from irradiated fuel is controlled in the case of Windscale by transporting the elements in sealed Multi-Element Bottles which are removed from the flask and used for pond storage at the reprocessing plant. The crud remains contained within the Multi-Element Bottles during transport and subsequent storage. In the case of La Hague, secondary containment is not utilised but intensive flushing of the inside of the flask takes place before the lid is removed and by this means, the inside of the flask is kept relatively clear of build-up of crud.

The transport of failed fuel is avoided. For transports to La Hague, stringent tests to establish the integrity of the fuel are applied to the filled flask before its departure.

Sweating out of contamination continues to cause difficulties relative to the cleanliness of the reactor pond and the standard of housekeeping which is maintained. The present solution is to aim to decontaminate the flasks to one-tenth of the permissible limit before departure from the reactor site. Wherever possible, a rigid metal skirt covering all the finned area of the flask is used and this device is successful in keeping the fins out of contact with the contaminating pond water. A plastic skirt is used in those cases where the design prevents the use of a metallic skirt.

There has been no significant increase over the past three years in public interest in irradiated fuel transport, largely because these operate smoothly and safely and do not attract public attenti

6. SAFETY

The safety record of irradiated fuel transport can be quickly dismissed since other authors at the symposium have contributed a considerable number of papers on the topic. From the point of

operating experience, the safety record remains 100%.

7. COMMERCIAL ARRANGEMENTS

Transport is marketed in conjunction with reprocessing by the European reprocessors and NTL as a specialized transporter of irradiated LWR fuel is a sub-contractor of these reprocessors. All the parties consider this to be a most satisfactory arrangement for a number of reasons:

i The reprocessor must be able to control the reception programme of flasks at the reprocessing site. Chaos would result if a number of independent unco-ordinated transporters were able to arrive at the reprocessing plant at the same time.

ii Only the reprocessor and not the transporter can guarantee to the reactors that flasks will be unloaded on arrival.

iii The reprocessor needs to exert a considerable influence on the design of flask to be handled. It would be impractical for the reprocessing plant to provide rapid turn-round if they had to handle flasks of several different designs for basically similar fuels. The tendency at reprocessing plants is now to establish the maximum possible degree of standardization and automation of flask handling to speed up flask turn-round and minimize operator radiation uptake.

iv Public relations are of the utmost importance to the transporter and the reprocessor but the latter is concerned mainly with local aspects within an area of say 100 km from the reprocessing plant. It is essential that the reprocessor should know precisely what transport activities are planned or are in progress in that area at any time so that he can respond quickly and effectively to any query or incident. This situation can only be achieved by a close understanding between the reprocessor and transporter which would be difficult to achieve simultaneously with several different transporters.

v On the other hand, the reprocessors recognize that the international transport of irradiated fuel is a highly specialized business requiring special knowledge, abilities, experience and resources in a number of different countries. They consider that this task can best be executed by a relatively small, highly specialized, flexible transport organization such as NTL.

Thus the main contractual partners are the reprocessor and the utility and there is no direct commercial relationship between the utility and the transporter. The reprocessor sub-contracts his transport responsibilities to the transporter who in practice will then develop the close working relationship with the reactor operator to build the bridge between the reactor and the reprocessing plant.

The objective of the transport programme has been to remove the fuel arising from the annual reactor discharge from the reactor

pond before the shut-down of the reactor in the following year. Since the fuel requires a minimum cooling period of 6 months, this gives less than 6 months for the transport campaign. During the 12 years of European LWR irradiated fuel transport, no reactor has been shut down because of the failure of the transporter to transport the fuel before the operating deadline.

8. FUTURE CAPABILITIES

All European LWR fuel which has been transported between 1968 and 1980 has been transported by the NTL fleet of 18 flasks of varying weights and fuel element capacities. Whilst many of these flasks will continue in operation for a number of years, COGEMA, who will receive over 90% of the European arisings from 1981 onwards, have ordered the next batch of flasks to meet this requirement. NTL will be the operator of these flasks as the sub-contractor of COGEMA and provide the same services as in the past. The new flasks to come into operation in 1981 will be of the TN12 type, designed by Transnucléaire, or its variants which cater for greater fuel length, or reactor weight restrictions.

9. THE ORGANISATION OF IRRADIATED FUEL TRANSPORT

The performance of European irradiated fuel transport between all the different countries requires an organisation spread out over the whole of the Continent. Nuclear Transport Limited has its headquarters in the United Kingdom but has strong resources in Germany and France and close links with specialists in other countries, including Transnuclear Incorporated in the U.S.A. The organisation works closely with the reprocessors and the reactor operators to provide the bridge between the two plants. The success of this organisation over the past 8 years is demonstrated by its performance to which a major contribution has been made by the close relationships and mutual confidence established with regulatory authorities of all kinds, executives and operatives of the road transport and railway organisations as well as national safety and inspection services.

DISCUSSION

Question by G. Cohendy, France: How do you reconcile your reference to the NTL2 flask as a candidate for the NTL flask museum with the fact that you have applied to the French competent authority for a new transport licence for this flask?

Answer: The NTL2 is obsolete not because it is unsafe but because it has become operationally obsolescent in relation to the operating procedures of the reprocessing plant. Its use has been accepted for the shut-down Gundremmingen reactor after which it will only be used for minor shipments such as single fuel pins.

CURRENT STATUS OF NUCLEAR MATERIALS TRANSPORTATION IN JAPAN

Shigebumi AOKI, (Tokyo Institute of Technology)

Satoshi FUKUDA, (Central Research Institute of Electric Power Industry)

Masayuki NOMURA, Akio YUKI and Tsuneo TAKEDA (Science and Technology Agency)

1. INTRODUCTION

The transportation of nuclear materials in Japan has been carried out satisfactorily under the new regulatory system, which came into force in March, 1979. Safety criteria for Japanese regulations are based on the IAEA Regulations for the Safe Transport of Radioactive Materials (1973 revised edition). Nuclear source materials are imported by sea-going vessels and transported on road to fuel fabrication plants. LWR's fresh fuel assemblies are mainly transported on road from fuel fabrication plants to nuclear power plants. And LWR's spent fuels are transported by sea-going vessels from reactor sites to the PNC* reprocessing plant at Tokai-mura and also to BNFL and COGEMA reprocessing plants in Europe. The risk assessment of nuclear materials has been performed for the sea and road transportation, respectively. The risk assessment of the sea transportation was already presented and that of road transportation has been carried out considering the result of the truck crash test. The demonstration test program on the reliability of spent fuel package has been performed on schedule. The 9 m drop tests of about 60 t package have been successfully carried out.

Current status of nuclear materials transportation in Japan will be reviewed in the following.

2. TRANSPORTATION OF NUCLEAR MATERIALS

In Japan, 12 BWRs, 9 PWRs, 1 GCR and 1 Advanced Thermal Reactor are now in operation, and 1 BWR and 3 PWRs are under construction. Certificates of design approval for 61 packages have been issued (23 for type A fissile class packages, 28 for type B(M) packages, 9 for type B(U) packages and 1 for special form as of June, 1980). The approved package designs are listed on Table 1. Total number of approved packagings, which are registered to STA prior to their use, is about 6370 (ca. 6220 for type A packagings and ca. 150 for type B packagings are of June, 1980). Type A packagings such as 21PF-1 and RCC-1 are used for the transportation of nuclear source materials and fresh fuel assemblies. Type B packagings such as 75T, EXCELLOX-3, EXCELLOX-3A, EXCELLOX-3B for LWR and TK for GCR are used for the transportation of spent fuels. Spent fuels are mainly transported by sea-going vessel "Hinoura-maru" for domestic transportation and by "Pacific Fisher" and "Pacific Swan" for overseas transportation.

Results of transportation up to 1979 are shown in Table 2.

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* Power Reactor and Nuclear Fuel Development Corporation

Table 1. List of Package Designs Approved by STA* (June 20, 1980)

No.	Identification Mark	Package Name	Description
1	J/ 1/AF	RCC-1	2 PWR fuel elements (INVALID)
2	J/ 2/B(U)F(Rev.1)	TN-9121/A,TN-9121	FBR fuel elements or Pu fuel rods for irradiation examination
3	J/ 3/B(M)F	S-B/23	Research reactor (KUR) spent fuel elements
4	J/ 4/B(U)F	GE model BP-2	Pu fuel rods for irradiation examination (INVALID)
5	J/ 5/B(M)F(Rev.2)	EZ-75T	7 PWR or 17 BWR spent fuel elements (INVALID)
6	J/ 6/AF(Rev.2)	RCC-1,RCC-3	2 PWR fuel elements
7	J/ 7/AF(Rev.1)	RCC-1	2 PWR fuel elements
8	J/ 8/B(M)	S-B/23	Spent fuel pieces for post irradiation examination (INVALID)
9	J/ 9/B(U)F(Rev.2)	DOT spec. 6M	Pu fuel pins for irradiation examination or PuO_2-UO_2 powder
10	J/10/AF(Rev.1)	NFI-II	2 PWR fuel elements
11	J/11/B(M)F	UNIFETCH L	Research reactor (TAIC reactor) spent fuel elements (INVALID)
12	J/12/B(M)F(Rev.2)	JFC-75Y20T	Research reactor (JAERI reactors) spent fuel elements
13	J/13/B(M)F(Rev.1)	JMTR W5200	Research reactor (JMTR) spent fuel elements
14	J/14/B(M)F(Rev.1)	EXCELLOX-3A	12 BWR spent fuel elements
15	J/15/B(M)	S-B/23	Spent fuel pieces for post irradiation examination (INVALID)
16	J/16/B(U)F(Rev.1)	TN-9180/A, TN-9180/B,TN-9180	2 ATR mixed oxide fuel elements
17	J/17/AF	OGL-77Y1.5T	Coated particle fuels for irradiation experiment
18	J/18/B(M)F(Rev.1)	EXCELLOX-3	14 BWR fuel elements
19	J/19/B(M)(Rev.2)	TK	Magnox spent fuel elements
20	J/20/AF	RAJ	2 BWR fuel elements
21	J/21/B(U)F(Rev.1)	EXCELLOX-3A	5 PWR fuel elements
22	J/22/AF	NT-II	2 ATR fuel elements (UO_2 fuel elements)
23	J/23/AF(Rev.1)	BU-J	UO_2 powder, bulk or scrap
24	J/24/AF(Rev.1)	BU-J,BU-5,BU-7	UO_2 powder, bulk or scrap
25	J/25/AF	21PF-1	UF_6 solid (30A or 30B cylinder)
26	J/26/AF	21PF-1	UF_6 solid (30A or 30B cylinder)
27	J/27/AF	21PF-1	UF_6 solid (30A or 30B cylinder)
28	J/28/AF	21PF-1'	UF_6 solid (30A or 30B cylinder)
29	J/29/AF(Rev.2)	DOT spec. 6M	High enriched U metal, U-Al alloy, UO_2 or graphite coated UO_2
30	J/30/B(U)F	OPU-77Y-50K	Pu metal or Pu-U alloy, etc.
31	J/31/AF	NT-III	Fuel elements for research reactor
32	J/32/AF	NSC-77Y-180K	UO_2 fuel pins for irradiation examination
33	J/33/B(M)F	UNIFETCH-L	Research reactor (JRR-2, JRR-4) spent fuel elements
34	J/34/B(U)	DOT spec. 6M	Dry-up samples of spent fuel solution
35	J/35/AF(Rev.1)	NFI-III	U-Al high enriched fuel elements for reactor or UO_2 powder

* STA : Science and Technology Agency

Table 1. List of Package Designs Approved by STA (Continued)

No.	Identification Mark	Package Name	Description
36	J/36/B(U)F	TN6-2	Irradiated fuel pins (INVALID)
37	J/37/AF	NT-VI	2 BWR fuel elements
38	J/38/B(M)F	JMHL-78T15T	U and/or Th samples for irradiation examination
39	J/39/B(M)F(Rev.1)	EXCELLOX-3B	14 BWR spent fuel elements
40	J/40/B(M)F	EXCELLOX-3B	5 PWR spent fuel elements
41	J/41/B(M)F	MO-1	2 PWR fuel elements (PuO_2-UO_2)
42	J/42/B(M)F	NH-25	2 BWR or 1 PWR fuel element(s) for post irradiation examination
43	J/43/AF	20PF-2	UF_6 solid (8A cylinder)
44	J/44/AF	NSFC-006	UO_2 fuel pin for irradiation examination
45	J/45/B(M)F(Rev.1)	JMHL-78T15T	U and/or Th samples for irradiation examination
46	J/46/B(M)F(Rev.1)	HZ-75T	17 BWR spent fuel elements
47	J/47/B(M)F(Rev.2)	HZ-75T	17 BWR spent fuel elements
48	J/48/B(M)F(Rev.1)	HZ-75T	7 PWR spent fuel elements
49	J/49/B(M)F(Rev.1)	HZ-75T	7 PWR spent fuel elements
50	J/50/B(U)F	DOT spec. 6M	High enriched U metal, U-Al alloy or UO_2 powder
51	J/51/B(M)F(Rev.1)	HZ-75T	7 PWR spent fuel elements
52	J/52/B(U)	DOT spec. 6M	Pu metal, Pu sample or Pu-U sample for chemical analysis
53	J/53/B(M)F	TN6-2	Irradiated UO_2 fuel elements for post examination
54	J/54/B(M)F	NH-25	2 BWR or 1 PWR spent fuel element(s) for post irradiation examination
55	J/55/B(M)F	PUB	PuO_2 powder
56	J/56/S	SS-P	Special formed Pu-238 source for X-ray luminescence analyzer
57	J/57/AF	NT-VII	UO_2, U_3O_8 powder, bulk or scraps
58	J/58/AF	NT-VIII	2 ATR fuel elements
59	J/59/B(M)F	NH-25	2 BWR or 1 PWR spent fuel element(s) for post irradiation examination
60	J/60/B(M)	HLW-79Y-4T	Liquid waste of reprocessing of spent fuel examination
61	J/61/B(M)F	JRC-80Y-20T	Research reactor (JAERI reactors) spent fuel elements

Table 2. Results of Nuclear Materials Transportation in Japan

	1974		1975		1976		1977		1978		1979	
	times	amounts	times	amounts	times	amounts	times	amounts	times	amounts	times	amounts
(1) UF_6	8	120 t	10	200 t	7	90 t	14	164 t	13	212 t	14	176 t
[redacted]O_2	53	160 t	105	230 t	116	460 t	96	426 t	62	236 t	53	292 t
[redacted]gnox fuel	2	53 t	3	49 t	3	49 t	3	43 t	4	53 t	3	50 t
(4) LWR's fresh fuel	26	270 t	20	160 t	18	200 t	51	849 t	35	418 t	37	298 t
(5) Spent fuel (Commertial reactor)	4	76 t	5	56 t	4	64 t	23	64 t	14	145 t	18	190 t
(6) Spent fuel (Research reactor)	3	41 kg	-	-	-	-	-	-	2	35 kg	-	-
(7) Miscellaneous	29	250 kg	42	229 kg	27	178 kg	60	207 kg	19	132 kg	36	281 kg

3. LICENSING EXPERIENCES

(1) Safety Criteria

It was in 1975 that Japan Atomic Energy Commission (JAEC) set up "Safety Criteria for the Transport of Radioactive Materials" based on "The IAEA Regulations for The Safe Transport of Radioactive Materials (1973 revised edition)", and at the same time made a decision that domestic regulations related to the transport of radioactive materials should be arranged to adopt the new criteria by the competent authorities.

Japanese regulations concerned are divided into two groups. The first group is concerned with nuclear materials, and the second with radioisotopes other than nuclear materials. The competent authorities adjusted these regulations to be based upon the IAEA Regulation revised in 1973 with some modifications, in compliance with the JAEC decision.

(2) Regulations

Transportation of nuclear materials is regulated for the most part by the STA and MOT regulations, which are based on the Law Concerning the Regulations of Nuclear Source Materials, Nuclear Fuel Materials and Reactors (for road transport), the Ship's Safety Law (for sea transport) and the Civil Aeronautics Law (for air transport).

After the latest amendment of the law for road transport in 1978, the relevant regulations and ordinances were amended and came into force in March, 1979. At the same time, the other laws for sea and air transport were amended and adjusted. The latest jurisdiction scheme is shown in Fig. 1.

(3) Licensing

Two steps of confirmation by the competent authorities are required before shipment, i.e. "the Confirmation of Package" and "the Confirmation of Transportation Method". The former confirmation consists of package design approval, packaging registration and confirmation of transportation. The latter includes compliance of Transport Index (TI) and compliance of loading. Application and approval procedure flow is shown in Fig. 2, as in the case of type B and fissile class nuclear material packages.

Designs for type A fissile class packages and for type B packages are sometimes required to be examined at the Technical Advisory Committee on Transport Package Safety, which is composed of 14 advisors appointed by the Minister of STA. The committee is made up of 4 Analysis Subcommittees, i.e. subcommittees on structure safety and material properties, on thermal safety, on containment safety and on shielding and criticality safety.

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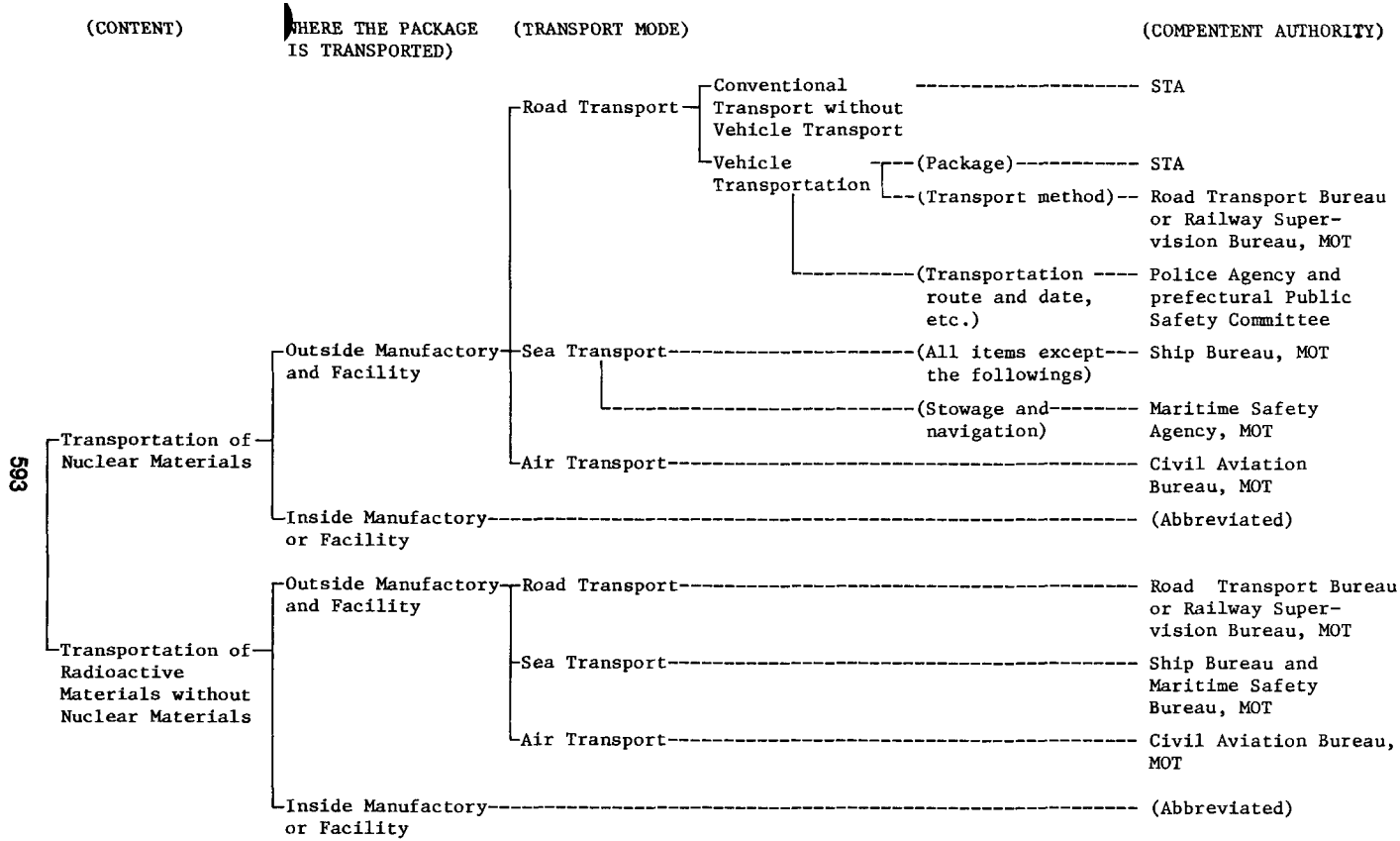
* Ministry of Transport



Note (1): STA; Science and Technology Agency, MOT; Ministry of Transport
Note (2): General information are available from MOT and National Police Agency.

Fig. 1 The Latest Jurisdiction Scheme and Competent Authorities in Japan

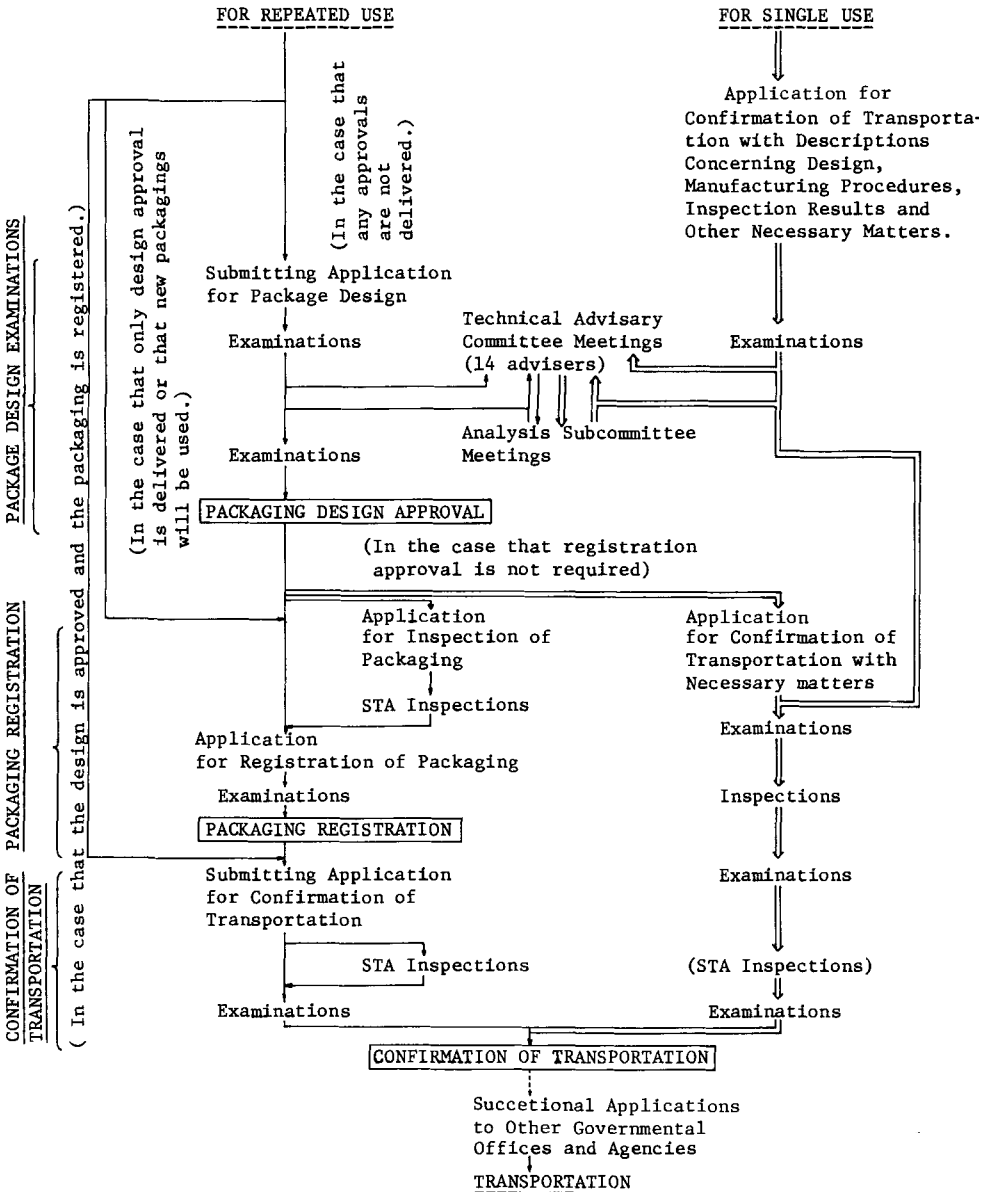


Fig. 2 Flow chart - Application and Approval for Type Band Fissile Class Nuclear Material Packages under the Jurisdiction of STA

4. TRANSPORT RISK ASSESSMENT

(1) Sea Transport of Spent Fuels

In 1978, safety assessment reports on sea transport of spent fuels in Japan were presented to JAEC[1], this assessments have been performed in the followings;

1) Investigations of marine accidents,
2) Probability of submersion of ships,
3) Leakage mechanism of radioactive materials and
4) Assessment of the exposure dosage to the general public.

Safety evaluations have resulted in the conclusion that, in the case of a cask submerged to a depth not less than 200 m, the exposure dosage of the general public is lower than the permissible exposure dosage of the general public at ordinary times set by the ICRP.

(2) Road Transport of Nuclear Fuels

In 1979 and 1980, risk assessment of road transport of nuclear fuel materials (UO_2, UF_6 and LWR fuel assemblies) has been performed.[2]

Radiation exposure under normal condition has been evaluated in terms of materials and shipping routes. And radiation dose received in accidents has been evaluated as follows.

1) Investigation of the probability of the traffic accident in domestic routes,
2) Estimation of the possible leakage of radioactive materials in accidents, and
3) Estimation of internal and external radiation exposure

The result under normal transport conditions was found to be sufficiently low, compared to the annual collective dose by natural radioactivity in Japan.

Radiation exposure has been evaluated using leakage model under severe accident condition. It is shown that individual dose is less than the dose limit recommended by ICRP and the collective dose is less than that from natural background. The detailed presentation will be made in this symposium.

5. SAFETY ANALYSIS AND TESTS FOR PACKAGE

Computer programs for analyses on integrity, criticality and shielding of packages have been developed in two institutes, JAERI* and CRIEPI**.

In addition, the basic researches of safety evaluations of packages, such as fire tests, drop tests and shielding tests, are also being carried out in some national laboratories, MEL***, FRL**** and SRI*****.[3],[4]

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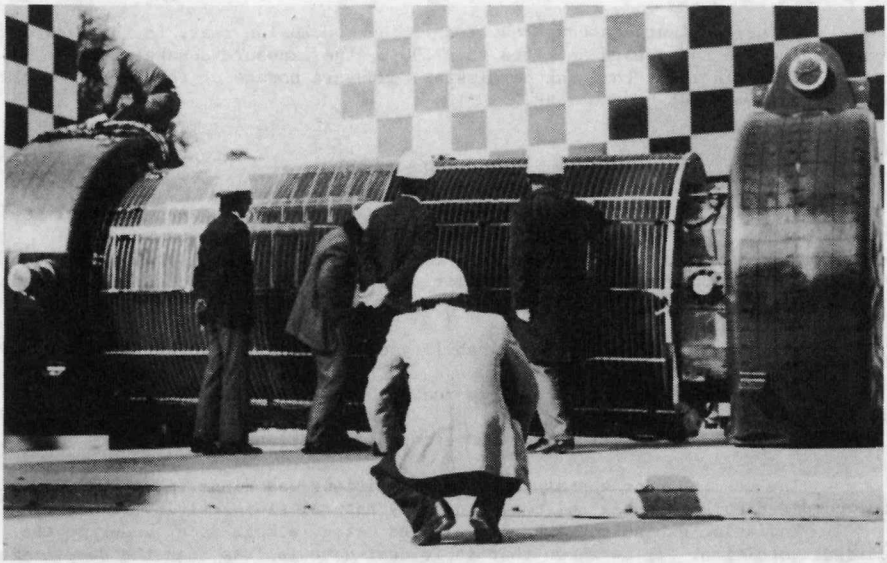
* Japan Atomic Energy Research Institute
** Central Research Institute of Electric Power Industry
*** Mechanical Engineering Laboratory
**** Fire Research Laboratory
***** Ship Research Institute

The demonstration test programs on the reliability of spent fuel package have been conducted by CRIEPI under the contract with STA. The programs include 1) 9 meter-drop test, 2) 800°C fire test, 3) shielding test, and 4) pressure resistance test using 60 t and 100 t spent fuel packages. The 9 meter-drop tests were carried out on a 60 t package with its longitudinal axis horizontal and vertical (Photo 1). The results showed that deceleration and deformation were in good agreement with those analytically obtained and the leaktightness of the package after the tests was retained.(5)

Photo. 1 Horizontal Drop Impact Test of 60 t Spent Fuel Cask



[REFERENCES]

(1) Advisory Committee on Nuclear Fuel Safety (Atomic Energy Commission): "Safety Assessment on Sea Transport of Spent Fuel in Japan", Sept. 1978.

(2) H. Shikata, et al.: "Safety Analysis on Transportation of Radioactive Materials by Truck in Japan" (in this symposium)

(3) S. Shimamura, et al.: "Impact Resistance of Shipping Containers for Radioactive Materials under Free Rall Drop Test" (in this symposium)

(4) H. Yamakoshi, et al.: "A Handy Calculation Method of Transport Index and Radiation Spectrum Outside of Cask" (in this symposium)

(5) T. Nagakura, et al.: "Test for Demonstrating the Reliability of Containers Used for Transport of Spent Fuel - Preliminary Test of 50-ton Class Spent Fuel Transporting Containers for their Impact Value, Obtained by Dropping them from a Height of 9 Meters" (in this symposium)

MANAGEMENT OF RADIOACTIVE MATERIALS TRANSPORTATION SAFETY FOR THE U.S. DEPARTMENT OF ENERGY - OAK RIDGE OPERATIONS OFFICE

William A. Pryor
U.S. Department of Energy

Oak Ridge Operations (ORO) is the largest and most diverse of the six field offices of the U.S. Department of Energy (USDOE). This office is responsible for production, research, and development associated with uranium enrichment, fuel elements for production reactors, isotopes, and national defense. Thus, the packaging and transport of radioactive materials are major activities associated with the operation of these facilities. ORO utilizes 22 different Type B packages for which Certificates of Compliance have been issued. These are in addition to the U.S. Department of Transportation Type B Specification Packages which are used for fissile and large quantity shipments. A variety of Type A packages are used to ship all forms of radioactive materials. Continuity in transport is maintained through a strong safety program to develop new package designs, to assure adherence to domestic and international transport regulations, and to provide radiological assistance when requested.

U.S. EXPERIENCE IN URANIUM HEXAFLUORIDE TRANSPORT

In 1966, following a two-year development and test program by Oak Ridge Gaseous Diffusion Plant (ORGDP) (operated for USDOE by Union Carbide Corporation-Nuclear Division), the family of overpacks with their contained 5-inch, 8-inch, 12-inch, and 30-inch-diameter Uranium Hexafluoride (UF_6) cylinders were approved for transport.[1,2,3] Up until this time, the largest diameter cylinder used for transporting enriched UF_6 for commercial application was 12-inch. It is significant to note that the first international shipment of leased enriched UF_6 in a 30-inch cylinder with a 4,800 lb. UF_6 capacity in an overpack was shipped from ORGDP on August 19, 1966. Its destination was the Federal Republic of Germany to fuel the nuclear power plant at Obrigheim. Since the overpack was too large to hide, it was painted in the tri-colors of red, white, and blue. On each side, in both German and English, were stenciled, "Enriched Uranium Hexafluoride for Federal Republic of Germany KWO Nuclear Power Station from USAEC" and "Safe Economical Nuclear Power." Figure 1 shows this shipment.

The enriched UF_6 for this nuclear power station was shipped in 32 30-inch-diameter cylinders. The same 2,600 pounds of ^{235}U at 2 percent enrichment would have required 340 of the smaller 12-inch-diameter cylinders. Thus, we can conclude that both safety and economy were enhanced in this arrangement.

Following this milestone in radioactive materials transport, on January 1, 1969, the United States established its Toll Enrichment Program to supply slightly enriched uranium for the world's nuclear power plants. Under toll enrichment, the customer first supplies natural UF_6 to one of the three GDPs as directed by USDOE. The UF_6 is fed into the cascades, enriched to the customer's requirements, and is withdrawn into the customer-owned 30-inch-diameter cylinders. The cylinders are then loaded into customer-owned overpacks for transport.

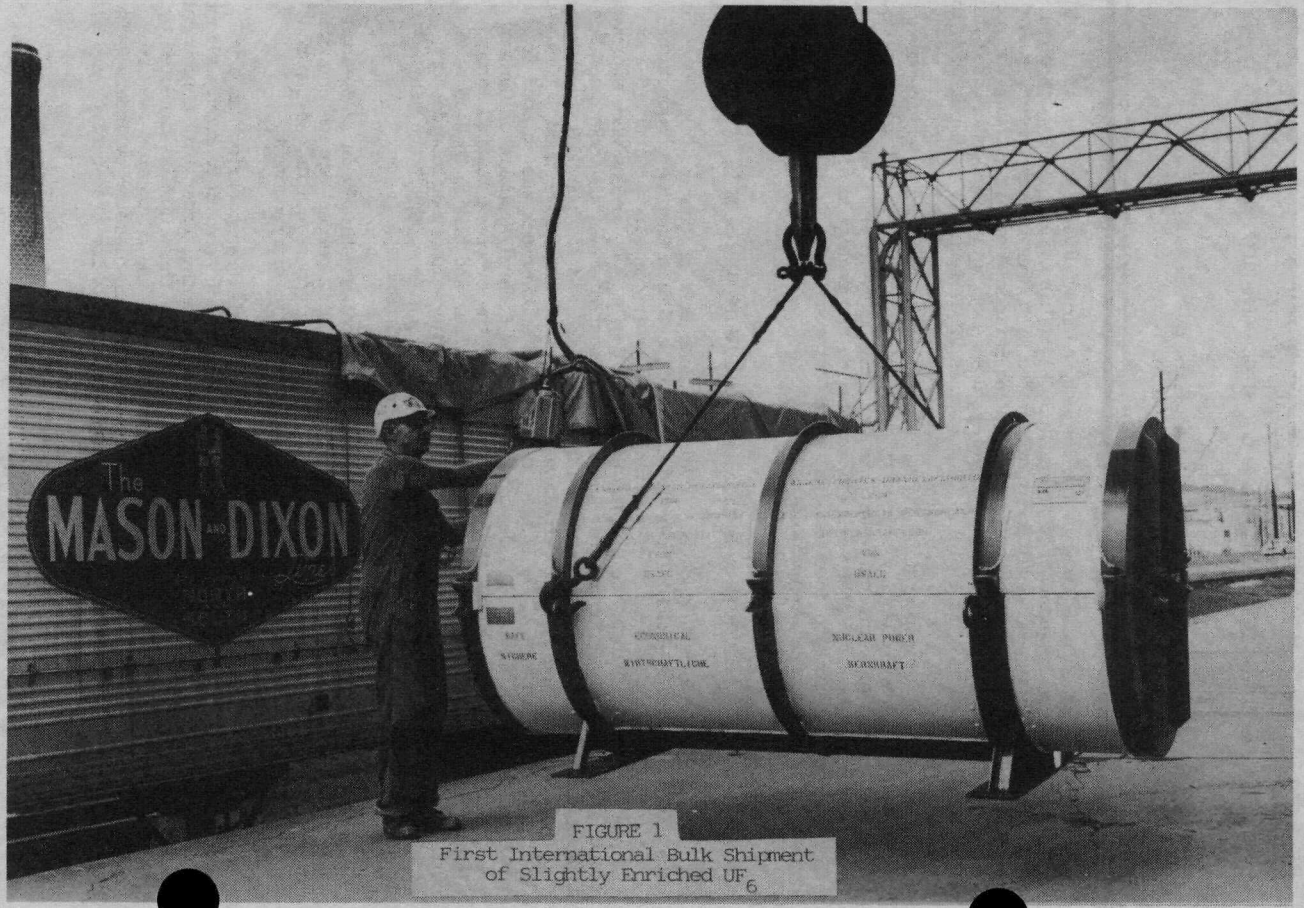


FIGURE 1
First International Bulk Shipment
of Slightly Enriched UF_6

The first eleven years of toll enrichment are summarized in Table I below:

TABLE I

Eleven Years of Toll Enrichment

	Natural U[a/ d/] Received	Enriched U[a/ e/] Shipped
Metric Tons	146,000	30,000
No. of Cylinders	>20,000[b/]	>20,000[c/]

a/ As UF_6
b/ 48-inch diameter
c/ 30-inch diameter
d/ Received at 3 locations
e/ Shipped from 2 locations

In addition to these shipments, the USDOE ships slightly enriched UF_6 up to 4.5 percent ^{235}U enrichment in the interplant flows from the GDP at Paducah, Kentucky (operated for USDOE by Union Carbide Corporation-Nuclear Division), to the GDPs at Oak Ridge, Tennessee, and Portsmouth, Ohio (operated for USDOE by Goodyear Atomic Corporation), in 48-inch-diameter (14-ton) cylinders in "Paducah Tiger Overpacks."[4] These overpacks have also been licensed by the U.S. Nuclear Regulatory Commission (USNRC) so that non-USDOE installations may use this type of overpack.

The USDOE recently entered into a contract with Nuclear Containers, Inc., to develop a new generation overpack to accommodate the 48-inch-diameter (14-ton) UF_6 cylinders. It is anticipated that the testing of a prototype will be completed early in 1981. UF_6 up to 4.5 percent ^{235}U enrichment will also be shipped in these overpacks.

SPECIAL TEST PROGRAMS

Foam Tests

A significant part of the UF_6 overpack development program included the development of an insulating material having satisfactory properties for impact and thermal protection. ORGDP developed a phenolic foam insulation which was strengthened by the inclusion of fiberglass fibers. This insulation is described in USDOE Material and Equipment Specification SP-9, Rev. 1.

As written, the chemical ingredients are specified under manufacturer trade names and chemical grades. Since SP-9 was originally developed, some of these ingredients are no longer available. Substitutes, some of which were recommended by the manufacturers as being the nearest equivalent, have been found.

Substitutions were required as follows:

TABLE II

Ingredient Substitutions in Phenolic Foam Insulations

	Material	Original	Substitute
1.	Surfactant	Unavailable	New*
2.	Fiberglas	Unavailable	New*
3.	Boric Anhydride	Reagent Grade**	Practical Grade
4.	Oxalic Acid	Reagent Grade	Technical Grade

*Recommended by manufacturer
**Available mesh sizes also different

Two 10-lb foam samples were prepared using the substitute surfactant and fiberglass in both cases, but using the reagent grade materials as originally specified in one case and the substitute grades for the other materials. The preparation procedure was the same in both cases.

Tests were conducted on each sample followed by the measurements and observations as indicated below:[5,6]

	Operation or Test	Measurement
1.	Free Foam	Heat of reaction (Exothermic)
2.	Thermal Exposure - 10 Minutes	Depth of Charring
3.	Water Absorption	Weight of Water
4.	Corrosive Nature of Wet Foam	pH of Solution

For the shock testing, two sets of five test models were prepared simulating overpacks containing UF_6 cylinders. Each consisted of a 5-gallon bucket, heavy wall steel pipe, and phenolic foam insulation. One set of five test models contained one type of insulation, and the second set contained the other type. The weight of each model was about 25 lbs.

Accelerometers were attached to each model, one on the inside of the pipe and the other on the outside. Free drops were made from heights of 10 feet onto a concrete floor with the longitudinal center line parallel to the floor and at a slight angle to the floor.

Comparisons were made of average values from the shock pulse time histories for five drops for each type of insulation which included:

Maximum positive Acceleration of Pulse Outside and Inside.

Positive Pulse Length - Outside and Inside.

Rise Time to Positive Maximum - Outside and Inside.

Positive Maximum Peak to Negative
Minimum Peak Acceleration - Outside and Inside

The results of these tests indicate the characteristics of the two foams are essentially the same; thus, the foam made with the substitute ingredients is now qualified. The test reports are now part of the specification.

LOW TEMPERATURE DROP TESTS

The operation of the gaseous diffusion complex is affected by a variety of conditions, one of the more significant being the utilization of electrical power. The USDOE has periodically cut back on the power procurement. In a low power configuration, there is reduction in the natural UF_6 fed to the GDPs. Thus beginning in early 1979, the incoming natural UF_6 from toll enrichment customers, which is normally fed within 90 days after arrival, was being stored. Since this UF_6 is shipped in customer-owned cylinders, a shortage of these cylinders was rapidly being created. To prevent the problem from becoming one of major proportion, the USDOE proposed to provide USDOE-owned cylinders to the UF_6 suppliers for shipment to the GDPs. This would eliminate the necessity of transferring the UF_6 on arrival to USDOE-owned cylinders and would minimize the holdup of customer-owned cylinders. The USDOE planned to supply cylinders which are normally used for the storage of UF_6 from the GDPs.

The domestic UF_6 suppliers are licensed by the USNRC, and the requirements of Title 10 CFR Part 71, "Packaging of Radioactive Materials for Transport," which specify a Type A package would apply to these licensed shipments.

The available cylinders were fabricated from ASTM A-285 Grade C steel. The USNRC was concerned about brittle fracture at low temperature. Thus, a test program was designed to test a Model 48 HX cylinder against the Type A drop requirements.[7] Since the gross weight of a loaded cylinder is greater than 30,000 lb., the required drop height is one foot.

For the test, depleted uranium metal and ice simulated the UF_6. The cylinder was immersed in a trichloroethane-dry ice bath to reach the test temperature. The inside temperatures reached -100°F. With an ambient temperature of about 20°F, the cylinder was removed and suspended over the drop pad at an angle of 35°. The lowest part of the cylinder was one foot above the drop pad. The cylinder shell in the vicinity of the point of impact was permitted to "warm up" to -20°F before the drop. There was some flattening in the impact area, however, there was not any failure of cylinder integrity. Deflection measurements were taken along the impact area. The cylinder was again returned to the bath.

A second one-foot drop was made with the cylinder in a horizontal position and with the lower skin temperature at -33°F. Deflection measurements were taken, and again, the cylinder retained its integrity. The deformation indicated that the steel will plastically yield at or below -20°F without the onset of a brittle fracture under test conditions.

A third drop at a 35° angle from a height of four feet resulted in the failure of the cylinder, however, the calculated energy of this drop was about four times that of the horizontal one-foot drop.

The USNRC issued Certificate of Compliance USA/9090/A for this cylinder. A restriction was included stating that shipments would be made only when the ambient temperature is greater than 32°F.

RADIOLOGICAL ASSISTANCE IN THE TRANSPORTATION ENVIRONMENT

The DOE maintains a Radiological Assistance Program to provide expert assistance in the event of emergencies involving radioactive materials. A large portion of this assistance involves radioactive materials in a transportation environment. The three incidents involving DOE-ORO radiological assistance described below ranged from providing a response team to assisting in the determination that no problem existed. No radiation injury to the public, to transportation workers or to emergency personnel resulted in any of the incidents. Some of the more subtle aspects of radiological assistance are described.

Incident No. 1

Location: Near Rockingham, North Carolina
Date: March 31, 1977
Transport Mode: Rail
Radioactive Material and Form: Natural uranium as solid UF_6 in four 14-ton capacity bare cylinders
Incident: Derailment with fire adjacent to one cylinder.

Twenty-nine cars of a 102-car train derailed including two flat cars each loaded with two trailers with a single UF_6 cylinder per trailer. One cylinder was adjacent to a burning boxcar. Two reports were received that (1) a military unit had monitored the cylinder and had found alpha contamination, and (2) a news reporter had been burned by UF_6 which had been released from a ruptured cylinder. Since personnel were able to come in direct contact with the cylinders (1) without detecting unusual odors (HF gas), (2) without seeing any yellowish-white solid or vapors, and (3) since the supposedly "released UF_6" was a brown-black color, it was very unlikely that any UF_6 had been released at all. The USDOE team dispatched to the site verified that no release had occurred. The team also assisted in the recovery of the cylinders.

Incident No. 2

Location: Sioux City, Iowa
Date: September 30, 1978 (Saturday)
Transport Mode: Air, Passenger
Radioactive Material and Form: ^{99}Mo solid
Incident: Wet Type A package identified as containing radioactive material, discovery made in the cargo terminal after departure of aircraft.

The call for assistance via Chemtrec (Manufacturing Chemists Association's Chemical Transportation Emergency Center) was received on a weekend during nonworking hours. All aspects were handled by telephone with tie-in to the USDOE-ORO Communication Center. After several calls, direct contact was made with airline's flight control center to obtain details and to answer their questions. The center requested advice as to whether the aircraft should: (1) return to Sioux City, Iowa, (2) be diverted to an alternate airport, or (3) be permitted to continue to its next destination.

In order to evaluate the situation, it was necessary to determine from the manifest and shipping papers: the isotope involved, form, quantity, and Transport Index. After this information was obtained, it appeared obvious that the wetness was not due to any release of the radioactive isotope which was solid. There had been a rain shower at the airport where the shipment originated. The flight control center was told that there was no reason to divert the aircraft from its

destination. However, as a further precaution, arrangements were made to have both the package and aircraft monitored on arrival. Since the next destination was outside the ORO Radiological Assistance Region, the situation was transferred appropriately to DOE's Chicago Region for followup. It was verified that no release had occurred.

Incident No. 3

Location: Knoxville, Tennessee
Date: February 13, 1976
Transport Mode: Air, Passenger
Radioactive Material and Form ^{133}Xe Gas
Incident: Package of radioactive material found to have high radiation levels.

An airline cargo handler, trained to monitor packages of radioactive material, discovered one Type A package out of several packages with unusually high radiation levels. The shipper, an ORO contractor, was informed and the package was retrieved prior to loading on the aircraft. When the package was opened for inspection, a hairline crack was discovered in the glass ampule containing the ^{133}Xe gas. The crack was probably caused by heat stress when the ampule was flame sealed. Although the gas had diffused through the crack, it had been retained by the packing inside the sealed can, but was outside the lead sheet used for radiation shielding. Improvements have been made in the filling operation and the packaging to prevent recurrence.

REFERENCES

1. A. J. Mallett and C. E. Newlon, Protective Shipping Packages for 30-Inch Diameter UF_6 Cylinders, K-1686 (April 13, 1967).

2. A. J. Mallett and C. E. Newlon, Protective Shipping Packages for 5-Inch Diameter UF_6 Cylinders, K-1716 (June 14, 1967).

3. A. J. Mallett and C. E. Newlon, Protective Shipping Packages for 8-Inch and 12-Inch Diameter UF_6 Cylinders, K-1714 (April 20, 1967).

4. D. H. Stitt, Safety Analysis Report on the "Paducah Tiger" Protective Overpacks for 10-Ton Cylinders of Uranium Hexafluoride, KY-668, (June 16, 1975).

5. C. E. Daugherty, G. E. Harris, and R. R. Wright, Evaluation of Material Used in Fire - Resistant Phenolic Foam, K/TL-729 (March 1, 1978).

6. C. R. Barlow and W. D. Strunk, Cylinder Overpack Phenolic Foam Shock Testing, K/P-6567 (April 20, 1980).

7. S. C. Blue, T. R. Odom, and D. H. Stitt, License Application for Type 48 HX Cylinder for Shipment of Natural Assay UF_6, KY-697 (February 16, 1979).

DISCUSSION

Question by D. H. Lohmann, United Kingdom: In incident no. 2 you report that cargo terminal staff reported a package containing a solid had been found to be leaking. Subsequent investigation showed that this was rain water. Do you get many false alarms of this type?

Answer: Yes, the most frequent type of radiological assistance requests, particularly from the airlines, are:
1. Wet packages suspected of leaking radioactive contents. These a[illegible] mostly type A.
2. Packages which have been damaged by ground handling equipment - also mostly type A.

Question by D. Haon, France: Have you performed thermal tests on UF_6 packages? You have not mentioned this subject in your paper.

Answer: Yes, two types of thermal tests on UF_6 packages have been performed:
1. For the family of UF_6 cylinders in type B overpacks;
2. Tests on small, bare UF_6 cylinders.

We recognised that bare UF_6 cylinders at elevated temperatures will fail due to hydraulic expansion. At even higher temperatures, UF_6 is highly reactive particularly with organic materials such as hydrocarbon oils, explosive reactions will occur.

EXPERIENCE OF A MANUFACTURER IN QUALITY ASSURANCE

by Henri BRUEL, ROBATEL SLPI (France)

INTRODUCTION

You surely guess all the new problems set by the application of a Quality Assurance system to a manufacturer already experienced in spent fuel transport packagings.

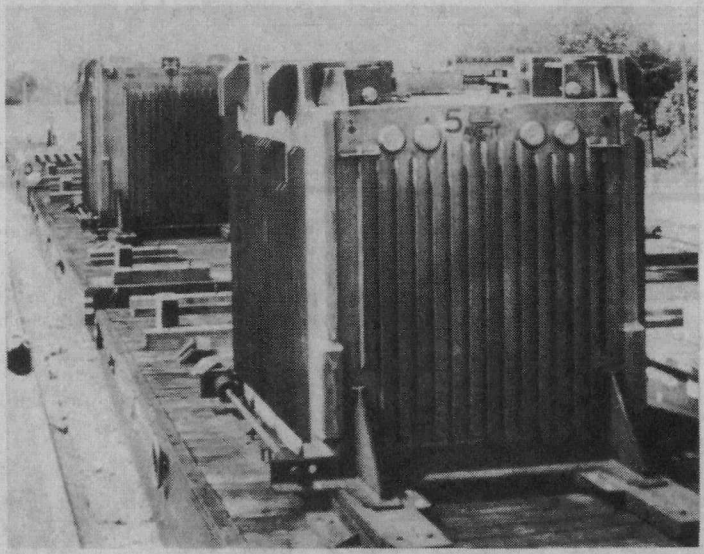
We wish to analyse these problems, the difficulties to be avoided and the corresponding conclusions.

First of all and for setting this experience, let me introduce our firm : ROBATEL SLPI, 400 employees, a modern plant built in 1970 near Lyons (France).

PACKAGING FOR PWR SPENT FUEL ELEMENTS

As soon as 1953, the firm started to work for the nuclear industry, manufacturing lead-clad steel 'coffins' (so were called the packagings at this time) due to transport the first radioactive materials. ROBATEL SLPI carried this transition out without any problem as their workshops were already used to supply very elaborated and controlled equipment such as centrifuges for the chemical industry and lead-clad steel tanks for liquid bromine transportation.

These last years, our firm manufactured and sold 33 type B packagings each weighing more than 20 tons, made of lead-clad steel.



50 TONS PACKAGINGS FOR GRAPHITE-GAS ELEMENTS

Most of above packagings are still circulating on roads and railways and, to our knowledge, none of them has ever caused human injury in spite of serious shocks in handling operations or road accidents.

That leads to the conclusion that the IAEA regulations, the conception of our own packagings, their manufacture, the severe controls performed by official organisms and the proper organization of the transports all together answer the safety rules imposed by the dangers that many people link to the nuclear field.

QUALITY ASSURANCE

Nevertheless, in 1974, we had to complete our usual quality control system and add the requirements of the US Quality Assurance.

In fact, ROBATEL SLPI won from the TRANSNUCLEAR group, an important order for the manufacture of several 33 ton lead-clad steel packagings for the US territory. It was the start of our relations with the US Nuclear Regulatory Commission which, during a visit to our plant, gave us some advice for the redaction of our Quality Assurance Manual (QAM). The setting of this QAM was rather long and difficult as we had to introduce in our former organization the 1 criterion of the 10 CFR Part 71, Appendix E.

Once the QAM had been approved by the NRC, a Quality Assurance Program (QAP) applied to this specific job had to be set.

Only then could be started the detailed design and the materials purchases, in conformity with Section III of the ASME Code. The first controls of the NRC took place in our works in May 1976,

i.e. two years after our first contact with this organism.

This means that the factor 'time' must be taken into account in the application of a Quality Assurance Program.

We must also say that the packaging was delivered with 2,200 kg of documents (some of them bearing the index 14) required for the compulsory 15 year recording of the 18 criterion. They tell the extremely detailed story of all the packaging manufacture.

CONTROLS OF THE N.R.C.

Our works received three times the visit of the N.R.C. agents, their main purpose being to compare the written procedures with the actual operations involved in the packaging manufacture.

Every single failure or anomaly is noted as a deviation. They are confirmed within the month following the inspection and the same delay is granted for answering the N.R.C., stating :

1. the corrective action intended to delete the anomaly
2. the action taken to prevent such a deviation to happen again
3. the dates expected for the implementation of both actions
4. the person responsible for this implementation.

This system is very clear and efficient but also motivating as the NRC publishes the results of these inspections :

- in its 'Public Document Room' at Washington, after the firm has been asked if this will not affect its industrial property rights

- and on a quaterly 'white book' addressed to all the manufacturers concerned and summarizing the results obtained by the firms inspected during the previous quater.

This last publication enables the firm to compare its own results to the average ones and to conclude whether it still has to progress or it has learnt its QA lesson ...

That is for the method which effiency is, in our opinion, illustrated by the results obtained by ROBATEL SLPI :

in May 1976	8 deviations
in June 1977	5 deviations
in July 1980	1 deviation

As far as the analysis of the deviations is concerned, we noted that, some of them are important enough to affect the equipment quality, others, based on the strict comparison between the written documents and their application, lead to the revision of the procedures themselves.

For example, one of our welders worked under an intensity of 120 amperes while the corresponding procedure set a maximum 100 amperes intensity. After examination, the welder was found right for that kind of welding and the corrective action consisted in enlarging the compulsory range of intensity. Of course, as required by Section IX of the ASME Code, this modification was duly documented.

An important care of the NRC was to exclude all corrosive agents and we had to prove that the dye-penetrant test and marking products were absolutely free of halogens.

The measurement and control instruments have also been inspected by the NRC agents. For example, the inspectors noted that the thermometer measuring the temperature of the stove for storing welding rods was not gauged as periodically required, even though the permissibile temperature varied from 90 to 150°C !

Then, I will recommend to the manufacturers to set a minimum list of measurement and control instruments, stating only those in direct connection with the equipment quality. The periodical gauging of these instruments is very expensive : for an ordinary double-meter, it costs ten times its purchase price.

At last, we have been asked for an important modification of our general organization set in our Manual : the disengagement of the Inspection Section form the hierarchical subordination of the Workshop Manager as the latter could hardly supervise both the manufacturing operations and their control.

THE LESSONS OF EXPERIENCE

It is undeniable that the purpose of the NRC inspections is to improve the final quality of the equipment and they lead to some improvements also in our methods of working.

But here is another thing that seems important to us after our experience and that illustrates the maxim "Leave well alone" :

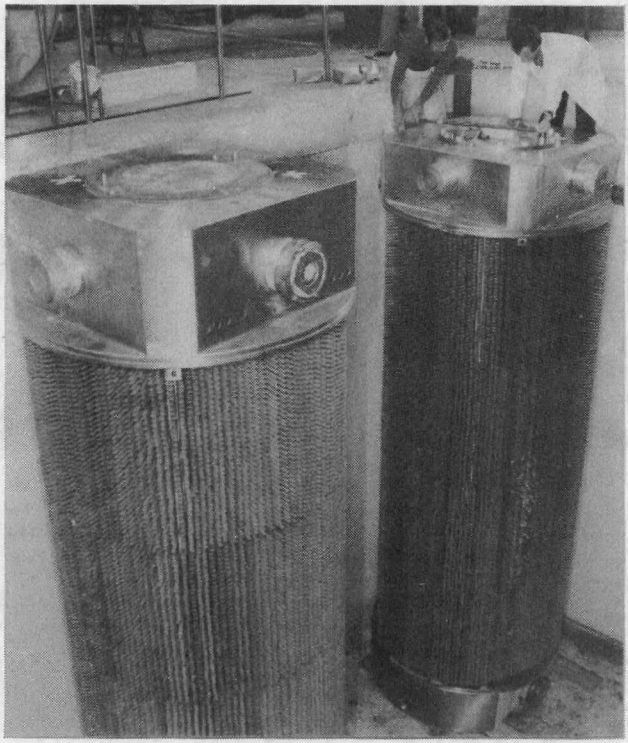
- it took us 4 years for manufacturing a packaging with a contractual delivery of 18 months, term agreed upon before we knew the requirements of the American QA system,

- the price increase is very important and out of measure compared to the small percentage generally allotted by some customers.

In our opinion, it is essential to define the level of quality to be reached. We all know that a 100 % safety does not exist. Aren't we presently setting very heavy QA systems for trying to approach these 100 % ?

The answer belongs to the users themselves which must face :

- the increased cost
- and the safety criterion imposed by the competent governmental agencies.



TWO PACKAGINGS UNDER CONTROL IN OUR POOL

Coming back to our own case, this experience lead us to set a new QAM as complete as the previous one and referring to the same 18 criterion, but offering the possibility - according to the level of quality selected by the customer - to apply only the required items.

I would also like to recommend to the future QA manufacturers to take care of two difficulties while they write their QAP :

. if the customer has to approve the manufacturing drawings, take the time necessary for this approval off the contractual delivery term

. give to a Technical Committee the right to decide - with all the required documentation, of course - to raise a non-conformity after measures have been taken to cope with it, but without waiting for the customer's agreement.

Both points are in fact the main reasons for late deliveries.

TN.9 TYPE PACKAGING LEAVING OUR WORKSHOPS

CONCLUSIONS

Generally speaking and as a conclusion, we wish that Quality Assurance be subject of international standard.

In fact, it is certain that Quality Assurance requirements vary according to the users or even to the Nations.

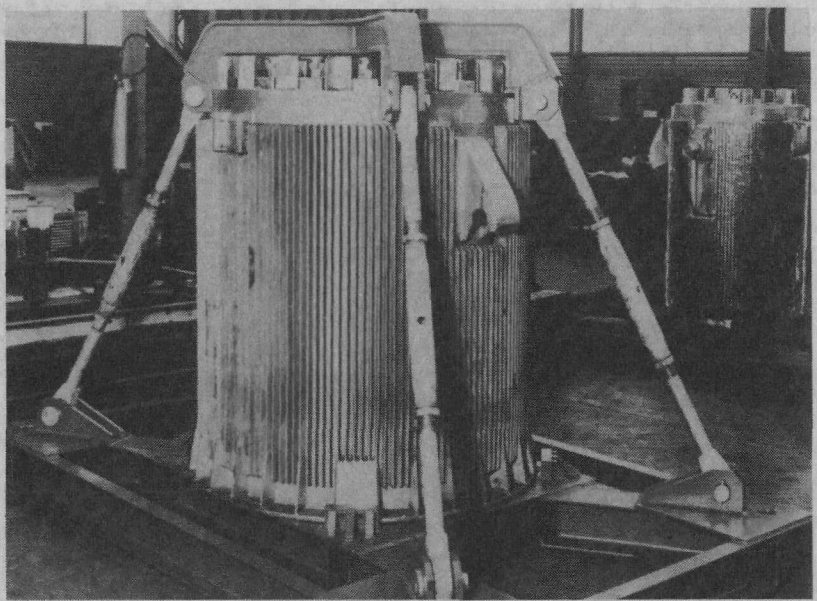
Some organisms are already working in Europe to set a common terminology. In France, a Quality Assurance system applied only to spent fuel transport packagings is under study.

An efficient job would be done if all these eminent specialists agreed to set a common document, avoiding nevertheless an easy solution that would consist in gathering all national requirements without considering the factors of costs and time.

A document "à la carte" would allow grown-up (i.e. competent and responsible) users to choose the level of quality to be reached by their products.

To be logical about safety requirements, not only the manufacture of the packaging should be controlled, but also its service. Periodical examinations would be necessary for checking the initial properties of the packaging after some servicing time.

As far as ROBATEL SLPI is concerned, we carry on our efforts towards the conception and the manufacture of reliable packagings offering the advantage to have already long lived and references.



" PEGASE " TYPE PACKAGINGS

These 'PEGASE' type packagings, made of lead-clad steel, were manufactured by ROBATEL SLPI and are still on use for the transportation of MTR fuel elements.

Twenty years later, ROBATEL SLPI still manufactures lead-clad steel packagings and this technique remains up-to-date, in our opinion.

In fact, it seems that recently the steel shieldings have known a great development, probably because of a bad result obtained during a fire test on a lead packaging. We can assure that it has been performed on a model of bad design or manufacture and cannot and must not condemn the lead shielded packagings.

Those who need to be convinced should refer to the conference presented by Mr COHENDY from the French ATOMIC ENERGY COMMISSION, entitled "Resistance to worse-than-regulatory fire conditions" which details a test performed with great care and a very elaborated equipment. This test demonstrated that the lead shielding reached a maximal temperature of 173°C after an exposure of 1.5 hour to a 1,100°C fire, conditions far worse than regulatory ones.

As a conclusion, we think that in using tested technics, calling for experienced manufacturers, the future users of transport packagings will have to set adequate Quality Assurance systems satisfying, at the selected level, the triple requirements of

QUALITY, PRICE AND DELIVERY TIME.

-oo0oo-

DISCUSSION

Question by R. F. Barker, Austria: Have you an estimate of the additional costs resulting from the quality assurance requirements?

Answer: The costs can be divided into two:
1. Final correction of the quality assurance manuel, and the QA programme intended for specific fabrication. This has taken two or three engineers 18 months.
2. Then, in order to fill in all the documents the QA programme requires (inspection reports, test reports, audits), we can count on the full-time occupation of three people during the total cask construction period, i.e. about two and a half years for the first packaging of the series.

TWELVE YEARS EXPERIENCE WITH THE TRANSPORT OF RADIO-ACTIVE WASTES IN A SALT MINE

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The Gesellschaft für Strahlen- und Umweltforschung mbH München runs with its Institut für Tieflagerung the former salt mine Asse since 1965, which is situated approx. 25 kms South-East of Braunschweig.

After the final disposal of radio-active waste in deep geological formations, and especially here in salt depositories, was recognized as a safe and sensible disposal concept, the Asse Pilot Repository received amongst others the task to develop optimum techniques and methods for the transport and disposal of radio-active wastes in a salt mine, and to demonstrate this in a technical long-time test. The result of this work is to flow into the plans of a great Federal Final Repository within a salt dome.
In the fulfillment of this task more than 125 000 containers with radio-active wastes were transported from 1967 to 1978 and disposed of (Fig. 1). Individually, the following are meant:

approx. 110 00 Drums (200 a. 400 l-Drums), low level wastes

approx. 15 000 Lost Shieldings
and approx. 1 300 Drums with intermediate level waste

In 1976 the legal foundation for the disposal Licences were changed through the amendment of the Atomic Law. The Licences expiring in 1977, and 1978, could not be extended or renewed. Since the beginning of January 1979 no more wastes are disposed of. However, all research work in the shaft mine Asse is continued, which serves the investigation of geological, hydrological and radiological characteristic dates for the final disposal of radio-active wastes in a salt mine.

The continuation of disposal cannot take place before the positive conclusion of a Plan Verification Procedure. As to the necessary time for this Licence procedure, no statement can be made at the time being. Permit me to give an optimistic judging, according to which approx. 5 years are probably necessary for this procedure.

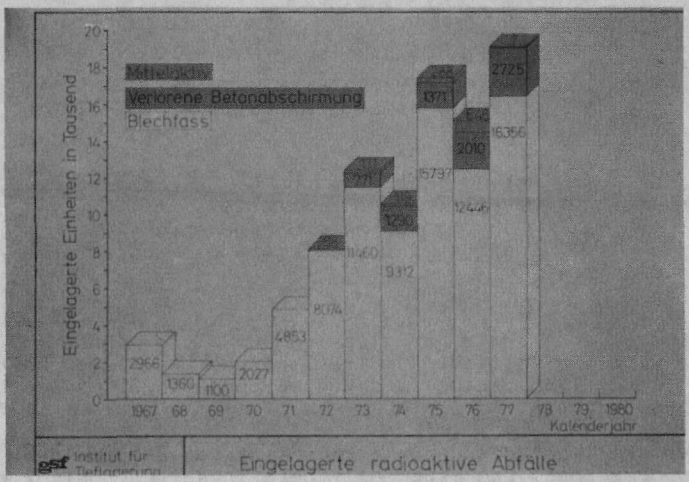

Fig. 1: Diagram of the disposed radioactive wastes

In the time from April 1967 to December 1978, it was possible to develop and test the transport systems during a period of 12 years, particularly for low-level radio-active wastes.

During a five year length of disposal practice emphasis was laid down on the development of transports above-ground. In the underground operation the transport system, which was inaugerated in 1972 was kept on with slight changes until the expiration of the Licence.

Firstly, I would like to explain the transport low-level radioactive wastes:

The transport path begins in the Shaft Hall with the taking over of the containers from the above-ground transport vehicles.

The normal drums, as well as the lost shieldings, were transported on in "House-to House" containers by the German Bundesbahn (Fig.2). Partly, concrete containers were brought to the Asse in deep-loading vehicles.

The meanwhile existing railway from the nearest station to the Asse salt mine was still being built in 1978.

For the unloading of the containers in the shaft hall a fork-lift-truck with Diesel drive, or battery-fed electric drive, was employed, which, according to requirements, was equipped with fork-arms or drum-grippers. The lifting capacity of the truck amounted to 2,5 to 3,5 to (Fig.3). On account of the high weight of the lost concrete shieldings, a 15 to Diesel truck was supplied with a loading-arm(Fig.4).

Fig. 2: Low level wastes in "House to House" containers

In order to extract the drums from the Bundesbahn containers, the drum-grippers must be very long, which leads to unfavourable load distribution, and makes heavy machines necessary in accordance.

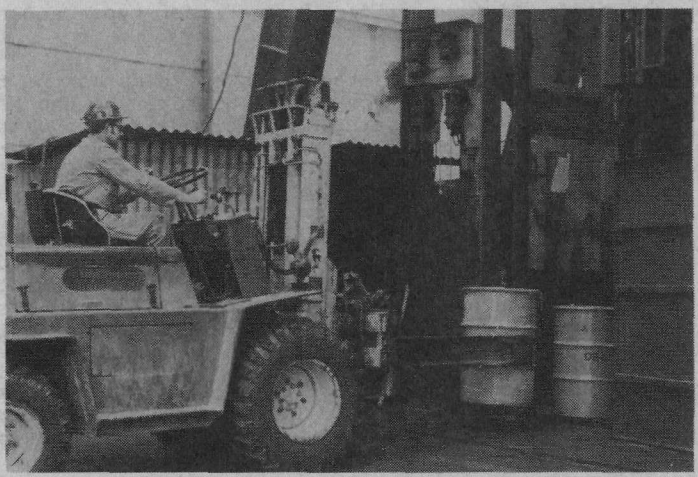

Fig. 3: Transport by fork-lift to the cage

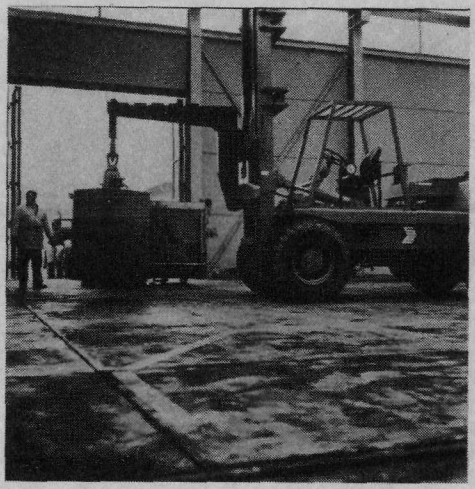

Fig. 4: Transport of lost concrete shieldings

The drums were taken over singularly and can then immediately be deposited onto the cage, or deposited on a storing spot in the shaft hall for a short period of time.

This storage spot is important, so that during the alternative delivery of drums and concrete containers a separation between the same can take place. The two different container types necessitate a re-arrangement of the cage.

The transport in the shaft shows the narrowest pass of the transport system. In this case consideration had to be taken on the Asse salt mine with reference to the existing shaft cross-section with only 5,5 m diameter. In this cross-section the hoisting system, the laid down second driving possibility, as well as the separation of the in- and outgoing air flow had to be accommodated.

In order to be able to accommodate a large transport cage in spite of the low cross-section, the conveyor plant was only fitted with one cage and a counter-weight. The cage is 1,2 wide, 2,3 m long and 5,0 m high. On this conveyor belt the transport of persons as well as also the transport of operational material and radio-active wastes must be carried out.

In order to increase the capacity for the transport of drums, two intermediate floors can be inserted. The transport of persons only takes place on the lower floor. Due to this, the transport of persons is separated from the drum transport, due to which the danger of the spreading of a possible contamination of the intermediate floor is avoided. Men-riding in cage during the running shift cannot be avoided in the operation of shaft-plants like the Asse. This is simply the result out of the various different assignments of the plant, namely place of research, disposal in a great technical

operation, and last not least, demonstrational object for a wide public, which finally led to a number of visitors of more than 10 000 persons per year. Solely for the conveying of the lost concrete shields the intermediate floors have to be removed.They are unloaded on the lower cage floor, after the bottom has been lined with a rubber matting. The rubber matting serves in the case of a contamination for an easier de-contamination and must be removed before a transport with persons takes place. The shaft entrances are secured by means of special lockings.

During a normal hoisting operation the winding-engine is interlocked in such a manner with gate control at the shaft-entrances, that the cage cannot be pulled away from the landing with open gates. A camera control of the shaft entrances supplied the conveying mechanic - aside from the assigned acustic and optical signals - via a monitor the possibility of direkt controlling of the working procedures at the shaft landing. The conveyor speed within the shaft amounts to 10 m per second, as well as for material, as also for the men-riding. The low-level radio active wastes are unloaded on the 750-m-level.

Here the following unloading devices are available:

Diesel fork-lift truck with 2,5 to lifting capacity
" " " " 5 to lifting capacity
Electro " " " 3,5 to lifting capacity
(battery-fed)

and a one-railed crane with a permissable hoisting capacity of 10 to.

The drums are unloaded with hydraulic braces out of the cage, and then immediately handed over to the ready-standing transport vehicles.

Here also it is advisable to have ready storing surfaces,in order to avoid waiting time for the shaft conveyors.

The lost concrete shieldings are unloaded with th- one-railed crane. The crab of this crane is not adjustable on the sides, but only in the longitudanal direction of the cage, the filling spot, resp. A jib of 3,23 m length makes the receival of the load possible into the cage.

For haulage underground scoop-trams are employed, which can also be utilized for loading- and transport purposes for the removal of rock salt material (Fig. 5). These mining machines are fitted with a crack steering device and permit, in spite of a construction length of 9,68 m an exterior turning radius of only 7,40 m. The driving speeds amount up to 30 km/h.

The shovel contents of the employed devices are different according to the types and run from 1,6 m^3 to 6,5 m^3. Here the general trend can be recognized, that with greater disposal amounts and ever-increasing transportation distances, the machines had to become larger in the case of new orders rendered.

Fig. 5: Transport of drums ba a scoop-tram

The greatest distance between shaft and disposal chamber amounted to 5oo m up to now, whereby the roads partly showed inclinations of 10%. In order to overcome larger distances shovel-loaders are no longer suitable. In this case pure transport vehicles with box fittings are necessary. In order to maintain the suitable mining low type of construction, the floor of the loading-box must be arranged as a plate tape.

In order to guarantee a rapid and undisturbed transport underground the floor of the roads must be given special attention. Wheras for salt transportation for short distances in a mine a road of rolled fine salt, which has been sprayed over with water is sufficient, a concrete road must be supplied for a more rapid and further transportation. In the Asse salt mine a concrete made of salt, cement and water are used. This additional material is available underground when new mining chambers are won.

The roadways normally have a cross-section of 15 m^2. They run horizontally on a level, whereas the connection galleries between the levels have to be mined with inclinations of 10 to 12%.
In order to minimize the encountering traffic as much as possible, it is often necessary, on the basis of operational measures, to separate the disposal transports from the operationally necessary salt transports. In practice this leads in the salt mine Asse thereto that disposal was carried out during the early shift, whereas the salt transports took place in the second and third shift. At certain times the disposal operation was extended into the second shift.
The unloading of the shovel loader in the disposal chambers was effected by means of simple tip up. The 200 1- and 400-1-drums are normally no longer being disposed of since 1975, but instead, these are tipped off on banks and lengthwise covered with salt. In this way, simultaneously with the transport, the disposal procedure was also solved (Fig. 6).

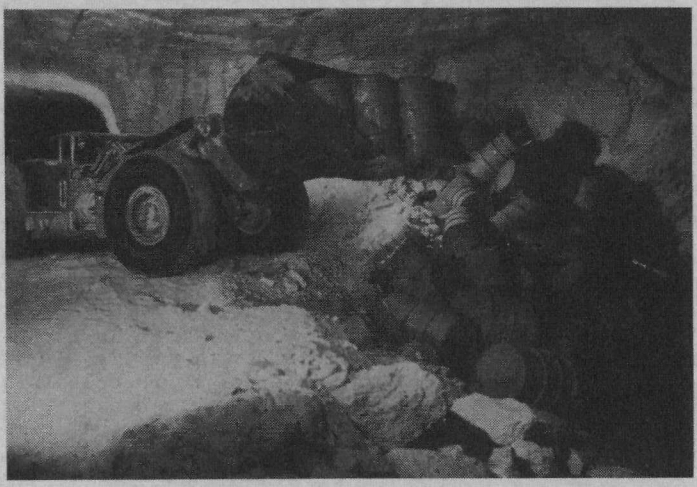

Fig. 6: Disposal of drums with low-level wastes by a scoop-tram

The lost concrete shieldings, however, had to be stapled after this (Fig. 7).

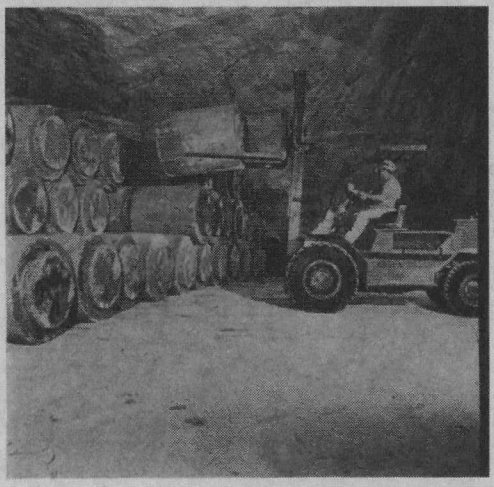

Fig. 7: Disposal of lost concrete shieldings by a fork-lift

The machines for the transport underground were normally available as mining machines for winning operations on the market. With view to their safety equipment they have all necessary devices, such as for example automatic fire-extinguishers, re-inforced brake systems etc. It must only be noted that the shovels are mad in smooth construction, i.e. do not possess protruding re-inforced ribs, or similar, in order to make a necessary de-contamination easier.

And now a short survey about the transport of intermediate level radioactive wastes:
They are solely fixed and packed in 200 l-drums, and are transported in shielded containers. Whereas during the first years the over-land transportation took place in single shielded containers, from 1975 onwards the Nuclear Research Center in Karlsruhe (KfK) developed a collected shielding container of the Type S7V, which can hold seven 200 l-drums in one interior roundabout (Fig.8). Due to the fact that the permissable transport capacity of the cage only amounts to 10 to load, a re-loading of the drums must take place in the shaft hall out of the collecting shielding container into single shielded containers (Fig.9). For movement of the loads a two-carrier-bridge crane plant is used with a crane road length of 25 m and a spreading width of 19 m. The crab possesses two separate hoisting works. One has a carrying capacity of 10 to and serves for the movement of the single shielding containers. The second lifting device with a capacity of 1 to is required for the re-loading of waste drums out of the collecting-into the single shielded container.

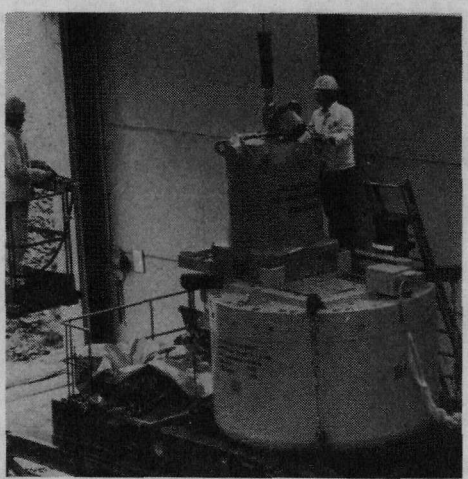

Fig. 8: Collected Shielding container of the type S7V

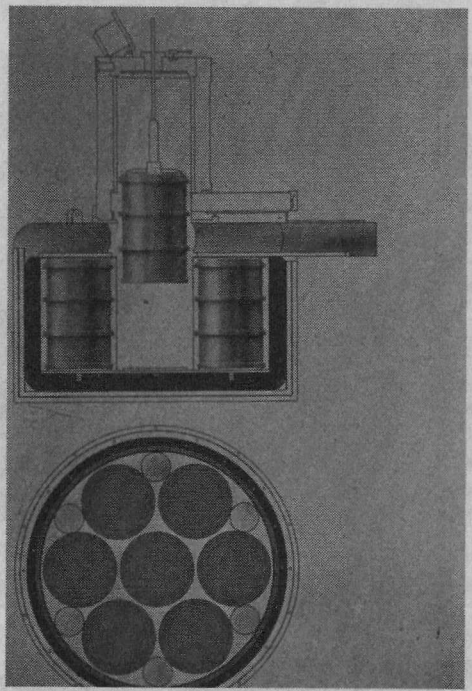

Fig. 9: Cross section through the S7V

After the change-over the single shielded container was brought to the cage on the loading arm of the 15 to-stacker. The unloading of the cage took place on the 490-m-level with the assistance of a bridge-crane.

One for this purpose re-built Magirus-truck took over the transport of the feeding chamber (Fig.10). With a double-crab which was installed there, the container was set onto the disposal bore hole. After opening of the slides the drum - by means of the crane rope - is released below out of the shielding container and deposited in the chamber.

When the drum had reached the existing drum-cone, it is electro-magnetically released and the rope retrieved. The empty shield container then had to be transported back above ground.

With the here described devices for the depositing of low- and intermediate level wastes a total of 25 000 loaded hoisting operations were carried out in the shaft in the time between 1967 and 1978.

The total stretch of the underground vehicles loaded with radioactive wastes runs up to approx. 11 000 km. For this 24 000 single drives were necessary. They were carried through to 25% in a gallary with an inclination of 10%.

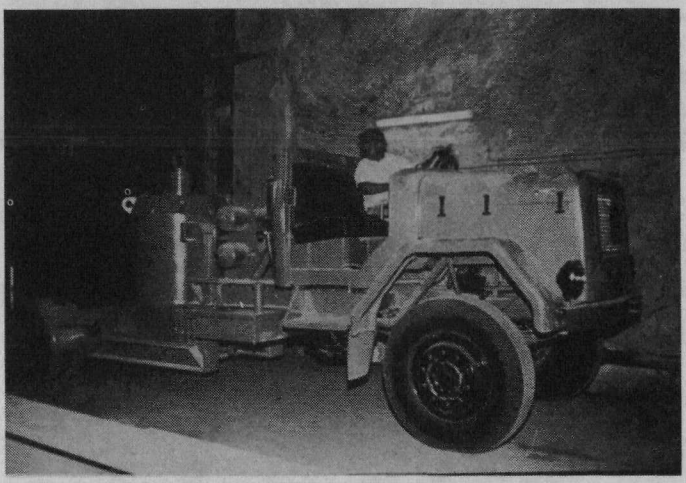

Fig. 10: Truck with one single shielded container

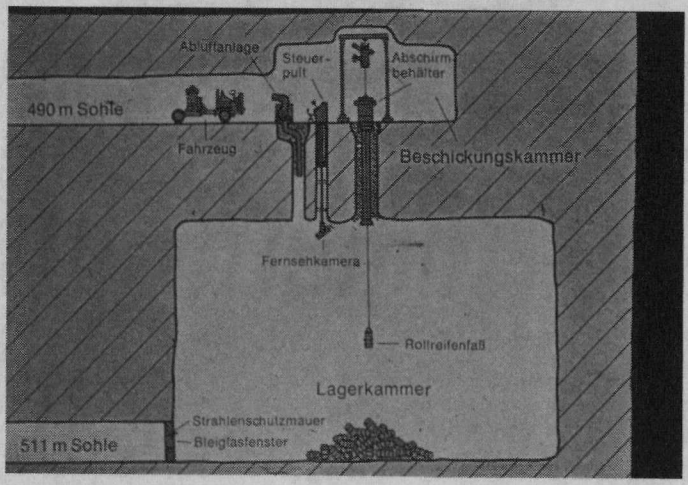

Fig.11: Cross section through the charging and disposal room for intermediate-level wastes.

At the turn-over spots in the shaft hall, in the filling spots and in the chambers approx. 400 000 single and approx. 400 000 single re-loading processes took place.

During the entire depositing time of 12 years we can gladly claim that only one hand-injury had to be reported as an accident in connection with the transport and depositing work.

All transport- and depositing processes underly the contamination surveyance. A few slight contaminations on the depositing spots and transport vehicles could be eliminated without difficulty.

The radiation dose rate of the transport personnel can only be determined with difficulty for the individual working processes. The total dosis anually rises with the number of single re-loading processes, and therewith with the number of deposited waste containers.

As in the year of 1978 30 000 containers formed the highest yearly amount which was transported in the Asse salt mine I would like to mention the dosis distribution of the entire staff for the year of 1978 as follows:

of 60 persons in question
the anual dosis in the case of 45 persons
steadily lay under the detection border.

In the case of 10 persons up to 600 mrem/Year
" " " " 0 persons up to 700 mrem/Year
" " " " 2 persons up to 800 mrem/Year
" " " " 2 persons up to 900 mrem/Year
" " " " 1 person up to 1000 mrem/Year.

The highest detectable single dosis per month amounted to 260 mrem.

The higher values came about during the unloading processes in the shaft-hall and in stacking work in the disposal room. The latter work can be charged to the depositing methods and not to the transport system, but can, however, not be extracted from the views of the dosis rate.

In summarizing it can be said that in the run of 12 years of practical depositing operation the de-veloped transportation system has proven to be fully successful. The most important expectations for this occupation, namely:

1. safe handling of the containers and machines with possible low dosis rates of the personnel,

2. undamaged containers, even with the relatively rough mining works,

3. efficiency of the transport system for the fulfillment of the waste disposal concept,

were realized.

In spite of this, in the sphere of transport of radioactive wastes in a salt mine, further research and development functions must still be undertaken:

For example: - Alteration of the re-loading mechanism at the lost concrete shieldings with the goal of further sinking of the personnel dosis rates, and for the increas e of the conveying capacity in the shaft.

- Simplifying of the time-robbing transport system for intermediate level wastes

- Development of greater Transport vehicles for longer transport roads for underground operation.

The possibility to re-arrange the shaft conveying system for larger containers does, unfortunately, not exist, because the shaft cross-section with a diameter of 5,5 m is, in comparison, small.

DISCUSSION

Question by G. Gaede, FR Germany: Did you have any corrosion problems with the different materials?

Answer: Up until now only mild steel drums have been used. The drums embedded in salt since 1965 at the 800 m level show no recognisable corrosion after 15 years. The reason is the extremely low humidity in a salt mine.

Question by K. Kirchhoff, FR Germany: 1. Do you have experience with internal corrosion caused by the waste material? 2. How do you check the containers for external dose rate and content?

Answer: 1. In a few cases corrosion was noticed from inside, caused by humidity remaining from not completely set concrete. 2. Examination of the casks was carried out only by externally us measurement processes.

EXPERIENCE WITH DRY CASK SHIPPING OF HTGR SPENT FUEL

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DENVER, COLORADO, USA

INTRODUCTION

The first significant experience for the General Atomic Company involving the use of equipment and facilities specifically designed for the dry loading, dry transport, dry unloading and dry storage of spent fuel elements from a commercial nuclear reactor was completed on October 13, 1980 when the last of 39 shipments from Fort St. Vrain arrived at the Irradiated Fuel Storage Facility.

Fort St. Vrain nuclear generating station with a capacity of 330 megawatts (electric) was designed for Public Service Company of Colorado by General Atomic Company. The Irradiated Fuel Storage Facility is a U.S. Department of Energy installation located at the Idaho National Engineering Laboratory and designed for the dry storage of eight core segments (one complete core is six segments) of spent fuel elements from Fort St. Vrain and the entire second core from the Peach Bottom I nuclear generating station. General Atomic Company previous experience in dry handling, shipping and storing of spent fuel elements was related to Peach Bottom I.

Peach Bottom I

General Atomic Company designed Peach Bottom I, a 40 megawatt (electric) High Temperature Gas Cooled Reactor (HTGR) demonstration plant for Philadelphia Electric Company. During the scheduled lifetime for this reactor, two cores of spent fuel elements were shipped to the Idaho National Engineering Laboratory for storage. Equipment and facilities normally used for the handling of spent fuel elements at Light Water Reactors (LWRs) were adapted for the handling of the spent fuel. Peach Bottom I had a storage pool but since the fuel elements with the graphite bodies could not be placed directly into the water, a remotely operated canning machine was used to seal each spent fuel element in a canister after removal from the core and before placement in the pool. During the spent fuel element loading procedure the model PB-1 cask was lowered into the pool at which time 18 spent fuel elements were placed into the PB-1 cask cavity basket, the closure was positioned in place, the cask raised to the surface of the pool and the closure bolts installed. Following this procedure the pool water was

drained from the cavity and the cask moved to the decontamination area where dry air was passed through the cask cavity to remove any remaining water. Final preparations, including a leakage test on the closure seals and a survey by the Health Physics personnel to verify compliance with all operating procedures and U.S. Department of Transportation regulations, prior to shipment were then completed. The PB-1 cask was then installed on the transport semi-trailer. A gross vehicle weight of approximately 105,000 pounds (47,627 kilograms) required that overweight permits be obtained for the 2200 mile (3540 kilometer) trip to the Idaho National Engineering Laboratory. Spent fuel from core number 1, which was shipped prior to the completion of the Irradiated Fuel Storage Facility, is stored in underground vaults. The PB-1 cask (Figure 1) has a closure at each end and a penetration through the top closure. These design features made it possible to up end the PB-1 cask on top of one of the underground vaults, open the penetration in the upper closure, attach a hoist cable to the basket, remove the lower closure and then lower the basket containing the 18 spent fuel elements into the underground storage vault. Each vault now contains 18 canistered, spent fuel elements from Peach Bottom I, Core 1 in the same basket used during transport in the PB-1 cask.

Spent fuel elements from Core 2 were shipped subsequent to the completion of the Irradiated Fuel Storage Facility and were unloaded from the PB-1 cask in the fuel handling core, removed from the individual canisters and placed in quantities of 12 in facility canisters for storage. The Peach Bottom I fuel elements (Figure 2) are 3.5 inches in diameter by 144 inches long (8.9 centimeters by 367.8 centimeters) and the storage canisters which were made from schedule 10 pipe are 18 inches in diameter (45.7 centimeters) have a cavity length of 129 inches (327.7 centimeters), thus it was necessary to remove the spent fuel elements from the canister and further to remove a portion of the graphite reflector from the top end of the fuel element by use of a remotely operated band saw. The spent fuel elements with a modified length of 126 inches (320 centimeters) were then placed in the storage canisters for transfer from the fuel handling cave into the storage room. It required 44 trips with the PB-1 cask to deliver the spent fuel elements from each core to the storage facility or a total of 88 shipments for the two cores. These shipments were made without any radiation or contamination problems and a minimum of transportation problems. The few troubles encountered during these two spent fuel shipping operations were due in part to travel restrictions because of the overweight permits. Some states do not allow overweight vehicles to operate during the hours of darkness and others stop travel by overweight vehicles on weekends and holidays. During the spring of the year certain roads were closed to overweight vehicles until the ground frozen during the winter has completely thawed. Most of the remaining troubles were related to maintaining the band saw in the fuel handling cave with a few delays due to other equipment problems, some at the reactor and some in transit (tractor or semi-trailer).

Fort St. Vrain

The second HTGR designed by General Atomic Company is the Fort St. Vrain nuclear generating station, a 330 megawatt (electric) plant owned and operated by Public Service Company of Colorado. Many significant design changes were incorporated in the design of this second HTGR, based on the experience gained from Peach Bottom I. Of particular interest are those involving the facilities and equipment related to the handling, transport and storage of spent fuel elements. In every case the equipment and facilities were specifically designed to avoid the use of a storage pool or any equipment (modified) normally used in conjunction with a storage pool.

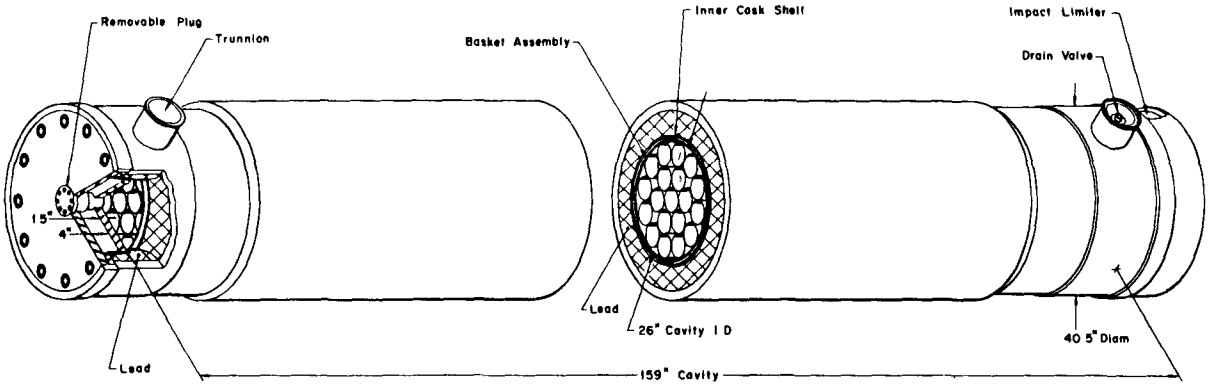

Figure 1. Model PB-1 Cask

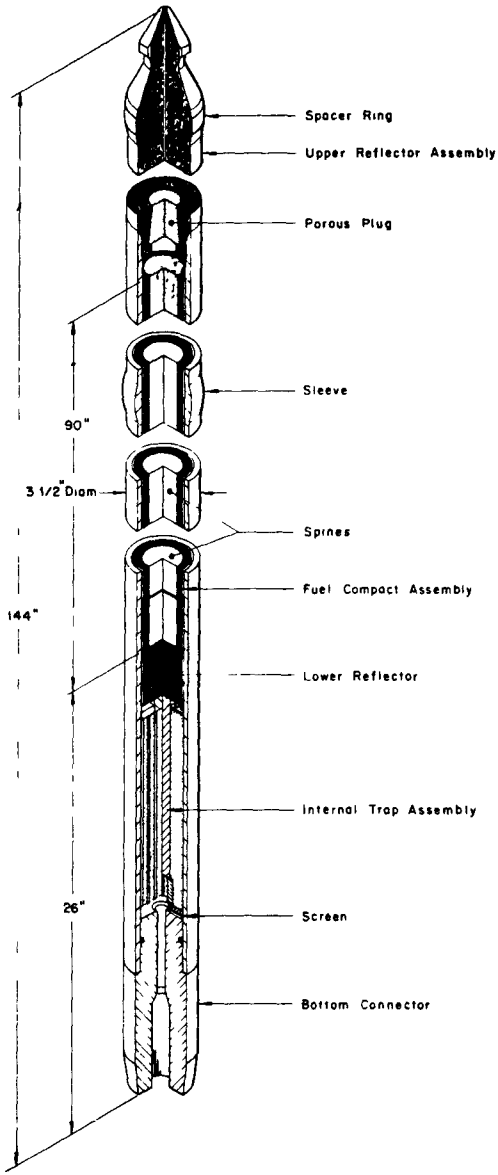

Figure 2. Peach Bottom I Fuel Element

Fort St. Vrain Fuel Elements

Fort St. Vrain fuel elements (Figure 3) are hexagonal blocks of graphite 31.2 inches (79.25 centimeters) tall and 14.2 inches (36.07 centimeters) across the flats. Both fissile and fertile fuel material, uranium (U-235) and thorium (Th-232), respectively, in the form of microspheres, are placed in the graphite block. Quantities of U-235 and Th-232 per fuel element vary depending on designated core locations with the maximum amounts being 1.31 kilograms of U-235 and 11.25 kilograms of Th-232. The maximum calculated burnup after six years in the reactor core will be 98,000 megawatt days per metric ton (MWD/MT), however, the spent fuel now being shipped from the first refueling experienced 175 equivalent full power days (EFPD) and approximately 6300 MWD/MT. Since the burnup for spent fuel elements from the first reactor cycle is well below the design values for the model FSV-1 cask, the fuel heat load and radiation levels are also well below those anticipated for spent fuel shipments from later reactor cycles. However, the measured external contact radiation levels of 10 to 15 milirem per hour (mrem/hr) are in good agreement with the anticipated values for these spent fuel elements from the first reactor cycle.

Model FSV-1 Cask

The FSV-1 cask (Figure 4) has an inner container that serves as the containment vessel to satisfy the requirements of Title 10 Code of Federal Regulations Part 71 (10CFR71). This inner container is within the cavity of the cask body which provides the gamma shielding (neutron shielding is not required since the Fort St. Vrain fuel has high enrichment), and also provides the structural strength to protect the containment vessel during the accident conditions of transport. Depleted uranium, 3.5 inches (8.89 centimeters) thick is contained between type 304 stainless steel shells which form the body of the cask. Type 304 stainless steel is also used for the containment vessel. Twelve high strength bolts attach the inner closure, which is made up of 4.15 inches (10.54 centimeters) of depleted uranium encased in type 304 stainless steel, to the containment vessel. Twenty-four high strength bolts attach the outer closure, fabricated from type HY140 alloy steel, to the cask body. Solid, type 304 stainless steel, 8.94 inches (22.71 centimeters) thick forms the base of the cask body, this thickness providing the necessary gamma shielding. An impact limiter made from multiple layers of marine grade, douglas fir plywood and encased in a protective sheath of aluminum provides both thermal and impact protection to the closure end of the cask.

Fuel Handling Machine

At Fort St. Vrain, fresh fuel elements are placed in the core and spent fuel elements are removed from the core by use of a computer controlled Fuel Handling Machine. This same Fuel Handling Machine is used to remove spent fuel elements from the fuel storage wells and place them, six at a time, into the FSV-1 cask. The Fuel Handling Machine has sufficient shielding, a combination of steel and lead shot, to limit the radiation at the surface to a maximum of 10 milirem per hour (mrem/hr) when it contains 35 full burn-up spent fuel elements without any decay time. Thus personnel may work in the near vicinity with little or no exposure during spent fuel shipping operations when the Fuel Handling Machine contains only six spent fuel elements with a minimum of 120 days decay time.

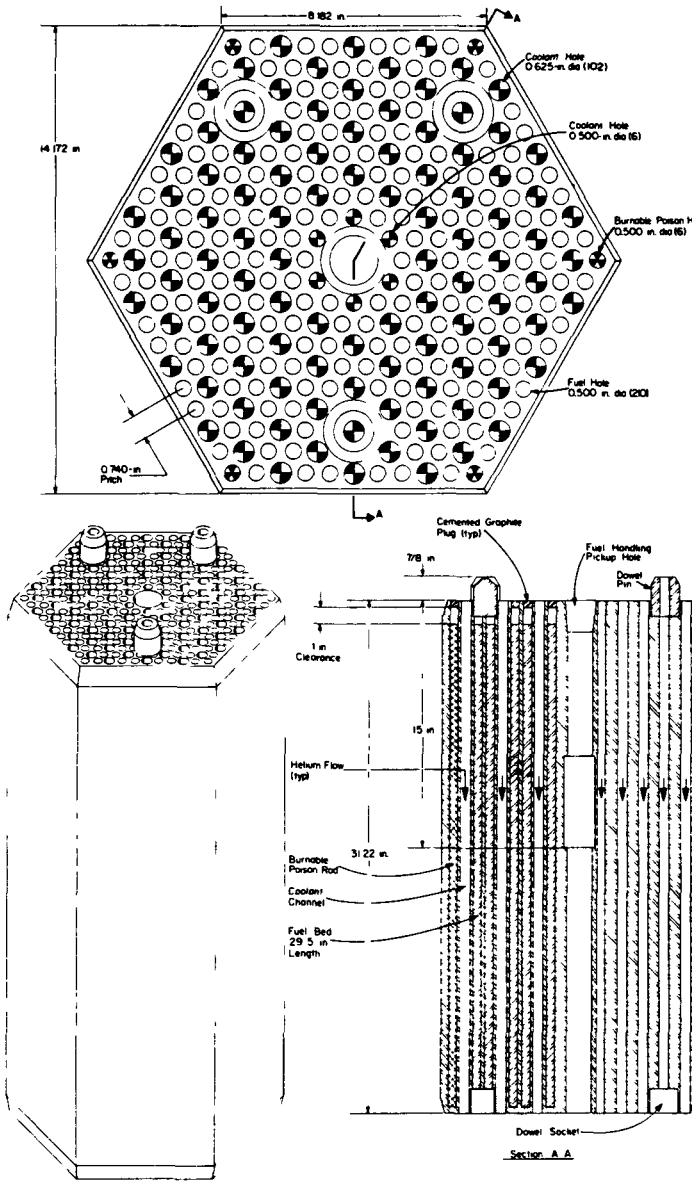

Figure 3. Fort St. Vrain Fuel Element

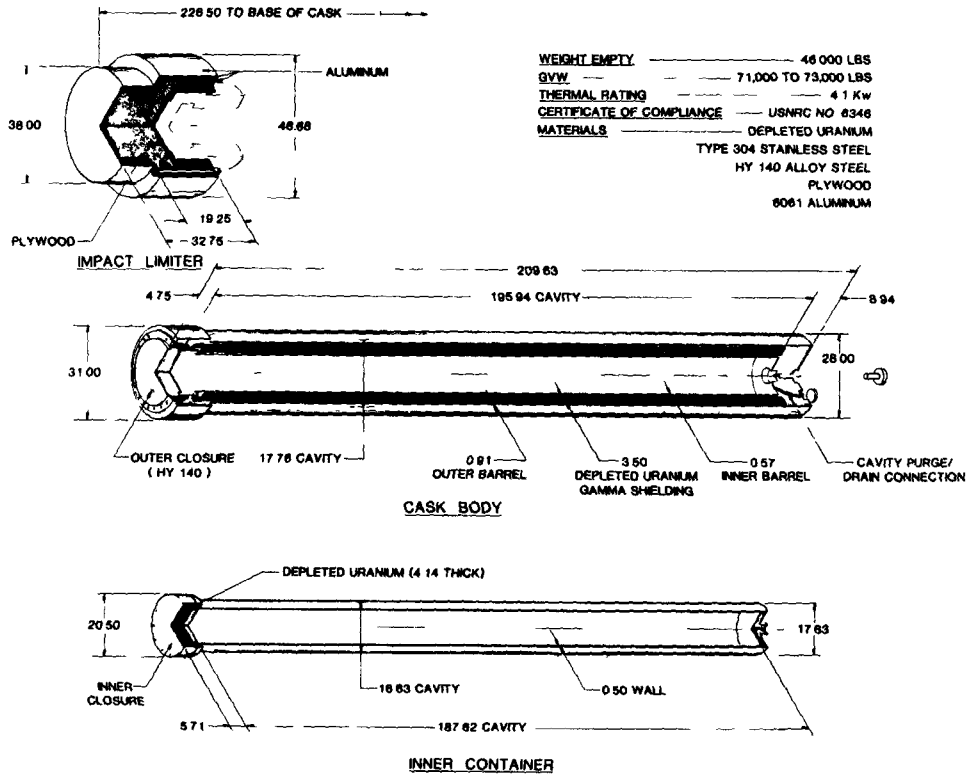

Figure 4. Model FSV-1 Cask

Reactor Isolation Valve

The Fuel Handling Machine is mated to the reactor vessel, the fuel storage wells or the FSV-1 cask by means of a Reactor Isolation Valve. The Reactor Isolation Valves are designed to provide for a seal at the top interface and at the bottom interface, and approximately 12 inches (30.5 centimeters) of shielding. A sliding gate allows for the transfer of fresh or spent fuel elements or such equipment as the inner closure of the FSV-1 cask or reactor control rod assemblies through the throat of the Reactor Isolation Valve.

Spent Fuel Storage

The Irradiated Fuel Storage Facility located at the Idaho National Engineering Laboratory is a U.S. Department of Energy installation which was designed to receive and store spent fuel elements from both Peach Bottom I and Fort St. Vrain. The major elements of this installation are the cask receiving bay, the handling cave and the storage room.

Spent fuel shipping casks such as the PB-1 and the FSV-1 are removed from the transport semi-trailer, suspended in a transfer car and then moved into the handling cave where the spent fuel elements are removed from the cask and prepared as necessary for storage. Since the body of the cask is suspended below the transfer car, only the top portion and the interior of the cask is exposed to any contamination present in the handling cave. Major equipment within the handling cave consists of a manipulator, television camera and during the Peach Bottom I, Core 2 spent fuel element handling, a band saw. Adjoining the fuel element cave is the air cooled storage room with positions for 636 canisters; 491 of which are designated for the storage of spent fuel elements from Fort St. Vrain.

Fort St. Vrain Spent Fuel Shipping

Upon arrival at Fort St. Vrain, an empty FSV-1 cask on its transport semi-trailer is moved into the truck bay of the reactor building. The impact limiter is then removed and the FSV-1 cask is released from the trailer and lifted to the refueling floor by the reactor building crane. Next the FSV-1 cask is lowered into the fuel loading port, where it is suspended with the outer closure just below floor level. This position allows a Reactor Isolation Valve to be mated to the FSV-1 cask, after the outer closure has been removed. The inner closure bolts are now removed and the remotely operated Transfer Cask is positioned on the Reactor Isolation Valve, and by means of a grapple removes the inner closure. Now the Auxiliary Transfer Cask is moved away and the Fuel Handling Machine containing six spent fuel elements is mounted on the Reactor Isolation Valve, the computer controlled sequence started and the six spent fuel elements placed in the FSV-1 cask. Since the FSV-1 cask now contains the highly radioactive spent fuel elements, the sliding gate in the Reactor Isolation Valve must be closed before the Fuel Handling Machine is moved. Following this move the Auxiliary Transfer Cask containing the inner closure is returned to the Reactor Isolation Valve, the sliding gate opened and the inner closure now lowered into place. Once the inner closure is positioned, the Reactor Isolation Valve can be removed since sufficient shielding is provided for personnel to complete the installation of the inner closure. After the bolts attaching the inner closure are torqued to the specified value, a leakage rate test is conducted to verify proper assembly of the inner closure to the inner container. The FSV-1 cask has a seal assembly with concentric silicone rubber seals. This arrangement provides an inner-space which is evacuated, through a test port, to a

pressure of one millimeter of mercury, the vacuum pump is then valved off and the rise in pressure is recorded after a specified time. A typical leakage rate for the FSV-1 cask seal assembly is 2.5×10^{-4} cm^3/sec at one atmosphere, which is due to diffusion of the air molecules through the silicone rubber and not an imperfect seal. This leakage rate of 2.5×10^{-4} cm^3/sec at one atmosphere is well below the allowable leakage rate calculated in accordance with a Regulatory Guide from the U.S. Nuclear Regulatory Commission. Following the successful leakage test, the test connection is removed, the test port is sealed and the outer closure is installed. Now the FSV-1 cask is returned to the transport semi-trailer, front and rear tie-downs secured and the impact limiter attached. Radiation levels and surface contamination are checked by the Health Physics personnel and when these are verified to be below the operational or regulatory limits the FSV-1 cask and transport semi-trailer is delivered to the carrier for the trip to the storage facility.

Transport

Actual hauling of FSV-1 cask and transport semi-trailer was performed by Tri-State Motor Transit Company using personnel, procedures and equipment qualified in accordance with U.S. Nuclear Regulatory Commission regulation 10CFR73.37. All 39 shipments were made on schedule without incident on the 690 mile (1110 kilometer) route. A spent fuel shipment was scheduled to arrive at the Irradiated Fuel Storage Facility each Monday and Thursday morning at 8:00 a.m. To meet this arrival time, the shipments left Fort St. Vrain the preceding afternoon about 3:00 p.m.

Unloading

After the semi-trailer was positioned in the cask receiving bay, the impact limiter was removed, the front and rear tie-downs released and the FSV-1 cask was removed from the semi-trailer and placed in the transfer car. Now the outer closure was removed, the sampling port in the inner closure connected to a constant air monitor and the air in the inner container (containment vessel) sampled for radioactivity. Following this step the bolts were removed from the inner closure and a lifting fixture attached. Now the FSV-1 cask was moved into the handling cave by the transfer car and the inner closure removed. The six spent fuel elements are now removed using a grapple suspended on the handling cave hoist. A remotely controlled television camera allows the operator to observe the interior of the FSV-1 cask and guide the grapple as necessary. As each spent fuel element is removed from the shipping cask it is placed in a storage canister which has a capacity of four of the Fort St. Vrain spent fuel elements. Full storage canisters are then transferred to the storage room. Once the FSV-1 cask is empty it is removed from the handling cave on the transfer car, the closures installed, then returned to semi-trailer and prepared for departure.

Review of Fort St. Vrain Experience

This first experience using the FSV-1 cask in conjunction with facilities and equipment all designed specifically for the dry handling, transport and storage of spent fuel elements has verified that the design objectives of ease of operation, low radiation exposure to operating personnel and virtually no contamination of the equipment have been obtained. Perhaps the most significant design change is the elimination of pool storage which in turn eliminates the draining, drying and decontamination of the spent fuel shipping cask after its

removal from the pool. The elimination of these steps makes a major contribution to the reduction in personnel exposure. Another significant factor in the reduced personnel exposure is the proper design of the shielding for the auxiliary equipment (Fuel Handling Machine, Auxiliary Transfer Cask and Reactor Isolation Valve) used to handle the spent fuel elements and the inner closure of the FSV-1 cask. During the shipment of spent fuel elements from segment 1, all operations were conducted where radiation levels were below 2.5 mrem/hr and most were conducted where there was no radiation. Since these spent fuel elements had low burnup and more than minimum decay time, the radiation levels are expected to increase during future spent fuel shipments, but personnel exposure will continue to remain extremely low.

Loading of spent fuel elements at the reactor was normally accomplished in the equivalent of one 8-hour shift, travel time to the storage facility was about 17 hours and unloading was normally accomplished in less than one 8-hour shift. In one case the total turn around time at the spent fuel storage facility was 2-1/2 hours.

The spent fuel elements from segment 1 of the Fort St. Vrain nuclear generating station were loaded into the FSV-1 cask, transported to the Irradiated Fuel Storage Facility, unloaded and placed in storage on schedule and without incident except for minor equipment problems. The success of this operation involving 39 shipments is a credit to the skill and dedication of the operators and managers and also verifies that well-designed and simplified systems have significant operational advantages.

Other Dry Shipment Activities

General Atomic Company has recently completed a study to determine the feasibilty of the dry transport of Light Water Reactor (LWR) spent fuel assemblies in the model FSV-1 cask. As a result of this study it was determined that a significant spectrum, based on burnup and decay time could be transported in the FSV-1 cask. The most restrictive parameters are the allowable fuel heat load of 4.1 kW and the lack of neutron shielding, however, one PWR fuel assembly with less than 30,000 MWD/MT burnup or two BWR fuel assemblies with less than 20,000 MWD/MT burnup, in each case with five years of decay time, can be transported in the model FSV-1 cask.

Since there are large quantities of LWR irradiated fuel assemblies presently located in reactor storage pools where they have decayed for many years, it appears cost effective to develop systems to facilitate dry transport to the future storage and for reprocessing facilities.

DISCUSSION

Question by M. S. T. Price, United Kingdom: We heard from Mr. Curtis in paper 035 of this session that it is not normal to ship LWR with broken fuel. In the case of the HTGR the coating on the particle is the equivalent of the clad. The broken particle fraction in the spent Peach Bottom 1 fuel must have been quite high - probably higher than will be found in any future HTGR discharged fuel. Can you comment on whether this high broken particle fraction caused any problems in spent fuel handling and shipping.

Answer: No problems were encountered during handling and shipping since the PB fuel elements were individually canned in sealed aluminium cannisters. This prevented any potential leakage of gases into the cask or pool. Handling of fuel at the receiving facility was accomplished in a hot cell under controlled ventilation conditions. Note that cask cavity sampling operations carried out at ICPP for the FSV spent fuel did not detect any measurable cavity off-gases, thus confirming minimum fuel breakage.

Question by S. Mika, FR Germany: During transport and handling, is a noble gas atmosphere maintained in the cask?

Answer: All fuel handling operations were carried out in a helium atmosphere. The cask container cavity was maintained at approximately 1/2 atm of pressure with helium. Air handling was used at the storage facility at Idaho.

Question by W. Denk, FR Germany: Why don't you need a neutron shield? Is it because of the increased self-shielding due to the carbon elements, or is the neutron source strength small enough (maybe due to special HTGR conditions)?

Answer: No neutron shielding is required due to the high enrichment nature of the HTGR fuel (93.5 %) thus producing very small amounts of neutron producing isotopes and nuclides.

CASK-FACILITY INTERFACES IN SPENT FUEL SHIPPING

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INTRODUCTION

The handling and transportation of spent fuel is receiving an increasing amount of attention in the commercial nuclear fuel cycle. Safeguarding and non-proliferation issues add to the standard transportation problems of package design, safety and regulatory and operational requirements. Although it has been demonstrated during the decade of the seventies that shipments of spent nuclear fuel can be moved routinely by truck, rail and water, there remain many problems associated with fuel cycle transport yet to be solved.

The Nuclear Assurance Corporation (NAC) owns and operates the Model NAC-1 spent fuel shipping casks. The casks have been in operation for almost six years and have been loaded or unloaded at over twenty nuclear facilities including nuclear generating stations, nuclear research laboratories and spent fuel storage sites. There are many significant differences in the cask handling facilities at each of the plants and a number of complex interface problems have been encountered. This area of cask/facility interfacing and cask handling capabilities at nuclear facilities is very significant to the conduct of an effective spent fuel shipping campaign.

CASK DESCRIPTION

The NAC casks have been used routinely throughout the United States to transport nuclear spent fuel, fuel rods and radioactive non-fuel-bearing reactor components. The NAC-1 cask has eight inches of equivalent lead shielding and 4-1/2 inches of neutron shielding. Impact protection is provided by crushable balsa wood impact limiters on each cask end. The top impact limiter is removable and remains with the transport trailer when the cask is off loaded at a facility. Vent and fill valves are simple ball type specially designed for sealing in a fire environment and are recessed in the heavy steel cask ends for protection. The neutron shield tanks and main cavity of the cask are protected from over pressure by rupture discs with the main cavity having a backup pressure relief valve. The cask is designed for wet shipment but is also approved for dry shipment of low heat rate fuel. One pair of lifting trunnions (top trunnion) are used for raising and lowering the cask on and off the transport trailer and for handling at facilities. The bottom trunnions are used only for rotation and tieing the cask to its trailer. The standard lifting yoke which interfaces the cask to a facility crane is a simple 2-arm lifting device.

The cask weighs approximately 48,500 lbs. when loaded with a PWR fuel assembly. When loaded onto the Tri-State Motor Transit Company transporting vehicle the gross vehicle weight is about 72,000 lbs.

OPERATING EXPERIENCE

During their six years of service the casks have been used at a variety of facilities including PWR and BWR type nuclear generating stations, com-

mercial research laboratories, national engineering and test facilities and waste burial sites. There are many individual differences in the facilities and procedures for cask handling at each of the sites serviced. Although most features are quite similar with respect to cask handling capabilities, there is a new requirement for interfacing at almost every facility visited. There is simply no standard design for the transportation interface at reactors of other sites. Crane systems differ greatly and sites are encountered that have oversize hooks too wide for the opening in the standard lifting yoke, several types and sizes of redundant hooks, cranes which have only two degrees of motion freedom (up-down and forward-backward), cranes without sufficient vertical movement for moving the casks to the various stations and/or plant restrictions on placing the hook and cables into the pool.

AIR PALLET SYSTEM

The cask handling facility at one nuclear generating station necessitates unloading the cask outside, hoisting the cask from its transporter up to the turbine deck, then transit the length of the turbine deck to the decontamination pad and spent fuel pool area. Although a bridge crane was installed initially for the cask movement, its use was forbidden after a review of the seismic conditions at the site and analysis of the cask drop accident during travel of the cask over the turbine deck. In lieu of the crane, the station resorted to the use of an air pallet system specifically designed for the NAC-1 cask. The air pallet measures 7 feet square, is attached and clamped to the lower impact limiter of the cask and, when air pressure is supplied, lifts approximately 1/2 inch off the deck for an air ride. The cask weight is distributed over the large base to give a low floor loading on the deck and the elevation of the cask above the deck is limited to three inches to satisfy the cask drop analysis. Some 200 spent fuel shipments have been successfully completed using the air pallet system and it is now a proven part of the cask handling facility.

NON FUEL BEARING COMPONENTS AND RELOAD FUEL

A spent fuel shipping cask is specifically designed for irradiated fuel but situations arise on occasion to extend its applications. For example, highly radioactive non-fuel-bearing components from the reactor such as packaged fuel channels, local power range monitors, control rod clusters and neutron sources must be transported to waste burial sites. A spent fuel cask can provide the type shielding required. Containers of such reactor components having a radiation field strength of the order of 7000 R/hr have been transported in the NAC-1 casks to burial sites and off loaded using mobile cranes. Special cask handling and personnel control procedures were formulated and executed to assure that no personnel overexposure was encountered during the burial operations.

One utility having damaged a fuel assembly during core discharge that required for reload, made arrangements with a second utility to obtain an assembly with similar partial burnup as a replacement. Special consideration was given to the transport of the reload fuel assembly in a spent fuel shipping cask. Based upon the recommendations of the fuel fabricator, spring loaded axial spacers were used with the fuel basket in the cask to restrain the assembly from moving axially or laterally during shipment. A mechanical recording accelerometer whose patented feature was its ability to record shock and impact from all directions was adapted and mounted on the cask. The record on chart paper gave a history during transit of the num-

ber, magnitude, direction, and time of occurrence of any impacts experienced. A post-shipment examination of the accelerometer record and a visual examination of the fuel assembly showed that specified shock levels had not been exceeded and the assembly was acceptable for use as reload fuel.

HORIZONTAL CASK HANDLING

The NAC-1 cask is normally upended, hoisted from the trailer and moved while suspended vertically from the crane. At one small reactor plant, the cask handling required entry to the containment building and fuel pool through an air lock. A strongback was designed and fabricated for use in lifting the cask off the trailer and placing it on a tracked dolly for horizontal movement through the airlock. Cask support on the dolly was arranged with the cask upper and lower impact limiter resting on saddles. Once through the airlock, the cask was upended and handled and loaded in the normal way. The procedure was reversed for the exit movement. Similar handling procedures were implemented at one research laboratory when crane height and entry door clearance required horizontal cask movement. In each case, trunnion supports were necessary inside the facility to permit upending of the cask after entry.

HOT CELL OPERATIONS

On occasion spent fuel is shipped to laboratories for study, experimentation and destructive analyses. Two of the national laboratories receiving spent fuel perform all cask handling and unloading operations inside hot cells. Unique procedures for cask operation were prepared, and practice runs performed to verify the procedural accuracy. The cask unloading was performed by the use of manipulators and remote controls inside the hot cell. The fuel assembly and the unloading operation was visually monitored, both through the hot cell window and by use of closed circuit television. All fuel grappling, fuel handling and cask lid removal and installation were performed by remote control. Special tooling required was primarily for interfacing with the manipulators. The cask was fully shrouded with polyethelene wrapping to minimize cask contamination from fuel handling or fallout of airborne particulate matter.

CASK LIFTING ADAPTORS

Special considerations of cask height, pool depth, crane height limitations and local operating procedures have resulted in the need for specially-designed cask lifting adapters and extension bars customized for some facilities. One such facility is restrictive for cask handling in that clearance in elevation is very limited when moving a cask from the trailer over curbing and bulkheads to the decontamination area and into the pool. A very shortcoupled adaptor was required connecting crane hook to cask yoke to provide the cask clearance needed during this operation. This facility, as well as many others, have established operating procedures that preclude lowering the crane hook and cables into the spent fuel pool. The objective is to keep the hook clean since it is used for a multitude of lifting tasks. Accordingly at this site, with the cask positioned on the intermediate level platform in the spent fuel pool, extension bars were required to permit lowering the cask to the cask set-down area at the bottom of the pool. Adaptors, extension bars and an auxiliary hook were provided and used in making fuel shipments from the site.

FUEL SHUFFLING

Multiple reactor sites normally have more than one pool for spent fuel storage purposes. In most cases the pools are not connected by transfer canals or other direct means of fuel shuffling between pools. In such instances fuel transfer can only be accomplished by use of shielded spent fuel casks. The NAC-1 casks have been used for extended periods of time for fuel transfer on site between spent fuel pools. At one site the fuel was loaded into the cask in the normal manner, the cask was loaded on the trailer and the shipment moved approximately 200 feet to the second pool. A second reactor site, having a different facility layout, was able to make the cask movement between pools by crane. In each case, every effort was made to minimize the fuel transfer times. Decontamination time was limited to cask spraydown and the use of a rubberized leakproof shroud to enclose the cask for the short move between pools. The cask preshipment testing, required for offsite shipments, was reduced to tightening of lid bolts and radiation surveys. Ten to twelve fuel assemblies per day could be transferred in this manner between reactor pools even though the movement required exiting from one auxiliary building and entering another.

UTILITY INTERFACES

Cask suppliers and transportation service contractors in many instances noted above are encountering complex utility interface problems. Facility design is only recently improving with regard to cask handling capability. Some plants will require changes if they are to utilize more than one type of cask and mode of transport. Facility design considerations at reactor plants must assure adequate provisions for transport vehicle receiving and parking, cask handling operations, fuel loading operations, cask and equipment decontamination and auxiliary services as needed for an efficient spent fuel shipment program.

CASK RECEIVING AND HANDLING

The cask transport vehicle upon arrival must be received, inspected and maneuvered into the loading dock or vehicle bay. When a large spent fuel shipping program is underway as many as three or four cask vehicles may be on-site at one time. Parking space is required and a yard tractor will prove helpful for movement of cask vehicles in and out of the fuel handling building. The vehicle bay must be large enough to receive the transport vehicle, provide storage space for impact limiters and cask lifting yokes and permit closure of entry doors. Upending of the cask on the vehicle requires that the crane travel be sufficient for the upend/lay down maneuver of the cask on the vehicle or, alternatively, the bay must be long enough to permit vehicle movement during the cask rotation.

Cask handling and loading is greatly expedited if the vehicle bay, decontamination area and cask loading area are adjacent to each other and the crane can move the cask to each station in sequence without restriction or need for hook to yoke interface equipment. Procedural prohibitions on placing the crane hook in the contaminated pool water is very restrictive and adds greatly to the interface equipment required and the time scheduled for cask loading. Even with a cask loading pit separate from the spent fuel pool, if the crane hook must be kept dry, the pit must be drained each time the cask is to be lifted in or out adding to the cask loading time.

CASK LOADING

Facility design must consider the type and size of the spent fuel cask to be employed. A frequent problem is encountered in the cask set down area of the spent fuel pit in that sufficient area is not avaiable for the cask-yoke assembly operations, engagement and disengagement of the yoke, and lift off and handling of the cask closure lid. Often these areas are also poorly lighted which further complicates operations. Frequently the cask loading pit or area is used for fuel inspection or similar type equipment that encroaches upon the space originally provided for cask handling.

OPERATIONAL PLANNING

Preparations for a fuel shipping campaign must include coverage of all those items that affect cask loading and cask turnaround times. Cask loading is adversely affected or delayed by malfunctioning grapples or fuel handling equipment, difficulties with fuel assembly identifications or decay heat release verification. Some fuel shipment programs have established an in-cask fuel leak test procedure that must be performed as a preshipment test which adds significantly to cask turnaround time.

The cask height, pool depth, and fuel handling system must be compatible so that the fuel assembly will clear the cask for loading and maintain the radiation level at the pool surface acceptable when the fuel assembly is at the loading elevation. In addition, instances have occurred of grapple interference with the cask structure or fuel basket so that the fuel assembly could not be seated in the cask and the grapple disengaged without an equipment modification.

Preshipment testing of the cask and its contents requires facilities and procedures for handling contaminated water, air and possibly helium or steam from the loaded casks. Fluid sampling and radioactivity measurments must be accomplished rapidly. Supply and disposal of large quantities of demineralized water is required for wet casks and handling helium, steam and fission gases may be necessary for dry casks. Routine and emergency cask cooling capabilities should be available for use as necessary to prevent coolant boiling or to control metal temperatures.

CONCLUSION

Progress is being made in the cask handling capabilities of the newer plants. The improvements are quite striking when a comparison is made with older plants that required opening and closing roof-level cask-access ports for cask entry to the spent fuel pool. After fuel loading the cask was again moved outside the fuel handling building to other adjacent facilties for decontamination and preparation for shipment. Time-consuming operations were involved in opening and closing the fuel building port two times for each shipment and following special precautions to allay environmental concerns during cask movements when the cask is not fully prepared for shipment.

Proper consideration for the facility design features noted will go far to enhance the efficiency of spent fuel shipping programs. Cask utilization will be improved and disruption of facility operation will be minimized. The safety of the spent fuel shipping operations is of paramount importance. These factors dictate that the utility, the A & E and the cask designer work together to assure that the cask-facility interface problems are addressed and that safe, efficient and economical transportation of spent fuel is provided.

Question by C. R. Davis, USA: What unloading times have been required for the NAC-1 cask? How does the time compare to dry unloading of the FSV-1 cask at ICPP? (Reported in the previous paper as 7 hours).

Answer: It depends upon the facility. The NAC cask has been turned around in as little as 2 hours and as long as 2 days.

Question by E. P. Goldfinch: Could you please tell me what the surface finish of the cask is, what decontaminants are used, whether there are any difficult areas and whether there is any increase in the level of contamination in transit.

Answer: The surface finish is bead blasted stainless steel. Decontaminants used are usually demineralised water - on occasion commercial decontamination agents. Difficult areas are around valve ports where contaminants can be trapped. Normally the cask surface is decontaminated to less than 1/10 allowable levels - on arrival the surface contamination is about double that at initiation of shipment.

EXPERIENCE IN THE INTERNATIONAL SHIPMENT OF IRRADIATED MTR FUEL ELEMENTS BY TRUCK, SHIP AND AIRCRAFT

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Th. Staake
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The paper is intended to provide a review of the shipments of irradiated MTR (Material Test Reactor) fuel elements, the selection of the casks, and the shipping equipment. The information presented is based on the experience which Transnuklear GmbH has been able to acquire during the last 12 years in the shipment of irradiated MTR fuel elements. During this time about 2.000 fuel elements have been shipped to various destinations for reprocessing:
about 1.000 fuel elements to the European reprocessing plants (Eurochemic/Belgium and CEA, Marcoule/France) and - since 1977 more than 1.000 fuel elements to the US-DOE "Savannah River" plant.

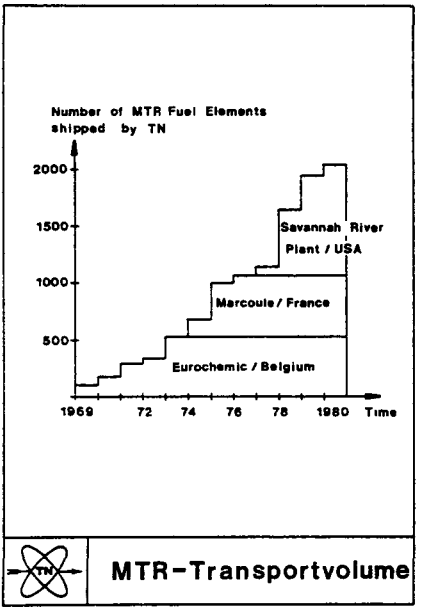


Fig. 1

Shipping casks

The internationally valid safety regulations - they are precisely defined in the "Regulations for the Safe Transport of Radioactive Material" issued by the IAEA, Vienna - are well known to be very demanding. The three essential tests of the shipping casks should be mentioned again here, and are the following:

- 9 m drop onto an unyielding surface, with the cask striking the surface at that point on the cask which results in maximum damage to it.
- 1 m drop onto a cylinder with a diameter of 15 cm, likewise with the cask striking the cylinder at that point on the cask which results in greatest damage to it.
- Fire Test:
  here, the cask is subjected to a temperature of 800°C for 30 minutes.

After all these tests, the cask must continue to fulfill its essential functions, namely:

- assurance of subcriticality
- shielding
- leaktightness

After a cask has successfully passed these tests, among others, it is qualified as a so-called type B(U) cask.
Transnuklear GmbH uses four type B - shipping casks for the shipping of irradiated MTR fuel elements. These are casks with shipping weights of 12 - 25 metric tons. An overview of the most important data on these casks is given in Fig. 2. During an expensive, time-consuming and laborious licensing process all casks have been approved for shipment by truck and in a special case also by aircraft in various

SHIPPING EQUIPMENT

Due to the relatively large shipping volume, as well as the demanding requirements for safety and physical protection of the material to be shipped, the selection of the shipping facilities had to be made with especially great care and from the following viewpoints:
- optimum safety
- minimum dwell time on public transportation routes
- control of the material during shipment and at transshipment points.

Following extensive studies, we have selected the following shipping systems:

Shipment by Truck

The shipping casks are mounted on standardized palettes or trailers which have been specially reinforced to hold the heavy specific loads: the casks in the 12 ton - class on 20 foot-flats, those in the 25 ton - class on 40 foot-trailers. The flats can then be mounted on trailers which have also been standardized.

The advantage of using these units is that this method of shipment is based on a system that has proven successful millions of times in international container traffice.

Transport by Ship

Here, the selection of the type of ship to be employed was made in favour of modern roll on - roll off (Ro-Ro) ships, which offer the great advantage that the casks together with the trailer can be drawn as a single unit up a loading ramp onto the ship and lashed in place there. After a trip of 10 - 12 days from Europe to the east coast of the USA, in the U.S. port the trailer is attached to a tractor specially equipped with security and safety equipment, and is then hauled under strict physical protective measures to the reprocessing plant.

Transport by Aircraft

The fact that shipment of irradiated fuel elements is also possible by aircraft was confirmed by a shipment from Berlin to Bremen. This shipment was made by a special cargo plane of the type "Hercules", which is not only capable of carrying such loads but also is fitted with a correspondingly large loading hatch. The so-called "Goslar" shipping cask was used here - this cask has a weight of about 120 kN and has been approved for shipment by aircraft.

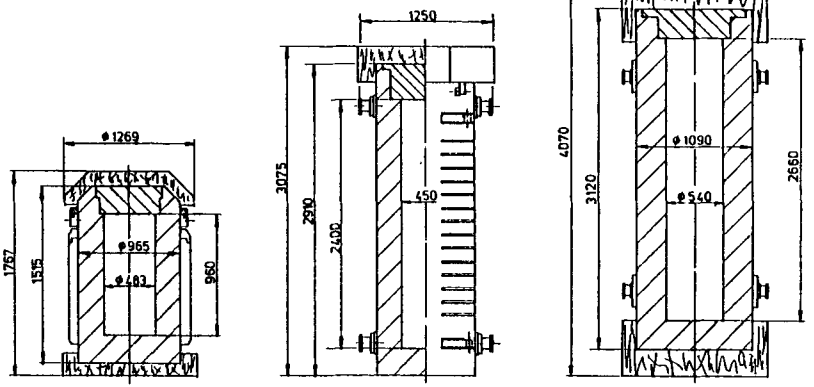
SHIPPING COORDINATION

Administration of the shipping operations proves to be extremely difficult, because, on the one hand, dwell times on public transportation routes must be minimized and intermediate storage is not permitted, while on the other hand arrival and departure times of ships can change at relatively short notice.

This becomes especially problematical with long preliminary road shipping distances and with the selection of expensive shipping facilities such as, for example, the aircraft, which must be firmly chartered in advance.

Serious problems are also presented by the fact that shipping companies change the itineraries of their ships at short notice and divert their ships to ports for which shipment approvals have to be applied and obtained within a few days. The fact that shipment of our cargo can nevertheless be continued is mainly the result of the cooperative efforts made by the responsible authorities in the relevant countries.

Fig. 2



type	Gosiar	TN-1	TN-7
available quantity	2	1	1
max. loaden weight (kN)	110	180	250
fuel element capacity	13	42	60

TN

MTR-Transport cask

AIRBORNE CONTAMINATION RELEASED DURING UNDERWATER UNLOADING OF A FAILED PWR SPENT FUEL ASSEMBLY

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Project Manager
BATTELLE
Columbus Laboratories

INTRODUCTION

In May 1980 a spent PWR fuel assembly containing several severely failed rods was shipped from the Connecticut Yankee, Haddam Neck Reactor to the Battelle Columbus Hot Cell Laboratory for postirradiation examination. Nuclear Assurance Corporation's spent fuel shipping cask NAC-1E was used for the shipment. This cask, when loaded, is a sealed unit licensed for transport of uncanned failed rods or assemblies. Thus, the assembly was carried in the cask without being placed in a failed fuel can. Also, the shipment was made dry; cask cavity environment was normal atmospheric air.

No unusual events occurred during transport. Condition of the cask on arrival was normal. As the first step in unloading, water was slowly backfilled into the cask until the internal cavity was filled. Upon sampling, this water was found to contain contamination levels much higher than usual.

During subsequent underwater unloading of the cask, excessive contamination was released into the pool water and airborne contamination spread throughout the high-bay cask handling area. Effective operating procedures minimized personnel exposure and contained the contamination within the high-bay work area. The contamination signature was identified as being that of high burnup spent fuel.

At this time it is believed that the contamination source term was finely divided spent fuel material in the form of particulates that were released through fractures in the fuel rod cladding. Such particulate material can be generated in two ways: first, pulverization of fuel material as a result of normal cracking within the fuel pellet during operation, and second, release of particulate material due to density and porosity changes occurring in the fuel as a result of exposed fuel being subjected to high temperatures.

During its operating cycle and, in this case, during transit the exposed fuel material was subjected to temperatures exceeding 500 F (260 C) while either fully submerged in water or within the dry (but moist) environment of the shipping cask. Under these conditions the fuel material is subject to porosity changes which led to the generation of particulate fines.[1]

In either case, such finely divided particulates are capable of becoming airborne directly or being entrained (but not necessarily wetted) in a water flow over the material surface.

An investigative research study is being initiated by Battelle-Columbus for the U.S. DOE to study the conditions relating to this contamination release and its implications to the handling, transport, and storage of spent fuel. This paper reviews the release event and summarizes the experimental study results obtained to date.

FUEL ASSEMBLY CHARACTERISTICS

The fuel assembly considered in this study is a 14 x 14 PWR assembly with stainless steel clad fuel rods. It had undergone three consecutive reactor operating cycles achieving an end-of-life burnup of approximately 36,000 MWD/MTU. It was discharged from the reactor in January, 1979, and was stored in the reactor spent fuel storage pool until April, 1980. At that time it was shipped to the hot cell laboratory for postirradiation examination.

During operation, this assembly experienced failure of several individual fuel rods. Postdischarge underwater optical examination in the storage pool revealed visible failures in seven individual rods. These failures consisted of longitudinal splits in the cladding which had opened to expose the interior fuel material. The splits varied in length from a few centimeters to as much as a meter with gaps of several millimeters.

Figure 1 is a view of the cladding failure in one rod as seen through the underwater periscope of the reactor spent fuel pool. Figure 2 is a view of the same failure area as seen through the optical telescope of the Battelle hot cell. Of course Figure 1 shows the rod before shipment to Battelle while Figure 2 is after the shipment.

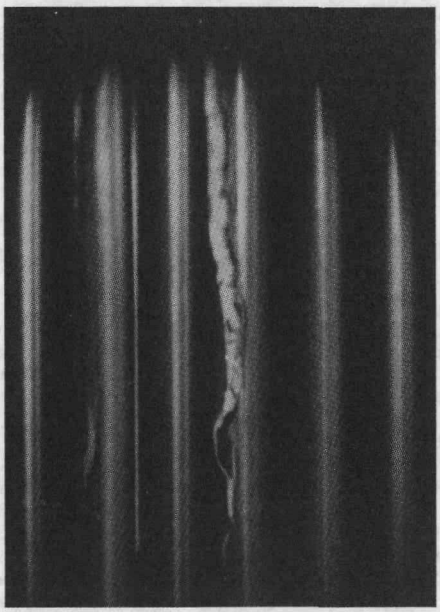

FIGURE 1. UNDERWATER VIEW OF FAILED ROD

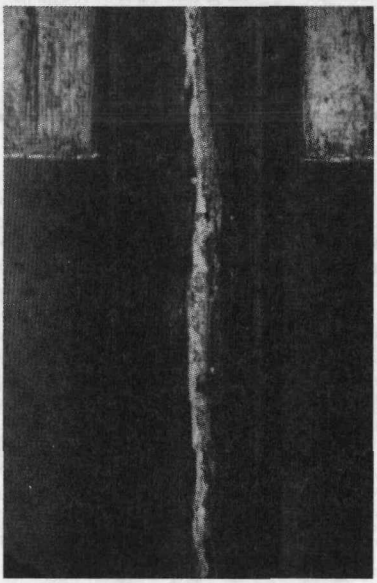

FIGURE 2. IN-CELL VIEW OF FAILED ROD

This degree of cladding failure would no doubt expose interior rod fuel material to reactor coolant water during operation and to pool water during storage. Thus, the failed rods would certainly have experienced some degree of water intake. This was the condition of the fuel assembly as it was packaged for shipment.

HANDLING AND TRANSPORTATION

When loaded and sealed for shipment, the cask cavity contained unpressurized normal atmospheric air. When the residual pool water was drained from the cask cavity, it was the first time since being initially loaded into the reactor that the fuel assembly had not been continuously submerged under water.

The fuel assembly was sealed in the cask for six days prior to its arriv and unloading at the hot cell site. Heat content of the assembly was calculat to be approximately 3 KW. Maximum fuel rod cladding surface temperatures reached during shipment were estimated to be approximately 220 C (430 F) for peripheral rods and 300 C (570 F) for center rods. If the heat content of the assembly had been 2.5 KW, it is estimated that these temperatures would have been approximately 200 C (390 F) and 270 C (520 F), respectively. In either case, these maximum temperatures are comparable to reactor operating temperatures, and no cladding degradation should be expected. However, these temperatures are sufficient to promote oxidation of the UO_2 fuel material.[1]

Upon arrival of the cask at its destination, a sample of the cask cavity atmosphere was taken. The cask was found to be still unpressurized, and the atmosphere had a low radioactive background and was essentially depleted of oxygen. Results from a mass-spectrographic analysis of the sample are given in Table 1.

TABLE 1. POSTSHIPMENT COMPOSITION OF CASK CAVITY ATMOSPHERE

Gas	Volume Percent
Nitrogen	95.6
Oxygen	1.4
Helium	0.27
Krypton	0.03
Xenon	0.28
Argon	1.37
Carbon dioxide	0.97
Others	0.08
Hydrogen	
Carbon monoxide	
CO	
NO_x	
HCl	

Depletion of the oxygen is of interest. At the temperatures reached by the fuel during transit, reoxidation of the UO_2 would have been possible. Specific oxides and the quantities produced are difficult to determine. For example, consider the following. Assuming the cask cavity void volume to be 350 liters, depletion of atmospheric oxygen would produce approximately 23 kg of U_3O_7 or 13 kg of U_4O_9 or equivalent quantities of other oxides. These and other possibly produced oxides are less dense, more porous, and have different crystal structures than UO_2. Consequently the physical and chemical changes associated with such reoxidation could produce finely divided particles within the spent fuel material.

POOL AND AIRBORNE CONTAMINATION

Water samples taken from the unloading pool along with those taken from the nearby air monitors were analyzed. Results of the analyses are given in Table 2. Both the water and airborne contaminants are seen to be fuel and fission products typical of spent PWR, UO_2 fuel.

Upon unloading the cask in the pool, the specific activity of the pool water increased from 1.1×10^{-4} μCi/ml to 1.1×10^{-2} μCi/ml. The major contributors to the increase were Ce^{144}, Ru^{106}, Cs^{134}, Cs^{137}, and detectable amounts of actinides. The Ce and Ru nuclides appeared to concentrate in the surface layer of the pool water. The actinide-to-fission product ratio was of such magnitude to indicate spent UO_2 fuel as the contamination source. No Co^{60} was included as a contaminant. In fact the normal pool water background concentration of Co^{60} was totally masked by the additional contamination.

Air monitor samples showed an airborne migration of the same basic fuel and fission product species. The relative solubility and wettability of the particulates are being studied.

TABLE 2. POOL WATER AND AIRBORNE CONTAMINANTS

		Percent of Total Activity					
	μCi/ml	Ce^{144}	Ru^{106}	Cs^{137}	Cs^{134}	Co^{60}	Actinides
Pool Water							
Before Event	1.1×10^{-4}	--	--	20	--	80	--
During Unloading	1.1×10^{-2}	24	10	38	28	--	Detectable
Surface Layer	--	36	20	23	20	--	Detectable
Six Weeks Later	1.1×10^{-2}	29	--	39	29	3	Trace
Six Months Later	2×10^{-3}	34	--	31	22	10	Trace
Airborne	--	47	24	15	11	--	Detectable
Typical UO_2 Spent Fuel	--	22	25	26	27	--	Detectable

Contaminating species found in surface smears taken throughout the high bay area duplicated those found in the airborne samples.

Upon subsequent examination, the interior surfaces of the cask cavity along with the surfaces of an internal supporting spacer appeared to be covered with a dark coating. A dark liquid residue remaining in the bottom of the cask cavity resembled water in which very fine particulates were suspended. This liquid and the contaminating film both gave extremely high radioactive readings. Both had the same physical appearance. Sampling and analyses revealed the dark suspended particles to be highly radioactive fuel/fission product materials. Results of these analyses are given in Table 3.

TABLE 3. SUSPENDED PARTICULATE MATERIAL ANALYSIS

Percent of Total Activity					
Ce^{144}	Ru^{106}	Cs^{137}	Cs^{134}	Co^{60}	Actinides
42	31	14	12	--	Detectable

From these combined analyses it is concluded that the contaminating material is fuel and fission products in the form of very fine particles. Specific mechanisms by which these particles were produced and by which they were transported are being investigated.

EXAMINATION OF THE FUEL

Porosity and density characteristics of the fuel material help to determine the susceptibility for generation of particulate forms. Exposed fuel material was examined by studying a transverse-cross-section sample taken from the exposed region of a failed fuel rod.

Those surfaces of the fuel which were exposed by the clad failure appear to be covered with a porous coating. At this time it is not clear whether this is a corrosive coating on the fuel material or an oxidized change-of-state layer of the fuel itself. Chemical and physical analyses of this material have not been completed. A microphotograph of the sample is shown in Figure 3. The coating material can be identified. A magnified view of the coating material is shown in the microphotograph reproduced in Figure 4.

FIGURE 3. SECTION THROUGH FAILED REGION OF ROD

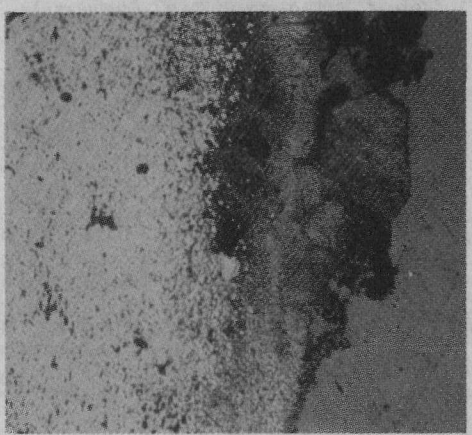

FIGURE 4. MAGNIFIED VIEW OF COATING ON EXPOSED FUEL

Fuel material samples like these are being examined metallographically, chemically, and by electron microscopic techniques to determine the characteristics of the material and to determine any physical or chemical changes which have taken place.

FINDINGS

Operationally, the handling, packaging, transport, and storage of failed fuel, either rods or assemblies, have the potential for the release of radionuclides leading to local contamination. Operational procedures should be adaptable to actions to ensure prevention of such contamination.

Although still in progress, the experimental investigation of radionuclide release from a failed fuel assembly indicates that the source term was particulate spent fuel material released from the failed rods. Such particulate matter was generated either by the normal cracking of fuel pellets during operation or (as a result of heat generation during transit) reoxidization of the fuel material generating a physical and chemical change-of-state. Resulting changes in density and porosity generated finely divided particulates. Such particulates are entrained in any water flow over the material surface. Because wetting of the particulates is not complete, they can be carried to the water surface, subsequently becoming airborne.

REFERENCES

(1) D. G. Boase and T. T. Vandergraff, "The Canadian Spent Fuel Cannister: Some Materials Aspects", Nuclear Technology, 32, 60 (January 1977).

DISCUSSION

Question by W. Heudorfer, FR Germany: 1. Is it possible that there was an error in the fabrication of the spent fuel pin? 2. Were there foreign substances in the transport container during transport?

Answer: Probably not. The fuel pins, even though failed, did not contain fabrication errors which caused the contamination or were the source of the contamination. 2. Again, probably not. The cask was relatively clean before loading and did not contact foreign substances during transport and handling.

Comment by K. Goldmann, USA: Your observation of a yellow coating on the failed fuel surfaces appears to be a further indication that the original UO_2 pellet material had oxidised to U_3O_8 since U_3O_8 has a yellow colour.

Question by K. Goldmann, USA: Did you fill the cask with water prior to immersion into the pool to avoid possible bubbles from boiling or air upon lid removal? Such bubbles could provide the mechanism for transporting solid material by entrainment to the top of the pool and then ejecting it into the ambient environment as the bubbles break through the surface.

Comment by the author: We believe the coating material on the fuel surfaces to be some form of reoxidised fuel material consisting of a mixture of various complex compounds and oxides including U_3O_8.

Answer: Yes, the cask was lock-filled with water prior to immersion in the pool. This was to both reduce the temperature of the assembly and to minimise bubble release upon lid removal.

This operation did minimise bubble release in that no large turbulent action occured when the lid was removed. However, mild turbulence was generated and some small bubbles along with a column of water resulting from cooler pool water exchanging with the warmer water in the cask cavity did rise to the surface.

It is believed that finely divided particulates of spent fuel material were entrained in this upward motion to the surface.

SESSION IX
MANUFACTURING PROCESSES AND MATERIALS

FRACTURE MECHANICAL PROPERTIES IN HEAVY SECTIONS OF FERRITIC NODULAR IRON CASTINGS

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1. Introduction

Storage and transportation casks must meet a lot of safety requirements, for example the IAEA regulations as type B transport containers. One important safety aspect is to prevent cracking of the casks in an accidental incident during transportation and storage time.

The presupposition is that a tough material is used for construction, especially because of the necessity to use a great wall thickness in order to shield the radiation activity. An interesting material for this application is ferritic nodular cast iron GGG 40.3, DIN 1693, part 2 (1).

In 1980 a cooperative investigation was finished about fracture mechanical properties in heavy sections of nodular iron castings. The work was performed by a group of German and Swiss producers and customers of great castings (2).

2. Production of the test castings

In ferritic nodular iron two types of test castings were produced. The dimensions and the approximate weight were as follows.

Test casting No.	Length x width x height mm	Modulus cm	Casting weight t
2	700 x 400 x 600	10,8	2,5
3	1050 x 640 x 700	15,5	5,5

Moulding was in cement sand. In order to determine the effect of more rapid solidification a 120-mm thick chill was applied to one of the longer sides. The melts were adjusted to the production conditions of the foundries. The mains frequency induction furnace was the preferred melting unit. Inoculation after the magnesium treatment was carried out immediately before casting the test blocks, the pouring temperature was kept in the range between 1300 and 1350 °C. Additionally mould inoculation was carried out. The chemical compositions of the test melts are given in Table 1. The castings were left in the as-cast condition.

Following non-destructive testing using ultrasonics, compact-tensions-test (CT) specimens for the fracture mechanics tests were machined according to fixed sampling layout, as well as tensile and impact test pieces and metallographic specimens. In particular a separate evaluation was scheduled for the sand-cast and the chill cast sides of the test castings.

3. Properties of the test castings

The mechanical properties were determined using tensile test pieces B 14 x 70 DIN 50 145. The internal properties of the castings were ranging between the following values.

		R_m N/mm^2	$R_{p\,0,2}$ N/mm^2	A_5 %	HB 30
chill side	min	376	211	16,4	117
	max	420	304	29,0	146
sand side	min	366	224	19,1	117
	max	405	305	27,9	147
center	min			6,2	

No significant differences were established between the sand-cast and the chill-cast sides.

Properties determined at different temperatures are presented in Figure 1.

The dependence on temperature of the notched impact energy was determined. Figure 2 shows the results. In the diagrams the transition temperatures are given, defined by an impact energy of 10 Joule.

The structure of the graphite and the matrix is summarized in Table 2. Here a higher nodule count could be determined at the chill side. The matrix was nearly entirely ferritic.

4. Static fracture mechanical properties

Compact tension test specimens with section sizes of 100 (CT 4), 150 (CT 6) and 200 mm (CT 8) were used for the investigations. The initation of a fatigue crack, the subsequent tensile test and its interpretation were carried out in accordance with ASTM-E 399 (1974). The interpretation differentiated between

- the critical stress-intensity factor (plane strain fracture toughness) K_{IC}, which assumes linear-elastic behaviour;
- the fracture toughness value K_Q, an estimated value, which was determined from the elasto-plastic-behaviour of the material;
- the K_{max}-value, which was calculated from the maximal load.

In some cases the critical crack opening displacement (COD) was measured; the basis of this tests is the determination of the displacement value δ_c at the tip of the fatigue crack in dependence on the load. In this way the stable crack propagation which sets in with increasing load can be measured, this was determined at the testing institutes (Gebr. Sulzer AG) by acoustic emission or (Kraftwerk-Union AG) by the DC electrical potential method.

These tests were carried out between - 140 and + 20 °C.

The results are summarized in Figure 3 a. Below - 60 °C the behaviour of the material is linear-elastic. With increasing load a spontaneous crack propagation almost always occurs (pop-in). With further loading the sample again stabilizes itself; the "pop-in" values are valid as K_{IC}-values. Their minimum values lie between 1000 $Nmm^{-3/2}$ at - 140 °C and 1400 $Nmm^{-3/2}$ at - 60 °C. Above - 60 °C the behaviour of the material is increasingly elastic-plastic. Valid K_{IC}-values can no longer be obtained even with larger sample dimensions, "pop-in" effects are not to be observed too. The K_Q-values which are now obtained increase continuously to about 3500 $Nmm^{-3/2}$ at room temperature.

A similar behaviour can be observed relating to the crack opening displacement (Figure 3 b). Below - 60 °C the lower values of the critical crack opening displacement remain practically constant at δ_c = 0,03 mm. Above this temperature this value increases continuously to 0,20 mm at - 10 °C.

A consideration of all the results separated into the sand- and chill-cast sides indicates that no significant differences in the fracture toughness can be determined between samples of the two different positions.

The fracture surfaces of the compact tension testpieces were retained after fracture and fractographic examination carried out using a scanning electron microscope.

In correspondence to the fracture mechanical properties entirely brittle fractures with cleavage planes could be observed at temperatures lower than - 60 °C. At higher temperatures increasingly ductile dimpled fracture portions occured.

5. Crack initiation and crack propagation under alternating loading

Compact tension test specimens of 50-mm section (CT 2), with a notch radius of 0,2-mm, were subjected to pulsating tensile stresses so as to investigate at what load and number of cycles the formation of an initial crack occured. The DC potential probe method was used to detect this, by means of which a crack of about 0,5 mm in depth can be determined. The relationship between the number of cycles for crack initiation and the stress intensity ΔK are shown in Figure 4. For comparison samples of the pearlitic grade GGG 60, in the as-cast condition and normalized as well as the unalloyed cast steel GS-C 25 were tested under the same conditions too. All the results form a close range of scatter. From

this its can be concluded that the graphite nodules laying at the base of the notch or directly below the surface seems not to promote the process of crack initiation.

If an already cracked sample is subjected to an increasing alternating load in air three different stages of behaviour can be observed

a) the existing crack does not grow further

b) the crack lengthens in a stable manner for a certain amount per alternating load

c) the crack grows very rapidly in a unstable manner.

The stable crack growth was measured with CT 2-samples. The results are presented in Figure 5. The given constants c and m can be inserted into the "Paris"-approximation

$$da/dN = c\ (\Delta K)^m$$

where da/dN is the crack growth per cycle in mm an ΔK the maximum stress intensity in $Nmm^{-3/2}$. It should be mentioned that all the growth rates are simular despite the origine of the casting (producer) and the position within the test block (sand or chill cooled).

Fractographic studies show that the crack growth can be approximately determined by measuring the distances of the grid lines. But low distance strain distribution caused by the nodules changes the growth rate as well as their direction so that clear fracture lines such as those in steel are rarely found.

6. Discussion of results

The fracture mechanical study of ferritic nodular iron in great sections indicates that the grade GGG 40.3 is a steel-like tough material suitable for the application of spent fuel casks. This could be confirmed by drop tests at + 20 °C and - 40 °C. Smaller casks with the scale relationship of 1 : 3 dropped down from a hight of 9 m on a thick steel plate. After this neither a crack formation within the cask walls nor a leakage in the sealing system could be detected.

Moreover the excellent casting properties of the nodular iron melt enable to produce the whole construction with the great wall thickness of 360-mm equipped with small and high cooling fins as one casting. So a steady and not interrupted heat flow from the storage cell to the surrounding air can be established.

(1) Keese, H.; W. Anspach; R. Christ and E. Schlich:

Storage of spent fuel in transport/storage casks

PATRAM 80, November 10 - 14, 1980, Berlin (West)
paper no. 055

(2) Motz, J. M. and 9 coworkers:

Fracture mechanical properties in heavy sections of nodular iron castings

Giessforsch. 32 (1980) No. 3
Giesserei 67 (1980) No. 20, page 628 - 637

Table 1: Chemical composition of the test castings

Test casting	C %	Si %	Mn %	P %	S %	Mg %	Ni %	Cu %
A 3	3,44	1,92	0,11	0,025	0,008	0,065	0,03	0,019
B 2	3,32	2,00	0,12	0,056	0,006	0,079	n.b.	0,035
E 2	3,42	1,79	0,25	0,016	0,003	0,053	0,15	0,03
E 3	3,41	2,32	0,19	0,021	0,001	0,037	0,21	0,04
F 2	3,31	2,17	0,16	0,03	0,006	0,07	n.b.	n.b.
G 2	3,52	2,15	0,16	0,019	0,007	0,055	0,04	0,04

Test casting	Mo %	As %	Pb %	Sb %	Ti %	V %	Cr %
A 3	0,015	0,020	n.b.	0,001	0,001	n.b.	n.b.
B 2	<0,05	0,004	n.b.	<0,001	n.b.	n.b.	0,016
E 2	n.b.[x)]	0,006	<0,001	<0,003	0,010	0,008	0,02
E 3	n.b.	0,011	0,001	0,007	0,02	0,015	0,06
F 2	n.b.	0,016	0,001	0,004	0,006	0,026	0,024
G 2	0,01	0,005	<0,001	0,001	0,013	0,013	0,03

Table 2: Microstructure in the test castings

Casting No.	Graphite formation form V and VI						Matrix	
	Sand side			Chill side			Pearlite	
	Amount %	Size	Nodule count mm^{-2}	Amount %	Size	Nodule count mm^{-2}	Sand side %	Chill side %
A 3	90-95	4	120	~100	5	250	1-2	<1
B 2	95	5/6	180	~100	6	220	<1	<1
E 2	90	4	120	~100	6/7	350	3-5	<1
E 3	80	5	150	95	6/7	350	~2	<1
F 2	95	5	180	n.b.[x)]	n.b.	n.b.	~1	n.b.
G 2	95	5	200	~100	6/7	400	1-2	<1

[x)] n.b. = not determined

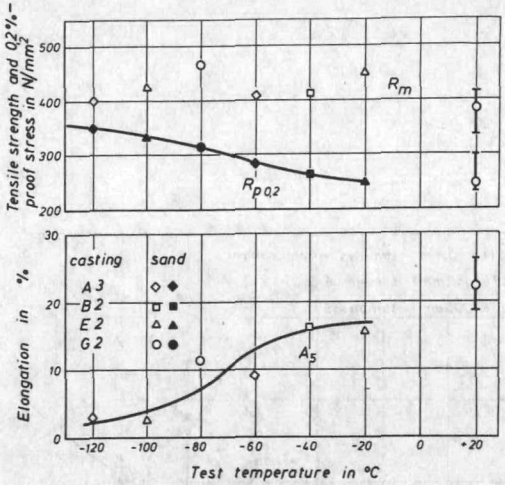


Figure 1: Tensile strength, 0.2 % proof stress and elongation (GGG 40) at different temperatures in heavy sections

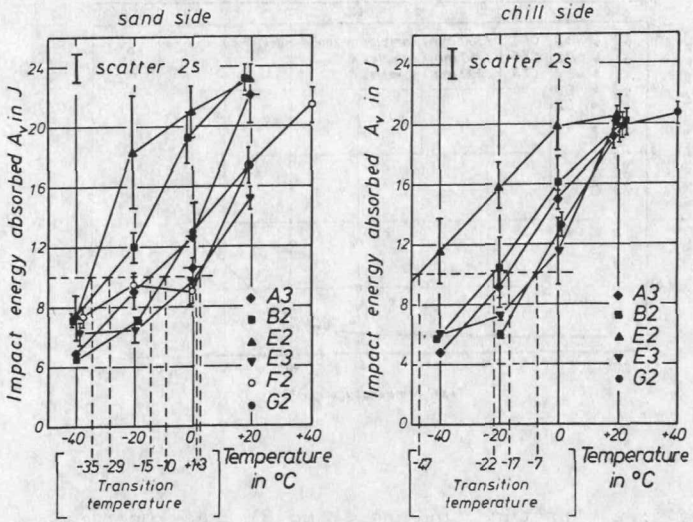


Figure 2: Relationship to temperature of the impact energy A_V (ISO-V) of the ferritic nodular iron in the test castings

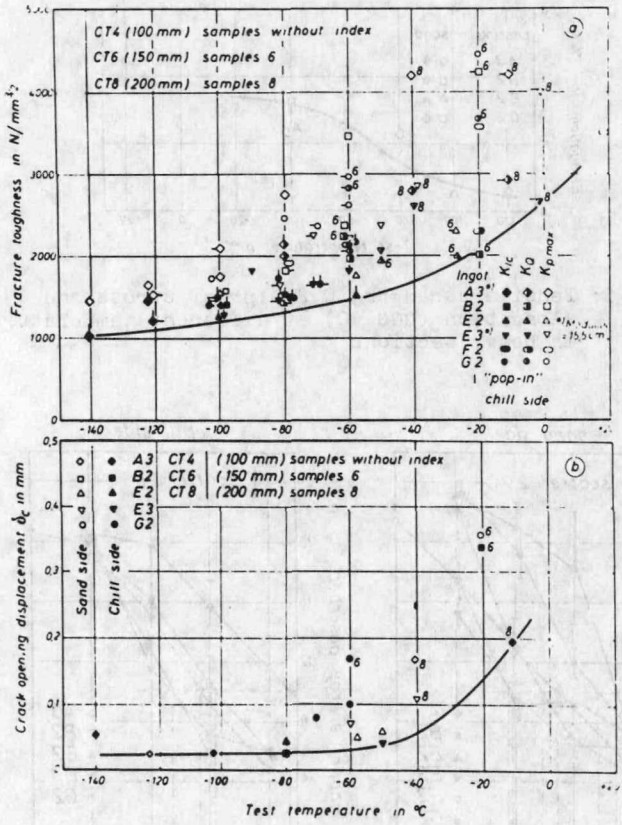


Figure 3: a) Fracture toughness and b) crack opening displacement (COD) of GGG 40 in heavy sections (DIN 1693, part 2) with a modulus of $\geq$ 10,8 cm

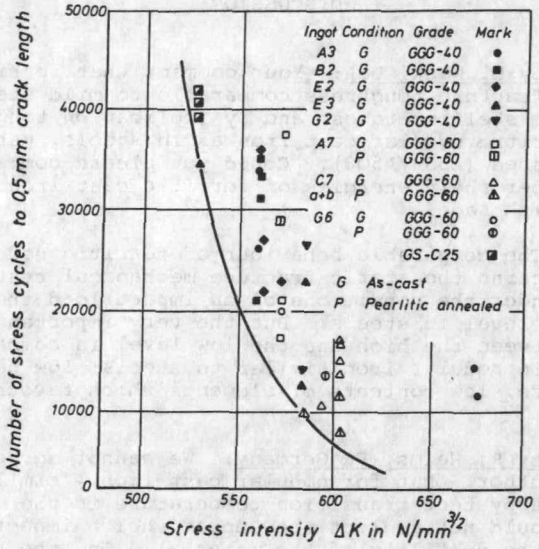


Figure 4: Load cycles and stress intensity for the initiation of a 0,5 mm deep crack in nodular cast iron of heavy section (CT 2 specimens with a notch radius of 0,2 mm)

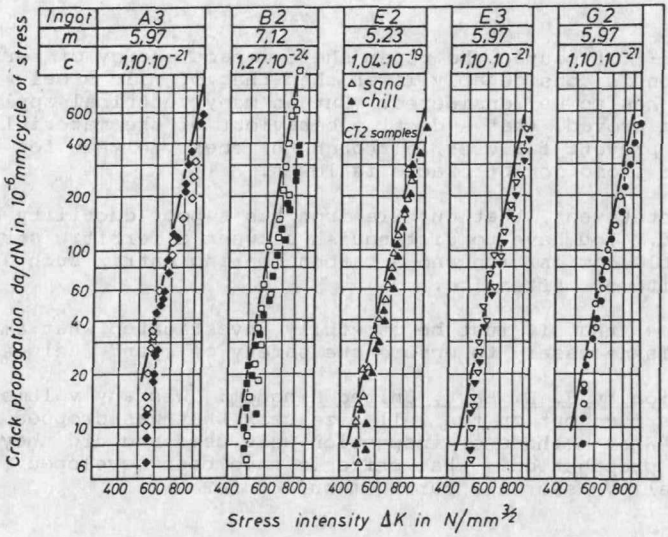


Figure 5: Crack propagation in ferritic nodular cast iron of great section size (modulus $\geq$ 10,8 cm)

DISCUSSION

Question by H. Rack, USA: Your comment that ferritic nodular cast iron has fracture toughness comparable to mild steel concerns me. Certainly upper shelf energies, and by implication toughness, are not as high in ferritic nodular cast iron as in steels, particularly pressure vessel grades (A533/A508). Could you please comment on the relatively low upper shelf energies of ferritic cast iron and how one might design with same?

Answer: The comparable behaviour of ferritic nodular iron and steel only concerns the static fracture mechanical results, not impact tests. Under the conditions of an impact load there is of cou[illegible] a higher shelf level in steels. But the very important transition temperature between the high and the low level is comparable and can be influenced in nodular iron similar to steels: low pearlite rate, small grain size, low contents of elements which favour embrittlement, etc.

Question by R. Helms, FR Germany: We cannot agree with the opinion of the author, that for nodular cast iron - similar as for steel - the Charpy test transition temperature is the most important value. One should not neglect the "upper shelf" impact energy, which is related to the ductility of the material. On the base of the "upper shelf" energy nodular cast iron has rather poor ductility, which indicates low resistance against the initiation of cracks where defects occur as well as against the proportion of cracks. Notwithstanding this point the main question is, what amount of ductility is required for transportation and storage casks for spent fuel elements?

Answer: Of course the upper shelf impact energy of ferritic nodular iron is considerably lower than that of mild steel. This limitation has to be considered. But in many technical applications it had been proved, that a ductile behaviour of the material is sufficient to prevent failures, although the energy niveau for the initiation and promotion of cracks is lower.

Your statement, that ductile iron has a poor ductility cannot be generalised. You have to distinguish between a ferritic structure of a suitable composition and a carbon bearing matrix such as pearlite, bainite or martensite.

I agree, that it must be carefully investigated what amount of ductility is necessary to ensure the safety of spent fuel casks.

Question by J. D. Hart, United Kingdom: Was any volumetric inspection carried out on the full size cask that was dropped from 9[illegible] at -40°C? What methods of inspection were used and did they dete[illegible] any defect propagation? What criteria have been developed for the routine quality assurance during manufacture?

Answer: After all the crash tests met-l-check and ultrasonic inspection were performed. Apart from the gaps in the cooling fins, no indications of cracking could be observed. In some crash tests strain gauges were applied on the body of the casks. No deformation could be registered outside the directly deformed zones. Fractographic examinations of the fins broken at -40°C showed a 50 % brittle 50 % ductile fracture. The quality assurance during manufacture is regulated by a "Quality Assurance Handbook" and the production and testing plans are approved by BAM.

Comment by H. Rack, USA: I would again like to point out the portance of understanding upper shelf behaviour of ferritic nodular st iron. A similar phenemenon with irradiated pressure vessel beltline materials is presently an active area of concern in the US. It appears that this may require implementation of elastic/plastic fracture mechanics procedures. The lower minimum operating cask temperature -20 to -40°C, relative to pressure vessels may also require that we consider this analysis procedure in general and particularly for ferritic nodular cast irons.

Answer: The applied COD procedure is a testing method for an elasto-plastic behaviour of the material especially if big CT specimens of 150 mm and 200 mm thickness are used. Between -60°C and room temperature an increasing elasto-plastic strain at the roof of the notch could be observed. According to this an augmentation of ductile fracture positions could be seen by fractographic examinations.

Question by G. Arendasy, Austria: 1. Is there any representative number of specimens investigated concerning ductility. 2. What is the guarantee for -40°C ductility?

Answer: 1. Three or four specimens for each temperature. 2. The normal guarantee for heavy section castings is regulated by DIN 1693 part 2. If other temperatures and values are requested, it can be regulated by agreement depending on the design of the casting.

POST TEST EVALUATION OF A FIRE TESTED RAIL SPENT FUEL CASK*

H. J. Rack and H. R. Yoshimura
Sandia National Laboratories,† Albuquerque, NM, USA

During January 1978, a 67 metric ton stainless steel-lead shielded spent fuel shipping container was exposed to a JP-4 jet fuel fire at Sandia National Laboratories. This test was intended to simulate a severe, but highly improbable, fire-associated accident condition. The shipping container used in this test was constructed in 1962. It was a double-walled, lead-shielded cylindrical vessel 3.96 m long and 1.5 m in diameter. The inner cask liner had been fabricated from two pieces of 9.4-mm-thick 304 stainless steel with one girth and one seam weld. The outer shell was manufactured from two pieces of 34.8-mm-thick 304 stainless steel while the top and bottom of the vessel were ACI type CF8 stainless steel castings. Initially, the cask utilized an auxiliary water cooling system consisting of 3.05-mm-thick-walled copper channels longitudinally welded to the outside of the cask inner cavity wall. These channels were subsequently replaced by 304 stainless steel channels when it was discovered that the original copper channels had been dissolved during the first lead pour. Additionally, copper fins were longitudinally welded to the inside of the outer shell to enhance the thermal path between the lead shielding and the outer shell.

Prior to the fire test, the cask/railcar system was subjected to a 131 km/h impact into a massive concrete barrier. The cask survived the impact without failure, suffering only minor cooling fin damage. Post-crash test inspection revealed that the outer shell of the cask had not been penetrated and that the cask had retained its internal cavity pressure.

After the crash test, the cask and railcar were moved to the fire site and placed over a specially constructed pool fire test facility. The railcar was supported by concrete pedestals located centrally in the pool, with wheel trucks under one end of the car to maintain its post impact orientation. The level of the fuel was 1.4 m below the lowest point of the cask. Flame temperatures during the fire test ranged from 1250 to 1475 K with the container showing no signs of degradation for at least 90 minutes. At about 100 minutes into the test, white smoke, indicative of lead combustion, became intermingled with the normal JP-4 fuel black-colored smoke. The fuel supply was then stopped and the fire self-extinguished at about 125 minutes. Because of the weakened condition of the concrete and metal supports and the railcar frame, the railcar toppled on its side at this point--severing all instrumentation connections. Internal thermocouples on the cask indicated, however, that complete lead melt had occurred prior to toppling (1).

Examination of the shipping container following the fire test showed that the cask inner shell, which was the containment boundary, was unperturbed. It did, however, reveal the presence of two macrofissures in the outer cask shell, Fig. 1. Both cracks had their major direction lying parallel to the longitudinal axis of the cask. Crack no. 1 was located within the seam weld fusion zone region while

*This work sponsored U. S. Department of Energy under Contract DE-AC04-76-DP00789.
†A U. S. Department of Energy Facility

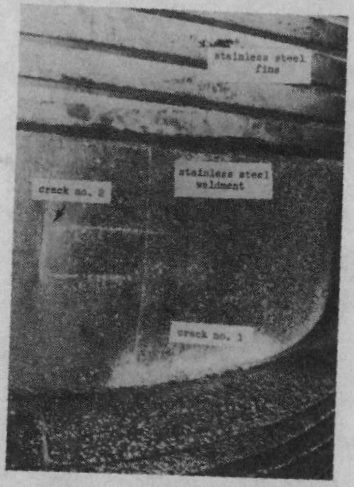


Fig. 1. Enlarged view of cask outer shell containing two macrofissures

crack no. 2 lay in the 304 stainless steel base metal. The latter crack also showed indications of lead seepage and gross plastic deformation (2).

Crack no. 1 was approximately 115 mm long and was located 12 mm from the centerline of the stainless steel seam weld. Examination of the internal surface of the outer shell indicated that the weldment had been fabricated using a backup strip. Chemical analysis of the base metal, fusion zone, and backup strip, Table 1, showed that they all met the requirements of 304 stainless steel although the higher chromium content of the weld fusion zone did suggest that 308 filler wire was used in its manufacture.

Table 1

Chemical Composition of Spent Fuel Shipping Cask Materials

	Element (Weight Percent)								
	Cr	Ni	C	Mn	Si	P	S	Mo	Cu
Cask Body	18.3	9.6	0.03	1.51	0.39	0.02	0.01	0.25	0.24
Fusion Zone	19.3	9.9	0.06	1.01	0.47	0.03	0.01	0.11	0.14
Backup Strip	18.0	9.8	0.06	1.1	0.49	0.024	0.01	0.2	0.24

Observation of the fracture surface showed that crack no. 1 had initiated on the external surface of the outer cask shell and had not completely penetrated the outer shell wall, Fig. 2. Further examination also showed that the original weld had not resulted in full weld penetration. Typically, approximately 20 percent of the outer shell thickness had not been joined by the seam welding procedure utilized for the original cask fabrication. Metallographic sections taken normal to this crack illustrated the multipass character of the weldment. Higher magnification examination revealed the presence of numerous internal microcracks, Fig. 3. These cracks tended to lie along the austenite-δ ferrite interface, changing direction when traversing from one weld puddle to the next. Comparison with previous studies of weld cracking in austenitic stainless steels (3) suggests that these internal microcracks, and by implication crack no. 1, are hot cracks.

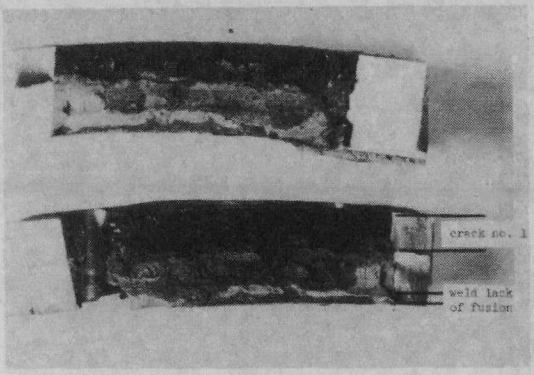


Fig. 2. Macroview of crack no. 1

Fig. 3. Optional micrograph illustrating internal cracking in stainless steel of weld fusion zone.
Magnification : 25x

Hot cracking of austenitic stainless steel weldments is generally associated with the presence of a low melting point liquid film at the austenite-δ ferrite interface. This liquid film reduces the local cohesive strength of the stainless steel weldment. Concurrent stress application then results in crack formation.

Scanning electron microscope observations support the aforementioned hypothesis that crack no. 1 was a hot crack formed in the weld fusion zone during the fire test. Elemental analysis of droplets lying on the fracture surface indicate that they contain substantial amounts of sulfur, probably in the form of complex alloy sulfides, Fig. 4. Comparing typical values for alloy sulfide melting points (1250-1475 K) (3) with measured cask temperatures suggests that these sulfides were molten under the present fire test conditions. The further requirement, an applied stress, was apparently supplied by internal pressurization of the fire-exposed spent fuel shipping cask. Indeed internal pressurization within the lead gamma shield cavity was accentuated in the present instance by a manufacturing oversight whereby passage holes from the shield cavity to appropriate expansion volumes originally provided for in the cask design were omitted during cask fabrication.

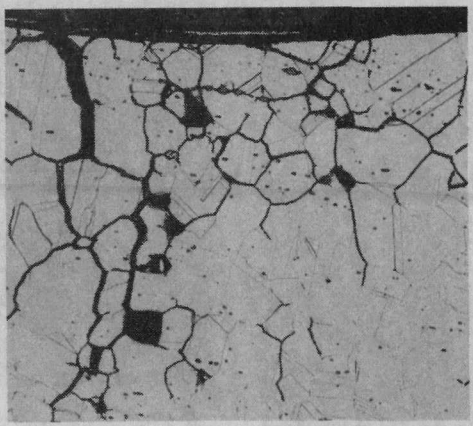


Fig. 6. Optional micrograph illustrating intergranular character of crack in stainless steel. Magnification : 100x

analysis of droplets observed in the copper-stainless steel fusion zone indicated that they contained appreciable quantities of iron and chromium. The form of these droplets and their chemistry suggest that they were formed by a high heat input welding process which caused localized melting of the 304 stainless steel during welding.

Consideration of the above suggests that crack no. 2 originated from micro-cracks formed during improper welding of the copper cooling fins to the stainless steel outer shell. This assumption was supported by elemental analysis of the crack no. 2 fracture surface, Fig. 8, where copper could be detected to a depth of

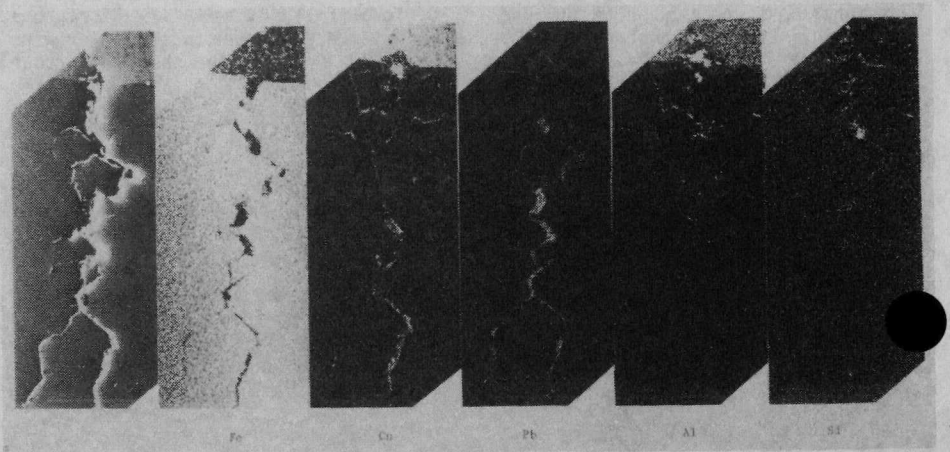
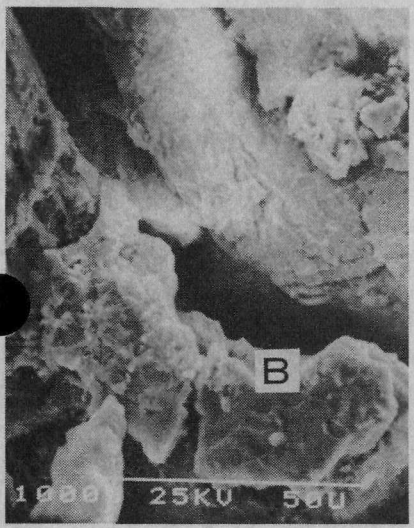
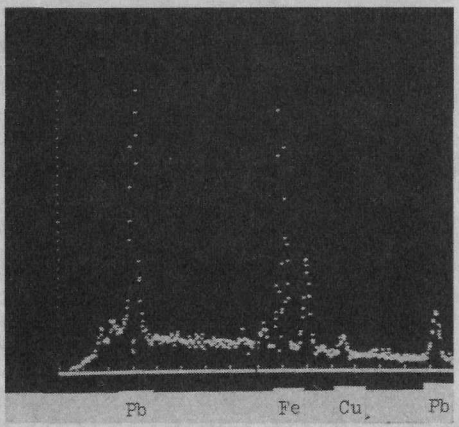


Fig. 7. Identical area elemental distribution photomicrograph of crack emanating from 304 stainless steel/copper joint. Magnification : 100x.



(a)



(b)

Fig. 8. (a) Scanning electron micrograph of crack no. 2 fracture surface and (b) elemental analysis of region B near mid-section of crack no. 2

approximately 17 mm, i.e., 50 pct. of the outer shell wall thickness. It appears then that final crack propagation through the outer shell wall occurred during the fire-test, the driving force for this crack propagation being supplied by the internal pressurization within the lead gamma shield alluded to previously.

Summarizing, a postmortem examination of a large rail-transported spent fuel shipping cask which had been exposed to a JP-4 fuel fire revealed the presence of two macrofissures in the outer cask shell. One, a part-through crack located within the seam weld fusion zone of the outer cask shell, is typical of hot cracks found in stainless steel weldments. The other, a through-crack, was apparently initiated during the formation of a copper-stainless steel dissimilar metal joint, with crack propagation through the cask outer shell having occurred during the fire-test.

REFERENCES

1. H. R. Yoshimura, "Full Scale Simulations of Accidents on Spent-Nuclear-Fuel Shipping Systems," Proceedings of the Fifth International Symposium: Packaging and Transportation of Radioactive Materials, Las Vegas, NV, May 7-12, 1978, pp. 463-476.
2. H. J. Rack and H. R. Yoshimura, "Postmortem Metallurgical Examination of A Fire-Exposed Spent Fuel Shipping Cask," SAND79-1424, Sandia National Laboratories, Albuquerque, NM, April 1980.
3. T. G. Gooch, "Welding Metallurgy of Stainless Steel," Stainless Steel, Iron and Steel Institute Publication No. 117, Iron and Steel Institute, London, 1969, pp. 77-84.
4. H. B. Bott, "How to Weld Copper to Stainless Steel and Mild Steel," Welding Journal, 38, 236 (1959).

DISCUSSION

Question by W. Gibson, Canada: What part, if any, do you think the impact crash test played on the crack propagation within the stainless steel wall of the cask compared to the final effect of the fire test?

Answer: Unfortunately, I cannot answer this question directly. All that one can say is that ultrasonic inspection after the crash test but prior to fire test could not locate any discontinuities. It should also be remembered that the test cask had been operational for many years prior to our receival. This operation may have extended original manufacturing defects. Again we could/did not detect by the NDE procedure used.

CASK MANUFACTURING METHODS AND QUALITY ASSURANCE PROBLEMS*

By

J. Henry Riddle and Howard D. Cross
Exxon Nuclear Company, Inc.

Amado A. Trujillo, Ronald B. Pope and H. J. Rack
Sandia National Laboratories

INTRODUCTION

Existing spent fuel and other radioactive material shipping casks have utilized a variety of construction methods and materials. Casks with steel structures and lead or depleted uranium gamma shields have been widely used for spent fuel transportation. All steel casks, depending upon a heavy steel wall for gamma shielding have also been utilized for processed waste shipping casks. Significant progress has been made recently in Europe in the development of large cast iron casks for spent fuel shipping and storage.

This paper presents the results of a study performed to identify and evaluate cask manufacturing methods and materials which are presently or could be readily available in the United States. An analysis of several candidate materials and processes was made to determine existing capabilities to efficiently manufacture spent fuel, transuranic and high-level waste shipping casks. Emphasis was placed on minimizing cask manufacturing costs while retaining cask maintainability, longevity and adequate quality assurance. Current manufacturing methods and technology as well as any improvements which can reasonably be expected to be available both domestically and internationally have been evaluated for use in the U.S. The major structural aspects of cask design and manufacture were emphasized; whereas detailed features such as covers, seals, lifting and handling trunions, and pressurized venting and sampling devices were addressed only to the extent they affect the major structural components.

The study includes a comparison of the economic and technical advantages to be gained from manufacturing casks specifically for application to long-cooled fuel (5 years or greater) and waste materials.

DESIGN EVALUATION

This phase of the study resulted in development of typical design concepts to be used as a basis for evaluation of manufacturing and QA problems and relative costs. Existing conceptual and licensed designs have been used for comparison purposes and as a basis to confirm the

* Work was done for U. S. Department of Energy under Contract DE-AC04-76-DP00 789 through Sandia National Laboratories Transportation Technology Center.

adequacy of the design concepts developed. Typical designs were developed for spent fuel, transuranic waste and high level commercial and defense waste road and rail shipping casks. The essential design features of a LWR fuel cask are shown in Figure 1. The design consists of a concentric arrangement of cavity wall, a gamma shield, a containment wall and a neutron shield. Designs of each of the four cask types are similar except that neutron shields are not required for defense waste and transuranic waste casks.

In all cases design alternatives using steel structures with a lead, depleted uranium, or all steel (or cast iron) gamma shield were developed. In the cases where a steel (or cast iron) gamma shield is utilized the gamma shielding function and inner and outer container functions can be combined in a single structure.

Gamma shield thicknesses were selected to comply with paragraphs 226 and 229 of IAEA Safety Series #6 (49 CFR 173.393 and 173.398). In the case of LWR fuel casks and high level commercial wastes, which require neutron shielding, the neutron shield was included in accordance with the same requirements. The forms of neutron shield considered included the conventional borated water shield as shown schematically in Figure 1, placement of boron carbide or other neutron absorbers in formed or drilled holes in the cask structure wall (Figure 2) and borated cast-in-place silicone rubber or other solid neutron absorbers as shown in Figure 3.

Design features affected by thermal requirements and structural (including accident) requirements were selected by comparison with existing designs, i.e., detailed re-evaluation of these requirements were not included in this study.

The designs developed have been based on the following payload requirements:

LWR Road Cask - 2 PWR/4 BWR Fuel Elements
LWR Rail Cask - 12 PWR/28 BWR Fuel Elements (PWR arrangement shown in Figure 4)
High Level Commercial Waste Road Cask - One Container at 750 Kg
High Level Commercial Waste Rail Cask - Seven Containers at 750 Kg Each
High Level Defense Waste Road Cask - One Container at 3040 Kg
High Level Defense Waste Rail Cask - Three Containers at 3040 Kg Each
Transuranic Waste Road Cask - One Container at 3200 Kg
Transuranic Waste Rail Cask - Three Containers at 3200 Kg Each

Maximum empty road cask weights were established as follows: LWR-22,100 Kg, HLCW-23,300 Kg, HLDW-20,180 Kg, TRU-20,400 Kg.

Maximum empty rail cask weights were established as follows: LWR-70,750 Kg, HLCW-77,770 Kg, HLDW-71,180 Kg, TRU-70,680 Kg. Cask size and weight limits were based on legal limits and practical considerations. The truck cask weight limit was based on a maximum gross vehicle weight of 36,300 Kg (80,000 lbs.) and the rail cask weight limit was based on a typical contemporary reactor plant crane limit of 90,700 Kg (100 U.S. tons).

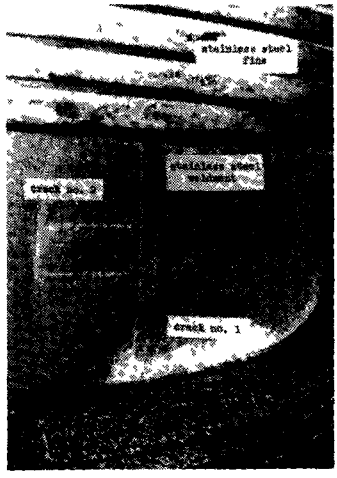


Fig. 1. Enlarged view of cask outer shell containing two macrofissures

crack no. 2 lay in the 304 stainless steel base metal. The latter crack also showed indications of lead seepage and gross plastic deformation (2).

Crack no. 1 was approximately 115 mm long and was located 12 mm from the centerline of the stainless steel seam weld. Examination of the internal surface of the outer shell indicated that the weldment had been fabricated using a backup strip. Chemical analysis of the base metal, fusion zone, and backup strip, Table 1, showed that they all met the requirements of 304 stainless steel although the higher chromium content of the weld fusion zone did suggest that 308 filler wire was used in its manufacture.

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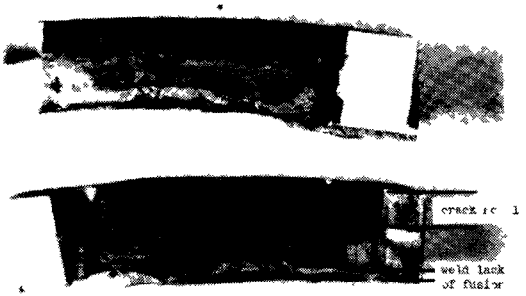

Fig. 2. Macroview of crack no. 1

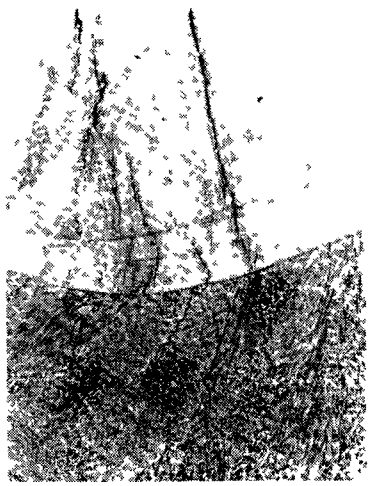

Fig. 3. Optional micrograph illustrating internal cracking in stainless steel of weld fusion zone.
Magnification : 25x

Hot cracking of austenitic stainless steel weldments is generally associated with the presence of a low melting point liquid film at the austenite-δ ferrite interface. This liquid film reduces the local cohesive strength of the stainless steel weldment. Concurrent stress application then results in crack formation.

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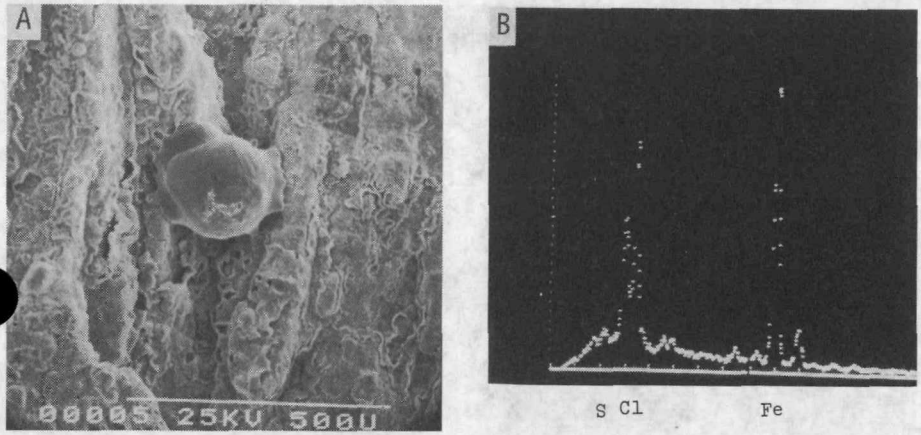


Fig. 4. (a) Scanning electron micrograph of crack no. 1 fracture surface and (b) elemental analysis photo of droplet shown in (a)

Crack no. 2, which had penetrated the cask outer shell, was approximately 70 mm long and was located in the stainless steel base metal. Visual observations of the inside surface of the outer shell immediately behind crack no. 2 confirmed that copper fins had been welded to the outer shell. Further study indicated that many of these copper/stainless steel weldments were cracked, Fig. 5. Detailed examination of these cracks showed that they were predominantly intergranular, Fig. 6, following the austenitic stainless steel grain boundaries. Microprobe examination also showed that the cracks contained significant quantities of copper and lead and lesser amounts of aluminum and silicon, Fig. 7.

Generally, successful welding of copper to stainless steel requires that low heat inputs be combined with the use of a nickel filler wire (4). Microprobe examination indicated that, in the present instance, a copper base filler wire had been used to join the copper cooling fins to the cask outer shell. Further

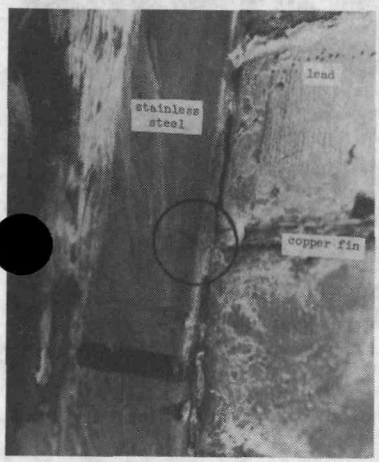


Fig. 5. Typical crack observed originating from copper-stainless steel weldment

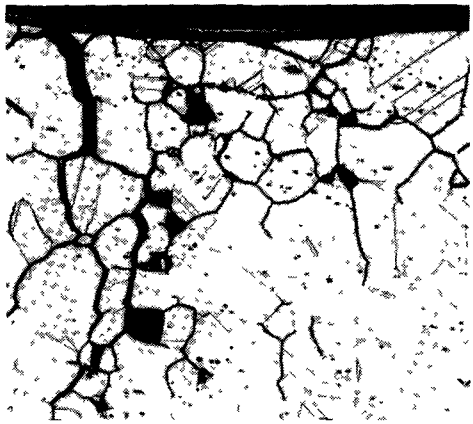


Fig. 6. Optional micrograph illustrating intergranular character of crack in stainless steel. Magnification : 100x

analysis of droplets observed in the copper-stainless steel fusion zone indicated that they contained appreciable quantities of iron and chromium. The form of these droplets and their chemistry suggest that they were formed by a high heat input welding process which caused localized melting of the 304 stainless steel during welding.

Consideration of the above suggests that crack no. 2 originated from micro-cracks formed during improper welding of the copper cooling fins to the stainless steel outer shell. This assumption was supported by elemental analysis of the crack no. 2 fracture surface, Fig. 8, where copper could be detected to a depth of

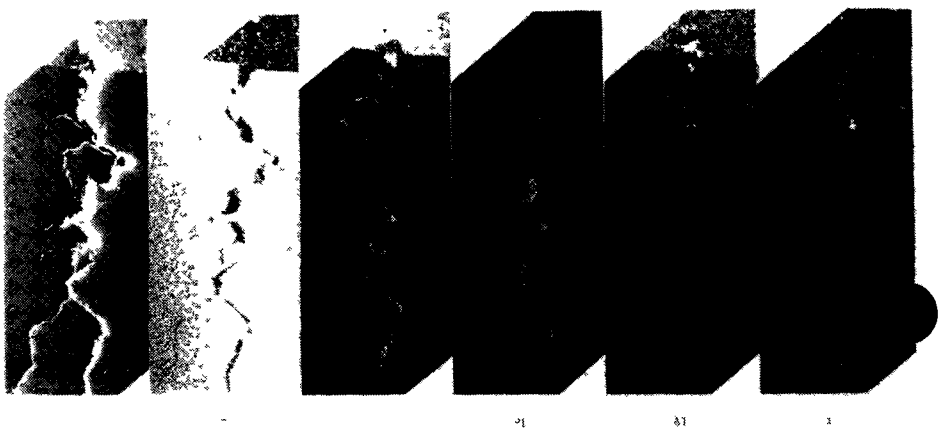
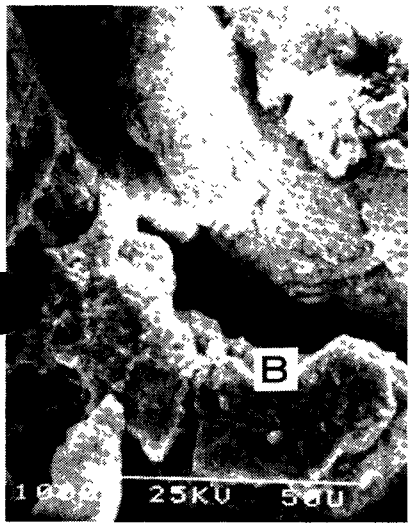
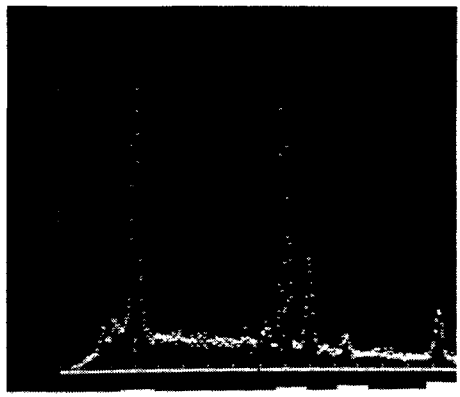


Fig. 7. Identical area elemental distribution photomicrograph of crack emanating from 304 stainless steel/copper joint. Magnification : 100x.



(a)



(b)

Fig. 8. (a) Scanning electron micrograph of crack no. 2 fracture surface and (b) elemental analysis of region B near mid-section of crack no. 2

approximately 17 mm, i.e., 50 pct. of the outer shell wall thickness. It appears then that final crack propagation through the outer shell wall occurred during the fire-test, the driving force for this crack propagation being supplied by the internal pressurization within the lead gamma shield alluded to previously.

Summarizing, a postmortem examination of a large rail-transported spent fuel shipping cask which had been exposed to a JP-4 fuel fire revealed the presence of two macrofissures in the outer cask shell. One, a part-through crack located within the seam weld fusion zone of the outer cask shell, is typical of hot cracks found in stainless steel weldments. The other, a through-crack, was apparently initiated during the formation of a copper-stainless steel dissimilar metal joint, with crack propagation through the cask outer shell having occurred during the fire-test.

REFERENCES

1. H. R. Yoshimura, "Full Scale Simulations of Accidents on Spent-Nuclear-Fuel Shipping Systems," Proceedings of the Fifth International Symposium: Packaging and Transportation of Radioactive Materials, Las Vegas, NV, May 7-12, 1978, pp. 463-476.
2. H. J. Rack and H. R. Yoshimura, "Postmortem Metallurgical Examination of A Fire-Exposed Spent Fuel Shipping Cask," SAND79-1424, Sandia National Laboratories, Albuquerque, NM, April 1980.
3. T. G. Gooch, "Welding Metallurgy of Stainless Steel," Stainless Steel, Iron and Steel Institute Publication No. 117, Iron and Steel Institute, London, 1969, pp. 77-84.
4. H. B. Bott, "How to Weld Copper to Stainless Steel and Mild Steel," Welding Journal, 38, 236 (1959).

DISCUSSION

Question by W. Gibson, Canada: What part, if any, do you think the impact crash test played on the crack propagation within the stainless steel wall of the cask compared to the final effect of the fire test?

Answer: Unfortunately, I cannot answer this question directly. All that one can say is that ultrasonic inspection after the crash test but prior to fire test could not locate any discontinuities. It should also be remembered that the test cask had been operational for many years prior to our receival. This operation may have extended original manufacturing defects. Again we could/did not detect by the NDE procedure used.

CASK MANUFACTURING METHODS AND QUALITY ASSURANCE PROBLEMS*

By

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INTRODUCTION

Existing spent fuel and other radioactive material shipping casks have utilized a variety of construction methods and materials. Casks with steel structures and lead or depleted uranium gamma shields have been widely used for spent fuel transportation. All steel casks, depending upon a heavy steel wall for gamma shielding have also been utilized for processed waste shipping casks. Significant progress has been made recently in Europe in the development of large cast iron casks for spent fuel shipping and storage.

This paper presents the results of a study performed to identify and evaluate cask manufacturing methods and materials which are presently or could be readily available in the United States. An analysis of several candidate materials and processes was made to determine existing capabilities to efficiently manufacture spent fuel, transuranic and high-level waste shipping casks. Emphasis was placed on minimizing cask manufacturing costs while retaining cask maintainability, longevity and adequate quality assurance. Current manufacturing methods and technology as well as any improvements which can reasonably be expected to be available both domestically and internationally have been evaluated for use in the U.S. The major structural aspects of cask design and manufacture were emphasized; whereas detailed features such as covers, seals, lifting and handling trunions, and pressurized venting and sampling devices were addressed only to the extent they affect the major structural components.

The study includes a comparison of the economic and technical advantages to be gained from manufacturing casks specifically for application to long-cooled fuel (5 years or greater) and waste materials.

DESIGN EVALUATION

This phase of the study resulted in development of typical design concepts to be used as a basis for evaluation of manufacturing and QA problems and relative costs. Existing conceptual and licensed designs have been used for comparison purposes and as a basis to confirm the

* Work was done for U. S. Department of Energy under Contract DE-AC04-76-DP00 789 through Sandia National Laboratories Transportation Technology Center.

adequacy of the design concepts developed. Typical designs were developed for spent fuel, transuranic waste and high level commercial and defense waste road and rail shipping casks. The essential design features of a LWR fuel cask are shown in Figure 1. The design consists of a concentric arrangement of cavity wall, a gamma shield, a containment wall and a neutron shield. Designs of each of the four cask types are similar except that neutron shields are not required for defense waste and transuranic waste casks.

In all cases design alternatives using steel structures with a lead, depleted uranium, or all steel (or cast iron) gamma shield were developed. In the cases where a steel (or cast iron) gamma shield is utilized the gamma shielding function and inner and outer container functions can be combined in a single structure.

Gamma shield thicknesses were selected to comply with paragraphs 226 and 229 of IAEA Safety Series #6 (49 CFR 173.393 and 173.398). In the case of LWR fuel casks and high level commercial wastes, which require neutron shielding, the neutron shield was included in accordance with the same requirements. The forms of neutron shield considered included the conventional borated water shield as shown schematically in Figure 1, placement of boron carbide or other neutron absorbers in formed or drilled holes in the cask structure wall (Figure 2) and borated cast-in-place silicone rubber or other solid neutron absorbers as shown in Figure 3.

Design features affected by thermal requirements and structural (including accident) requirements were selected by comparison with existing designs, i.e., detailed re-evaluation of these requirements were not included in this study.

The designs developed have been based on the following payload requirements:

LWR Road Cask - 2 PWR/4 BWR Fuel Elements
LWR Rail Cask - 12 PWR/28 BWR Fuel Elements (PWR arrangement shown in Figure 4)
High Level Commercial Waste Road Cask - One Container at 750 Kg
High Level Commercial Waste Rail Cask - Seven Containers at 750 Kg Each
High Level Defense Waste Road Cask - One Container at 3040 Kg
High Level Defense Waste Rail Cask - Three Containers at 3040 Kg Each
Transuranic Waste Road Cask - One Container at 3200 Kg
Transuranic Waste Rail Cask - Three Containers at 3200 Kg Each

Maximum empty road cask weights were established as follows: LWR-22,100 Kg, HLCW-23,300 Kg, HLDW-20,180 Kg, TRU-20,400 Kg.

Maximum empty rail cask weights were established as follows: LWR-70,750 Kg, HLCW-77,770 Kg, HLDW-71,180 Kg, TRU-70,680 Kg. Cask size and weight limits were based on legal limits and practical considerations. The truck cask weight limit was based on a maximum gross vehicle weight of 36,300 Kg (80,000 lbs.) and the rail cask weight limit was based on a typical contemporary reactor plant crane limit of 90,700 Kg (100 U.S. tons).

The cask cavity dimensions and the required structural and gamma shield wall thicknesses are shown in Table 1. The resulting truck and rail cask weight estimates for LWR fuel casks are given in Table 2. This table shows that the payload objectives for one year cooled fuel can only be achieved with a uranium gamma shielded cask design. However, the payload objective can be achieved with five year cooled fuel using the all steel (or cast iron) or lead gamma shield with either borated water or silicone rubber neutron shields. Neutron shields placed in drilled holes in the cask body increase cask weights sufficiently to make this design over weight for all cases. The five year cooled fuel allows a steel or iron wall thickness reduction of about 20%, which in turn results in a cask weight reduction of almost 30%. Similar results were obtained for the high level commercial waste casks. The payload objectives are readily achievable for the defense and transuranic waste casks in all configurations.

MATERIAL AND MANUFACTURING EVALUATION

This phase of the study addressed the U. S. industrial capability to manufacture the basic cask designs developed during the design evaluation phase. This evaluation included consideration of the various technical and quality requirements in addition to manufacturing capacity considerations. In the case of steel components all conventional manufacturing methods including fabricated plate, forgings and castings were considered. The requirements of and limitations imposed on manufacturing techniques and some design arrangements by the Non-Destructive Examination requirements were considered, both for fabrication as well as in-service examination. The objective of this phase of the study was to develop a series of recommendations identifying preferred fabrication techniques for each type of cask in both rail and truck sizes.

Low Temperature Ductility

A design objective was to develop cask concepts that can be used at ambient temperatures as low as minus 40°C, which for this study was assumed to result in a requirement that the Nil Ductility Transition (NDT) temperature for the structural components be lower than minus 73°C. This requirement conservatively assumes that the outer structure of the cask would be at ambient temperature, i.e., no credit is taken for the insulating effect of the neutron shield material. Of the range of materials evaluated to date, austenitic stainless steels appear to be the only materials clearly capable of meeting this requirement. Low alloy steels also show some potential for meeting this requirement, particularly in thinner sections. Development of nodular cast iron and ferritic stainless steel castings meeting these requirements is being vigorously pursued in Europe; these developments are reported elsewhere during this symposium.

Uranium Shielded Casks

Although uranium is a very efficient gamma shield it's use in future cask designs in the U. S. is prejudiced by the lack of manufacturing capability. There is no longer a domestic U. S. industrial

capability to cast and machine the large uranium shield sections that would be required for rail casks in particular. The projected market for shipping casks may not be sufficient to financially justify re-establishing this manufacturing capability. Furthermore this manufacturing evaluation has shown that the uranium shielded casks are significantly more expensive than either lead shielded or all steel casks because of the high cost of the finished uranium shield components and because of the high assembly labor requirement resulting from the design complexity.

The uranium shielded LWR casks do have the advantage of being the only designs capable of meeting the payload goals for one year cooled fuel. However, the U. S. industry need for the foreseeable future will be predominantly shipment of five-year cooled and older fuel. Therefore, evaluations of cask alternatives have been based on the design of the 5 year cooled fuel and waste cask designs.

All Steel Casks

Very heavy wall sections, set by shielding requirements are required for the all steel casks. For example, a rail cask for 5 year cooled LWR fuel requires a wall thickness of about 22 cm. These heavy walls present some manufacturing problems and in several cases preclude the use of some processes because of manufacturing size limits. These size limitations result both from basic materials limitations (plate thickness, billet size, etc.) and manufacturing capacity limitations. A composite ranking of the available processes was developed considering cost, product quality, availability and manufacturing limitations. The resulting ranking of the fabrication processes for wall sections of less than and greater than 10 cm are as follows:

Rank	Wall Thickness Less Than 10 cm	Wall Thickness Greater Than 10 cm
1	Formed & welded plate	Forging
2	Forging	Formed and welded plate
3	Casting	Casting

Obtaining the small bend radius in thick walled sections required (especially for truck casks) makes the formed and welded plate fabrication process impractical. Availability of high quality stainless steel plate is limited to about 6 inch maximum thickness. Application of stainless steel plate to most of the all-steel designs is therefore prevented, and this method is therefore limited to the thinner wall sections including the inner and outer steel walls of the uranium and lead shielded casks.

High integrity forging in the heavy sections required for shipping casks are available in both stainless and low alloy steels. The complete body section of the largest casks (LWR rail casks) could be forged in one piece from low alloy steels at selected locations in the U.S. At least two pieces would be required for stainless steel forgings for comparable applications because of billet size limitations and the higher forming pressures that are required. However for both materials, high quality forgings can be obtained in sections large enough to minimize

the amount of assembly welding and the attendant nondestructive examination. This is likely to provide additional advantages as the requirements for In-Service Inspection (ISI) develop.

The simple geometry involved with the shipping cask structures considered in this study does not pose any restrictions on the use of castings for the body structure but at the same time does not particularly favor the casting technique. The restrictions that do apply involve the foundry melting/pouring capacity and the equipment required for handling the poured casting. United States foundries have the pouring capacity to make high quality steel castings (stainless and low alloy steel) of the required weight but are generally limited to a finished casting height of about 250 cm. Consequently three or more sections welded together would be required for the large all-steel casks. Steel castings suffer a disadvantage in comparison to forgings in that a substantial amount of relatively expensive weld repair (so-called upgrading) can be expected before the castings will pass demanding NDE requirements.

The technology for production of very large and high quality nodular iron castings similar to the recent German developments is not well developed in the U.S. However the foundry capacity for making the very large castings is available.

Laminated Steel Casks

Due to the high cost of stainless and low alloy steel the use of laminated steel sections was included in this evaluation. The basic design of such a cask is similar to a lead or uranium shielded cask where the shielding would be provided by a low grade non-structural steel or cast iron section sandwiched between stainless steel cavity and containment walls. To obtain the required heat transfer properties in the finished cask it was assumed that all sections would be machined and shrunk-fit together. A cost evaluation of this concept shows that the design is not competitive with the solid steel casks due to the added labor costs and suffers the additional disadvantage of presenting some NDE problems.

Lead Shielded Casks

Lead is the most common of all shielding materials in use in current transportation casks. A number of potential future cask fabricators in the U.S. have expertise in casting large lead shielded casks and other lead shielded structures.

There are areas of concern regarding the use of lead as a shield material, principally in the design for resistance to fire accidents, which are beyond the scope of this study but must be addressed in the future. For the purpose of this study the assumption was made that lead shielded casks will be licensible. The manufacturability must take into account the design and fabrication complexities involved with providing expansion areas within the cask wall to accept molten lead in the event of a severe fire accident.

This manufacturing study has shown that the costs of lead shielded casks are substantially lower than uranium shielded casks and somewhat lower than the all steel cask designs.

QUALITY ASSURANCE

Evaluation of quality assurance/non-destructive examination methods has been somewhat hampered in that specific design & fabrication code requirements do not exist in the U.S. at this time. While volumetric examination techniques do exist which allow NDE of both base metal and weld areas the adequacy of these techniques for shipping cask applications have yet to be verified. However, it has been concluded that acceptable surface examination techniques, where called for, exist. Adequate methods are not presently developed for measuring clad overlay thickness (stainless overlay on low alloy or carbon steel) or for locating and measuring gaps between layers in laminated vessels. Gamma scanning techniques are commonly used on lead shielded casks to detect voids in the lead shield but are time consuming and consequently expensive.

CONCLUSIONS

The conclusions that have been drawn from this study, which is still in progress are as follows:

1. If spent fuel payloads are in fact cooled for five or more years prior to transportation there is a significant advantage to be gained from cask designs which are tailored to this longer cooling time.

2. The most economic cask design is that which utilizes a lead gamma shield.

3. The all-steel, single wall casks show substantial potential benefits resulting from ready availability and the cost and inspection advantages that result from the design simplicity.

4. The future use of depleted uranium shielded casks in the United States is expected to be low as a result of the high cost and lack of manufacturing facilities.

5. Of the three types of neutron absorber systems evaluated, the borated cast-in-place silicone rubber shows significant technical and cost advantages.

6. Design and fabrication code requirements and regulatory requirements yet to be developed in the United States are likely to have a profound effect on the next generation of transportation casks. The specific areas of potential major impact include: (a) the licensability of lead shielding; (b) final development of low temperature material ductility requirements; (c) fabrication quality requirements and the attendant inspection requirements; (d) development of in-service inspection requirements.

TABLE 1

Cask Dimensional Requirements

Cask Type	Mode Of Shipment	Cavity Dia.(cm)	Cavity Ht.(cm)	Aging/ Cooling Period (Years)	Wall Thickness Dimensions (cm) All Steel	Steel/Lead	Steel/Uranium
LWR Spent Fuel	Truck (2/4)	24.4 x 50.8 Rectangular	452	1	25.4	3.5/16.0	3.5/8.9
				5	20.3	3.5/11.4	3.5/6.4
	Rail (12/28)	127	477.5	1	29.0	5.1/18.0	5.1/9.9
				5	22.1	5.1/13.0	5.1/7.6
Commercial Reprocessing	Truck	35.6	315	1	29.0	3.5/19.1	3.5/10.4
				5	22.9	3.5/14.5	3.5/7.9
	Rail	111.8	315	1	32.0	5.1/22.1	5.1/11.9
				5	25.4	5.1/18.0	5.1/9.4
Defense Reprocessing	Truck	35.6	315	1	18.3	3.5/9.4	3.5/5.1
				5	17.5	3.5/8.4	3.5/4.6
	Rail	111.8	315	1	20.3	5.1/11.4	5.1/6.1
				5	19.6	5.1/9.9	5.1/5.6
Transuranic	Truck	71.1	467.4	1	9.9	3.5/5.6	3.5/3.0
				5	8.9	3.5/4.6	3.5/2.5
	Rail	142.2	467.4	1	10.9	5.1/6.1	5.1/3.0
				5	9.9	5.1/5.6	5.1/3.0

TABLE 2

LWR Cask Weights

Weights In Metric Tons

Mode	Cooling Period	Gamma Shield / Neutron Shield: Steel / Water	Steel / Silicone	Lead / Water	Lead / Silicone	Uranium / Water	Uranium / Silicone
Truck	1 Year	33*	30*	31*	27*	24*	21
	5 Year	24*	22	20	18	17	15
Rail	1 Year	77*	73*	84*	73*	71	62
	5 Year	55	53	64	54	57	49

* Overweight cask.

Figure 1
TYPICAL LWR CASK

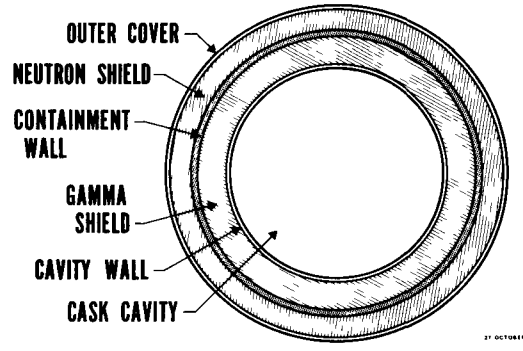


Figure 2
CASK with NEUTRON SHIELD HOLES BORED IN CASK BODY

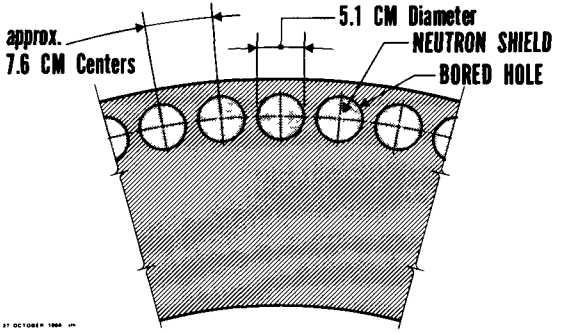


Figure 3
CASK with BORATED CAST-IN-PLACE SILICONE NEUTRON SHIELD

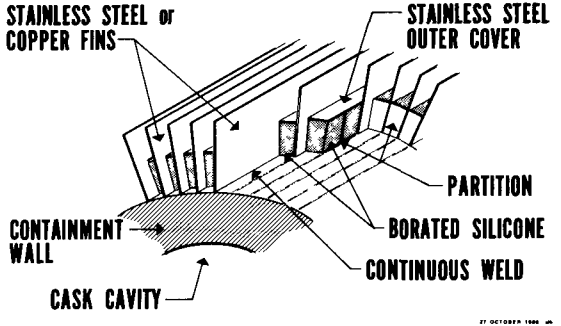
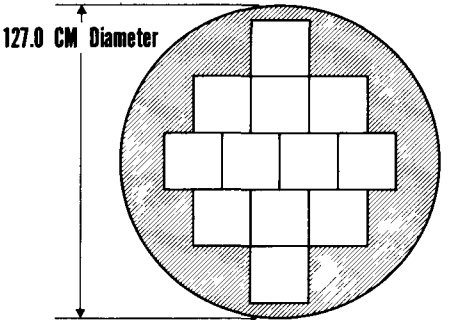


Figure 4
TWELVE (12) ELEMENT PWR RAIL CASK CAVITY ARRANGEMENT



DISCUSSION

Question by R. Sieber, FR Germany: Did you come to a conclusion in the described study in respect to the future use of lead shielded casks? Did the study show whether inspection during fabrication, i.e. quality assurance, and reinspection poses serious problems?

Answer: The future use of lead shielded casks will be heavily dependent on the requirements which are developed for both initial production inspection and in-service inspection. Volumetric examination techniques for the laminated casks including lead and uranium gamma shielded casks are not well developed.

PRODUCTION OF CASTOR CASKS FOR THE TRANSPORTATION AND STORAGE OF SPENT FUEL ELEMENT BUNDLES - MANAGEMENT AND QUALITY ASSURANCE

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1. INTRODUCTION

This paper reports first experience in the production of CASTOR casks, casks of the third generation, cast casks, which are nickel-plated and which fit the shape of the material to be packed.

With regard to design and quality assurance accompanying manufacture, CASTOR transport and storage casks must comply with traffic law requirements and, in the case of use for intermediate storage, also with atomic law requirements. Quality assurance is carried out by means of work and testing plans, whereby, in each case, depending on safety relevance and type of test, a two or three fold coverage is made. The safety relevance, i.e. assignment of testing, is in this case not only derived from the specifications for the component, but the safety margins also play an important role.

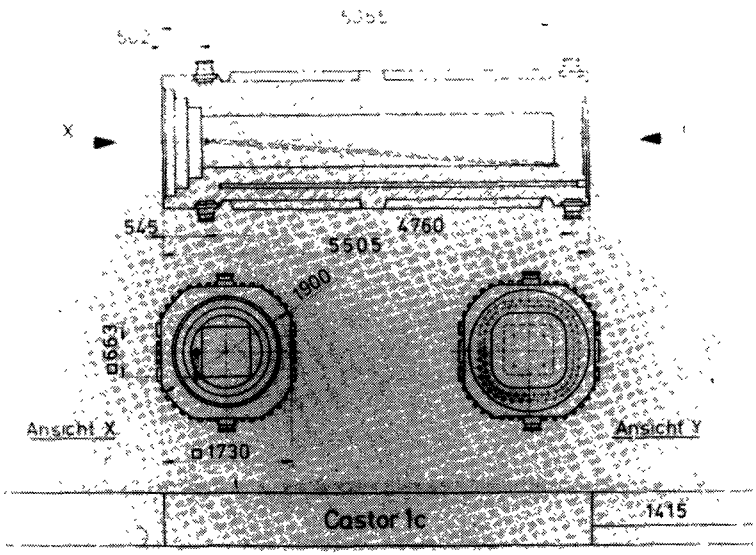

Figure 1

The main component of the CASTOR cask is the cavity. Fabrication and quality assurance is demonstrated by the manufacture of this component.

Figure 1 shows a diagram of the cask, where in particular its quadrilateral cross-section which fits the form the the fuel elements can be seen. The reason for this shape is to save weight but still permitting extensive shielding and optimum heat dissipation. Particular attention should be paid to the decommissioning connection which, in the case of a quadrilateral construction, can be positioned at an incline in the wall.

The other components necessary for completion are not considered individually. In each case they link together and are assembled with the cask cavity.

2. PRODUCTION PREPARATION

In order to establish an intermediate storage facility designed for 120 t fuel on the site of a nuclear power plant, a first series of 8 CASTOR transport and storage casks for 24 t spent fuel should be produced. Delivery begins within 6 months after technical clearance and should be finished 9-10 months later. This means that the casks are produced in 14-day cycles; two casks must be cast a month and subjected to further processing.

In order to guarantee this procedure, it is necessary to work with several moulding devices and to operate these devices alternately. Furthermore, 800-900 t selected starting material must be available for the smelt, as well as the required quantities of special moulding materials.

3. PRODUCTION

Casting of five casks has already taken place. The remainder are expected to be carried out this year or the beginning of 1981.

To give an idea of the procedure followed, the amount of time necessary for the individual fabrication steps follows.

A period of about 14 days is needed for moulding; casting can be carried out in less than 3 minutes. Cooling follows and takes about 3 weeks. Another 4-5 weeks are necessary for cleaning and testing. The preparation of a cask cavity with regard to casting therefore takes about 10 weeks.

Manufacture of the mould, smelting of the cast iron, preparation the cast iron (the injection processes) and casting itself is ed down in work and testing plans and in operating instructions, which are cross-checked at the test-relevant witness points.

A smelting plan is established for the smelting process, as due to the furnace capacity (about 10-20 t/h) the cast iron is to be collected to make a quantity of about 100 t available for casting. Analysis of the smelting process is continuously checked and recorded, the casting temperature is checked and recorded before casting.

Casting is carried out vertical to the cask axis; here it is particularly important that the rise in the liquid iron takes place unimpeded and that the gases can escape. Segregations are separated off during the smelting process into the machining allowance part.

The casting process is followed by the controlled cooling period. Cooling is supervised by measuring techniques and takes about three weeks.

After cooling the casting is drained, i.e. the unfinished piece is removed from the moulding region and exposed. The cleaning and dressing operations follow, i.e. the adhering sand which remains is removed from the casting which is at this point still at a temperature of 100°C. The inlet and pouring systems, i.e. the region out of wh[illegible] on solidification, material flows into the casting and in which seg[illegible]regations collect, are cut off. After cleaning and cutting off the pouring systems, which takes 2-3 weeks, a first inspection of the quality of the casting is made. This mechanical pre-examination of the cast samples is carried out internally, i.e. by the manufacturer. The first complete inspection covers:

- visual inspection
- gauging
- structure inspection
- destructive and non-destructive materials testing.

After successful completion of these examinations, which take about one week, the cavity is internally cleared to such an extent that it can be tested by the customer and, where necessary, by experts designated by the official consultants. In addition to this, the cast samples are stamped and taken away and cut into pieces according to the sample-taking plan and submitted to destructive materials testing. In the case of non-destructive materials testing, the cast which in the meantime has been finely polished, undergoes an ultra-sonic test and, in so far as is necessary, an inspection for surface cracks. At the same time gauging of the casting is carried out.

The cask is released for machining when the values laid down in the material data sheets and test specifications have been attained.

4. MACHINING

In the machining process, the cavity casting and the unworked parts for the lid, trunnions, screws, plugs, cover plates etc. are joined together. In this paper, priority is given to further processing of the cavity.

As a first step, renewed gauging of the cask is performed and reference surfaces for processing are established. The external contours are then reworked and the seats for sealing and the trunnions are machined. The dimensions for galvanic nickel plating are already taken into account. On completion of, or parallel to the processing of the surface, approximately 80 deep hole borings are made for the moderator piles and the decommissioning hole. The first machining

phase finishes with the intermediate assembly of the moderator piles and sealing of the moderator pile borings.

The next step is the galvanic nickel plating, whereby a rust-proof, wear-resistant surface at least 1 mm thick (finished thickness) is galvanically coated so that it is adhesive and completely free of pores. On the external surface it encloses the head and base regions. and covers the complete internal surface including the decommissioning boring.

After galvanic plating, renewed machining is carried out, which, however, only affects the sealing seat region, in order to produce the required accuracy of gauge and surface quality.

All operations are monitored with regard to quality assurance during this phase. This means that in each case after mechanical machining, surveillance by gauging follows and the surface quality of the cask to be galvanically nickel plated is inspected and recorded.

The galvanic plating itself is carried out by means of a pre-tested process. The outcome of the plating with regard to quality, adhesion, absence of pores, hardness of the coating, thickness of the plating, is 100 % checked and recorded.

The mechanical machining process ends with reworking of the galvanically plated cask and processing of the other unworked pieces as well as quality assurance measures.

5. ASSEMBLY

Assembly is carried out by means of an assembly plan. To this end manufacturing steps and also working specifications, e.g. torques to be expended, installation dimensions, etc. are recorded and specified. The assembly ends with the commissioning tests, which are carried out under the supervision of the local expert designated by the authorities. They cover pressure testing, as well as the leakage test which starts as a pressure rise method and, after positive results, is concluded as a helium leak test with a nominal value of $\leq 10^{-7}$ mbar l/s leak rate.

Handling tests, such as insertion of the fuel element basket, fitting of the lid, etc., are carried out automatically on completion, i.e. assembly, and are not repeated individually.

Before delivery takes place, the epoxy decontamination coating is applied in the regions which are not nickel-plated, whereby the transport and storage cask of CASTOR type is ready for its intended purpose. The values for assembly and tests are recorded.

DISCUSSION

Question by Mr.Grek, FR Germany: How do you guarantee the possibility of reinspection for the wall thickness with regard to the deep hole borings?

Answer: As already stated in the paper, the possibility of reinspection with regard to surface crack testing is 100 % guaranteed. An ultra-sonic test is made more difficult by deep hole boring. Appropriate records will be made before delivery and these will be checked.

Question by K. Goldmann, USA: You mentioned that you use NDE meth for detecting internal and external flaws. What procedures do you use to remove and repair unacceptable defects? What has been your experience to date in that regard?

Answer: In order to establish that there are any external surface defects, we use a surface crack test, e.g. dye penetration test. If defects are found, there will be a yes or no decision, i.e. it will be determined whether the defect is inside or outside the permissible tolerence limits. The casting will be discarded if the defect is outside these limits.

Question by E. Schlich, FR Germany: 1. Do you use gaskets on the GGG 40.3 surface or on the nickel plating? 2. Is it true that nickel plating is applied from inside to the surface without interruption?

Answer: 1. Sealing is exclusively on the nickel layer. 2. The nickel coating covers those surfaces which have to be galvanically plated without interruption. This is for both internal and external regions.

Question by J. D. Hart, United Kingdom: I understood that the fins are not machined. Do you experience any difficulties with the trapping of radioactive contamination even though the surfaces are painted? If it is necessary to machine the outside of the body, I would have thought that the cost incentive would be eroded. Can you tell me what is the present cost advantage of cast flasks over forged flasks?

Answer: The fin region is not machined. It is coated with a classical, well-proven multi-layer decontamination dye so that the surface is smooth and free of pores. Experience with such decontamination coating shows that in the case of contamination, cleaning can be carried out as well as, or even better than, with stainless steel. Moreover, the fin region is covered with an additional de contamination shirt. Machining of the lid region would be possi but is not necessary under these conditions.

Costs for CASTOR casks in contrast to forged steel casks with comparable capacity, but smaller shielding capability, amount to about 1/4 to 1/3 of normal construction.

Question by G. Arendasy, Austria: Is welding of surface failures foreseen?

Answer: No, it is not foreseen.

DEVELOPMENT OF A NON-VOLATILE BORON CARBIDE-COPPER CERMET NEUTRON SHIELD FOR HIGH PERFORMANCE SHIPPING CASKS*

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INTRODUCTION

Many conventional spent nuclear fuel shipping casks utilize liquids for neutron shields. In most cases the liquid is water or a mixture of ethylene-glycol and water. These liquids are inexpensive, readily available and are easily contained in cavities. However these shields are usually located near or at the outer surface of the shipping cask, making the liquid neutron shield vulnerable to damage should a severe accident occur. U.S. Nuclear Regulatory Commission regulations force a cask designer to assume that all of the liquid neutron shield will be lost in such an accident. Consequently sufficient material must survive severe accidents to provide suitable post-accident shielding. This, of necessity, increases the thickness and weight of the cask wall and therefore reduces the potential spent nuclear fuel payload.

Sandia National Laboratories has recently conducted a conceptual design study[1] for a new generation of shipping casks to transport the more radioactive spent fuel from Liquid Metal Fast Breeder Reactors. In that study a non-volatile neutron shield of boron carbide and copper was proposed. This cermet (ceramic-metal) shield combines a good neutron absorbing material, B_4C with a good thermal conductor material, Cu. It should be noted that neither of these materials is liquid and no additional liquid is required for neutron shielding. However, the conceptual design study[1] used a target composition for this cermet of 75 vol pct B_4C and 25 vol pct Cu, without having demonstrated the feasibility of fabricating this composition.

Powder metallurgy processing was chosen as the fabrication technology for this cermet because of the requirements of: 1) a uniform dispersion of B_4C in a continuous matrix of copper, and 2) full theoretical density (no pores). Cermets [redacted]ing these requirements should provide optimum neutron opacity and thermal [redacted]uctivity. The powder metallurgy approach is ideally suited for uniformly combining such different materials. Hot isostatic pressing of the powders provides suitably high pressures to achieve full density.

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**U.S. Department of Energy facilities.

This report documents our efforts to demonstrate the feasibility of achieving a uniform dispersion of boron carbide in a continuous copper matrix with full theoretical density.

EXPERIMENTAL PROCEDURE

The investigation consisted of two phases: I. Evaluation of the effect that B_4C content had on the density, strength, thermal conductivity, and continuity of the copper matrix, and II. Evaluation of the influence that consolidation parameters had on the density and thermal conductivity. In both parts, technical grade boron carbide powder was used along with commercially pure copper powder. As seen in Figure 1, the boron carbide powder was irregular while the copper powder was spherical. The boron carbide particles were screened and only those particles less than 149 microns were used. Similarly the copper was screened but only those particles less than 44 microns were used. The resultant size distributions are plotted in Figure 2 and show the extent of particle size overlap. The powders

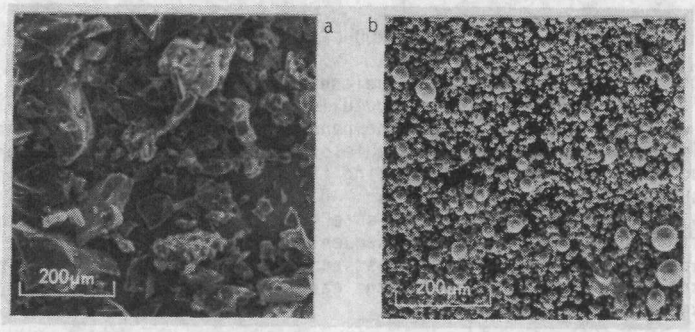

Figure 1. As-Received Powders: (a) Technical Grade B_4C, < 149 microns; (b) Commercial Purity Copper, < 44 microns

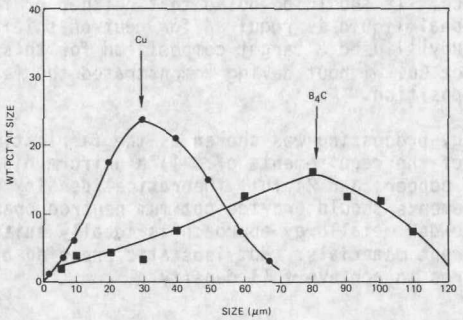

Figure 2. Particle Size Distribution for the As-Received Powders

were blended and loaded into stainless steel cans, which were then evacuated and sealed. Consolidation was done by hot isostatic pressing (HIP). In Phase I, the composition varied from zero to 80 vol pct B_4C with the compaction parameters fixed at 800°C, 100 MPa and 3 hr at temperature and pressure. In Phase II, consolidation temperatures of 700, 800 and 1000°C, consolidation pressures of 35, 103 and 138 MPa, and the times at temperature and pressure of 1, 3 and 6 hr were examined for both the 70 and 80 vol pct B_4C compositions.

RESULTS

Phase I

Optical microscopy revealed that a continuous copper matrix existed for samples with B_4C contents of 50 vol pct or less (see Figure 3).

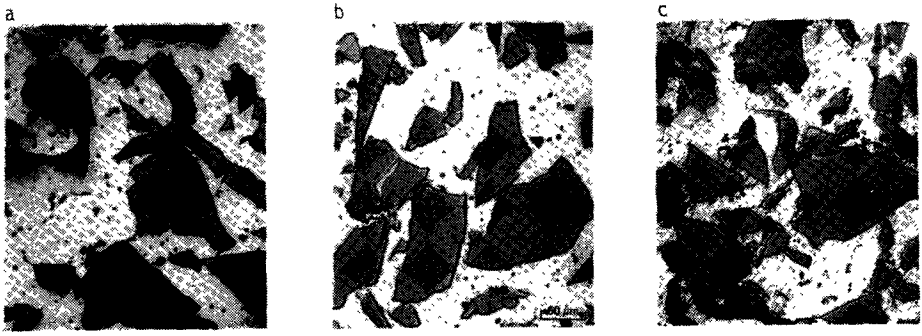

Figure 3. Optical Microstructure: (a) 40 vol pct B_4C, (b) 50 vol pct B_4C, and (c) 60 vol pct B_4C

The density of the compacted B_4C-Cu cermet was calculated as a percentage of the theoretical density and is plotted as a function of composition in Figure 4. Full theoretical density was achieved for B_4C contents of 60 vol pct or less. At the higher B_4C contents, variations in particle size distribution was seen to have some effect on the density, but it was difficult to single out which size distribution was optimum. The size variations are listed in Table I. Simple screening operations as well as more sophisticated operations like gap-grading[2] were used. The coarse particle size range was -149+105 microns, the medium particle size range was -74+53 microns and the small particle size range was -44 microns.

Thermal conductivity, measured by a comparator technique, ranged from atts/m-K for 80 vol pct B_4C to ~ 360 watts/m-K for zero vol pct B_4C (all per) and is plotted in Figure 5.

The compressive yield strength shows a maximum as a function of composition for samples in the as-received condition. The maximum of 193 MPa occurs at 60 vol pct B_4C as shown in Figure 6. Note that the compressive strength measured after a 2-hour anneal at 800°C does not exhibit a maximum. On the other hand, the transverse rupture strength (measured per ASTM-378-65) exhibits a monotonic decrease with increasing B_4C content as shown in Figure 7.

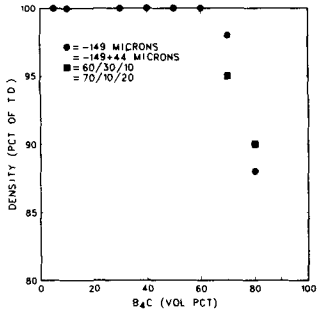

Figure 4. Density As a Function of Composition Showing Full Density Achieved for $\leq$ 60 vol pct B_4C. Note effect of particle size at higher B_4C Contents.

Table I
Density of B_4C-Cu Cermets for Various Particle Size Distributions

B_4C Content (vol pct)	ρ (pct T.D.)	B_4C Size (microns)	Cu Size (microns)
70	97.5	-149	-44
70	97.0	-149+44	-44
70	95.6	-149	-44
70	95.4	- 88	-44
70	94.9	60/30/10*	-44
70	91.0	- 44	-44
70	90.6	- 44	-44
70	89.9	-149	-44
80	93.6	70/10/20*	-44
80	92.0	-149 + 44	-44
80	90.3	50/20/30*	-44
80	89.5	60/30/10*	-44
80	88.2	-149	-44
80	87.7	-149	-44
80	84.7	- 44	-44
80	84.5	-149	-44
80	83.6	0/70/30*	-44
80	82.6	- 44	-44
80	81.0	- 44	-149 + 44
80	79.4	- 44	-149 + 44
80	74.3	- 88	-44

*Percentages of coarse, medium and fine particle sizes; see text for details.

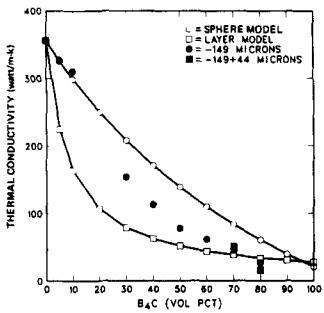

Figure 5. Thermal Conductivity As a Function of Composition. Comparison of data with calculations for two models.

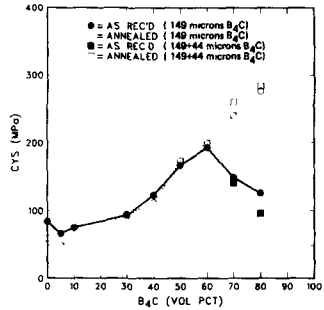

Figure 6. Compressive Yield Strength As a Function of Composition for Both As-Received and Annealed Samples

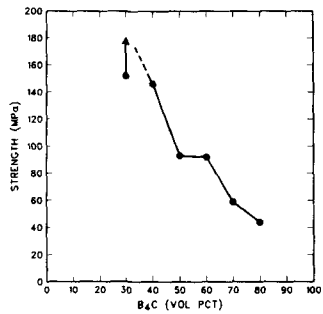

Figure 7. Transverse Rupture Strength As a Function of Composition. Strength for the 30 vol pt B_4C was greater than 152 MPa.

Phase II

The density and thermal conductivity after consolidation is shown in Table II. Densities ranged from 81 to 95 pct of theoretical density for the 70 vol pct B_4C but only 70 to 89 pct of theoretical density for the 80 vol pct B_4C. Thermal conductivity ranged from 2.4 to 8.4 watts/m-K for the 70 vol pct B_4C samples and from 1.1 to 4.6 watt/m-K for the 80 vol pct B_4C. The highest values of density and thermal conductivity are indicated with an asterisk.

DISCUSSION

These results show that fully dense B_4C-Cu cermets can be fabricated with up to 60 vol pct B_4C. For 70 vol pct B_4C the best density was 98% of T.D. using the as-received particle size distributions while for the 80 vol pct B_4C the best density was only 94% of T.D. In contrast to the 70 vol pct B_4C, the 80 vol pct B_4C had the highest density when gap grading for optimizing packing

Table II
Density and Thermal Conductivity of B_4C-Cu Cermets for Various Consolidation Conditions

B_4C Content (vol pct)	HIP T T(°C)	HIP P (MPa)	HIP t (hr)	ρ (pct T.D.)	K (watt/m-K)
70	700	35	1	82	3.6
70	700	35	6	81	3.7
70	700	138	1	92	8.4*
70	700	138	6	90	7.2*
70	1000	35	1	81	3.3
70	1000	35	6	82	2.4
70	1000	138	1	95*	6.2
70	1000	138	6	94*	6.9
70	800	103	3	90	2.8
70	800	103	3	98	14.0
80	700	35	1	70	1.1
80	700	35	6	73	1.5
80	700	138	1	89*	2.7
80	700	138	6	85	1.9
80	1000	35	1	74	1.7
80	1000	35	6	75	1.5
80	1000	138	1	88*	3.2*
80	1000	138	6	88*	4.6*
80	800	103	3	85	2.8
80	800	103	3	88	4.0
80	800	103	3	88	-

density was used (an increase from 88 pct T.D. to 94 pct T.D. was observed). However, the simple screening of the fine B_4C (those particles less than 44 microns) also had a significant impact on the density. This simple operation is considered to be the least time consuming and thus should be the least costly method of increasing the density of 80 vol pct B_4C cermets.

The measured thermal conductivity values were relatively low for the 50 to 80 vol pct B_4C sample, being more like the thermal conductivity of B_4C than that of Cu (see Figure 5). Two different models of the influence of volume fraction B_4C on the thermal conductivity of the B_4C-Cu cermets have been examined in an attempt to explain these results. In Case I, a homogeneous distribution of B_4C spheres in a continuous matrix of copper is assumed. The thermal conductivity, k, is then given by[3]

$$k = k_c \left[\frac{1 + 2V_B \left(\frac{1-k'}{1+2k'} \right)}{1 - V_B \left(\frac{1-k'}{1+k'} \right)} \right]$$

where k_c = thermal conductivity of copper

$$k' = \frac{k_c}{k_B}$$

k_B = thermal conductivity of B_4C

V_B = volume fraction of B_4C.

A plot of the thermal conductivity predicted by this model is given by the open circles in Figure 5. It can be seen that this model only applies for low conte of B_4C, suggesting that a continuous conduction path has not been achieved even for the fully dense compositions at 30, 40 and 50 vol pct B_4C. In Case II, alternating parallel layers of B_4C and Cu are assumed. This case represents the opposite extreme of Case I, in that it assumes no connection whatsoever of the more conducting species (copper). The thermal conductivity in this latter instance is then given by[3]

$$k = \frac{1}{\left(\frac{1-V_B}{k_c}\right) + \left(\frac{V_B}{k_B}\right)}$$

which is plotted with the open squares in Figure 5. This plot suggests that for 60 to 80 vol pct B_4C, the material behaves more like unconnected layers of Cu than like a continuous copper matrix. At the intermediate B_4C contents, the measured values lie between the two calculated values suggesting transitional behavior. Clearly there is room for improving the thermal conductivity at all B_4C contents greater than or equal to 30 vol pct.

In addition to the lack of copper matrix continuity, the presence of porosity can also reduce the thermal conductivity. This is most evident for the 70 and 80 vol pct B_4 samples which have considerable porosity. However, increasing the density of the 80 vol pct B_4C samples did not result in an increase in thermal conductivity although an increase was observed for increased density of 70 vol pct B_4C. Furthermore, the porosity cannot be used to explain the reduced thermal conductivity for samples in the 30 to 60 vol pct B_4C range since all samples were found to have achieved full theoretical density. Additional effort needs to be made to determine the extent of continuity and/or uniformity of the copper matrix. The important points are that the measured thermal conductivity values might be considered as typical and that the calculations based on the two models provide anticipated scatter in the data. For design purposes the measured values are sufficient while the development work to improve the values can be carried out. One additional point is that the predicted thermal conductivity drops off rather quickly with increased B_4C content even for the ideal case of a continuous matrix, and at 40 vol pct B_4C is less than half that of copper. In the 50 to 80 vol pct B_4C range there appears to be little extra benefit derived from the use of copper as a conducting matrix.

Strength determinations were made by two techniques as a means of assessing the integrity of the compacted cermets. The transverse rupture test provides an indication of mechanical behavior in a three-point bending mode. As seen in Figure 7 the strength increases with decreasing B_4C content with a significant strength enhancement occurring for B_4C contents less than 50 vol pct. Compressive yield strength shows a peak at 60 vol pct B_4C, consistent with the density measurements. As expected, the fully dense compacts are stronger than those with porosity. Comparison of the compressive yield strength and the transverse rupture strength suggests optimum behavior in the 40 to 50 vol pct B_4C range, with the 40 vol pct B_4C having a 123 MPa compressive yield strength and a 146 MPa transverse rupture strength. It is interesting to note that after annealing, the 70 and 80 vol pct B_4C samples had a higher compressive yield strength. No transverse rupture data is available on annealed samples, but the compressive yield strength behavior suggests that increasing the consolidation time might provide higher strengths for the higher B_4C compositions.

In the parametric study (phase II) of this investigation, the temperature, pressure and time were varied and samples were prepared for both the 70 and 80 vol pct B_4C compositions. Only these compositions were studied because full density was not achieved for consolidation at the nominal (phase I) 800°C, 103 MPa, 3-hour conditions. The parametric variables were chosen using the strategy of experimentation based on statistical analysis techniques.[4] Examination of Table II shows that although full theoretical density was not achieved for either composition, an indication of the recommended consolidation conditions was obtained. Generally,

the thermal conductivity is directly related to the percent of theoretical density, although there is a good deal of scatter in the data. For each composition, samples compacted at the highest pressure had the highest thermal conductivity. For the 70 vol pct B_4C sample 700°C provided the best results whereas for the 80 vol pct B_4C, 1000°C provided the best results. However, contrary to the predictions of the strength assessment of Phase I of this study, increased time at temperature did not provide consistently significant increases in the thermal conductivity. In fact the parametric study suggests that time is of minor importance and only one hour is sufficient to achieve the density and thermal conductivity representative of each temperature-pressure combination.

CONCLUSIONS

Significant progress has been made in fabricating boron carbide-copper neutron shields by powder metallurgy techniques:

1. Full theoretical density is readily achieved for B_4C contents less than 60 vol pct.

2. The best combination of density, thermal conductivity and mechanical properties have been observed for the 40 to 50 vol pct B_4C-Cu composition.

3. Based on theoretical calculations, there is promise for enhancing the thermal conductivity for the 30 to 60 vol pct B_4C compositions by adjusting the consolidation parameters.

4. Full theoretical density for 70 and 80 vol pct B_4C compositions was not achieved by varying the consolidation parameters and particle size, but pressure is the most important consolidation parameter.

5. Future efforts for optimizing the thermal conductivity and density should concentrate on improving the continuity of the copper matrix perhaps by coating the B_4C with copper prior to consolidation.

REFERENCES

1. George C. Allen, Jr., R. G. Eakes, J. M. Freedman, R. B. Pope, S. A. Dupree and W. P. Schimmel, Jr., "Conceptual Designs for LMFBR Spent Fuel Shipping Casks."SAND77-1483, Sandia National Laboratories, Livermore, March 1978.

2. R. K. McGeary, Journal of the American Ceramic Society 44, p. 513 (1961).

3. W. D. Kingery, Introduction to Ceramics, (J. Wiley & Sons, 1960).

4. "Strategy of Experimentation," DuPont Technical Publication, E. I. DuPont de Nemours and Co., Wilmington, Delaware, October 1975.

DISCUSSION

Question by P. E. Eggers, USA: How does the neutron shielding effectiveness of your B_4C/copper cermet compare with that of other neutron shielding alternatives such as borated silicone or water as a reference point?

Answer: On a per unit thickness basis, the B_4C/Cu cermet is not as effective as an undamaged water neutron shield and would be similar in shielding efficiency to an undamaged borosilicone rubber material of the type described by R. B. Pope in an earlier paper. However, the presence, and thus the effectiveness, of a B_4C/Cu shield can be assured following a fire accident. In comparison, a water shield would be expected to be totally voided and a borosilicone rubber shield would be partially degraded.

SESSION X
QUALITY ASSURANCE AND MAINTENANCE

SETTING - UP OF QUALITY ASSURANCE IN FRANCE

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FONTENAY-aux-ROSES (FRANCE)
Monsieur Pierre WARNIEZ - - idem -

Should I confess, as a preamble, that writing this paper caused me some troubles ? I think I should, because these troubles seem to be proportional to the topic and, had not the organizers kindly insist, I would perhaps have given up.

It is not uninteresting to go a little into the motivation of these troubles. In fact, the topic does exist, it is well defined and important. Why then should I be embarrassed for teckling it ? This is because, just as it can be - relatively ! - easy to talk about a well-known, well "run in" topic, so it is difficult to start considerations relating to a field in which a big development has still to be done.

That is the reason why I preferred to peak about the "setting up" of quality assurance in France, thus meaning that we did in no way pretend to present the situation in our country as a model but on the contrary to describe how we proceeded up to now, the difficulties encountered and foreseen, the results obtained and those still expected.

It seemed to me useful first to go up to the sources, in order to analyze the gradual awareness of this problem.

For us, which are specialists fo safe transport, these sources are the successive editions of the IAEA Regulations. If one reads all the various handbooks of Safety Series n° 6, one can notice at once that our early precursors of the sixties were already dealing with quality assurance - or at least began to consider its bases - just as Monsieur Jourdain (said Molière) was making prose : without being aware of it ! The "recommandations for initial tests and inspections and, later on, for periodical tests and inspections in service" given in the first edition of the Regulations, this of 1961, were already part of the essential criteria of the not yet born quality assurance ! (1)

As early as 1964, the Regulations refer to "conformance to standards of model of the methods and materials used for the packaging manufacture" and to "inspections by the competent authority". (2)

The 1967 revision takes over exactly the same provisions (3), but, by 1973, appears for the first time the concept of quality control, clearly expressed if not explicit ; it insists especially on the maintenance of quality control during the life of the packaging (4). Even if these provisions do not seem to add something new to the preceeding ones, they are very significant by the only fact that they are set under the sign of quality. They are made still more important by the way in which they are explicated in the Advisory Material for the application of the Regulations (5). In fact, I consider this text as the basis of all official doctrine in this field.

To complete this somewhat historical review, I shall point out that the current revised edition of the Regulations is the first one to speak namely of quality assurance (6). The earlier provisions have been reinforced and, in particular, the revised edition of the Advisory Material, which is presently being printed, develops and specifies the preceeding texts in view of the results obtained by the Technical Review Committee which held a meeting in Rome in October 1978, and especially by the relevant working group. It should also be mentionned that another specialized working group met in Vienna in September 1980 in the Framework of the Advisory Group in charge of working out the first proposals relating to the revision of the 1973 Regulations.

All this allows to find out that the problem of what has become quality assurance has always been kept in the mind of the Regulations "designers" and to estimate the distance covered from the origin.

What has been done in France at regulatory and practical levels ?

As soon as 1956, there was within the Regulations for the Transport of Dangerous Materials an embryo of regulation for the transport of radioactive materials. It covered exactly four and a half pages and it is worth noting that, though the volume was small, it was yet stipulated that "the consignor should attest on the consignment declaration that the packaging is conforming to this specified for class IV b materials" (in France, we still speak about class IV b instead of class 7).

I am not going to impose on you a comparative history of the successive texts issued on this subject, and I will only tell you that, since 1966, our Regulations indicate that "the Minister can order inspections of the packaging even during its manufacture". Even if the Minister has not had this possibility for a long time, this sentence - which is not from me - seems to me rather premonitory : as a matter of fact, 14 years ago, somebody in France (some of you have known him) had been aware of the double aspect of quality assurance, exerted successively during the packaging manufacture, to obtain the required quality, and during its use, to maintain it. Besides, I regret that we don't speak more accurately of quality compliance (to specifications) in the first case and of quality maintenance (from the original state) in the second case.

We must admit that it is only around 1977 that we considered the practical application of this sentence of the Regulations - But how to apply it ?

First, it should be specified that the Minister in question is the Minister of Transports and that he delegates his powers, concerning the transport of dangerous materials - including radioactive materials, which is not the case in all countries - to a higher civil servant who then becomes the Competent Authority. Concerning the transport of radioactive materials, the French Competent Authority itself relies upon a small group of advisers, members of the Commissariat à l'Energie Atomique. I am myself one of these advisers.

The transport of radioactive materials in France is characterized by the importance of those concerning the fuel cycle ; in this respect, the transport of spent fuel and plutonium oxide should, of course, have a privileged place.

The significance of radiopharmaceuticals transport should also be taken into account. The Office des Radioéléments Ionisants (Office for Ionizing

Radioelements) (which is part of the Commissariat à l'Energie Atomique) proceeds to hundreds of consignments per day and, though it is the main consignor, it is not the only one. However, we can say that it is, by far, the main user of type A packagings, even though consignments of this type do not represent the majority if its consignments.

At last, a very special type of transport should not be neglected. I mean the radiogammagraphy instruments, the transport flow of which is not well known but certainly very large. Fortunately, these devices are working instruments before being transport packagings and their specifications, concerning quality assurance, are considered in view of their main use - which gives guarantees that we estimate now sufficient.

It is the same for type A packagings upon which, within the structures peculiar to the Commissariat à l'Energie Atomique, and without refering to the Competent Authority, we can have an action, relatively unformal indeed but such as it also gives us sufficient guarantees.

Of course, quality assurance could possibly affect any type of packaging. However, it did not seem justified to us to deal in priority, in this field, with low specific activity materials and low-level solid radioactive materials, in view of the big problems arising in other respects.

Thus, we came to concentrating our action on the packagings, constituting parts of the package models which require an approval from the compétent authority (excluding radiogammagraphy instruements, at least for the moment).

According to the prescriptions of the 1973 Regulations, the approval certificate should contain a reference to the instructions for use and maintenance of the packaging. We have interpreted this prescription in the facts by introducing in our approval certificate indications, summary indeed, but explicit on both of these points. This forms one of the basis of quality assurance during the packaging useful life- what I have called quality maintenance.

The Regulations also provide (if necessary, they specify) for an expiration date of the approval certificate validity. When we set up, at the beginning of 1976, the new models of approval certificates - that is certificates conforming to the 1973 Regulations - we established their validity period at three years. When this period ended for the first certificates delivered, we were just beginning to set up the first measures concerning quality assurance. We thought it was a good opportunity to lay a foundation stone, in a way, and we put as a condition to any validation prorogation that the packaging owner should certify its good state. This resulted in numerous controls which allowed- at least - to make the people involved sensitive to the problem.

[redacted] this occasion, we became aware of another problem, obviously very important. [redacted]e initial approval certificate relates to a package model and the aim at this stage is essentially, as I said before, a matter of conformance to specifications. As soon as the various specimens of the approved model are in service, the scope of quality assurance becomes a question of maintaining this quality with regard to the level determined and obtained at the origin ; therefore, it is no longer a matter of model but an individual matter that must be taken into account in the Regulations - In the meantime, we have been led, as a

conservative measure, to prorogate approval certificates only for some specimens of packaging, duly indexed by their individual numbers.

That is what we have done as a conservative measure, I say it again, and strictly applying the Regulations prescriptions - but not all the advisory material of the current handbook n° 37.

Of course, we did not stop at that point and, at the beginning of 1978, we started to study a project of codification of quality assurance concerning the safety of packagings for the transport of radioactive materials.

This project has been worked out in the closest possible cooperation with packaging designers and manufacturers, when dealing with quality conformance and with the users, when dealing with quality amintenance. For we do not believe that it should be possible to come to a satisfactory result in such a difficult field by taking unilateral provisions, even - and especially - if they are concretized by official texts. Therefore, this project is for the moment the subject of a gradual setting up and, to tell the truth, of a "run in" which might still last a few months. Only when it is entirely perfected, it will be taken up by an official text. In this respect, French administrative rules are flexible enough to allow a progression with relatively noiseless tread. To have the force of a law, the text should be published under the form of a decree in the French Official Gazette. But we will take care not to take this way, which make uneasy any further alteration. On the contrary, we will advise the Competent Authority to issue the text in the form of a memorandum, the publishing of which in the Gazette is not compulsory and which can be, consequantly, much more easily modified. Only when this text has proved itself will it be issued in the form of a decree.

By that time, we will be able to say that the setting up of quality assurance in France is completed. Then, of course, its strict application will have to be checked.

It is obvious that our project does not pretend to be particulary original. In fact, we have widely taken into account the texts already existing in this field, abroad as well as at IAEA (in particular the Advisory Material relating to safe transport (5), (6), and texts such as the Codes of Practive relating to nuclear power plants).

Therefore, I think it would not be very interesting to impose on you, if not the integral text, a least a summary. Moreover, that would be quite impossible in view of the time granted to me.

Since we are dealing with a "setting up", I should not make a conclusion. However, I think I can say that all this shows - if ever it was necessary - that the problem of quality assurance is a difficult one. As it is an imperious necessity, it seems justified to proceed step by step, and we think the way chosen in France is the good one, even the best one if one considers the national conditions.

RÉFÉRENCES

(1) I.A.E.A. Safety Series n° 6 - Regulations for the Safe Transport of Radioactive Materials - 1961, paragraph 16.2.1.1., last section.

(2) I.A.E.A. Safety Series n° 6 - Regulations for the Safe Transport of Radioactive Materials - 1964 Revised Edition, published in 1965 - paragraph C 6.2.2. d) and Appendix IV I.1.1.

(3) I.A.E.A. Safety Series n° 6 - R-gulations for the Safe Transport of Radioactive Materials - 1967 Revised Edition - paragraph C 6.2.2. d) and Appendix IV I.1.1.

(4) I.A.E.A. Safety Series n° 6 - Regulations for the Safe Transport of Radioactive Materials - 1973 Revised Edition - paragraph 839.

(5) I.A.E.A. Safety Series n° 37 - Advisory Material for the Application of the I.A.E.A. Transport Regulations - 1974 - paragraphs 820 - 822.

(6) I.A.E.A. Safety Series n° 6 - Regulations for the Safe Transport of Radioactive Materials - 1973 - Revised Edition (As Amended) - paragraph 839.

DISCUSSION

Question by H. W. Curtis, United Kingdom: As users of packages we have always applied correct procedures of inspection in their manufacture, operation and maintenance. What will be the contribution to real safety of this magic called quality assurance? Can you assure us that old containers made before the development of quality assurance will not be penalised by the French competent authority?

Answer: I do not think the contribution of quality assurance to safety can be described as "magical", as I have said that it was taken into account even in the very first editions of the regulati even if the actual word was not mentioned. I cannot speak for the competent French authority but it seems obvious to me that future resolutions pertaining to the regulations on the subject will be applied with understanding - even if the "grandfather clause" is not formally explained.

Question by K. E. Wieser, FR Germany: Which role is foreseen in the framework of the French quality assurance system for the different parties (manufacturer, applicant, official inspectors, competent authority)?

Answer: The manufacturer will be responsible, vis-à-vis the competent authority, for the quality of the package prototype and new packaging models. The applicant will be responsible for maintaining this quality. The way in which the competent authority will exercise its powers has not yet been defined. However, it will probably be similar to the way all transportation of dangerous goods is checks, i.e. by a body of specialised inspectors (the competence of whom could perhaps be extended to quality maintenance inspection, if not to obtaining this quality).

THE QUALITY ASSURANCE PROGRAM IN SMALL SERIES CONSTRUCTION OF UF_6 PACKAGES AND THE ROLE OF COMPETENT AUTHORITY

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Introduction

With the development of nuclear power programs in various countries, Competent Authorities devoted a greater attention to the aspects of nuclear safety and radiation protection even if they are less known to the public opinion, with the aim to reach more adequate standards of safety for workers and general public.

Along this line the Italian National Committee for Nuclear Energy (CNEN), as Competent Authority, requires that all the activities in the nuclear field undergo quality assurance programs as indicated in an ad hoc document "Guida tecnica n° 8". For this reason a quality assurance program for packages carrying radioactive and fissile materials is required to assure an adequate design, construction, use and maintenance during the package life.

The first application of a Quality Assurance program in the field of transport packagings was carried out by CNEN on the construction of UF_6 packages, 30 B type. This construction follows the requirements of ANSI 14/1/71 (2) standards.

These packagings are built in two parts an external overpack and an internal cylinder used to store UF_6 also for long period of time.

The Quality Assurance program was applied to the construction in small series of these internal cylinders.

The program was developed in accordance with the fabricant, who set up his internal organization following the provisions of the Guida Tecnica n° 8.

Quality Assurance Programme of the Licence

The licensee presented to the C.A. a quality assurance programme in form of a proposal and liable therefore to be modified in accordance with the C.A. requirements.

The main points of this programme are indicated in the following:

- the licensee, who is responsible of the Quality Assurance programme, provides an organizational chart denoting in a clear way the areas and the responsibility of the major functional organizations performing quality related activities in the design, procurement, fabrication, testing, inspection and auditing.

The positions within the organization which have written delegated authority to stop work or to control further processing or use of nonconforming items must be also identified;

- during the design stage measures are taken to ensure that the procedures and drawings established, including changes thereto, are followed. These procedures ensure the compliance of the packages to the applicable requirements; the ANSI standards 14/1/1971 are taken as reference. We wish to underline that these cylinders have to follow either the requirements of the international transport Regulations as well as the national requirements existing on the italian territory for pressure vessels with regard to the filling or emptying procedures.
  The national regulations for pressure vessels require the approval by the Associazione Nazionale per il Controllo della Combustione (ANCC), which acts as Competent Authority in this field.
- cryteria to be adopted to in the choice of contractors must be indicated, having in mind that cylinders will be fabricated by firms having their own internal quality control;
- measures to be established to control materials, parts or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include procedures for identification, documentation, segregation, deposition and notification to affected organizations. Items which do not conform shall be reviewed

and accepted, rejected, repaired or reworked in accordance with the Competent Authority.

- For this purpose the licensee hasestablished during the construction phase a plan of intervention agreed with the fabricant where the check points are clearly identified as well the responsability of the actions to be taken. This plan is illustrated in the table. The two different competent Authorities CNEN and ANCC are indicated. ANCC intervenes with gonne hold and inspection points relevant to qualify the cylinders as pressure wessels. This allows CNEN not to intervene in such stages and to carry out random checks along the whole construction cycle (the fabricant has to inform when the different stages of construction take place). These random checks have been carried out during the construction of the first series of the cylinders in the phases indicated in the table. These intervention allow CNEN to checks the conformance with the requirements.
- Documents which must be submitted with each cylinder, in order to have a complete knowledge of the mechanical characteristics have been also established.

The Role of Competent Authority

We wish to stress some points of this exercise of the quality assurance in the field of transport packages:

1) first of all the Competent Authority required and obtained the right to intervene in the preparatory stage of the quality assurance programme which the licesee was preparing for its own activity. The Competent Authority was successful in this request as the licensee was in that time revising its activity in the field of transport packages and was organizing ex novo its internal organization. We wish to stress that the intervention of the Competent Authority in this phase contributed to a great extent to the success of the operation;
2) secondly the Competent Authority during the package design approval stage took the advantage in taking part in the Quality Assurance programme.

The conformity of the design to the standard requirement of the IAEA Transport Regulations was, in any case, assured according to the provisions of these Regulations; the Quality Assurance programme added some fundamental points including all those planned and systematic actions necessary to provide adequate confidence that the package will perform satisfactorily in service. The quality assurance programme is, therefore, strongly interconnected with the approval and it is also mentioned in the package design approval certificate;

3) finally we wish to stress that the Quality Assurance programme can be utilized also during the life time of the package either in terms of periodical checks of the package itself and of its components (such as valves etc.).

Conclusion

The conclusions which can be withdrawn from this experience can be summarized as follows:

a) it is possible to introduce quality assurance programs which can be of help to the Competent Authority in the development of its activity without imposing a heavy burden to the Competent Authority;

b) these quality assurance programs can be prepared in a simpler form than those adopted in other types of activity;

c) it seems to us useful to introduce in the revision of the Regulations the outlines and the most important steps of a quality assurance program for packages transporting radioactive materials.

This study would contribute to the improvement, in the opinion of the general public, of the efforts carried out to garantee the respect of the safety standards and in the same time will assure a more wide acceptance at international level of the unilateral type B(U) concept.

OPERATION DESCRIPTIVE	REQUIREMENTS ASME	Q.C.D.	LICENCEE	COMPETENT AUTHORITY ANCC	CNEN
Identification of materials	C	C	C	C	
Plate cutting/ /forming		D			
Welding / /fit up	V-RR	V-RR	V	V-R	(*)
Half coupling welding		LL	V	L	
Head/course assembling/ /welding	RR-C	RR-C-V	V	R	(*)
Find dimensional check	D				
Hydraulic test	CC	CC		CC	
Internal cleaning		V	V		(*)
Leak test-Volume det.		C			(*)
Painting		D-C	D-V		
Weighting		C	C		
Final check		VV	CC		

SYMBOLS	HOLD POINT	INSPECTION POINT
Dimensional check	DD	D
Visual check	VV	V
Certification	CC	C
Rx examination	RR	R
Liquid penetrant	LL	L

(*) random checks

QUALITY ASSURANCE MEASURES FOR RAM-PACKAGINGS IN THE FEDERAL REPUBLIC OF GERMANY

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First it should be pointed out that, due to the numerous different types of packagings for the transport of radioactive materials, the structure of Quality Assurance Measures (QAM) will be dealt with rather than details of the packagings themselves. This paper refers mainly to type B packagings.

Quality assurance comprises all measures - from design, manufacture, to the operation of a system - which are necessary for this system to satisfy pre-determined requirements arising from both the use to which the packaging is put and the protective aims of the packaging, and also for this to be guaranteed throughout the whole lifetime of the packaging.

For these reasons, the "IAEA Regulations for the Safe Transport of Radioactive Materials, Section VIII, Para. 839, require that a Quality Assurance Programme (QAP) be established for the design, manufacture and maintenance of packagings for the transport of radioactive materials and also foresee their compliance with the appropriate criteria of all relevant regulatory ordinances.

In the Federal Republic of Germany the special regulations for the shipment of class 7 material coming from the ordinances for road, rail and maritime traffic in connection with the "Guide to the Approval of a Packaging Prototype for the Transport of Radioactive Materials" [1] form the basis for the establishment of QAM for the manufacture and maintenance of packagings. Because - in contrast to other fields of technology - there is no technical regulatory basis for QAM in the field of RAM packagings, the basic principles of other regulations are used, such as those of pressure vessel manufacture (AD-Merkblatter) [2], the Safety Rules of the Nuclear Technology Board [3] and Appendix E, Code of Federal Regulations, Title 10, Chapter 7 [4].

Depending on the "life history" of a packaging, QAM have to cover the following three areas:

- design, calculations, structure of the construction, including choice of material and manufacturing procedures;
- serial production;
- operation, handling, maintenance and reinspection of the serial containers.

For packagings already in existence, the grandfather clause applies, i.e. a QAP consisting at least of handling and maintenance procedures and with reinspection routines will be accepted.

PHASE 1 - FROM DESIGN TO THE CERTIFICATE OF APPROVAL

The first part - the development of the packaging design - ends with the manufacture of a prototype (either a model or of original size), which has to prove in the following tests that it meets the required criteria with regard to mechanical and thermal resistance, leaktightness, radiological safety, operability and testability. During these tests and the safety-technical investigations by the "Bundesanstalt fur Materialprufung (BAM)", changes in the construction or the choice of other materials were often necessary to obtain an optimisation of design with regard to safety technology or - mostly at the request of the applicant - for economic reasons. Therefore, the approval procedure and the resulting cooperation between BAM and the applicant must be regarded as an important contribution to quality assurance with the aim of preventing design or system originated defects and sources of error.

It should be pointed out, that determination of prototype characteristics might serve to define serial values to be guaranteed, because the aim of all QAM is to prove that the safety-relevant data of a standard container are at least as good as those of the approved prototype or even better. Here it can clearly be seen that QAM for prototype approved systems also have to maintain defined standards.

The basis for the certificate of approval by the "Physikalisch-Technische Bundesanstalt (PTB)" is an official BAM test report, which includes the results of the BAM tests and calculations as well as the QAM for manufacture and maintenance of the packagings. All changes in, and deviations from, the design and QAM after approval have to be newly approved by BAM. If the changes are not safety relevant, there is only an informal approval, otherwise an expert's opinion is formally required as well as official release.

PHASE 2 - SERIAL PRODUCTION

The general structure of QAM for manufacture and operation can be seen in Fig. 1.

The applicant - this is usually the user or manufacturer of a packaging - has to submit the QAM for manufacture during the approval procedure mostly in the form of a Quality Assurance Programme (QAP). This QAP also has to go through an approval procedure and becomes part of the official test report. The QAP will - according to its importance - also be approved like the design.

During the approval procedure of the QAP and the QAM for operation and reinspection procedures, BAM partly enlists the help of other technical supervisory organisations such as TÜV and others which have enough appropriate experience in this field. The complexity of the QAP - and generally of the QAM - strongly depends on the type of packaging and its contents.

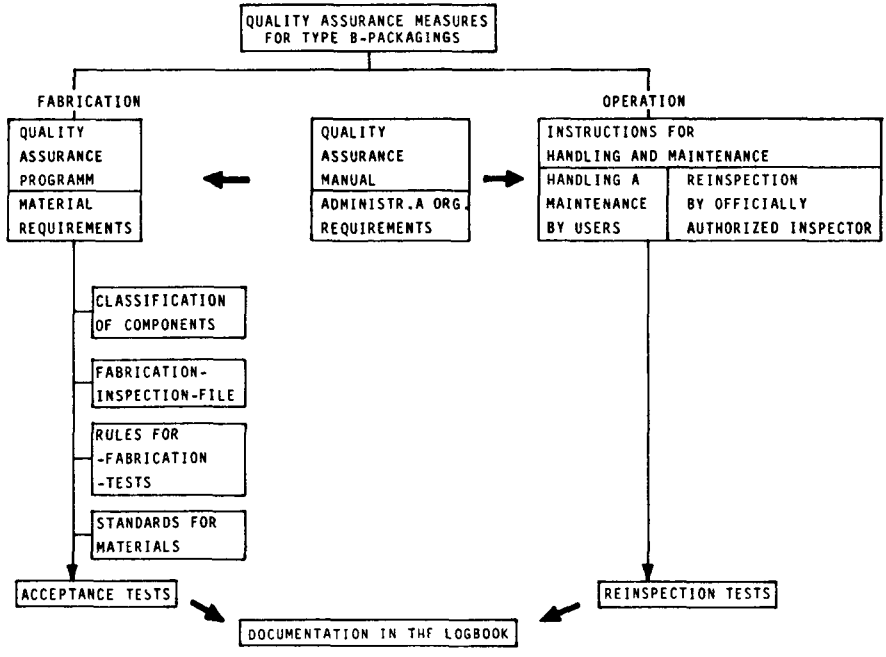

Fig. 1

Because BAM is extensively involved in the testing and investigation of transport and storage casks designed in the Federal Republic of Germany, the following remarks depend on the experience BAM has had during development of the QAM for these types of packagings.

The QAP for manufacture proved to be the most extensive and complex part of the QAM. It must contain all information, instructions and material requirements concerning the container components, their fabrication and assembly as well as specifications for the joint cooperation of all responsable, to supervise and approve the obtainment of all safety-relevant functions and properties.

To identify the object where quality has to be assured, the QAP starts with a description of the packaging and its function. This has to comprise a listing of all components with their materials, production form and method of manufacture.

Based on this, one can start to classify the components with regard to the safety relevance of the required tests and their extension. The result of classification is the division of the components into requirement steps such as:

- Step 1: of very significant safety-technical relevance;

- Step 2: of significant safety-technical relevance;

- Step 3: of minor safety-technical relevance.

Depending on the requirement step, proof of the manufacture tests has to be graded. For requirement step 1, the so-called three-fold test is required in the Federal Republic of Germany resulting in a test certificate according to DIN 50 049 [5], which must testify that the manufacturer, the applicant and the official expert have carried out the same test individually. At a requirement step lower down the scale, the test can be carried out so that the official expert only supervises and testifies correct testing at the manufacturer's. At requirement step 3, proof by the manufacturer and the applicant is sufficient.

The actual core of the QAP is the manufacturing and test sequence plan which covers the detailed performance of the manufacturing and testing steps together with the following data:

- manufacture specifications and regulations to be applied;

- testing specifications to be applied;

- testing bodies (manufacturer, applicant, official expert);

- type of proof certificate according to DIN 50 049.

For complex packagings it is advisable to establish manufacturing and test sequence plans for individual components and to lay down the QAM for the final assembly with appropriate manufacture and test specifications in assembly specifications.

The measures listed in the manufacture, machining and testing as well as specification of the test scopes have to be explicitly included in the QAP.

Besides specification of material and component properties and exact specification of the test scopes, the definition of allowable magnitudes of error, and/or tolerated deviation from the nominal condition such as:

- size tolerance,

- leak rates,

- number, magnitude and type of volume and surface errors,

- admissibility of repair or filler welding

are an essential part of the testing and material specifications.

Generally the definition of material data, manufacturing and testing specifications for certain products of normal metallic material such as sheet metal, bolts, forge and cast parts of certain dimensions, cause no difficulties because the generally accepted rules of technology such as the DIN standards, "AD-Merkblatter," or

VDTÜV material data sheets can be referred to. But in many other cases such specifications have to be established jointly by BAM and the applicant during the approval procedure. This situation arose for example in the appraisal of a new design of spent fuel element transport and storage casks made of nodular cast iron, where there were no directly applicable standards and test specifications for castings of such size.

After approval of prototypes (model and original size) under extreme test conditions (e.g. 9 m drop test at -40°C) accompanied by mechanical-technological material testing, the general suitability of the material for the intended purpose could be proved. So the material data sheets and sample plans for destructive testing could be established.

Another important item was the problem of non-destructive testing by ultrasonics (US). Here also a detailed US test programme could be established after fundamental investigations by the respective BAM department, a programme which must ensure greatest possible completeness of the tested cast and recognition and exclusion of failures which are not permissible.

Another important part of the QAP is a description of the quality assurance system. It must include the basis for planning, enforcement, organisation and administration of QAM for the manufacturer and applicant. With regard to the importance of smooth communication between all parties involved in the QA and because of the required strict separation of the responsibilites of manufacturer, applicant and subcontractor for testing, the establishment of a special catalogue for measures, e.g. a quality assurance manual, is recommended in which the customer also stipulates, besides the responsibilities of his own personnel, the qualifications and responsibilities of manufacturers, subcontractors and of the test personnel.

PHASE 3 - OPERATION, MAINTENANCE AND REINSPECTION

The afore-mentioned German "Zulassungsrichtlinie" also requires an operation and maintenance procedure as part of the application documents. These procedures on the one hand include all specifications for the consignor, which have to ensure safe containment of the radioactive contents, such as the sequence of handling operations, visual control of the sealing system, torques of the lid bolts, tests prior to shipment etc. as well as safe handling of the packaging. On the other hand, all measures are stipulated which are suitable for maintaining or reestablishing the properties of the packaging such as changing of the gaskets, renewal of the paint, functional testing of mechanisms, etc.

Referring to the legally regulated practice of other fields of dangerous goods containment, it is absolutely essential, in our opinion, that, to complete the above mentioned complexes, recurrent inspection after certain periods is carried out, supervised and testified by an official expert. These tests have to prove that each serial packaging meets the requirements for its approval during its whole lifetime. The tests comprise, for instance, checking identification documents, non-destructive testing, function, leakage and load tests.

DOCUMENTATION

Here, only documentation of manufacture and audits, together with reinspection will be considered. In this respect the "logbook" plays an important role. This logbook has to be established by the official expert for each individual serial packaging after all approval tests prior to being put into service. It covers all records which characterise the history of a packaging and two copies are issued, one copy remaining with the operator. On request it must be submitted to the proper authorities. In detail, the following documents have to be filed:

- official expert's certificate of manufacturing tests and audits according to the
- manufacturing and test sequence plans, also filed, in which all data concerning test records, certificates, etc., are included.

These documents prove conformity of the serial packaging to the approved design:

- reinspection plan; this serves as working instructions and refers to test standards to be met;
- official expert's certificates for reinspection work carried out;
- results of extraordinary testing after repair work, if necessary, (or equivalent).

By summarising the above mentioned sections the logbook has a handy format; on the other hand all references are included which allow access to individual test results. Especially for international traffic, all authorities involved can quickly assess the actual condition of each individual type B packaging.

As a closing remark one comment should be made. The most important precondition for the success of quality assurance is not a measure directly accessible by means of expertise and testing; it is the responsibility combined with expert authority of all people involved in manufacture and quality assurance.

REFERENCES

1. Richtlinien fur die Zulassung des Baumusters von Versandstucken zur Beforderung radioaktiver Stoffe, Verkehrsblatt Amtlicher Teil, Heft 20, Bonn 1977.
2. AD-Merkblatter (German Pressure Vessel Code), Ausgabe 1977, Vereinigung der Technischen Überwachungs-Vereine e.V., Essen 1977.
3. Sicherheitstechnischen Regeln des Kerntechnischen Ausschusses, Entwurf 6/78 der KTA 1401 "Allg. Anforderungen and die QS".
4. Code of Federal Regulations, Title 10 Energy, Chapter 71, Packaging of Radioactive Material for Transport and Transportation of Radioactive Material under certain Conditions, as amended by Supplement no. 26, 1980, Rules and Regulations, United States Nuclear Regulatory Commission, Washington, DC 20555.

5. DIN 50 049 - Bescheinigungen uber Werkstoffprufungen (Certificates on Material Tests - Certificats des Essais Matériaux), 1972.

DISCUSSION

Question by L. B. Shappert, USA: With regard to ultra-sonic inspection of a thick wall section of nodular cask iron, how small a flaw do you feel you can find?

Answer: We all know that the detection of a flaw in a thick wall by ultra-sonic inspection depends considerably on flaw geometry, the direction of the ultra-sonic beam, the detector characteristic, etc. During tests we could find a circular flaw of 5 mm diameter (bore-hole/perpendicular direction).

Question by G. Malkoske, Canada: With regard to the prototype to be manufactured for testing, to what extent do you audit prototype package quality records prior to testing? Do you require detailed records or a certificate of compliance from the manufacturer?

Answer: A complete quality assurance programme was not set up before prototype testing in all cases, the establishment of a QA programme and the development of a package design are processes which run parallel. But quality assurance in prototype manufacturing has an important role when forming a quality standard basis by using the prototype properties as a comparison with the properties of packages manufactured later on.

Comment by R. Genevray, France: Quality assurance is a very good tool, but the most important item for quality is technical know-how (including skilled workers and satisfactory equipment).

To add an inspector .. to an inspector .. to an inspector does not seem to be a good solution. In every item there are some parts which cannot be controlled in satisfactory conditions. Therefore too much time is spent in the regions with easy access ... and the rest remains in the dark. The total cost becomes prohibitive.

It is a better solution to use the money to improve the design and the conditions of fabrication. A good complete inspection by a skilled man and, from time to time, a check by another one, seems to be satisfactory.

Answer: Naturally the most important thing in "assuring qua assurance" is the quality of manufacturing and the experience of persons who are involved in manufacturing. But the system of QA as pointed out in the paper has for many years been quite similar to the usual system of important techniques in Germany (e.g. pressure vessel manufacturing) and we are of the opinion that it at least has to be acceptable for packages of radioactive materials too.

SERVICING AND MAINTENANCE OF URANIUM HEXAFLUORIDE (UF_6) TRANSPORT CONTAINERS

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FRANCE

- INTRODUCTION

The paper that I am about to present to you describes a facility designed for industrial scale servicing and maintenance of UF_6 containers :

Type 30 B : 2278 kg 736 liters

Type 48 Y : 12500 kg 4040 liters

Type 48 Z : 13750 kg 4445 liters

CONTENTS

1 - INTRODUCTION

2 - APPLICABLE REGULATIONS

3 - DESCRIPTION

4 - QUALITY ASSURANCE

5 - CONCLUSION

The facility is located in southeastern France at the COGEMA plant in Pierrelatte. It has been designed for an initial throughput rate of 1500 to 2000 containers annually based on a normal 8-hour working day, or about 10 units of all types per day. If necessary, it will be possible to increase the throughput to 3500 or 4000 containers a year.

Construction of the facility is nearing completion, and it will be fully operational by February 1981, although it has been running at partial capacity since July 1980.

Why did COGEMA feel it necessary to set up a facility of this type, and why was it located at Pierrelatte ?

The site was selected because of the immediate proximity of two gaseous diffusion uranium enrichment plants (EURODIF and COGEMA), as well as the availability of decontamination and effluent processing facilities and qualified personnel.

The need for such an installation can be summarized in three points :

- there are some 8 000 to 10 000 such containers in France at the present time ;
- these containers are considered as pressure vessels and, as such, must be submitted to routine hydrostatic proof testing every five years ;
- UF_6 containers are liable to be subjected to harsh conditions in use or in transport, and must be inspected, assayed and reconditioned if necessary following any incident - particularly during transport.

2 - APPLICABLE REGULATIONS

I shall not burden this discussion with an exhaustive list of the regulatory measures applicable to UF_6 container servicing and maintenance, but it is worth briefly mentioning the following :

- French pressure vessel regulations,
- ANSI N14.1 - 1971,
- ORO 651, issue 4.

We are also following with considerable interest the preparation of the "International Guide for Uranium Hexafluoride Transport Containers", still at the early draft stage under the reference ISO/DP/7195, published in 1980.

However, two points may be considered as of major importance in this area in France.

- First, the decree dated March 24, 1978, and applicable since January 1980, especially Article 16 covering welding during manufacturing and repair operations, and which requires qualification of the welding process by an approved French organization.
- Second, a waiver recently granted by the French Ministry of Industry concerning the 5-year inspection requirement. This waiver stipulates

that the containers need not be emptied simply for the purpose of this inspection, which may be postponed up to 10 years at the latest, provided, that from the 5th to the 10th year the container is submitted annually to visual examination and ultrasonic thickness testing. If the container is emptied between the 5th and 10th year it must, of course, be proof tested.

3 - DESCRIPTION

I shall now briefly describe the facility and its capabilities.

The facility is designed to handle all testing, maintenance or repair operations that may be required on UF_6 containers, including the following :

- internal degassing
- external decontamination by steam cleaning or shot blasting,
- internal cleaning,
- painting,
- borescope examination,
- X-ray examination,
- ultrasonic examination of weld seams and residual thickness,
- regulatory hydrostatic proof testing,
- drying,
- valve and plug reassembly,
- leak testing,
- taring,
- all necessary appraisal or repair operations.

For these purposes, the facility is subdivided into several stations, some of which are extensively automated, interconnected by container transfer systems designed to minimize handling requirements. In chronclogical order, a standard container servicing cycle (i.e. internal and external cleaning followed by regulatory proof testing) includes the following operations.

3.1 - Internal Degassing

At a specialized work station the container is heated to about 100°C and

pumped to an internal vacuum for maximum UF_6 recovery. The container is weighed before and after the operation to check the amount of product thus removed, and compared with the original tare to determine the residual amount. This information is entered on the container log card. Degassing is mandatory prior to all servicing operations.

3.2 - Input Storage Area

The 500-container storage zone located to the north of the maintenance shop is used to prevent unnecessary personnel exposure to radiation hazards by allowing the residual activity of certain containers to decay prior to servicing, and also as a buffer storage to ensure uniform plant input. Handling provisions include two traveling cranes and a monorail system for unloading the container delivery vehicles and supplying the plant input station.

3.3 - Internal Cleaning

As I mentioned a few moments ago, statutory regulations require hydraulic proof testing at 28 bars every five years, except for containers in storage which may be tested at up to 10-year intervals.

Both in order to prevent criticality hazards and to avoid contaminating excessive amounts of water, it is essential to eliminate the heel from the containers before filling them with water for hydraulic proof testing. Experience has also shown that internal cleaning is sometimes necessary to prevent any deterioration in the quality of the material transported, or to prevent irradiation from exceeding transport standards.

The cleaning process is therefore an important one, and the cleaning facility was designed and built for maximum automatic operation. The cleaning facility basically consists of :

- an input lock,
- a handling area,
- a weighing station,
- 3 cleaning cells, each capable of accommodating 2 containers,
- a control room with master display console,
- a cell ventilation station,
- a liquid waste processing station.

The internal cleaning process uses high-pressure water sprayed from a rotating nozzle assembly automatically inserted into the container though the valve port (model 30B) or the plug orifice (model 48Y). The water sprayed through the nozzle is immediately recovered by a drain system connected to the bottom of the container, and the outflow is continuously monitored.

The chronological sequence is as follows :

a - Weighing

Based on the tare engraved on the identification plate and the actual measured weight (which should be identical with the log card entry) the inspector decides whether the container is to be processed. At present, if the difference exceeds 5 kg the container is returned to the degassing station.

b - Internal Cleaning

When weigh-station approval is received, the container is transferred on a special handling dolly to the cleaning cell. A photocell and roller system is used to rotate the container and to ensure proper operation. The operator disassembles the valve and plug, then connects the water drain system and the cleaning adapter, secured by an electromagnet. (The drain system is also used to apply suction pressure to the container in order to prevent any aerosol release on venting). Following these manual operations, the operator leaves the cell.

The actual cleaning sequence is controlled from a console outside the cell but providing full visual control of the operations. These include .

- raising the container upright for complete draining
- initiation of the wash cycle
- initiation of the rinse cycle
- lowering the container to the horizontal position.

The wash and rinse water is recovered for processing.
Absolute filters are provided in all cell ventilation lines.

After, cleaning, the operator returns to the cell, opens the container and examines it with a borescope, then temporarily closes the container and moves it on the dolly to the handling room.

The complete process has been approved by the Nuclear Safety Dept.

3.4 - Shotblasting & Painting

The containers are shotblasted for two purposes : to eliminate the existing paint and any traces of corrosion, and to remove all external contamination.

The container is transferred on the monorail from the handling area to the nearby shotblasting station. Shotblasting is fully automatic. The operator selects the coverage area as required : i.e. the entire container surface, the cylindrical portion only, or one or both end plates only.

The shot is recovered, filtered and calibrated on each cycle.
The container is then transported on the monorail to the adjacent painting station, which is also fully automatic : the operator again selects the appropriate robot mechanisms for overall or selective coverage.

After painting, the container is sent to the drying zone prior to proof testing.

3.5 - Hydrostatic Proof Testing

The proof testing facility consists basically of the following :

- a pretest storage area,
- a storage area for tested containers awaiting shipment,
- 2 ultrasonic thickness measurement devices,
- 3 hydraulic proof testers with provision for drying and reassembly of valves and plugs,
- a weighing station,
- a test water recovery system,
- a ventilation system.

The entire facility is serviced by an overhead handling system.

The test procedure includes the following steps in chronological order :

a - Ultrasonic thickness test, on an automatic device equipped with thickness measurement sensors.

. 1 tester for 48 Y containers with 28 simultaneous measurement points (21 on the cylindrical portion, 7 on the end covers) ;

. 1 tester for 30 B containers with 25 simultaneous measurement points (18 on the cylindrical portion, 7 on the end covers)

The results are output to a hard copy printer.

If the measured thickness at any point is less than 12.7 mm on 48 Y containers, or 8 mm on 30B containers, the container is scrapped.

b - Installation on the hydraulic tester. The facility includes 20 bays for 48 Y containers and 10 bays for 30B containers.

c - Borescope examination (mandatory prior to hydraulic proof testing)

d - Installation of test equipment, connection of water supply system.

e - Pressurisation to 28 bars, examination and approval stamping by the certifying engineer.

f - Draining, drying and inspection.

g - Valve and plug reassembly

h - Leak testing at 7 bars with dry deoiled air or nitrogen

i - Depressurization to 0.2 bar relative pressure.

The container is then removed from the test bay and weighed.

The new tare value is engraved on the identification plate.

The valve cover is installed, and the container is sent to the storage area awaiting shipment.

4 - QUALITY ASSURANCE

COGEMA, a wholly-owned subsidiary of the CEA, has decided on the gradual implementation of a Quality Assurance System in all of its plants and facilities The Quality Assurance Program for the UF_6 container facility is currently being drafted.

5 - CONCLUSION

I may conclude by pointing out that this servicing and maintenance facility is fully justified by the size of the French container stocks, that it is capable of meeting regulatory test requirements and that it guarantees the reliability of these containers, which are called upon to travel extensively.

This facility should be considered as a service station for COGEMA, Electricité de France, EURODIF and, of course, is also available to other clients irrespective of their nationality.

DISCUSSION

Question by E. Steinebach, The Netherlands: Please give information about the background of changing the best time for UF_6 cylinders from 5 up to 10 years because this is not in accordance with ORO and ANSI.

Answer: The containers are pressure vessels and are therefore submitted to French legislature regulations for such vessels. Periodic inspections including hydrostatic proof tests are scheduled every five years. The French Ministry has ruled that proof test intervals may be extended up to 10 years for vessels in storage which are not empty at the routine 5 year deadline. In such cases, visual and ultrasonic examinations are required on an annual basis. The profit test is mandatory after 10 years in all cases.